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Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Proposed Technical Specification Change Request Regarding Revised ANO-2 Pressure/Temperature and Low Temperature Overpressure Protection Limits for 32 Effective Full Power Years

Gentlemen:

Attached for your review and approval is a proposed change to the Arkansas Nuclear One, Unit 2 (ANO-2) Technical Specification (TS) 3.4.9, Pressure/Temperature Limits and TS 3.4.12; Low Temperature Overpressure Protection (LTOP) Limits. The primary change being requested is to update the existing pressure/temperature (P/T) limits from 21 to 32 effective full power years (EFPY) and to include additional restrictions in the LTOP technical specifications.

The current P/T limits contained in TS 3.4.9 are currently shown for 21 EFPY. However, the need to change the curves ahead of 21 EFPY was identified in Entergy letter dated June 18, 1997 (2CAN069709) which was in response to a request for additional information for Generic Letter 92-01, Revision 1; "*Reactor Vessel Structural Integrity*". Based on additional reviews of chemistry data for the ANO-2 vessel, the original chemistry data and chemistry factors were changed where the existing P/T curves in the ANO-2 TSs became less conservative. It was determined that the actual curves contained in TS 3.4.9 could only be credited through approximately 17 EFPY. After shutdown for the next refueling outage in the spring of 2002, ANO-2 will be just less than 17 EFPY.

Analyses to date had been based on the single vessel specimen which was removed at the end of cycle 2 commensurate with 1.69 EFPY. Therefore, in the ANO-2 fall 2000 outage a second vessel specimen was removed and analyzed. The results of the specimen analysis are contained in Framatome report BAW-2399, "*Analysis of Capsule W-104 Reactor Vessel Material Surveillance Program for ANO-2*" (September 2001) which is provided as Enclosure 2 to this letter. In accordance with TS SR 4.4.9.1.2, the revised specimen evaluation was used to develop new heatup/criticality, cooldown and inservice hydrostatic test curves which are contained in the proposed TS Figures 3.4-2A, 3.4-2B and 3.4-2C, respectively. The proposed change is discussed in Attachment 1 and a summary report of the fracture toughness evaluation to 32 EFPY is contained in BAW-2405, "*Appendix G Pressure-Temperature Limits for 32 EFPY, Using ASME Code Cases for ANO-2*", September 2001 which is provided in Enclosure 1.

10CFR50.60 requires that pressure/temperature (P/T) limits be established for reactor pressure vessels during normal operating and hydrostatic or leak rate testing conditions using the criteria of 10CFR50, Appendix G. Appendix G of 10CFR50 specifies that the requirements for these limits are the ASME Section XI, Appendix G limits. The new P/T analyses credit the use of Code Cases N-588 and N-640. Code Case N-641 similar to Code Case N-588, permits the postulation of a circumferentially oriented flaw (in lieu of an axially oriented flaw) for the evaluation of the circumferential welds in RPV P/T limit curves. Also, Code Case N-641, similar to Code Case N-640, permits the use of an alternate reference fracture toughness (K_{IC} fracture toughness curve instead of K_{IA} fracture toughness curve) for reactor vessel materials in determining the P/T limits. It is our understanding that Code Case N-641 has incorporated the requirements of both Code Cases N-588 and N-640 such that an exemption is only required to use Code Case N-641. Therefore, by this letter Entergy requests an exemption to 10CFR50.60 and 10CFRPart 50, Appendix G under the guidance of 10CFR50.12. The specific justification for the exemption is contained in Attachment 2.

Similar P/T analysis was performed for the LTOP conditions. The current TS vent path size of 6.38in^2, the enable temperature of 220°F, and the lift setpoint of 430 psig were assumed in the revised analysis. The LTOP transients were previously performed to account for the replacement steam generators and power uprate. These analyses demonstrated that the energy addition event was the most limiting and the peak transient pressure was determined to be 541.2 psia. To ensure that this analysis is still valid for the new P/T work, a comparison of the inputs was made. The heatup and cooldown rates remain the same and the maximum number of operating reactor coolant pumps while in LTOP conditions remain the same. Therefore, the transient analyses remain valid for 32 EFPY. In addition, Code Case N-641 was used for LTOP reanalysis.

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using the standards of 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal. Commitments identified in this submittal are contained in Attachment 3 of this letter.

As discussed above, the existing P/T curves are acceptable to approximately 17 EFPY, which will be after startup from the upcoming 2R15 refueling outage. However, even though it is acceptable to startup from the outage with the current curves, we are requesting that the proposed amendment be approved prior to the end of the outage. Therefore, Entergy Operations requests that NRC approval of this amendment request be granted by April 26, 2002 and the effective date for this change be within 60 days of approval. Although this request is neither exigent nor emergency, your prompt review is requested.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 30, 2001.

Very truly yours,

CGA/sab
Attachments
Enclosures

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PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT TWO

DOCKET NO. 50-368

Proposed Technical Specification Change Request Regarding Revised ANO-2 Pressure/Temperature and LTOP Limits for 32 EFPY

DESCRIPTION OF PROPOSED CHANGES

The proposed changes will establish new pressure/temperature (P/T) limits for 32 effective full power years (EFPY). A second reactor vessel specimen was removed and analyzed consistent with TS surveillance requirement 4.4.9.1.2 to continue to ensure that plant operations are appropriate to ensure reactor vessel fracture toughness. As a result of new specimen and RT_{NDT} analyses, new heatup, cooldown and inservice hydrostatic test curves were developed. The low temperature overpressure protection (LTOP) conditions were reanalyzed and the current technical specification (TS) conditions were confirmed. A summary report developed by Framatome for the revised fluence and fracture toughness analyses of the Arkansas Nuclear One, Unit 2 (ANO-2) reactor vessel are contained in BAW-2405, "*Appendix G Pressure-Temperature Limits for 32 EFPY, Using ASME Code Cases for ANO-2*", September 2001 which is provided in Enclosure 1. The following are the specific changes that are proposed:

Pressure/Temperature Limits

TS LCO 3.4.9.1: An editorial change to move the criticality condition to coincide with heatup condition is proposed since the criticality curve is reflected on the same figure as that for heatup.

TS LCO 3.4.9.1a: The labeling on the curves as curves A, B, C, and D which represent heatup rates of 50°F, 60°F, 70°F, and 80°F per hour respectively, has been removed from Figure 3.4-2A. Therefore, this labeling scheme is no longer appropriate in the LCO.

TS LCO 3.4.9.1b: For normal cooldown operation (Figure 3.4-2B) the following temperature dependant rates for ramped (constant) and stepped cooldown transients were used in the evaluation and are reflected in the TS LCO. These analyses are the same as those currently used except that no instrument uncertainty is being applied.

$200\ ^oF < T_C$	100^oF/hr (constant) or 50^oF in any half hour period (step)
$120\ ^oF \leq T_C \leq 200\ ^oF$	60^oF/hr (constant) or 30^oF in any half hour period (step)
$T_C < 120\ ^oF$	25^oF/hr (constant) or 12.5^oF in any half hour period (step)

TS Figures 3.4-2A, 3.4-2B and 3.4-2C have been revised to reflect new P/T limits for 32 EFPY for heatup/criticality, cooldown, and inservice hydrostatic test conditions, respectively.

These curves do not contain instrument uncertainty. Instrument uncertainty will be added to the operating limits controlled by station procedures.

The P/T Limit Bases have been modified to reflect the new P/T limits and analyses for 32 EFPY. Of particular note is the removal of the analytical results tables (Tables B 3/4.4-1 and B 3/4.4-2). This level of fracture toughness details and specific composite curve information is contained in the summary analysis of the enclosed BAW-2405 report. This information is excessive detail and is not consistent with the level of detail proposed in the Bases of NUREG 1432, Revision 2, "*Standard Technical Specifications for CE Plants"*.

LTOP Limits

LCO 3.4.12: This LCO has been modified to require that no more than one high pressure safety injection (HPSI) pump be capable of injecting into the reactor coolant system (RCS). Due to revisions in the design basis LTOP analysis for discharge through the LTOP relief valves, the remaining two HPSI pumps will be required to be in pull-to-lock to prevent injection into the RCS.

Action "e" to 3.4.12 has been added to state: "With more than one HPSI pump capable of injecting into the RCS, immediately initiate action to verify a maximum of one HPSI pump capable of injecting into the RCS." This action ensures that LCO 3.4.12 is fulfilled for design basis compliance. This TS action is consistent with the actions required of NUREG-1432 for having more than one HPSI aligned.

The LTOP system Bases has been modified to reflect the additional requirements to have only one HPSI pump available for injection and having an adequate void space in the pressurizer.

BACKGROUND

Existing TS Established P/T Limits and GL 92-01 Results

The period of applicability for the current TS P/T limits is 21 EFPY which was approved in Operating License Amendment 124 dated September 10, 1991. The current fluence extrapolation is based on the 1.69 EFPY capsule work which was removed at the end of cycle 2. The energy and spatial distribution of neutron flux in the reactor were calculated using the DOT IV 4.3 computer program. The RSIC Data Library BUGLE-80, and neutron absorption, scattering, and fission cross sections used were those supplied by this library. A summary report of the analysis of this specimen was provided in letter dated February 8, 1985, "*Reactor Vessel Surveillance Capsule Summary Report*" (2CAN028503). The second specimen was removed from the reactor vessel during 2R14on November 13, 2000 after ~15.7 EFPY of operation. The results of the second specimen analysis are contained in Framatome report BAW-2399, "*Analysis of Capsule W-104 Reactor Vessel Material Surveillance Program for ANO-2"* (September 2001) which is provided as Enclosure 2.

The ANO-2 reactor vessel is made up of six beltline plates and eight welds. These are identified as:

C8009-1, -2, -3	Intermediate Shell Plates
C8010-1, -2, -3	Lower Shell Plates
2-203-A, B, C	Intermediate Shell Longitudinal Welds
3-203-A, B, C	Lower Shell Longitudinal Welds
8-203	Upper / Intermediate Shell Girth Weld
9-203	Lower / Intermediate Shell Girth Weld

As part of the efforts to respond to Generic Letter 92-01, Revision 1, Supplement 1, "*Reactor Vessel Structural Integrity,"* a vessel specific summary of the evaluated fabrication records was developed. The chemistry properties were revised based upon new databases that were developed from the additional material copper/nickel content. The copper and nickel data for the six ANO-2 reactor vessel beltline plates was obtained from ABB-CE Report A-PENG-ER-002, Revision 0, "*The Reactor Vessel Group Records Evaluation Program Phase II Final Report for the ANO-2 Reactor Pressure Vessel Plates, Forgings, Welds, and Cladding*," October 1995. Therefore, several vessel evaluations were performed to identify the impact on current limits.

The table below provides a comparison of the best estimate original and revised copper and nickel values for the beltline materials listed above (the weld 8-203 is not included in this evaluation since the fluence to this material is lower than any of the other components).

Comparison Of Original To Revised Best-Estimate Cu/Ni Content

COMPONENT	ORIGINAL Cu (%)	ORIGINAL Ni (%)	REVISED Cu (%)	REVISED Ni (%)
C8009-1	0.12	0.63	0.098	0.605
C8009-2	0.08	0.59	0.085	0.600
C8009-3	0.08	0.60	0.096	0.580
C8010-1	0.08	0.59	0.085	0.585
C8010-2	0.07	0.66	0.083	0.668
C8010-3	0.07	0.65	0.080	0.653
2-203-A, B, C	0.05	0.18	0.046	0.082
3-203-A, B, C	0.05	0.18	0.046	0.082
9-203	0.05	0.08	0.045	0.087

The previous P/T and LTOP limits were based on plate C8009-1 being the limiting component. Two classes of components, plates, and welds were reviewed in this evaluation. The limiting component is the one with the highest adjusted reference temperature (ART) value at the end of the period under consideration at the quarter and three-quarter vessel thickness (¼T and ¾T) locations. The ART values were re-calculated in accordance with Regulatory Guide 1.99, Revision 2. Using the best-estimate chemistry values a new chemistry factor (CF) was obtained for each plate.

Using the equation provided in Regulatory Guide 1.99, Revision 2, the fluence at the ¼T and ¾T locations are 2.33E+19 n/cm^2 and 9.06E+18 n/cm^2, respectively. As a result, the ¼T fluence factor was determined to be 1.23 and the ¾T fluence factor is 0.97 at 21 EFPY. The fluence factor did not change due to the Generic Letter 92-01 effort.

Based on the new chemistry data, the limiting component for the beltline components was revised. The previous limiting plate was C8009-1; however, the limiting component is now shown to be plate C8010-1. In addition, revised chemistry factors were determined for each beltline weld. The revised chemistry factors for the welds were lower than the original chemistry factors. As such, the welds were not determined to be the limiting component of the reactor vessel. As a result, the period of applicability had to be revised from 21 EFPY to ~17 EFPY.

Existing ANO-2 LTOP Analysis and Need for Change

The ANO-2 LTOP system consists of a single discharge header from the pressurizer which feeds two redundant pressure relief valves each with two upstream motor operated isolation block valves. The relief valves are operator aligned to the RCS during cooldown by opening the isolation valves and conversely are isolated during heatup by closing the isolation block valves. The LTOP isolation valves provide the RCS boundary in operating modes 1, 2 and 3. Each LTOP relief valve provides a minimum of 6.38 in^2 opening which has the capacity to relieve the design basis event. The relief valve setting is 430 psig (445 psia). An alarm will notify the operator if the temperature drops below the enable temperature during cooldown and the isolation valves are not fully open.

In Operating License Amendment 180 dated March 7, 1987 the LTOP restrictions were added as a new TS 3.4.12 per the guidance of Generic Letter 90-06, "*Power-Operated Relief Valve and Block Valve Reliability*" and Generic Issue 94, "*Additional Low-Temperature Overpressure Protection*". The LTOP design basis event was a simultaneous injection of two HPSI pumps and all three charging pumps to the water-solid RCS. The analyses assumed that the safety injection tanks were either isolated or depressurized.

Subsequent to that change, in Entergy letter dated December 21, 1999 (2CAN129907), Entergy notified the NRC of a change to the Bases of TS 3.4.12. The change to ANO-2 TS Bases revised the limiting design basis event for this system. As a result of installation of the replacement steam generators in the fall of 2000 (2R14) and the implementation of changes for power uprate, the design basis transients that determine LTOP system requirements were changed. The design basis event analyses considered two postulated limiting overpressure events:

1) mass addition (simultaneous injection of two HPSI pumps and all three charging pumps to a water-solid RCS) and,

2) energy addition (the start of an idle reactor coolant pump, under water solid conditions, with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperature)

These events assumed the most limiting operating conditions and system configurations. The energy addition event produced the highest peak pressure. Therefore, the TS Bases were modified to reflect this new event.

During the fall 2000 refueling outage (2R14), Entergy identified a concern that the back pressure that would occur at the discharge of a LTOP relief valve when the LTOP valve lifted could under some operating conditions be a higher back pressure than was assumed in the LTOP analysis. Specifically, when the LTOP is in service (the upstream isolation valves open) and with pressurizer temperatures elevated, flashing could occur in the LTOP relief valve discharge line that would increase the discharge pressure above the original analysis assumed value. The results of the new analysis indicated that two phase flow would occur in the discharge line, which results in higher flow losses in the line to the Quench Tank than were previously analyzed. This condition was documented in a site condition report. It was determined that the existing relief valve bellows and the operational characteristics of the LTOP valves were not sufficient for the flowrate requirements credited in the LTOP analyses without additional operational restrictions. Administrative controls for controlling HPSI pumps and other precautionary measures were established to ensure adequate operation of the relief valves during a potential LTOP design basis event.

New relief valve bellows are being installed in 2R15 to ensure long term mitigation of the design basis event.

DISCUSSION OF CHANGE

Evaluation of the Second ANO-2 Reactor Vessel Specimen

10CFR50, Appendix H, *"Reactor Vessel Material Surveillance Program Requirements,"* defines the material surveillance program required to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment. Fracture toughness test data are obtained from material specimens contained in capsules that are periodically withdrawn from the reactor vessel. These data permit determination of the conditions under which the vessel can be operated with adequate safety margins against non-ductile fracture throughout its service life.

The reactor vessel surveillance program for ANO- 2 included six capsules designed to monitor the effects of neutron and thermal environment on the materials of the reactor pressure vessel core region. The capsules, which were inserted into the reactor vessel before initial plant startup, were positioned inside the reactor vessel between the core support barrel and the vessel wall. Capsule W-104 was irradiated in the 104° position during the time of irradiation in the reactor vessel (cycles 1 through 14).

Capsule W-104 was removed during the 2R14-refueling outage (November of 2000). The capsule contained Charpy V-notch (CVN) impact test specimens fabricated from one base metal plate (SA-533, Grade B, Class 1), heat-affected-zone (HAZ) material, a weld metal representative of the ANO- 2 reactor vessel beltline region intermediate and lower shell longitudinal welds, and a Standard Reference Material (SRM). The SRM is a standard heat of SA-533, Grade B, Class 1 material. The tensile test specimens were fabricated from the same base metal plate, HAZ, and weld metal. The number of specimens of each material contained in Capsule W-104, the chemical compositions of the surveillance materials, and the location of the individual specimens within the capsule is contained in Framatome report BAW-2399, "*Analysis of Capsule W-104 Reactor Vessel Material Surveillance Program for ANO-2*" (September 2001) which is provided as Enclosure 2. This report is provided in accordance with 10CFR50, Appendix H.

10CFR50, Appendix G, also requires a minimum initial Charpy upper shelf energy (C_vUSE) of 75 ft-lbs for all beltline region materials unless it is demonstrated that lower values of upper-shelf fracture energy will provide an adequate margin of safety against fracture equivalent to those required by ASME Section XI, Appendix G. No action is required for a material that does not meet the initial 75 ft-lbs requirement provided that the irradiation embrittlement does not cause the C_vUSE to drop below 50 ft-lbs. The results of the capsule analysis for CvUSE to 32 EFPY confirmed that the 50 ft-lbs minimum requirement was met.

Fluence Evaluation Process under BAW-2241

Framatome ANP, Inc. has developed a calculational based fluence analysis methodology in Topical Report BAW-2241P-A, Revision 1, "*Fluence and Uncertainty Methodology*" dated April, 1999 which closely predicts the fast neutron fluence in the reactor vessel using surveillance capsule dosimetry to verify the fluence predictions. This methodology was developed through a full-scale benchmark experiment which demonstrated that the accuracy of a fluence analysis that employs the Framatome ANP methodology would be unbiased and have a precision well within the Regulatory Guide 1.190 limit of 20%.

The NRC safety evaluation report (SER) to BAW-2241P-A, Revision 1 concluded that "the proposed methodology is acceptable for referencing in licensing applications for determining the pressure vessel fluence of Westinghouse, CE, and B&W designed reactors." In addition, there were three limitations imposed in the SER for Revision 0 of the topical. These limitations involved analysis of reactor designs not included in the 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. As discussed in the Enclosure 2, the approach performed for ANO-2 is consistent with the Framatome ANP methodology and therefore meets the limitations described in the NRC SER.

The Framatome ANP methodology was used to calculate the neutron fluence exposure to the 104º (W-104) capsule of the ANO Unit 2 nuclear reactor. The methodology was also used to estimate fluences on the inner surface of the reactor vessel, as well as at specified weld locations on the vessel surface. The fast neutron fluence (E>1 MeV) at each location was calculated in accordance with the requirements of Regulatory Guide 1.190. The energy-dependent flux on the capsule was used to determine the calculated activity of each dosimeter. Neutron transport calculations in two-dimensional geometry were used to obtain energy dependent flux distributions throughout the core. Reactor conditions were representative of an average over the cycle 1-9 and 10-14 irradiation periods. These periods were separated in order to adequately represent a water temperature reduction that occurred between cycles 9 and 10. Geometric detail was selected to explicitly represent the dosimeter holder and the reactor vessel.

The dosimetry of the W-104 capsule was located in the reactor for a total irradiation time of 5726 effective full power days (EFPDs) for cycles 1-14. The rated thermal power for the fourteen cycles was 2815 MWt. The fluence must be determined for the center of the dosimetry capsule in order to allow for analysis of the Charpy and tensile specimens. As a result, the entire cycle 1-14 analysis results in a maximum capsule fluence of 2.937E+19 n/cm. Based on an extrapolated flux incorporating power uprate conditions, the projected end-of-life (32 EFPY) peak fast fluence of the ANO-2 reactor vessel beltline region clad surface was determined to be 3.791E+19 n/cm^2.

Changes to the Pressure/Temperature TSs for 32 EFPY

10CFR50, Appendix G, *"Fracture Toughness Requirements,"* specifies minimum fracture toughness requirements for the ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary (RCPB) of light water-cooled power reactors and provides specific guidelines for determining the pressure-temperature limitations. The fracture toughness and operational requirements are specified to provide adequate safety margins during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{Ia} or K_{Ic} curve as applicable). The K_{Ia} curve appears in Appendix G of ASME Code Section XI. ASME Code Case N-641 permits the use of the K_{Ic} curve as given in Appendix A of ASME Code Section XI. When a given material is indexed to the K_{Ic} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Plant operating limits can then be determined using these allowable stress intensity factors.

A method for guarding against non-ductile fracture in reactor vessels is described in Appendix G to Section III of the ASME Code, *"Nuclear Power Plant Components"* and Appendix G to Section XI, *"Rules for Inservice Inspection"*. The application of Appendix G to the ASME Code is established in the requirements of 10CFR50, Appendix G. This method uses fracture mechanics concepts and the reference nil-ductility temperature, RT_{NDT}, which is defined as the greater of the drop weight NDT temperature (in accordance with ASTM E 208-81, "*Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels*") or the temperature that is 60°F below that at which the material exhibits 50 ft-lbs and 35 mils lateral expansion. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve), which appears in Appendix G of Section III and Section XI of ASME Code. The K_{IR} curve is a lower bound of dynamic and crack arrest fracture toughness data obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for the material as a function of temperature. The operating limits can then be determined using these allowable stress intensity factors.

The RT_{NDT} and, in turn, the operating limits, are adjusted to account for the effects of irradiation on the fracture toughness of the reactor vessel materials. The irradiation embrittlement and the resultant changes in mechanical properties of a given pressure vessel steel can be monitored by a surveillance program in which surveillance capsules containing prepared specimens of the reactor vessel materials are periodically removed from the operating nuclear reactor and the specimens are tested. The increase in the Charpy V-notch 30 ft-lb temperature is added to the original RT_{NDT} to adjust it for irradiation embrittlement. The adjusted RT_{NDT} is used to index the material to the K_{IR} curve which, in turn, is used to set operating limits for the nuclear power plant. These new limits take into account the effects of irradiation on the reactor vessel materials.

The ART for the reactor vessel beltline region materials is calculated in accordance with Regulatory Guide 1.99, Revision 2. The ART is calculated by adding the initial RT_{NDT}, the predicted radiation-induced shift in RT_{NDT} (ΔRT_{NDT}), and a margin term to cover the uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and the calculational procedures. The predicted radiation induced ΔRT_{NDT} is calculated using the respective reactor vessel beltline materials copper and nickel contents and the neutron fluence applicable to 32 EFPY including an estimated increase in flux due to a proposed power uprate. The ¼T and ¾T wall locations for each beltline material are determined by adding the minimum thickness of the cladding to the distance into the base metal at the ¼T and ¾T locations. The ¼T and ¾T ART results for the ANO-2 reactor vessel beltline region materials applicable to 32 EFPY are presented in the enclosed Framatome Report BAW-2405. Based on these results, the controlling beltline material for the ANO-2 reactor vessel is the lower shell plate C-8010-1. The applicability of 32 EFPY is also consistent with the removal schedule for the next capsule at 30 EFPY as shown in ANO-2 SAR Table 5.2-12.

Regulatory Guide 1.99, Revision 2, Position 2.1 requires that if there is evidence that the copper/nickel content of the surveillance specimen differs from that of the vessel, the measured values of ΔRT_{NDT} should be adjusted by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance specimen. The surveillance data would be fitted to obtain the relationship of ΔRT_{NDT} to fluence by calculating the chemistry factor for the best fit by ratioing each adjusted ΔRT_{NDT} by its corresponding fluence factor. The results of the second specimen evaluation has been performed in accordance with this position and is reported in Table 3-2 of the enclosed BAW-2405 report.

For each analyzed transient and steady state condition, the allowable pressure is determined as a function of reactor coolant temperature considering postulated flaws in the reactor vessel beltline, inlet nozzle, outlet nozzle, and closure head. In the beltline region, flaws are presumed to be present at the ¼T and ¾T locations of the controlling material (shell plate or weld), as defined by the fluence adjusted RT_{NDT}.

The P/T curves provided in this proposed amendment are adjusted for sensor location but does not include instrument uncertainty. Protection against non-ductile failure is ensured by using these curves to limit the reactor coolant pressure. The P/T limits for normal heatup (including criticality core limits) at 32 EFPY are provided on the revised TS Figure 3.4-2A. The criticality limit temperature is 190°F, which is based on the RT_{NDT} of the closure flange material (30°F) plus 160°F which is larger than the 175°F value that corresponds to the inservice hydrostatic test limit pressure of 2500 psig. Considering all the ramp and step cooldown transient scenarios, composite cooldown P/T limits for cooldown are shown in TS Figure 3.4-2B. The inservice hydrostatic test P/T limits are shown in TS Figure 3.4-2C. Acceptable pressure and temperature combinations for reactor operation are below and to the right of the pressure-temperature limit curves.

For normal heatup operation, TS Figure 3.4-2A, four ramped heatup transient conditions are considered in the evaluation. These transient conditions are simulated by increasing the reactor coolant system (RCS) cold leg temperature from 50°F to 560°F at constant rates of 50, 60, 70 and 80°F/hr. The inservice hydrostatic test condition contained in TS Figure 3.4-2C is also evaluated using the RCS cold leg temperature ranges, at a ramp rate of 10°F/hr.

For normal cooldown operation, Figure 3.4-2B, the following temperature dependant rates for ramped and stepped cooldown transients are considered in the evaluation. Except for these cooldown rates not including a margin for instrument uncertainty they are identical to those used in the current TS P/T curves contained in the TSs. The assumptions also include that only two RCPs are in operation while in the LTOP region of the curves consistent with the current TSs.

Actual RCS Cold Leg Temperature	Maximum Cooldown Rate
$200°F < T_C$	100°F/hr (constant) or 50°F in any half hour period (step)
$120°F \le T_C \le 200°F$	60°F/hr (constant) or 30°F in any half hour period (step)
$T_C < 120°F$	25°F/hr (constant) or 12.5°F in any half hour period (step)

The above approach utilizes the methodology of Code Case N-588, "*Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels*" (December 12, 1997) and Code Case N-640, "*Alternative Reference Fracture Toughness for Development of P-T Curves*" (February 26, 1999). As discussed below, these code cases have been incorporated into Code Case N-641, "*Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements*" (January 27, 2000).

Changes to LTOP Protection TSs for 32 EFPY

The isothermal pressure/temperature results developed for K_{Ic} measure of fracture toughness is used to develop LTOP P/T limits. The approach for determining new LTOP limits also utilizes Code Case N-641. The LTOP systems shall be effective at coolant temperatures less than 200°F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT}

+ 50°F, whichever is greater. Since the RT_{NDT} of the controlling beltline material is 113°F, the metal temperature at the ¼T depth from the inside surface of the beltline region is RT_{NDT} + 50°F or 163°F. During normal plant heatup the metal temperature is lower and lags the coolant temperature. The maximum temperature difference occurs during the maximum plant heatup rate at 80°F/hr when the corresponding coolant temperature is 186.4°F. The minimum LTOP enable temperature for ANO-2 is therefore the greater of 186.4°F or 200°F. Per Code Case N-641, the LTOP system must also limit the maximum pressure in the vessel to 100% of the pressure associated with the P/T limits when K_{IC} is used for fracture toughness.

The LTOP transients were performed to account for the replacement steam generators and power uprate. The current TS vent path size of 6.38in^2, the enable temperature of 220°F (includes instrument uncertainty), and the lift setpoint of 430 psig were assumed in the analysis. These analyses demonstrate that the energy addition event is the most limiting with the peak transient pressure being 541.2 psia. To ensure that this analysis is still valid for the new P/T work, a comparison of the inputs was made. The heatup and cooldown rates remain the same and the maximum number of operating RCPs while in LTOP conditions remain the same (2 RCPs). The transient analysis remains valid.

The LTOP PT limits have changed due to the vessel P/T work. The new minimum LTOP P/T limit is 607.7 psia using K_{IC} methodology. Therefore, the current enable temperature, vent path size and lift setpoint of the relief valves remain valid in the current TSs.

However, as a result of a condition identified in the last outage dealing with backpressure on the relief valve discharge, additional accident assumptions are necessary to mitigate a design basis LTOP event. The existing relief valve bellows were determined to not allow sufficient discharge without damaging the bellows due to backpressure on the valves. As a result, a new bellows design will be installed in 2R15. Even with the installation of new bellows, the modification will not restore the valves to the original design assumptions used in the LTOP analyses. Therefore, two additional operator actions need to be credited for the LTOP valve design which are reflected in the TS Bases. The credited operator actions are:

1) Operating restriction to assure two HPSI pumps are in pull-to-lock while LTOP conditions are enabled, and
2) Operating restriction to assure that the pressurizer water volume is less than 910 ft^3 when starting a reactor coolant pump.

The first operating restriction for HPSI pumps has historically been under procedural control even though not previously required by analysis. The second limitation on pressurizer level is only associated with the starting of an RCP while in LTOP conditions. Even though this action has not been explicitly defined in procedures for this mode, the retention of a pressurizer bubble is practiced. These two new operator actions will assure acceptable results under LTOP conditions. The ANO procedures will be revised to clarify the use of these two restrictions upon replacement of the relief valve bellows.

Therefore, LTOP TS 3.4.12 is being modified to credit the requirement to only have one HPSI pump available for injection. This action had been previously prescribed in the standard TS per GL 90-06, but was not assumed necessary when the original LTOP TS were proposed and issued in Operating License Amendment 180. The proposed changes for restricting HPSI pumps are also in accordance with the guidance of NUREG-1432, Revision 2. The operating restriction to limit the water volume in the pressurizer to less than 910 ft^3 will be controlled by procedural restrictions and documented in the Bases of the TS as an operator action in support of the relief valve design basis. This action is associated with RCP starts while in the LTOP condition and is not part of the standard TSs.

Evaluation of Pressurized Thermal Shock

A pressurized thermal shock (PTS) evaluation for the ANO-2 reactor vessel beltline materials was performed in accordance with 10CFR50.61. The results of the PTS evaluation are shown in Table 4-1 of the enclosed BAW-2405 topical report. These results demonstrate that the ANO-2 reactor vessel beltline materials will not exceed the PTS screening criteria before 32 EFPY. The controlling beltline material for the ANO-2 reactor vessel with respect to PTS is the lower shell plate C-8010-1, with a RT_{PTS} value of 118.8°F that is well below the PTS screening criterion of 270°F.

Exemption to 10CFR50.12 for Application of Code Case N-641

10CFR50.60 requires that pressure/temperature (P/T) limits be established for reactor pressure vessels during normal operating and hydrostatic or leak rate testing conditions. Specifically, 10CFRPart 50, Appendix G, states that "The appropriate requirements on both the pressure/temperature limits and the minimum permissible temperature must be met for all conditions." Appendix G of 10 CFR Part 50 specifies that the requirements for these limits are the ASME, Section XI, Appendix G limits.

Code Case N-641, similar to Code Case N-588 permits the postulation of a circumferentially oriented flaw (in lieu of an axially oriented flaw) for the evaluation of the circumferential welds in RPV P/T limit curves. Also, Code Case N-641 similar to Code Case N-640 permits the use of an alternate reference fracture toughness (K_{IC} fracture toughness curve instead of K_{IA} fracture toughness curve) for reactor vessel materials in determining the P/T limits. Since the pressure stresses on a circumferentially oriented flaw are lower than the pressure stresses on an axially oriented flaw by a factor of 2, postulating a circumferentially oriented flaw for the evaluation of the circumferential welds (as permitted by Code Case N-641) in establishing the P/T limits would be more realistic than the methodology currently endorsed by 10 CFR Part 50, Appendix G. Therefore, an exemption per 10CFR50.12 to allow the use of Code Case N-641 is required in lieu of the requirements of 10CFR50.60. The specific justification for the exemption is contained in Attachment 2.

Similar exemptions for the use of Code Case N-641 have been granted by the NRC for North Anna Power Station in May of 2001 and for Point Beach Nuclear Plant in October of 2000.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Entergy Operations, Inc. is proposing that the Arkansas Nuclear One Unit 2 (ANO-2) Operating License be amended to modify the pressure/temperature (P/T) limits from 21 to 32 effective full power years (EFPY) as reflected in ANO-2 Technical Specification 3.4.9. This includes changing the heatup/criticality, cooldown, and inservice hydrostatic pressure test operational curves. A vessel specimen was retrieved during the last ANO-2 refueling outage and tested in accordance with acceptable industry standards for determining the vessel nil ductility of the irradiated vessel plates. The new limits consider updated vessel chemistry data as a result of further information gained in the resolution of Generic Letter 92-01, Revision 1, Supplement 1. The new chemistry data is more conservative than that established for the initial curve development and the results from the testing of the recently pulled vessel specimen. Analyses were performed in accordance with Regulatory Guide 1.99, Revision 2 using recent code cases for performing fracture toughness.

In addition, transient analyses were performed for the low temperature overpressure (LTOP) operating region and were determined to be bounded by the existing analysis. However, based on the restriction of the LTOP system to perform full design discharge, operator actions must be credited to mitigate the event. This includes operator action to only allow one high pressure safety injection (HPSI) pump available for injection. Therefore, the limiting condition for operation for TS 3.4.12 and associated action statements have been modified to include HPSI pump restrictions.

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the criteria in 10CFR50.92(c). A discussion of these criteria as they relate to this amendment request follows:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The probability of occurrence of an accident previously evaluated for ANO-2 is not altered by the proposed amendment to the technical specifications (TSs). The accidents remain the same as currently analyzed in the ANO-2 Safety Analysis Report (SAR) as a result of changes to the P/T limits as well as those for LTOP. The new P/T and LTOP limits were based on NRC accepted methodologies along with ASME Code alternatives. The proposed changes do not impact the integrity of the reactor coolant pressure boundary (RCPB) (i.e. there is no change to the operating pressure, materials, loadings, etc.) as a result of this change. In addition, there is no increase in the potential for the occurrence of a loss of coolant accident. The probability of any design basis accident is not affected by this change, nor are the consequences of any design basis accident (DBA) affected by this proposed change. The proposed P/T limit curves and the LTOP limits are not considered to be an initiator or contributor to any accident currently evaluated in the ANO-2 SAR. These new limits ensure the long term integrity of the RCPB.

Fracture toughness test data are obtained from material specimens contained in capsules that are periodically withdrawn from the reactor vessel. These data permit determination of the conditions under which the vessel can be operated with adequate safety margins against non-ductile fracture throughout its service life. A new reactor vessel specimen capsule was withdrawn at the most recent refueling outage and was analyzed to predict the fracture toughness requirements using projected neutron fluence calculations. For each analyzed transient and steady state condition, the allowable pressure is determined as a function of reactor coolant temperature considering postulated flaws in the reactor vessel beltline, inlet nozzle, outlet nozzle, and closure head.

The predicted radiation induced ΔRT_{NDT} was calculated using the respective reactor vessel beltline materials copper and nickel contents and the neutron fluence applicable to 32 EFPY including an estimated increase in flux due to a proposed power uprate. The RT_{NDT} and, in turn, the operating limits for ANO-2 were adjusted to account for the effects of irradiation on the fracture toughness of the reactor vessel materials. Therefore, new operating limits are established which are represented in the revised operating curves for heatup/criticality, cooldown and inservice hydrostatic testing contained in the technical specifications.

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

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed changes to the P/T and LTOP limits will not create a new accident scenario. The requirements to have P/T and LTOP protection are part of the licensing basis of ANO-2. The proposed changes reflect the change in vessel material properties acknowledged and managed by regulation and the best data available in response to NRC Generic Letter 92-01, Revision 1. The approach used meets NRC and ASME regulations and guidelines. The calculational methodology for fluence is based on an NRC approved Framatome ANP approach. Therefore, the adjusted reference temperatures for fracture toughness are consistent with that previously provided to the NRC. The data analysis for the vessel specimen removed at 2R14 (approximately 15.7 EFPY of exposure) confirms that the vessel materials are responding as predicted.

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

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

The existing P/T curves and LTOP limits in the technical specifications are reaching their expiration period for the number of years at effective full power operation. The revision of the P/T limits and curves will ensure that ANO-2 continues to operate within the operating margins allowed by 10CFR50.60 and the ASME Code. The material properties used in the analysis are based on results established through CE material reports for copper and nickel content. The application of ASME Code Case N-641 presents alternative procedures for calculating P/T and LTOP temperatures and pressures in lieu of that established for ASME Section XI, Appendix G-2215. This Code alternative allows certain assumptions to be conservatively reduced. However, the procedures allowed by Code Case N-641 still provide significant conservatism and ensure an adequate margin of safety in the development of P/T operating and pressure test limits to prevent non-ductile fractures.

Therefore, this change does <u>not</u> involve a significant reduction in the margin of safety.

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does <u>not</u> involve a significant hazards consideration.

ENVIRONMENTAL IMPACT EVALUATION

10CFR51.22(c) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released off-site, or (3) result in a significant increase in individual or cumulative occupational radiation exposure. Entergy Operations, Inc. has reviewed this license amendment and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9). Pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the evaluation.

2. As discussed in the significant hazards evaluation, this change does not result in a significant change or significant increase in the radiological doses for any Design Basis Accident. The proposed license amendment does not result in a significant

change in the types or a significant increase in the amounts of any effluents that may be released off-site.

3. The proposed license amendment does not result in a significant increase to the individual or cumulative occupational radiation exposure because the proposed change to update the operating P/T and LTOP limits do not impact the exposure of plant personnel.

Justification for ASME Section XI Code Case N-641 Exemption Request

The following information provides the basis for the exemption request to 10CFR50.60 for use of ASME Section XI Code Case N-641, "*Alternative Pressure/Temperature Relationship and Low Temperature Overpressure Protection System Requirements, Section XI, Division I",* in lieu of the methods specified in 10CFR50, Appendix G.

The requested exemption will allow use of ASME Code Case N-641 to (a) determine stress intensity factors for postulated circumferential defects in circumferential welds, and for postulated axial defects in plates, forgings, and axial welds; and (b) use the K_{IC} fracture toughness curve shown on ASME XI, Appendix A, Figure A-2200-1, in lieu of the K_{IA} fracture toughness curve of ASME XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness.

10CFR50.12 states that the Commission may grant an exemption from requirements contained in 10CFR50 provided that:

1. The requested exemption is authorized by law:

   No law exists which precludes the activities covered by this exemption request. 10CFR50.60(b) allows the use of alternatives to 10CFR50, Appendices G and H when an exemption is granted by the Commission under 10CFR50.12.

2. The requested exemption does not present an undue risk to the public health and safety :

10CFR50, Appendix G, requires, in part, that Article G-2120 of ASME Section XI, Appendix G, be used to determine the maximum postulated defects in reactor pressure vessels when determining pressure/temperature limits for the vessel. These limits are determined for normal operation and pressure test conditions.

Article G-2120 specifies, in part, that the postulated defect be in the surface of the vessel material and normal to the direction of maximum stress. ASME Section XI, Appendix G, also provides a methodology to determine the stress intensity factors for a maximum postulated defect normal to the maximum stress. The purpose of this article is, in part, to prevent non-ductile fractures by providing procedures to identify the most limiting postulated fractures to be considered in the development of pressure/temperature(P/T) limits.

Due to progress made in NDE techniques over the last thirty years, it is very unlikely to have large, undetected defects present in the beltline region of reactor vessels. It is further unlikely to have axial cracks originating from a circumferential weld perpendicular to the weld seam orientation in reactor vessels. Both experience and engineering studies indicate that the primary degradation mechanism affecting the beltline region of the reactor vessel is neutron embrittlement. No other service induced degradation mechanism exists at a pressurized water reactor to cause a prior existing defect located in the beltline region of the reactor vessel to grow while in service. Based on these considerations, and the fact that the pressure/temperature limit for reactor operation is the limiting pressure for any of the materials in the vessel, it is not necessary to include additional conservatism in the assumed flaw orientation for circumferential welds. ASME Section XI, Code Case N-641, and a previous Section XI, Appendix G Code change correct this inconsistency in assumed flaw orientation for circumferential welds in vessels when calculating operating P/T limits.

Code Case N-641 provides benefits in terms of calculating P/T limits by revising the Section XI, Appendix G reference flaw orientation for circumferential welds in reactor vessels. The reference flaw is a postulated flaw that accounts for the possibility of a prior existing defect that may have gone undetected during the fabrication process. When considering a reference flaw with respect to a weld, the reference flaw would represent any prior existing defect that may have been introduced during fabrication. Thus, the intended application of a reference flaw is to account for prior existing defects that could physically exist within the geometry of the weldment. The currently endorsed ASME Section XI, Appendix G approach mandates consideration of an axial reference flaw in circumferential welds for purposes of calculating P/T limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the length of the flaw is 1.5 times the vessel thickness, which is much longer than the width of the reactor vessel girth weld. The possibility that an axial flaw may extend from a circumferential weld into a plate/forging or axial weld is already adequately covered by the requirement that axial defects be postulated in plates/forging and axial welds.

ASME Code Case N-641 reflects fabrication and NDE experience by allowing consideration of maximum postulated defects oriented circumferentially within the welds. Code Case N-641 also provides appropriate procedures to determine limiting circumferential weld defects and associated stress intensity factors for use in developing P/T limits per ASME Section XI, Appendix G procedures. The procedures allowed by Code Case N-641 are conservative and provide a margin of safety in the development of P/T operating and pressure test limits which will prevent nonductile fractures.

The revised P/T limits and LTOP limits being proposed for ANO-2 have been developed using the K_{IC} fracture toughness curve shown on ASME Section XI, Appendix A, Figure A-2200-1, in lieu of the K_{IA} fracture toughness curve of ASME XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. Use of the K_{IC} curve in determining the lower bound fracture toughness in the development of P/T operating limits curve is more technically correct than the K_{IA} curve. The K_{IC} curve models the slow heatup and cooldown process of a reactor vessel.

Use of this approach is justified by the initial conservatism of the K_{IA} curve when the curve was codified in 1974. This initial conservatism was necessary due to limited knowledge of reactor pressure vessel materials over time and usage. Since 1974, additional knowledge has been gained about the affect of usage on reactor pressure vessel materials. The additional knowledge demonstrates the lower bound on fracture toughness provided by the KI curve provides a margin of safety that is adequate to protect the public health and safety from potential reactor pressure vessel failure.

The LTOP analyses for ANO-2 were also performed using the method provided in Code Case N-641. Use of the Code Case N-641 methodology in the determination of the LTOP conditions is more technically correct than the generic value included in earlier versions of ASME Section XI and eliminates inconsistencies in the margin of safety between reactor vessels of various geometries.

Code Case N-641 provides bounding reactor vessel low temperature integrity protection during LTOP design basis transients. The LTOP lift setpoint utilizes 100% of the pressure determined to satisfy Appendix G, paragraph G-2215 of ASME Section XI, Division 1, as a design limit. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor vessel failure.

P/T curves based on Code Case N-641 will enhance overall plant safety by opening the pressure/temperature operating window with the greatest safety benefit in the region of low temperature operations. The primary safety benefit in opening the low temperature operating window is a reduction in the challenges to LTOP valves.

The proposed P/T limits include restrictions on allowable operating conditions and equipment operability requirements to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure/temperature limits specified in TS 3.4.9. Therefore, this exemption does not present an undue risk to the public health and safety.

3. The requested exemption will not endanger the common defense and security:

   The common defense and security are not endangered by this exemption request.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10CFR50.60:

   Pursuant to 10CFR50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This exemption meets the special circumstances of paragraphs:

10CFR50.12(a)(2)(ii) demonstrates that the underlying purpose of the regulation will continue to be achieved; (a)(2)(iii) would result in undue hardship or other cost that are significant if the regulation is enforced and; (a)(2)(v) will provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulations.

<u>10CFR50.12(a)(2)(ii), Underlying Purpose Of The Regulation Will Continue To Be Achieved:</u> The underlying purpose of 10CFR50, Appendix G and ASME XI, Appendix G, is to satisfy the requirement that the reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure that when stressed the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized, and the P/T operating and test curves provide margin in consideration of uncertainties in determining the effects of irradiation on material properties.

Application of Code Case N-641 to determine P/T operating and hydrostatic test limit curves per ASME Section XI, Appendix G, provides appropriate procedures to determine limiting maximum postulated defects and considering those defects in the P/T limits. This application of the code case maintains the margin of safety originally contemplated for reactor pressure vessel materials.

Therefore, use of Code Case N-641, as described above, satisfies the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable level of safety.

<u>10CFR50.12(a)(2)(iii), Result In Undue Hardship Or Other Cost:</u> The P/T operating window is defined by the P/T operating and test curves developed in accordance with the ASME Section XI, Appendix G procedure. Continued operation with these more restrictive P/T curves without the relief provided by ASME Code Case N-641 would unnecessarily restrict the pressure/ temperature and LTOP operating window for ANO-2. Use of Code Case N-641 will minimize the potential for RCP impeller cavitation wear while operating in the LTOP region and reduce the potential for undesired actuation of LTOP relief valves. Use of ASME Code Case N-641 in the development of the proposed P/T curves and LTOP setpoint and enable temperature alleviates any unnecessary burden. Implementation of the proposed P/T curves and LTOP parameters as allowed by ASME Code Case N-641 does not reduce the margin of safety originally contemplated by either the NRC or ASME.

Compliance with the specified requirements of 10CFR50.60 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-641 allows:

- Postulation of a circumferential defect in circumferential welds is appropriate in lieu of requiring the defect to be oriented across the weld from one plate or forging to the adjoining plate or forging. This circumstance was not considered at the time ASME Section XI, Appendix G was developed and imposes restrictions on P/T operating limits beyond those originally contemplated.

- A reduction in the fracture toughness lower bound is appropriate in lieu of the ASME Section XI, Appendix G, in the determination of reactor coolant pressure/temperature limits. This proposed alternative is acceptable because the Code Case maintains the relative margin of safety commensurate with that which existed at the time ASME Section XI, Appendix G, was approved in 1974. Therefore, application of Code Case N-641 for ANO-2 will ensure an acceptable margin of safety. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor vessel failure.

<u>10CFR50.12(a)(2)(v), Licensee Has Made Good Faith Efforts To Comply With The Regulations.:</u> The exemption provides only temporary relief from the applicable regulation and Entergy has made a good faith effort to comply with the regulation. We request that the exemption be granted until such time that the NRC generically approves ASME Code Case N-641 for use by the nuclear industry. However, to retain sufficient P/T operating margin to the end of the proposed ANO-2 P/T limits, we require an exemption to use Code Case N-641.

Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure/temperature limits specified in the proposed amendment. Therefore, this exemption does not present an undue risk to the public health and safety.

In aggregate, these proposed alternatives are acceptable because the Code Case maintains the relative margin of safety commensurate with that which existed at the time ASME Section XI, Appendix G, was approved in 1974. Therefore, application of Code Case N-641 for ANO-2 will ensure an acceptable margin of safety. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor vessel failure.

Commitments Contained in this Letter

Commitment	Type		Scheduled Completion Date (If Required)
	One Time Action	Continuing Compliance	
Entergy will replace the existing LTOP relief valves bellows with bellows having improved discharge capability	X		2R15 (Spring 2002)
The following restrictions will be further clarified in the ANO-2 Heatup and Cooldown procedures upon replacement of the relief valve bellows: 1) assure two HPSI pumps are in pull-to-lock while LTOP conditions are enabled, and 2) assure that the pressurizer water volume is less than 910 ft^3 when starting a reactor coolant pump.		X	2R15 (Spring 2002)
Instrument uncertainty for the new pressure/temperature limits in TS Figures 3.4-2A, 3.4-2B, and 3.4-2C will be added in the figures contained in station procedures.		X	2R15 (Spring 2002)

MARKUP OF PROPOSED TECHNICAL SPECIFICATION CHANGES

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

b. A maximum cooldown rate based on:

RCS Temperature (Tc)	Maximum Cooldown Rate
T_C > ~~220~~200°F	100°F per hour (constant) or 50°F in any half hour period (step)
~~140~~120°F ≤ T_C ≤ ~~220~~200°F	60°F per hour (constant) or 30°F in any half hour period (step)
T_C < ~~140~~120°F	25°F per hour (constant) or 12.5°F in any half hour period (step)

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

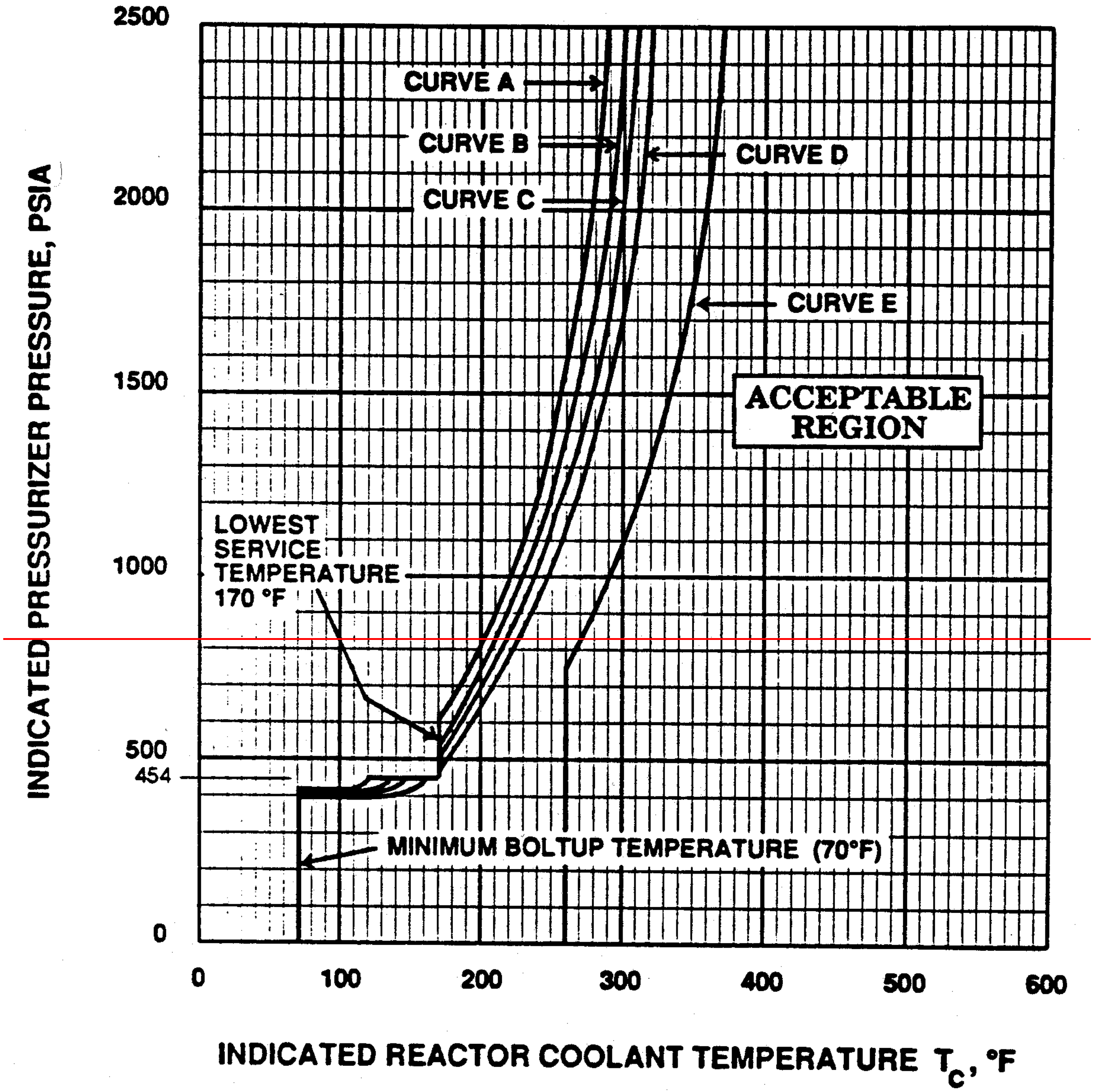
APPLICABILITY: At all times.

ACTION:

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

~~Figure 3.4-2A~~

~~ARKANSAS NUCLEAR ONE UNIT 2~~
~~HEATUP CURVE - 21 EFPY~~
~~REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS~~

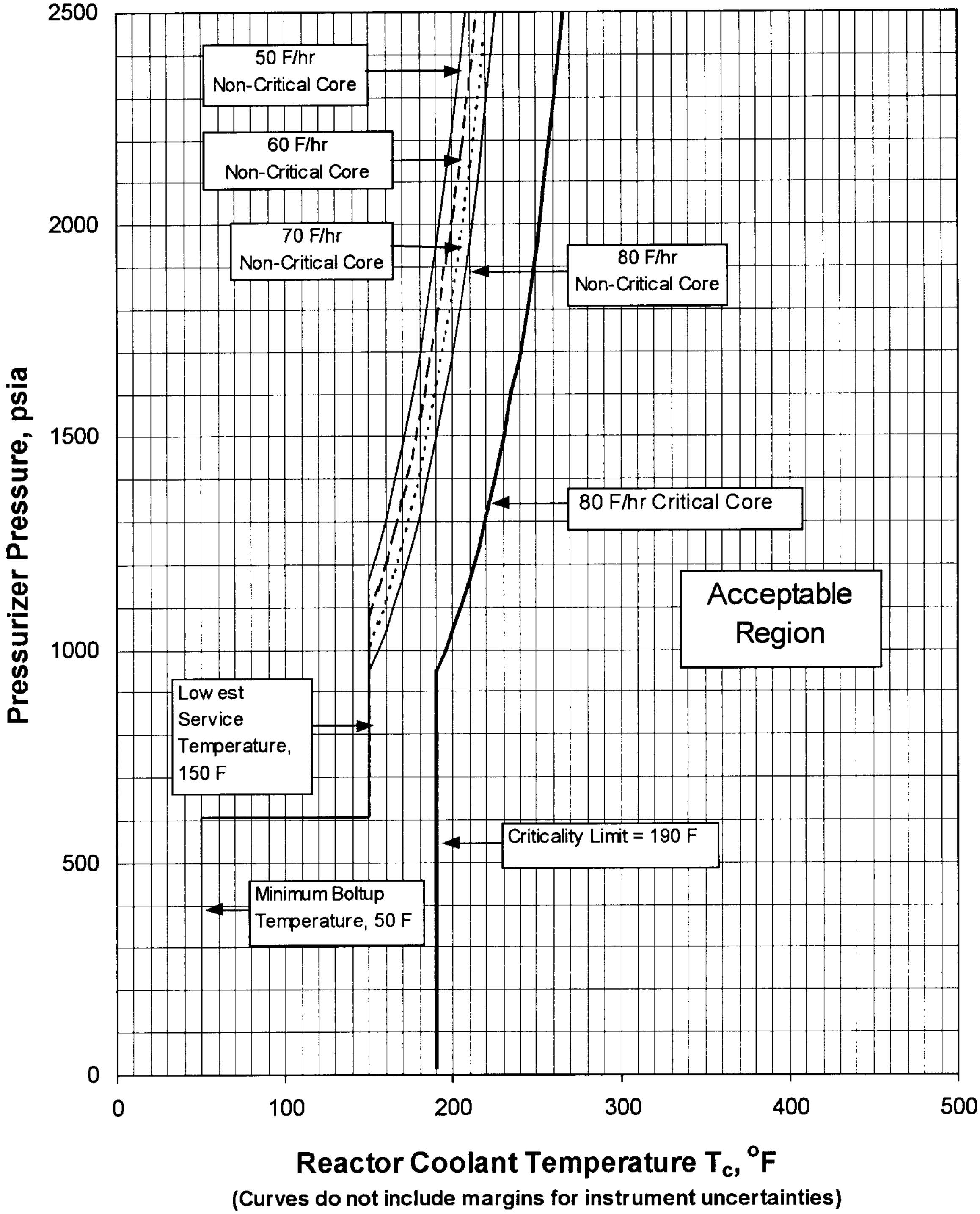


H/U LIMIT CURVE	H/U RATE LIMIT
A	50°F/HR
B	60°F/HR
C	70°F/HR
D	80°F/HR (NON-CRITICAL CORE)
E	80°F/HR (CRITICAL CORE)

Figure 3.4-2A

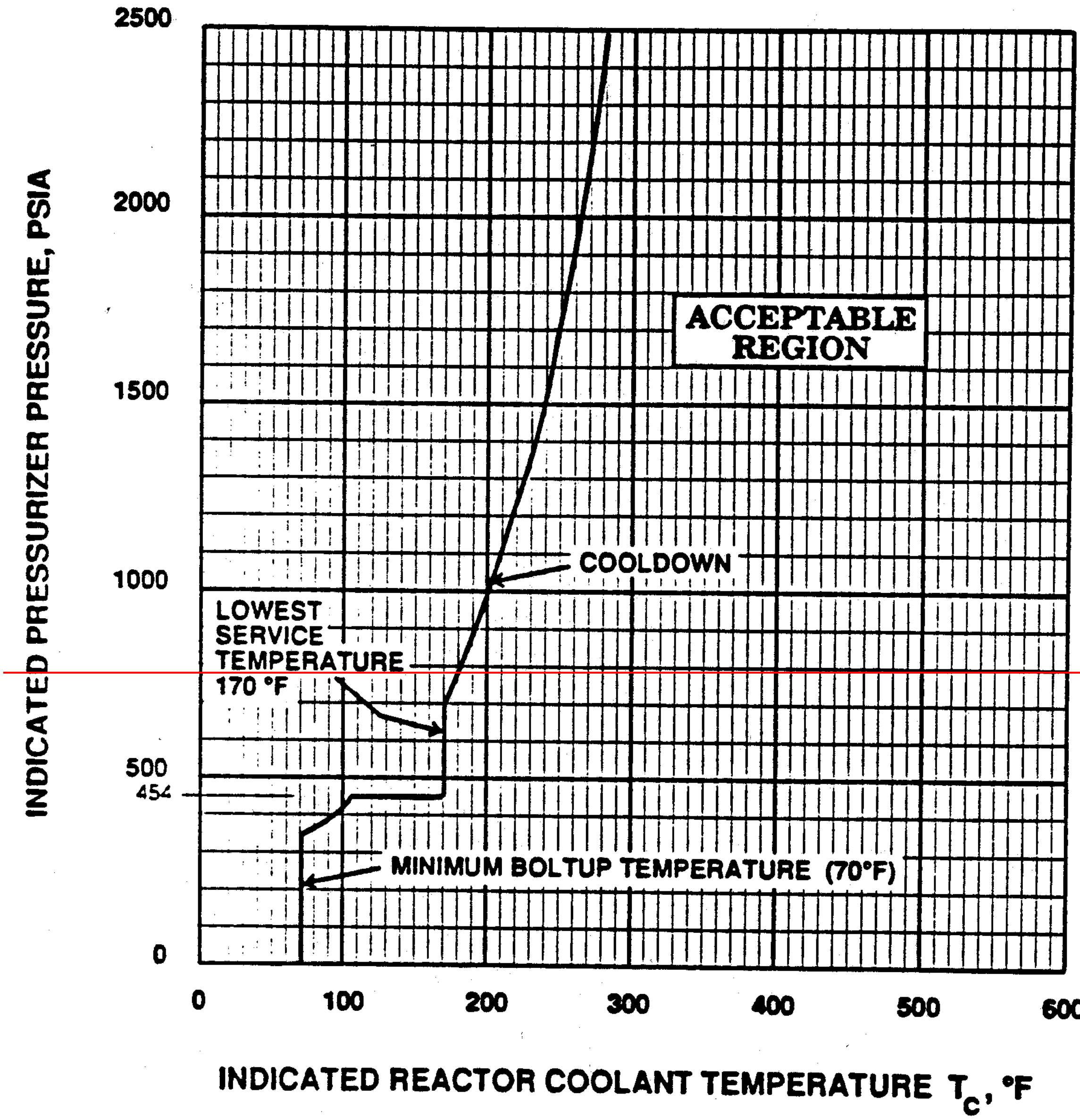
HEATUP CURVE - 32 EFPY

REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS



~~Figure 3.4-2B~~

~~ARKANSAS NUCLEAR ONE UNIT 2~~
~~COOLDOWN CURVE - 21 EFPY~~
~~REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS~~



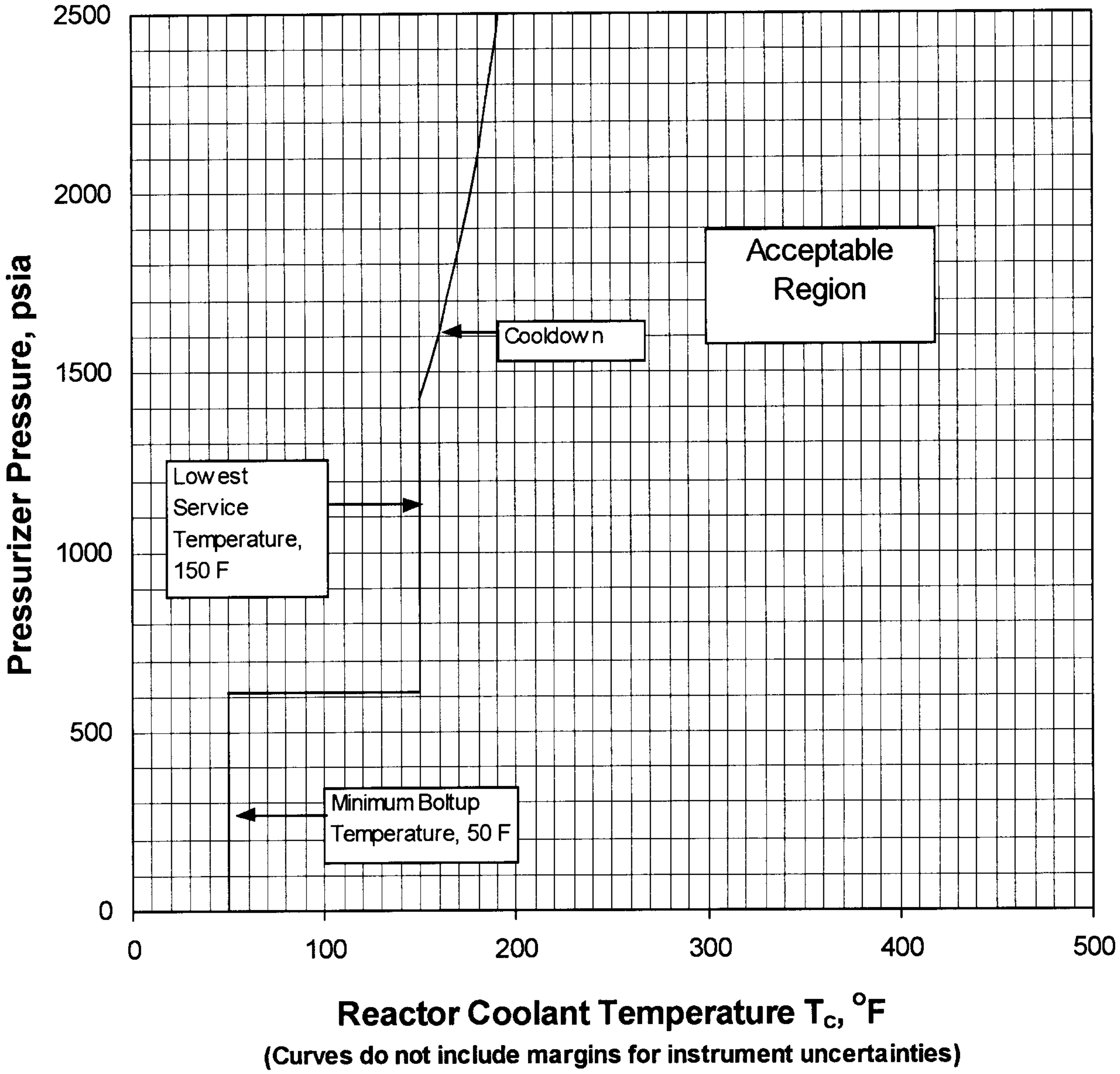
RCS TEMP (Tc)	C/D RATE	STEP *
T > 220°F	100°F/HR	≤ 50°F IN ANY 1/2 HR PERIOD
140F ≤ T ≤ 220°F	60°F/HR	≤ 30°F IN ANY 1/2 HR PERIOD
T < 140°F	25°F/HR	≤ 12.5°F IN ANY 1/2 HR PERIOD

* Not to exceed the specified instantaneous decrease in temperature with a subsequent thirty minute hold

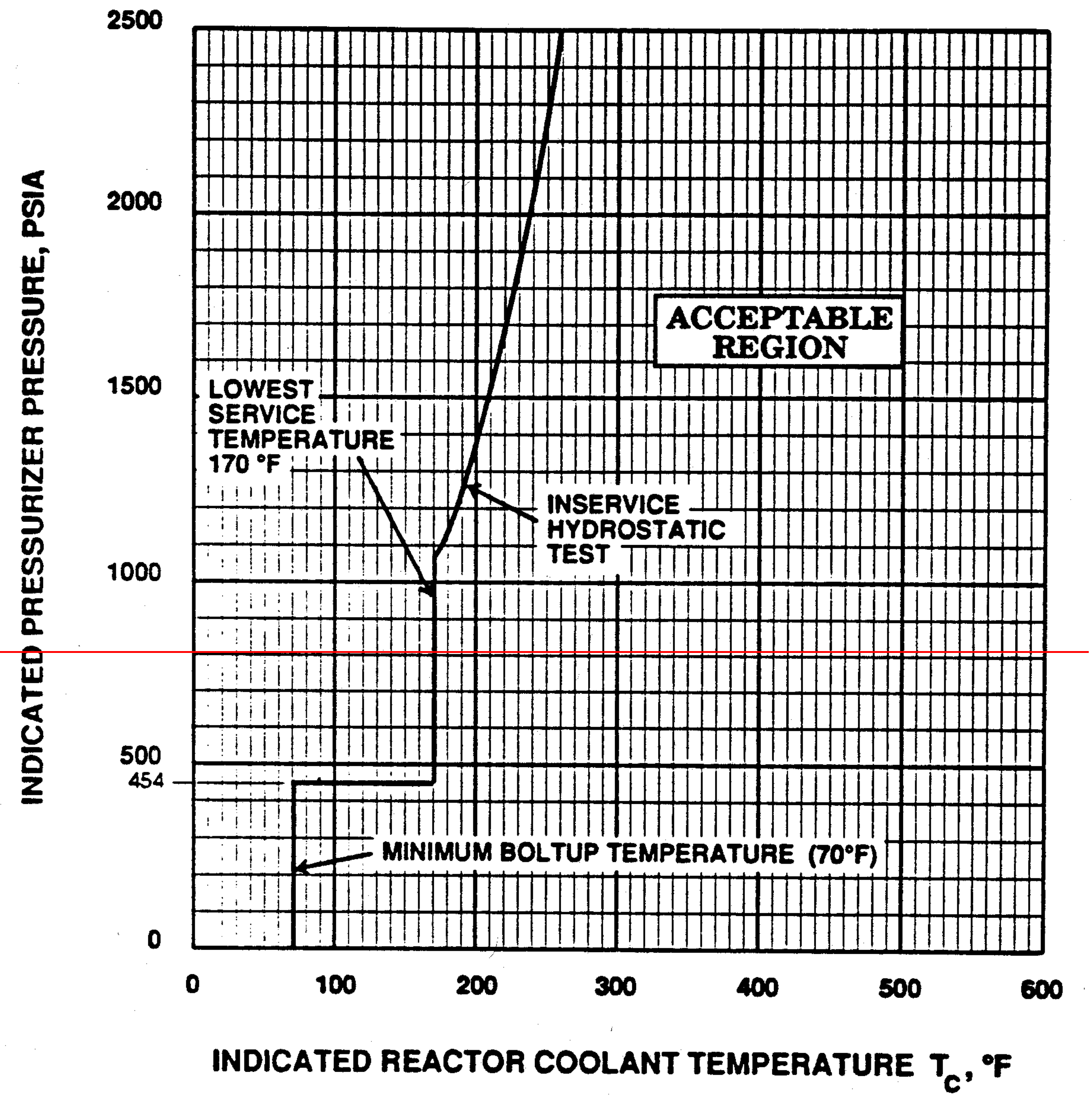
Figure 3.4-2B

COOLDOWN CURVE - 32 EFPY

REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS



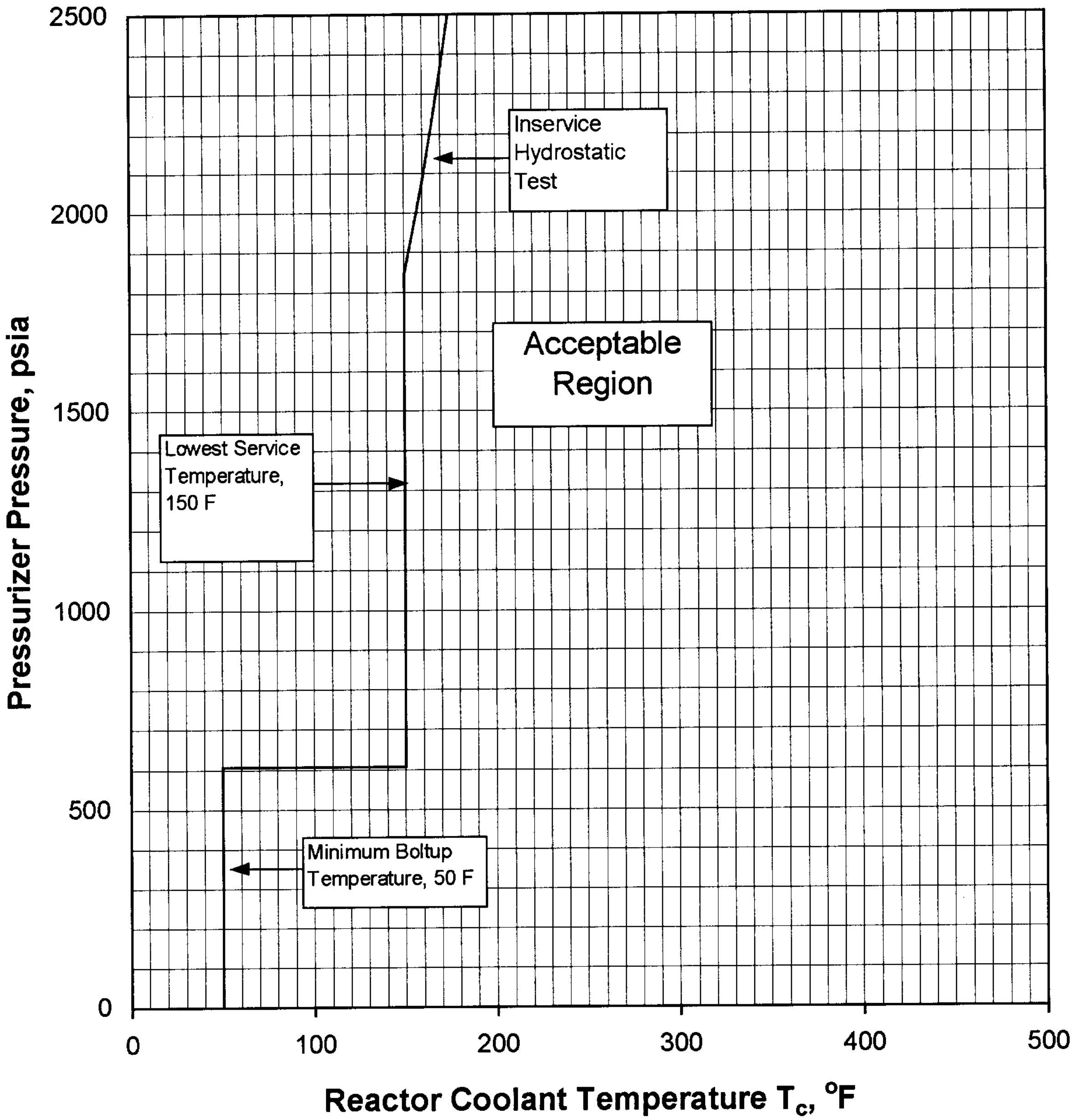
~~Figure 3.4-2C~~
~~ARKANSAS NUCLEAR ONE UNIT 2~~
~~INSERVICE HYDROSTATIC TEST CURVE - 21 EFPY~~
~~REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS~~



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

Figure 3.4-2C

INSERVICE HYDROSTATIC TEST CURVE - 32 EFPY
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS



(Curves do not include margins for instrument uncertainties)

LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.12 The LTOP system shall be OPERABLE with each SIT isolated that is pressurized to ≥ 300 psig, and a maximum of one HPSI pump capable of injecting into the RCS and:

a. Two LTOP relief valves with a lift setting of ≤ 430 psig, or

b. The Reactor Coolant System depressurized with an RCS vent path ≥ 6.38 square inches.

APPLICABILITY: MODE 4 with T_C ≤ 220°F, MODE 5, MODE 6 with reactor vessel head in place.

ACTION:

a. With one LTOP relief valve inoperable in MODE 4, restore the inoperable valve to OPERABLE status within 7 days or depressurize and vent the RCS through a ≥ 6.38 square inch vent path within the next 8 hours.

b. With one LTOP relief valve inoperable in MODE 5 or 6, restore the inoperable relief valve to OPERABLE status within 24 hours or depressurize and vent the RCS through a ≥ 6.38 square inch vent path within the next 8 hours.

c. With both LTOP relief valves inoperable, depressurize and vent the RCS through a ≥ 6.38 square inch vent path within 8 hours.

d. With a SIT not isolated and pressurized to ≥ 300 psig, isolate the affected SIT within 1 hour. If the affected SIT is not isolated within 1 hour, either:

(1) Depressurize the SIT to < 300 psig within the next 12 hours, or

(2) Increase cold leg temperature to > 220°F within the next 12 hours.

e. With more than one HPSI pump capable of injecting into the RCS, immediately initiate action to verify a maximum of one HPSI pump capable of injecting into the RCS.

~~e~~f. The provisions of Specification 3.0.4 are not applicable.

steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Arkansas Nuclear One site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 µCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Reducing Tavg to < 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2.1.5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates do not exceed the design assumptions and satisfy the stress limits for cyclic operation.

Operation within the limits of the appropriate heatup and cooldown curves assure the integrity of the reactor vessel against fracture induced by combined thermal and pressure stresses. As the vessel is subjected to increasing fluence, the toughness of the limiting material continues to decline, and even more restrictive pressure/temperature limits must be observed. ~~The current limits, Figures 3.4-2A, 3.4-2B and 3.42-C are for up to and including 21 Effective Full Power Years (EFPY) of operation.~~

The reactor vessel materials have been tested to determine their initial RT_{NDT}~~; the results of these test are shown in Table B 3/4.4-1~~. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. The heatup/criticality,~~ and~~ cooldown, and hydrostatic test limit curves for 32 EFPY shown on Figure 3.4-2A, 3.4-2B and 3.4-2C include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for the location and for possible errors in the pressure and temperature sensing instruments. It should be noted that the location adjustment considered the operation of three RCPs from a RCS temperature ~~of 70°F and above~~above the LTOP enable temperature and a maximum of two RCPs operating while below the LTOP enable temperature. ~~The heatup, cooldown, and hydrostatic test limits are presented in tabular form in Table B 3/4.4-2.~~ Instrument uncertainty in these curves is not included, but is added in station procedures.

The shift in the limiting material fracture toughness, as represented by RT_{NDT}, is calculated using Regulatory Guide 1.99, Revision 2. For ~~21~~ 32 EFPY, at the 1/4t position, the adjusted reference temperature (ART) value is ~~111~~112.7°F. At the 3/4t position the ART value is ~~96~~98.8°F. These values are conservatively based on a reactor vessel inner surface fluence of ~~3.74~~3.79 x 10^{19} nvt. The fluence at the 1/4t point is ~~2.33~~2.29 x 10^{19} nvt and the fluence of the 3/4t point is ~~9.06~~8.92 x 10^{18} nvt. These values are used with procedures developed in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G and Code Case N-641 to calculate heatup and cooldown limits in accordance with the requirements of 10 CFR Part 50, Appendix G.

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

To develop composite pressure/temperature limit for the cooldown event, the isothermal pressure/temperature limits must be calculated. The isothermal pressure/temperature limit is then compared to the pressure/temperature limit associated with both the constant cooldown rate and the corresponding step change rate (an instantaneous drop in temperature followed by a hold period). The more restrictive allowable pressure/temperature limit is chosen resulting in a composite limit curve for the reactor vessel beltline.

Both 10CFR Part 50, Appendix G and ASME Code Section III, Appendix G, require the development of pressure/temperature limits which are applicable to inservice hydrostatic tests. The minimum temperature for the inservice hydrostatic test pressure can be determined by entering the curve at the test pressure ~~(1.1 times normal operating pressure)~~ and locating the corresponding temperature. This curve is shown for ~~21~~ 32 EFPY on Figure 3.4-2C.

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

~~TABLE B 3/4.4-1~~

~~REACTOR VESSEL TOUGHNESS~~

Piece No.	Code No.	Material	Vessel Location	Drop Weight Results	Temperature of Charpy V-Notch @ 30 ft - lb	@ 50 ft - lb	Minimum Upper Shelf Cv energy for Longitudinal Direction-ft lb
Reference Dwg.	E-234-775						
203-02	C-8001	SA-508 Cl. 2	Vessel Flange Forging	+30°F	-50°F	-20°F	133
232-01	C-8013	"	Bottom Head Plate	-10°F	-56°F	-21°F	137
232-02	C-8014	A-533-B Cl. 1	Bottom Head Plate	-20°F	-40°F	-1°F	152
205-	C-8015-1	SA-508 Cl. 2	Inlet Nozzle Forging	+30°F	-50°F	-25°F	148
	C-8015-2	"	Inlet Nozzle Forging	+10°F	-50°F	-25°F	126
	C-8015-3	"	Inlet Nozzle Forging	+10°F	-80°F	-60°F	140
	C-8015-4	"	Inlet Nozzle Forging	+30°F	-60°F	-30°F	154
	C-8016-1	SA-508 Cl. 2	Outlet Nozzle Forging	0°F	-60°F	-38°F	131
205-	C-8016-2	"	Outlet Nozzle Forging	-10°F	-28°F	+38°F	114
205-03	C-8019-1	SA-508 Cl. 1	Inlet Nozzle Ex. Forg.	-40°F	-40°F	-20°F	163
205-03	C-8019-2	"	Inlet Nozzle Ex. Forg.	-40°F	-50°F	-35°F	154
205-03	C-8019-3	"	Inlet Nozzle Ex. Forg.	-20°F	-48°F	-35°F	184
205-03	C-8019-4	"	Inlet Nozzle Ex. Forg.	-20°F	-50°F	-25°F	145
205-07	C-8020-1	SA-508 Cl. 1	Outlet Nozzle Ex. Forg.	-30°F	-30°F	+10°F	153
205-07	C-8020-2	"	Outlet Nozzle Ex. Forg.	-30°F	-40°F	+10°F	148
215-01	C-8008-1	A-533 Gr. B. Cl.1	Upper Shell Plate	0°F	-8°F	+18°F	153
215-01	C-8008-2	"	Upper Shell Plate	+10°F	+24°F	+68°F	105
215-01	C-8008-3	"	Upper Shell Plate	0°F	+34°F	+56°F	109
215-02	C-8009-1	A-533 Gr. B. Cl.1	Inter. Shell Plate	-30°F	-18°F	+6°F	146
215-02	C-8009-2	"	Inter. Shell Plate	0°F	-24°F	0°F	142
215-02	C-8009-3	"	Inter. Shell Plate	-10°F	- 8°F	+32°F	134
215-03	C-8010-1	"	Lower Shell Plate	-20°F	+7°F	+36°F	138
215-02	C-8010-2	"	Lower Shell Plate	-30°F	-27°F	-5°F	144
215-03	C-8010-3	"	Lower Shell Plate	-30°F	-34°F	-10°F	150

~~TABLE B 3/4.4-1~~

~~REACTOR VESSEL TOUGHNESS~~

~~Piece No.~~	~~Code No.~~	~~Material~~	~~Vessel Location~~	~~Drop Weight Results~~	~~Temperature of Charpy V-Notch @ 30 ft - lb~~	~~@ 50 ft - lb~~	~~Minimum Upper Shelf Cv energy for Longitudinal Direction-ft lb~~
~~Reference Dwg.~~	~~E-234-776~~						
~~209-02~~	~~C-8002~~	~~SA-508 Cl. 1~~	~~Closure Head Flange Forging~~	~~+10°F~~	~~-80°F~~	~~-52°F~~	~~162~~
~~231-01A~~	~~C-8012~~	~~A-533 Gr. B Cl.1~~	~~Closure Head Peels~~	~~+10°F~~	~~-30°F~~	~~-1°F~~	~~139~~
~~B~~	~~"~~	~~"~~	~~Closure Head Peels~~	~~+10°F~~	~~-30°F~~	~~-1°F~~	~~139~~
~~C~~	~~"~~	~~"~~	~~Closure Head Peels~~	~~+10°F~~	~~-30°F~~	~~-1°F~~	~~139~~
~~D~~	~~"~~	~~"~~	~~Closure Head Peels~~	~~+10°F~~	~~-30°F~~	~~-1°F~~	~~139~~
~~231-02~~	~~C-8011~~	~~"~~	~~Closure Head Dome~~	~~+10°F~~	~~-25°F~~	~~+34°F~~	~~100~~

~~TABLE B 3/4.4-2~~
~~ARKANSAS NUCLEAR ONE UNIT 2~~
~~21 EFPY - TECHNICAL SPECIFICATION~~
~~PRESSURE-TEMPERATURE LIMITS~~

CURVE RCS TEMPERATURE DEG. F	COOLDOWN BELTLINE COMPOSITE CURVE P-ALLOWABLE (PSIA)	HEATUP BELTLINE COMPOSITE CURVE P-ALLOWABLE (PSIA) ISO THERMAL	50 F/ HOUR	60 F/ HOUR	70 F/ HOUR	80 F/ HOUR	HYDROSTATIC BELTLINE COMPOSITE P-ALLOWABLE (PSIA)
70	358.6	448.4	433.8	419.7	406.5	393.8	655.0
72.5	368.6	–	–	–	–	–	–
80	–	464.0	433.8	419.7	406.5	393.8	675.8
82.5	378.6	–	–	–	–	–	–
85	388.6	–	–	–	–	–	–
90	–	482.0	433.8	419.7	406.5	393.8	699.8
95	408.6	–	–	–	–	–	–
97.5	418.6	–	–	–	–	–	–
100	–	502.8	433.8	419.7	406.5	393.8	727.5
102.5	428.6	–	–	–	–	–	–
107.5	448.6	–	–	–	–	–	–
110	–	526.8	439.2	420.2	406.5	393.8	759.5
120	488.6	554.6	452.6	427.9	409.3	393.8	796.6
130	–	586.7	472.1	442.1	418.4	399.1	839.4
132.5	528.6	–	–	–	–	–	–
140	498.6	623.9	497.8	462.6	434.1	409.7	889.0
150	558.6	666.8	529.5	489.2	455.7	427.0	946.2
160	628.6	716.4	567.7	522.4	483.5	449.7	1012.4
170	698.6	773.8	613.9	562.4	518.1	479.5	1088.9
180	788.6	840.2	668.0	610.1	560.1	515.4	1177.4
190	888.6	916.9	731.0	666.3	609.8	559.6	1279.6
200	998.6	1005.5	803.9	732.1	668.2	611.3	1397.8
210	1108.0	1108.0	888.2	808.7	737.4	673.4	1528.2
220	1226.5	1226.5	987.6	897.8	817.9	744.9	1678.0
230	1363.4	1363.4	1102.3	1001.2	911.1	830.0	1851.3
240	1521.8	1521.8	1234.4	1121.1	1019.7	927.3	2051.5
250	1704.8	1704.8	1386.7	1259.9	1146.1	1042.4	2283.0
260	1916.5	1916.5	1562.1	1420.5	1291.9	1173.7	2550.7
270	2161.1	2161.1	1768.1	1606.4	1460.0	1328.4	–
280	2443.9	2443.9	2005.6	1821.3	1656.6	1504.7	–
290	2770.8	2770.8	2279.1	2070.0	1883.2	1712.2	–
300	–	–	2594.0	2357.5	2144.3	1948.5	–
310	–	–	–	2690.0	2447.0	2226.3	–
320	–	–	–	–	2798.3	2542.5	–

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

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

The minimum boltup temperature is the minimum allowable temperature at pressures below 20% of the pre-operational system hydrostatic test pressure. The minimum is defined as the initial RT_{NDT} for the material of the higher stressed region of the reactor vessel plus any effects for irradiation per Article G-2222 of Section III of the ASME Boiler and Pressure Vessel Code. The initial reference temperature of the reactor vessel and closure head flanges was determined using the certified material test reports and Branch Technical Position MTEB 5-2. The maximum initial RT_{NDT} associated with the stressed region of the vessel flange is 30°F. The minimum boltup temperature of 30°F ~~including temperature instrument uncertainty~~plus a 20°F conservatism is ~~30°F + 20°F =~~ 50°F. ~~However, for additional conservatism, a minimum boltup temperature of 70°F is utilized.~~

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

The limitations imposed on the pressurizer heatup and cooldown rates are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

Next page is B 3/4.4-11

3/4.4.12 LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

Low temperature overpressure protection (LTOP) of the RCS, including the reactor vessel, is provided by redundant relief valves on the pressurizer which discharge from a single discharge header. Each relief valve is isolated from the RCS by two motor operated block valves. Each LTOP relief valve is a direct action, spring-loaded relief valve, with orifice area of 6.38 in^2 and a lift setting of ≤ 430 psig, and is capable of protecting the RCS from overpressurization when the transient is either (1) the start of an idle reactor coolant pump~~, under water-solid conditions~~ with the pressurizer volume at < 910 ft^3, and with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperature (energy addition event), or (2) simultaneous injection of ~~two~~ one HPSI pump~~s~~ and all three charging pumps to the water-solid RCS (mass addition event). The limiting LTOP design basis event is the energy addition event. The analyses assume that the safety injection tanks (SITs) are either isolated or depressurized such that they are unable to challenge the LTOP relief setpoints.

Since neither the LTOP relief valves nor the RCS vent is analyzed for the pressure transient produced from SIT injection, the LCO requires each SIT that is pressurized to ≥ 300 psig to be isolated. The isolated SITs must have their discharge valves closed and the associated MOV power supply breaker in the open position. The individual SITs may be unisolated when pressurized to < 300 psig. The associated instrumentation uncertainty is not included in the 300 psig value and therefore, the procedural value for unisolating the SITs with the LTOPs in service will be reduced.

The LTOP system, in combination with the RCS heatup and cooldown limitations of LCO 3.4.9.1 and administrative restrictions on RCP operation, provides assurance that the reactor vessel non-ductile fracture limits are not exceeded during the design basis event at low RCS temperatures. These non-ductile fracture limits are identified as LTOP pressure-temperature (P-T) limits, which were specifically developed to provide a basis for the LTOP system. These LTOP P-T limits, along with the LTOP enable temperature, were developed using guidance provided in ASME Code Section XI, Division 1, Code Case N-~~514~~ 641. This code case allows using an alternate means of determining LTOP P/T condition ~~that mandates that~~ but limits "~~LTOP systems shall limit~~ the maximum pressure in the vessel to ~~110~~100% of the pressure ~~determined to satisfy Appendix G, paragraph G-2215 of Section XI, Division 1"~~using the K_{IC} approach allowed by the Code Case.

The enable temperature of the LTOP isolation valves is based on any RCS cold leg temperature reaching 220°F (including a 20°F uncertainty). Although each relief valve is capable of mitigating the design basis LTOP event, both LTOP relief valves are required to be OPERABLE below the enable temperature to meet the single failure criterion of NRC Branch Technical Position RSB 5-2, unless any RCS vent path of 6.38 in^2 (equivalent relief valve orifice area) or larger is maintained.

Enclosure 1 to
2CAN100101

BAW-2405

September 2001

Appendix G Pressure-Temperature Limits for 32 EFPY, Using ASME Code Cases for ANO-2

Framatome ANP

Enclosure 2 to
2CAN100101

BAW-2399

September 2001

Analysis of Capsule W-104

Reactor Vessel Material Surveillance Program

Arkansas Nuclear One Unit 2