Attachment 4

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Pressure-Temperature Limits at 54 EFPY"

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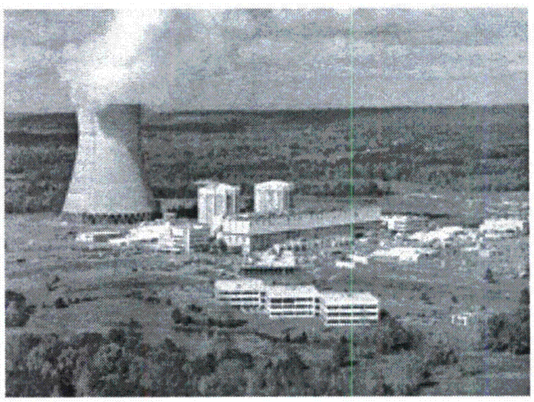
# Arkansas Nuclear One (ANO) Unit 1 Pressure-Temperature Limits at 54 EFPY



ANP-3300
June 2014

# Arkansas Nuclear One (ANO) Unit 1 Pressure-Temperature Limits at 54 EFPY

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**Prepared for**

**Entergy**

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## 1.0 INTRODUCTION

This report provides Reactor Coolant Pressure Boundary (RCPB) Technical Specification Pressure-Temperature (P-T) operating limits for Arkansas Nuclear One Unit 1 (ANO-1) at 54 effective full-power years (EFPY) of operation. The P-T limits are established in accordance with the requirements of 10 CFR Part 50, Appendix G [1]. These P-T limits are generated for normal operation heatup, normal operation cooldown, inservice leak and hydrostatic test (ISLH) conditions, and reactor core operations. These limits are expressed in the form of curves of allowable pressure versus temperature. The uncorrected P-T limits for ANO-1 were determined for 54 effective full power years (EFPY) of operation. Pressure correction factors were determined between pressure sensor locations in the reactor coolant system (RCS) hot leg and various regions of the reactor vessel (RV). In addition, the minimum temperature for core criticality is determined to satisfy the regulatory requirements of 10 CFR Part 50, Appendix G [1].

## 2.0 BACKGROUND

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. The three areas of the reactor pressure vessel addressed in the present report are the beltline shell region, the reactor coolant nozzles, and the closure head flange region.

A method for guarding against brittle fracture in reactor pressure vessels is described in Appendix G of the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" [2]. 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 ASTM E208 [3]) or the temperature at which the material exhibits 50 ft-lbs absorbed energy and 35 mils lateral expansion minus 60°F. The $RT_{NDT}$ of a given material is used to index that material to a reference stress intensity factor curve ($K_{Ic}$). The $K_{Ic}$ curve appears in Appendix G of ASME Code Section XI [2]. 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 pressure-temperature limits can then be determined using these allowable stress intensity factors.

The beltline region of the reactor vessel is the most highly exposed to neutron irradiation. The general effects of fast neutron irradiation on the mechanical properties of low-alloy ferritic steels such as SA-533, Grade B Class 1, and SA-508, Class 2 forging material used in the fabrication of the ANO-1 reactor vessel and inlet and outlet nozzles, are well characterized and documented in the literature. The

effects of irradiation on these steels include an increase in the yield and ultimate strengths and a decrease in ductility. The most significant effect, however, is an increase in the temperature associated with the transition from brittle to ductile fracture and a reduction in the Charpy upper-shelf energy value.

Pressure-temperature limits for the ANO-1 reactor vessel are developed in accordance with the requirements of 10 CFR Part 50, Appendix G [1], utilizing the analytical methods and flaw acceptance criteria of topical report BAW-10046A, Revision 2 [4] and ASME Code Section XI, Appendix G [2].

The ANO-1 reactor vessel contains both longitudinally and circumferentially oriented welds as shown in Figure 2-1. Therefore, the P-T limits for ANO-1 are based on the postulation of both longitudinal (axial) and circumferential flaws in the most limiting axial and circumferential welds

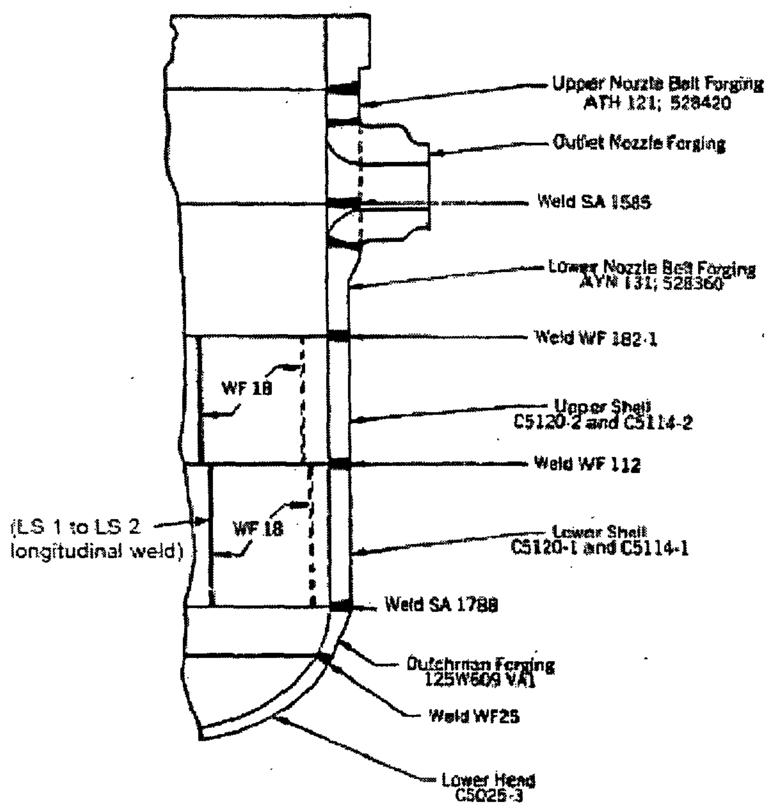



**Figure 2-1: The Location and Identification of Materials Used for ANO-1 RV**

## 3.0 ADJUSTED NIL-DUCTILITY TRANSITION REFERENCE TEMPERATURES

The $RT_{NDT}$ of the reactor vessel materials, and in turn, the pressure-temperature limits of a reactor vessel, must be adjusted to account for the effects of irradiation. The adjusted $RT_{NDT}$ (ART) values are calculated by adding a radiation-induced $\Delta RT_{NDT}$ to the initial $RT_{NDT}$ plus a margin term using Regulatory Guide 1.99 Revision 2 [5] to predict the radiation induced $\Delta RT_{NDT}$ values as a function of the material's copper and nickel content and neutron fluence. The projected fluence values at 54 EFPY are based on NRC approved Topical Report BAW-2241P-A, Revision 2 [6], which complies with Regulatory Guide 1.190 [7].

The 54 EFPY ¼ t (t - thickness of the section) and ¾ t ART values for the ANO-1 reactor vessel beltline base and weld materials are listed in Table 3-1 and Table 3-2 respectively. These values were calculated in accordance with Regulatory Guide 1.99, Revision 2 [5]. The calculation of the ART values for the weld metals used the following information from BAW-2308 Revision 1A and 2A [8]; the initial $RT_{NDT}$, *the associated standard deviation and the added chemistry factor requirement.* Entergy has made an exemption request to the NRC [9] to utilize BAW-2308 Revision 1A and 2A for determining the ART values for the Linde 80 weld metals for the ANO-1 unit. Table 3-3 summarizes the limiting ART values for ANO-1 used in the calculation of P-T limits.

The highest ART values for the ANO-1 reactor vessel is at the Lower Shell 1 to Lower Shell 2 longitudinal weld, WF-18, with an ART value of 166.8°F at the ¼t wall location and an ART value of 121.6°F at the ¾t wall location. The limiting ART values are listed in Table 3-3.

Note that the ART values of the longitudinal welds joining Upper Shell 1 to Upper Shell 2 (WF-18) are similar to the ART values of the longitudinal welds joining Lower Shell 1 to Lower Shell 2 (WF-18). The ART values for the Lower Shell 1 to Lower Shell 2 longitudinal welds are used as the limiting ART values because they are bounding. Also note that the limiting weld for the 54 EFPY P-T limits, WF-18, is different from the limiting weld for the 32 EFPY P-T limits, WF-112, as a result of using the inputs from BAW-2308.

## Table 3-1: Summary of ANO-1 RV Forging and Plate Data and Adjusted Reference Temperature Results at 54 EFPY

| Base Metal Identification | | | | Chemistry | | CF | Initial $RT_{NDT}$ (°F) | Projected 54 EFPY Fluence (n/cm$^2$) | | | $\Delta RT_{NDT}$ (°F) at 54 EFPY | | Margin (°F) | | ART (°F) at 54 EFPY | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Beltline Forgings or Plates | Material Type | Material ID | Heat No. | Cu wt% | Ni wt% | *[Note A]* | *[Note B]* | Wetted Surface | ¼ T | ¾ T | ¼ T | ¾ T | ¼ T | ¾ T | ¼ T | ¾ T |
| LNBF at start of 12" thickness | ASTM A508 Cl. 2 | AYN 131 | 528360 | 0.03 | 0.70 | 20.0 | 27.5 | 1.13E+18 | 5.34E+17 | 1.26E+17 | 6.1 | 2.6 | 26.5 | 25.9 | 60.1 | 56.0 |
| LNBF at start of 8.44" thickness | ASTM A508 Cl. 2 | AYN 131 | 528360 | 0.03 | 0.70 | 20.0 | 27.5 | 1.45E+18 | 8.48E+17 | 3.08E+17 | 7.7 | 4.5 | 26.9 | 26.2 | 62.1 | 58.1 |
| LNBF at LNBF to Upper Shell Weld | ASTM A508 Cl. 2 | AYN 131 | 528360 | 0.03 | 0.70 | 20.0 | 27.5 | 1.22E+19 | 7.14E+18 | 2.59E+18 | 18.1 | 12.7 | 31.5 | 28.7 | 77.1 | 68.9 |
| Upper Shell Plate 1 | SA 533 Gr. B Cl. 1 | C5120-2 | C5120-2 | 0.17 | 0.55 | 122.75 | -10 | 1.35E+19 | 7.90E+18 | 2.87E+18 | 114.6 | 80.9 | 34.0 | 34.0 | 138.6 | 104.9 |
| Upper Shell Plate 2 | SA 533 Gr. B Cl. 1 | C5114-2 | C5114-2 | 0.15 | 0.52 | 105.6 | -10 | 1.35E+19 | 7.90E+18 | 2.87E+18 | 98.6 | 69.6 | 34.0 | 34.0 | 122.6 | 93.6 |
| Lower Shell Plate 1 at 8.44" thickness | SA 533 Gr. B Cl. 1 | C5120-1 | C5120-1 | 0.17 | 0.55 | 122.75 | -10 | 1.33E+19 | 7.78E+18 | 2.83E+18 | 114.1 | 80.4 | 34.0 | 34.0 | 138.1 | 104.4 |
| Lower Shell Plate 2 at 8.44" thickness | SA 533 Gr. B Cl. 1 | C5114-1 | C5114-1 | 0.15 | 0.52 | 105.6 | 0 | 1.33E+19 | 7.78E+18 | 2.83E+18 | 98.2 | 69.2 | 34.0 | 34.0 | 132.2 | 103.2 |

LNBF = Lower Nozzle Belt Forging

*Notes:*

A. Chemistry Factor is calculated per Regulatory Guide 1.99, Revision 2 [5], Table 2 (linear interpolation allowed).

B. Initial $RT_{NDT}$ for the Lower Nozzle Belt Forging is a generic mean value for pre-1971 A508 Class 2 forgings manufactured by Ladish Company; Initial $RT_{NDT}$ values for Upper and Lower Shell Plates are measured values.

## Table 3-2: Summary of ANO-1 RV Weld Data and Adjusted Reference Temperature Results at 54 EFPY (BAW-2308 Inputs)

| Weld Metal Identification | | | Chemistry *[Note C]* | | Chem. Factor | Initial $RT_{NDT}$ (°F) | Projected 54 EFPY Fluence (n/cm²) | | | $\Delta RT_{NDT}$ (°F) at 54 EFPY | | Margin (°F) at 54 EFPY | | ART (°F) at 54 EFPY | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Beltline Welds | Material Acceptance No. | Wire Heat No. | Cu wt% | Ni wt% | *[Note D]* | *[Note E]* | Wetted Surface | ¼ T | ¾ T | ¼ T | ¾ T | ¼ T | ¾ T | ¼ T | ¾ T |
| LNBF to US Circ. Weld | WF-182-1 | 821T44 | 0.24 | 0.63 | 177.95 | -84.2 | 1.22E+19 | 7.14E+18 | 2.59E+18 | 161.1 | 112.7 | 59.2 | 59.2 | 136.1 | 87.7 |
| US 1 to US 2 Long. Welds (2) | WF-18 | 8T1762 | 0.19 | 0.57 | 167.0 | -48.6 | 1.08E+19 | 6.32E+18 | 2.29E+18 | 145.5 | 100.7 | 66.6 | 66.6 | 163.5 | 118.6 |
| US to LS Circ. Weld | WF-112 | 406L44 | 0.27 | 0.59 | 182.55 | -98.0 | 1.30E+19 | 7.60E+18 | 2.76E+18 | 168.5 | 118.5 | 60.6 | 60.6 | 131.1 | 81.1 |
| LS 1 to LS 2 Long. Welds (2) | WF-18 | 8T1762 | 0.19 | 0.57 | 167.0 | -48.6 | 1.16E+19 | 6.79E+18 | 2.46E+18 | 148.8 | 103.6 | 66.6 | 66.6 | 166.8 | 121.6 |

LNBF = Lower Nozzle Belt Forging
US = Upper Shell
LS = Lower Shell
Circ. = Circumferential
Long. = Longitudinal

*Notes:*

C. Cu wt% and Ni wt% weld wire heat best-estimates.

D. Chemistry Factor is calculated per Regulatory Guide 1.99, Revision 2 [5], Table 1 (linear interpolation allowed) with a minimum of 167°F per BAW-2308 [8].

E. Initial $RT_{NDT}$ is a heat-specific value calculated for Linde 80 weld metals in BAW-2308 [8]; A license exemption request per 10 CFR 50.12 has been made to the NRC [9] to use these values.

**Table 3-3: Limiting Adjusted Reference Temperatures for ANO-1 RV**

| Vessel Component | Material ID | ¼T ART | ¾T ART |
|---|---|---|---|
| Lower Shell 1 to Lower Shell 2 Longitudinal Weld | WF-18 | 166.8 | 121.6 |

## 4.0 DESIGN BASIS FOR PRESSURE-TEMPERATURE LIMITS

Essential analytical parameters used in the preparation of the ANO-1 P-T limits are described below.

### 4.1 Material Properties

Table 4-1 describes the material properties used in the development of the P-T limits for the ANO-1 unit. The RV material properties are obtained from Section II of ASME B&PV Code [2].

**Table 4-1: Reactor Vessel Steel and Cladding Material Properties**

| Temp. (°F) | Elastic Modulus E ($10^6$ psi) | Thermal(2) Expansion $\alpha$ ($10^{-6}$ in/in/°F) | Thermal Conductivity, k (Btu-in/hr-ft²-°F) | Specific Heat, $C_p$ (Btu/lb-°F) | Density $\rho$ (lb/ft³) | Thermal Conductivity for Cladding Material (Btu-in/hr-ft²-°F) |
|---|---|---|---|---|---|---|
| 70 | 29.2 | 7.0 | 282.0 | 0.105 | 490.9 | 103.2 |
| 100 | 29.0 | 7.1 | 283.2 | 0.107 | 490.5 | 104.4 |
| 150 | 28.8 | 7.2 | 283.2 | 0.110 | 489.9 | 108.0 |
| 200 | 28.5 | 7.3 | 283.2 | 0.114 | 489.2 | 111.6 |
| 250 | 28.3 | 7.3 | 282.0 | 0.116 | 488.6 | 115.2 |
| 300 | 28.0 | 7.4 | 280.8 | 0.120 | 487.9 | 117.6 |
| 350 | 27.7 | 7.5 | 279.6 | 0.123 | 487.3 | 121.2 |
| 400 | 27.4 | 7.6 | 277.2 | 0.126 | 486.7 | 124.8 |
| 450 | 27.2 | 7.6 | 274.8 | 0.128 | 486.0 | 127.2 |
| 500 | 27.0 | 7.7 | 272.4 | 0.131 | 485.4 | 130.8 |
| 550 | 26.7 | 7.8 | 270.0 | 0.134 | 484.7 | 133.2 |
| 600 | 26.4 | 7.8 | 266.4 | 0.136 | 484.1 | 135.6 |
| 650 | 25.9 | 7.9 | 262.8 | 0.139 | 483.4 | 139.2 |
| 700 | 25.3 | 7.9 | 259.2 | 0.142 | 482.8 | 141.6 |

### 4.2 Postulated Flaws

a. Postulated Reactor Vessel Beltline Flaws

Semi-elliptical surface flaws that are $^1/_4$ t deep and $1^1/_2$ t long are postulated on the inside (known as ¼ t flaw) and outside surfaces (known as ¾ t flaw) of the reactor vessel beltline region. A longitudinal flaw is postulated in the base metal and the longitudinal seam welds and a circumferential flaw is postulated in the circumferential welds.

b. Postulated Nozzle Corner Flaw

A 1/4 $t_{NB}$ ( $t_{NB}$ - the thickness at the nozzle belt) deep corner flaw is postulated on the inside surface of the reactor vessel inlet and outlet nozzles and core flood nozzle corner.

### 4.3 Upper Shelf Toughness

A maximum value of 200 ksi$\sqrt{\text{in}}$ is assumed for the upper shelf fracture toughness ($K_{Ic}$) of the reactor vessel beltline. For the nozzle forging materials, no "cut-off" limit is assumed.

### 4.4 Uncorrected Reactor Vessel Closure Head Limits

Pressure-temperature limits for the reactor vessel head-to-flange closure region for normal operation and In-Service Leak and Hydrotest (ISLH) operation were derived for the ANO-1 reactor vessel closure head based on the $K_{Ic}$ fracture toughness curve. The Pressure-Temperature limits derived for the reactor vessel head-to-flange satisfy the minimum temperature requirements specified in Table 1 of Appendix G to 10CFR Part 50[1].

### 4.5 Convection Film Coefficient

A value of 1000 BTU/hr-ft²-°F was used for an effective convection heat transfer film coefficient at the cladding to base metal interface for all the times during heatup and cooldown when any Reactor Coolant Pumps (RCP) are in use. When no reactor coolant pumps are running (i.e., when the reactor coolant temperature is 250 °F or less), a value of 430 BTU/hr-ft²-°F was used as an effective film coefficient at the cladding-to-base metal interface. The outside surface is modeled as a perfectly insulated boundary.

### 4.6 Reactor Coolant Temperature-Time Histories

Ramped transients are modeled for normal operation heatup. Both ramped and stepped transient definitions are used for normal cooldown. The normal heatup and cooldown transients are also used to simulate the reactor coolant transients used for inservice leak and hydrostatic (ISLH) pressure testing.

#### 4.6.1 Heatup Transients

The following three sets of normal heatup transients were analyzed:

60 °F - 84 °F: 15 °F/hr

84 °F - 570 °F; the three different ramp rates are: 50 °F/hr, 70 °F/hr and 90 °F/hr

#### 4.6.2 Cooldown Transients

For the analysis of the normal cooldown P-T limits, the cooldown transients were analyzed for a step transient as well as a ramp transient.

Initiation of the decay heat removal system (DHRS) occurs at a reactor coolant temperature of 270° F. DHRS initiation was modeled as a step change from 270 °F to 249 °F, with a hold at 249 °F for one minute, followed by a step temperature increase to 263 °F.

The cooldown transients were analyzed with the last Reactor Coolant Pump (RCP) tripping at three different temperatures (at 255 °F, at 200 °F, and at 175 °F). For each of these transient cases, the fourth RCP trip was simulated by a 25 °F temperature decrease in 20 seconds. This 25 °F change in temperature, at the time of the fourth RCP trip, occurs as the reactor coolant transitions from a state of RCP forced flow to one controlled by the DHRS.

Cooldown with last RCP Trip at 255 °F:

The step cooldown transient is defined as follows:

570 °F - 280 °F: 50 °F steps with 30 minute hold periods or equivalent

280 °F - 150 °F: 25 °F steps with 30 minute hold periods or equivalent

at 270 °F: DHRS initiation as described above

at 255 °F: 25 °F ramp in 20 seconds

150 °F - 60 °F: 25 °F steps with 60 minute hold periods or equivalent

The ramp cooldown transient is defined as follows:

570 °F - 280 °F: 100 °F/hr ramp

280 °F - 150 °F: 50 °F/hr ramp

at 270 °F: DHRS initiation as described above

at 255 °F: 25 °F ramp in 20 seconds (to simulate the tripping of the fourth RCP )

150 °F - 60 °F: 25 °F/hr ramp

Cooldown with last RCP trip at 200 °F:

The 200 °F value was selected as an intermediate value between 255 °F and 175 °F. Similar to the above transient, the fourth RCP trip was modeled by a 25 °F ramp in 20 seconds.

Acid Reducing Phase Cooldown with last RCP trip at 175 °F:

The purpose of this low temperature pump operation is to provide circulation throughout the RCS for acid reduction and control of water chemistry prior to completion of shutdown. For this special cooldown case involving an Acid Reducing Phase (last RCP trip at 175 °F), the cooldown transients are similar to the RCP trip at 255 °F. At 175 °F the fourth RCP trip is simulated by a 25 °F ramp in 20 seconds followed by a hold at 150 °F for 2 hours.

## 5.0 TECHNICAL BASIS FOR PRESSURE-TEMPERATURE LIMITS

Pressure-temperature limits are developed using an analytical approach that is in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G [2]. Additional requirements are contained in Table 1 of Appendix G to Title 10, Code of Federal Regulations, Part 50 [1]. The analytical techniques used to calculate P-T limits are based on approved linear elastic fracture mechanics methodology described in topical report BAW-10046A, Revision 2 [4]. The fundamental equation used to calculate the allowable pressure is

$$P_{allow} = \frac{K_{IR} - K_{IT}}{SF \times \hat{K}_{IP}}$$

where, $P_{allow}$ = allowable pressure

$K_{IR}$ = reference stress intensity factor ( $K_{Ic}$ )

$K_{IT}$ = thermal stress intensity factor

$\hat{K}_{IP}$ = unit pressure stress intensity factor (due to 1 psig)

$SF$ = safety factor

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, core flood nozzle, and closure head. In the beltline region, flaws are postulated to be present at the $^1/_4$t and $^3/_4$t locations of the controlling material (shell forging or circumferential weld), as defined by

the fluence adjusted $RT_{NDT}$. The reactor vessel nozzle flaws are located at the inside juncture (corner) with the nozzle shell, and the closure head flaw is located near the outside juncture with the head flange. P-T limits for the beltline and nozzle regions are calculated using a safety factor of 2 for normal operation and 1.5 for ISLH operation. The P-T limit curves presented consist of the allowable pressures for the controlling beltline flaw, inlet and outlet nozzles, core flood nozzle, and closure head, as a function of fluid temperature. These curves have been "smoothed", as necessary, to eliminate irregularities associated with the startup of the first reactor coolant pump during heatup and the initiation of decay heat removal during cooldown. After the initial determination of the P-T limit curves, location specific curves were adjusted for sensor location. No instrument error correction has been applied. The final results include the determination of a minimum/lower bound P-T curve.

The criticality limit temperature is obtained by determining the maximum required ISLH test temperature at a pressure of 2500 psig (approximately 10% above the normal operating pressure). The ISLH analysis considers the most limiting heatup and cooldown transients. The approach satisfies the requirement of Item 2.d in Table 1 of 10 CFR 50, Appendix G [1]. It requires the minimum temperature to be the larger of minimum permissible temperature for inservice system hydrostatic pressure test (259.5°F) or the $RT_{NDT}$ of the closure flange material + 160°F (110°F). Hence, the criticality limit temperature is 259.5°F.

Various aspects of the calculation procedures utilized in the development of P-T limits are discussed below.

### 5.1 Fracture Toughness

The fracture toughness of reactor vessel steels is expressed as a function of crack-tip temperature, T, indexed to the adjusted reference temperature of the material, $RT_{NDT}$. Pressure-Temperature limits developed in accordance with ASME Code, Section XI, Appendix G [2], which permits the use of $K_{IC}$ fracture toughness,

$$K_{Ic} = 33.2 + 20.734 \exp [0.02 (T - RT_{NDT})]$$

The upper shelf fracture toughness is limited to an upper bound value of 200 ksi√in for the reactor vessel welds and shell base metal. No such "cut-off" limit is used for the fracture toughness of the reactor vessel nozzles. The crack-tip temperature needed for these fracture toughness equations is obtained from the results of a transient thermal analysis, described below.

### 5.2 Thermal Analysis and Thermal Stress Intensity Factor

Through-wall temperature distributions are determined by solving the one-dimensional transient axisymmetric heat conduction equation,

$$\rho C_p \frac{\partial T}{\partial t} = k\left(\frac{\partial^2 T}{\partial r^2} + \frac{1}{r}\frac{\partial T}{\partial r}\right),$$

subject to the following boundary conditions:

at the inside surface, where r = $R_i$,

$$-k\frac{\partial T}{\partial r} = h(T_w - T_b)$$

at the outside surface, where r = $R_o$,

$$\frac{\partial T}{\partial r} = 0$$

where,

$\rho$ = density

$C_p$ = specific heat

$k$ = thermal conductivity

$T$ = temperature

$r$ = radial coordinate

$t$ = time

$h$ = convection heat transfer coefficient

$T_w$ = wall temperature

$T_b$ = bulk coolant temperature

$R_i$ = inside radius of vessel

$R_o$ = outside radius of vessel

The above equation is solved numerically using a finite difference technique to determine the temperature at 17 points through the wall as a function of time for prescribed changes in the bulk fluid temperature, such as multi-rate ramp and step changes for heatup and cooldown transients.

Thermal stress intensity factors are determined for a radial thermal gradient considering the through-wall temperature distribution at each solution time point. Through-wall thermal stress distributions are determined by trapezoidal integration of the following expression:

Thermal hoop stresses:

$$\sigma_\theta(r) = \frac{E\alpha}{1-\nu}\frac{1}{r^2}\left(\frac{r^2+R_i^2}{R_o^2-R_i^2}\int_{R_i}^{R_o} T\,r\,dr + \int_{R_i}^{r} T\,r\,dr - T\,r^2\right) \qquad \text{[Ref. 10, Eqn (255)]}$$

The thermal stress distribution is then expressed by the following polynomial:

$$\sigma(x) = C_0 + C_1\,(x/a) + C_2\,(x/a)^2 + C_3\,(x/a)^3\,,$$

where,

x = is a dummy variable that represents the radial distance from the appropriate (i.e., inside or outside) surface, in.

a = the flaw depth, in.

The thermal stress intensity factors are defined by the following relationships:

For a $^1/_4$ t inside surface flaw during cooldown,

$$K_{It} = (1.0359\,C_0 + 0.6322\,C_1 + 0.4753\,C_2 + 0.3855\,C_3)\sqrt{\pi a}$$

For a $^1/_4$ t outside surface flaw during heatup,

$$K_{It} = (1.043\,C_0 + 0.630\,C_1 + 0.481\,C_2 + 0.401\,C_3)\sqrt{\pi a}$$

### 5.3 Unit Pressure Stress Intensity Factor for Reactor Vessel Beltline Region

The membrane stress intensity factor in the reactor vessel shell due to a unit pressure load is

$K_{Im} = M_m \times R_i / t$

where

$R_i$ = vessel inner radius, in.

t = vessel wall thickness, in.

For a longitudinal ¼ -thickness x $^3/_2$ -thickness semi-elliptical surface flaw:

*at the inside surface,*

$M_m$ = 1.85 for $\sqrt{t} < 2$

= 0.926 $\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$

$= 3.21 \quad$ for $\sqrt{t} > 3.464$

at the outside surface,

$M_m = 1.77 \quad$ for $\sqrt{t} < 2$

$= 0.893\sqrt{t} \quad$ for $2 \leq \sqrt{t} \leq 3.464$

$= 3.09 \quad$ for $\sqrt{t} > 3.464$

### 5.4 Unit Pressure Stress Intensity Factor for Reactor Vessel Nozzles

*Considering a nozzle as a hole in a shell,* WRC Bulletin 175 [11] presents the following method for estimating stress intensity factors for a nozzle corner flaw:

$$K_{Im} = \sigma\sqrt{\pi a}\, F(a/r_n)$$

where

$\sigma = R_i / t$

$R_i =$ nozzle belt shell inner radius, in.

$t =$ nozzle belt shell wall thickness, in.

$a =$ flaw depth, in.

$r_n =$ apparent radius of nozzle, in.

$= r_i + 0.29r_c$

$r_i =$ inner radius of nozzle, in.

$r_c =$ nozzle corner radius, in.

and

$$F(a/r_n) = 2.5 - 6.108(a/r_n) + 12(a/r_n)^2 - 9.1664(a/r_n)^3$$

## 6.0 PRESSURE CORRECTIONS

The uncorrected P-T limits are calculated at the required locations or components in the RCS. Although both wide and low range pressure taps are located in the hot legs, they are both modeled at the same node in the thermal hydraulics model, and, therefore, only one set of location corrections is used. The uncorrected P-T limits are corrected to this single location. Location correction factors were determined for various temperatures and pump combinations. The limiting correction factors at various temperature

ranges were then determined for beltline, nozzle, and closure head locations, as tabulated in Table 6-1 for ANO-1.

**Table 6-1: Limiting Location Pressure Corrections Factors for ANO-1**

| Temperature Range, °F | 50-99 | | 100-249 | | 250-349 | | 350-449 | | 450-532[1] | |
|---|---|---|---|---|---|---|---|---|---|---|
| | ΔP, psi | RCP[2] | ΔP, psi | RCP[2] | ΔP, psi | RCP[2] | ΔP, psi | RCP[2] | ΔP, psi | RCP[2] |
| Beltline | 22 | 0/0 | 109 | 2/1 | 122 | 2/2 | 116 | 2/2 | 108 | 2/2 |
| Outlet Nozzle | 17 | 0/0 | 71 | 2/0 | 69 | 2/0 | 47 | 2/2 | 44 | 2/2 |
| RVCH | 14 | 0/0 | 67 | 2/0 | 66 | 2/0 | N/A | - | N/A | - |
| Core Flood Nozzle | 17 | 0/0 | 106 | 2/1 | 122 | 2/2 | 116 | 2/2 | 107 | 2/2 |

1) The correction factor is used for temperatures above 532°F since the values are bounding for higher temperatures

2) The definition of RCP combinations used here are as follows: 0/0 - no pumps operating; 2/2 - all pumps operating; 2/0 - both pumps of loop A operating, both pumps of loop B are turned off; 2/1 - two pumps of loop A and one pump of loop B operating, one pump of loop B turned off.

## 7.0 SUMMARY OF RESULTS

The following is a summary of results for the ANO-1 P-T limits at 54 EFPY. The allowable pressures are corrected for location only. Correction due to instrument uncertainty is not included.

Maintaining the reactor coolant system pressure below the upper limit of the pressure-temperature limit curves ensures protection against non-ductile failure. Acceptable pressure and temperature combinations for reactor vessel operation are below and to the right of the applicable P-T limit curves. These P-T limit curves have been adjusted based on the pressure differential between point of system pressure measurement and the point in the reactor vessel that establishes the controlling unadjusted pressure limit. The P-T limit curves provided in Figure 7-1 through Figure 7-3 have not been corrected for instrument error. The reactor is not permitted to be critical until the pressure-temperature combinations are, as a minimum, to the right of the criticality curve. The numerical values for the Technical Specification P-T curves provided in Figure 7-1 through Figure 7-3 are shown in Table 7-1 through Table 7-4. The

operational constraints for these curves are tabulated in Table 7-5 and Table 7-6. These Technical Specification P-T curves meet all the pressure and temperature requirements for the reactor pressure vessel listed in Table 1 of 10CFR Part 50, Appendix G[1].

The Tech. Spec. P-T limits for normal heatup for ANO Unit 1 are shown in Table 7-1. The Tech. Spec. P-T limits for normal cooldown for ANO-1 are determined by the limiting allowable pressure at every calculated temperature, as shown in Table 7-3. The Tech. Spec. P-T limits for ISLH heatup are shown in Table 7-4. The criticality limit temperature corresponding to a pressure of 2500 psig is determined through interpolation of the ISLH heatup data in Table 7-4. As shown in Table 7-2(a), the criticality limit temperatures for ANO-1, is 259.5°F. The criticality-limit P-T limits are shown in Table 7-2(b).

In BAW-10046A Rev. 2 [4], the RCS piping and control rod drive motor tube (both parts of RCS pressure boundary) are qualified by establishing Lowest Service Temperature (LST) requirements in lieu of Appendix G analysis. The maximum allowable pressure for RCS piping during normal operation for temperatures up to 150°F is 20% of pre-service hydro-test minus the pressure correction factor [4]. It has been demonstrated that the limiting component at low temperature is the RVCH which removes the requirement to include the LST of RCS piping in the P-T limits [4]. It has also been demonstrated that a LST of 40°F for the control rod drive mechanism motor tube satisfies the ASME Code and 10 CFR Appendix G requirements.

The Low Temperature Overpressure Protection (LTOP) enable temperature for 54 EFPY is determined as 248°F plus any instrument/measurement uncertainty. This is 14°F lower than the current (32 EFPY) LTOP enable temperature of 262°F.

The LTOP pressure limit is determined as 563.8 psig. This value, after adjustment for measurement and opening uncertainty, is to be used for the ERV (Electronic Relief Valve) setpoint whenever the RCS temperature is below the LTOP enable temperature.

**Table 7-1: Tech. Spec. P-T Limits for Normal Heatup**

| | Governing Adjusted Pressure | | |
|---|---|---|---|
| Fluid Temperature | at 50°F /hr | at 70°F /hr | at 90°F /hr |
| (°F) | (psig) | (psig) | (psig) |
| 60 | 586 | 583 | 581 |
| 65 | 586 | 583 | 581 |
| 70 | 586 | 583 | 581 |
| 75 | 586 | 583 | 581 |
| 80 | 586 | 583 | 581 |
| 84 | 586 | 583 | 581 |
| 89 | 586 | 583 | 581 |
| 94 | 586 | 583 | 581 |
| 99 | 586 | 583 | 581 |
| 104 | 586 | 583 | 581 |
| 109 | 593 | 589 | 586 |
| 114 | 601 | 596 | 593 |
| 119 | 610 | 604 | 593 |
| 123 | 619 | 611 | 593 |
| 124 | 621 | 613 | 593 |
| 129 | 632 | 623 | 596 |
| 134 | 645 | 634 | 601 |
| 139 | 659 | 647 | 609 |
| 144 | 675 | 660 | 621 |
| 149 | 693 | 676 | 634 |
| 154 | 712 | 692 | 651 |
| 159 | 734 | 711 | 671 |
| 164 | 757 | 732 | 694 |
| 169 | 783 | 754 | 720 |
| 174 | 812 | 780 | 749 |
| 179 | 844 | 807 | 779 |
| 184 | 880 | 838 | 806 |
| 189 | 919 | 872 | 836 |
| 194 | 962 | 910 | 869 |
| 199 | 1009 | 951 | 905 |
| 204 | 1062 | 997 | 946 |
| 209 | 1120 | 1047 | 990 |
| 214 | 1185 | 1103 | 1039 |
| 219 | 1256 | 1164 | 1092 |

**Figure 7-1 Tech. Spec. P-T Limits for Normal Heatup (continued)**

| | Governing Adjusted Pressure | | |
|---|---|---|---|
| Fluid Temperature | at 50°F /hr | at 70°F /hr | at 90°F /hr |
| 224 | 1335 | 1233 | 1152 |
| 229 | 1422 | 1308 | 1218 |
| 234 | 1518 | 1391 | 1290 |
| 239 | 1624 | 1483 | 1370 |
| 244 | 1742 | 1585 | 1459 |
| 249 | 1871 | 1697 | 1557 |
| 254 | 2038 | 1845 | 1683 |
| 259 | 2215 | 2005 | 1830 |
| 264 | 2403 | 2173 | 1980 |
| 269 | 2608 | 2355 | 2142 |
| 274 | 2835 | 2555 | 2320 |
| 279 | 3024 | 2775 | 2515 |
| 284 | 3024 | 3018 | 2729 |
| 289 | 3024 | 3024 | 2966 |
| 294 | - | - | 3024 |
| 299 | - | - | 3024 |
| 304 | - | - | 3024 |

## Table 7-2: Tech. Spec. Criticality Limit P-T Limits

### (a) Criticality Limit Determination

| Criticality Limit Temp. at 2500 psig during ISLH | |
|---|---|
| Pressure | Temp. |
| (psig) | (°F) |
| 2480* | 259* |
| 2681* | 264* |
| Interpolating: | |
| **2500** | **259.5** |

*From Table 7-4

### (b) Criticality Limit P-T Limits

| Fluid Temp | Governing Adjusted Pressure |
|---|---|
| (°F) | (psig) |
| 259.5 | 0 |
| 259.5 | 1098 |
| 264 | 1152 |
| 269 | 1218 |
| 274 | 1290 |
| 279 | 1370 |
| 284 | 1459 |
| 289 | 1557 |
| 294 | 1683 |
| 299 | 1830 |
| 304 | 1980 |
| 309 | 2142 |
| 314 | 2320 |
| 319 | 2515 |
| 324 | 2729 |
| 329 | 2966 |
| 334 | 3024 |
| 339 | 3024 |
| 344 | 3024 |

**Table 7-3: Tech. Spec. P-T Limits for Normal Cooldown**

| Fluid Temp. | Governing Adjusted Pressure |
|---|---|
| (°F) | (psig) |
| 60 | 535 |
| 65 | 535 |
| 70 | 535 |
| 75 | 535 |
| 80 | 535 |
| 85 | 535 |
| 90 | 535 |
| 95 | 535 |
| 100 | 535 |
| 105 | 547 |
| 110 | 559 |
| 115 | 573 |
| 120 | 587 |
| 123 | 597 |
| 125 | 603 |
| 130 | 619 |
| 135 | 637 |
| 140 | 657 |
| 145 | 680 |
| 150 | 717 |
| 155 | 752 |
| 160 | 806 |
| 165 | 854 |
| 170 | 904 |
| 175 | 936 |
| 180 | 993 |
| 185 | 1045 |
| 190 | 1103 |
| 193 | 1140 |
| 198 | 1207 |
| 203 | 1281 |
| 208 | 1364 |
| 213 | 1454 |
| 218 | 1555 |
| 223 | 1666 |

**Table 7-3: Tech. Spec. P-T Limits for Normal Cooldown (continued)**

| | |
|---|---|
| 228 | 1788 |
| 233 | 1924 |
| 238 | 2074 |
| 243 | 2239 |
| 248 | 2422 |
| 253 | 2514 |
| 255 | 2532 |
| 260 | 2532 |
| 265 | 2532 |
| 270 | 2532 |
| 340 | 2532 |
| 345 | 2532 |
| 350 | 2532 |
| 355 | 2532 |
| 360 | 2532 |
| 365 | 2532 |
| 370 | 2532 |
| 380 | 2532 |
| 385 | 2538 |
| 390 | 2544 |
| 395 | 2551 |
| 400 | 2558 |
| 405 | 2565 |
| 410 | 2573 |
| 415 | 2581 |
| 420 | 2589 |
| 425 | 2598 |
| 430 | 2602 |
| 435 | 2612 |
| 440 | 2622 |
| 445 | 2633 |
| 450 | 2647 |
| 455 | 2659 |
| 460 | 2671 |
| 465 | 2684 |
| 470 | 2697 |
| 475 | 2711 |
| 480 | 2725 |
| 485 | 2739 |

**Table 7-3: Tech. Spec. P-T Limits for Normal Cooldown (continued)**

| | |
|---|---|
| 490 | 2755 |
| 495 | 2770 |
| 500 | 2787 |
| 505 | 2803 |
| 510 | 2822 |
| 515 | 2840 |
| 520 | 2858 |
| 525 | 2878 |
| 530 | 2888 |
| 535 | 2918 |
| 540 | 2939 |
| 545 | 2960 |
| 550 | 2981 |
| 555 | 3002 |
| 560 | 3022 |
| 565 | 3040 |
| 570 | 3049 |

**Table 7-4: Tech. Spec. P-T Limits for ISLH HU/CD - Composite Curve**

| Fluid Temp. | Governing Adjusted Pressure |
|---|---|
| (°F) | (psig) |
| 60 | 668 |
| 65 | 750 |
| 70 | 750 |
| 75 | 750 |
| 80 | 750 |
| 85 | 750 |
| 90 | 750 |
| 95 | 750 |
| 100 | 750 |
| 105 | 765 |
| 110 | 782 |
| 115 | 800 |
| 120 | 819 |
| 124 | 827 |
| 129 | 831 |
| 134 | 838 |
| 139 | 849 |
| 144 | 864 |
| 149 | 882 |
| 154 | 904 |
| 159 | 931 |
| 164 | 961 |
| 169 | 996 |
| 174 | 1035 |
| 179 | 1076 |
| 184 | 1112 |
| 189 | 1151 |
| 194 | 1195 |
| 199 | 1244 |
| 204 | 1297 |
| 209 | 1356 |
| 214 | 1421 |
| 219 | 1493 |
| 224 | 1572 |
| 229 | 1660 |

**Table 7-4: Tech. Spec. P-T Limits for ISLH HU/CD - Composite Curve (continued)**

| Fluid Temp. | Governing Adjusted Pressure |
|---|---|
| (°F) | (psig) |
| 234 | 1757 |
| 239 | 1864 |
| 244 | 1982 |
| 249 | 2112 |
| 254 | 2284 |
| 259 | 2480 |
| 264 | 2681 |
| 269 | 2897 |
| 274 | 3133 |
| 279 | 3393 |
| 284 | 3512 |
| 289 | 3512 |
| 294 | 3512 |
| 299 | 3512 |
| 304 | 3512 |
| 309 | 3512 |
| 440 | 3512 |
| 445 | 3526 |
| 450 | 3544 |
| 455 | 3560 |
| 460 | 3576 |
| 465 | 3593 |
| 470 | 3611 |
| 475 | 3629 |
| 480 | 3648 |
| 485 | 3667 |
| 490 | 3687 |
| 495 | 3708 |
| 500 | 3730 |
| 505 | 3753 |
| 510 | 3778 |
| 515 | 3801 |
| 520 | 3826 |
| 525 | 3852 |
| 530 | 3866 |

**Table 7-4: Tech. Spec. P-T Limits for ISLH HU/CD - Composite Curve (continued)**

| Fluid Temp. | Governing Adjusted Pressure |
|---|---|
| (°F) | (psig) |
| 535 | 3905 |
| 540 | 3933 |
| 545 | 3961 |
| 550 | 3990 |
| 555 | 4018 |
| 560 | 4044 |
| 565 | 4067 |
| 570 | 4079 |

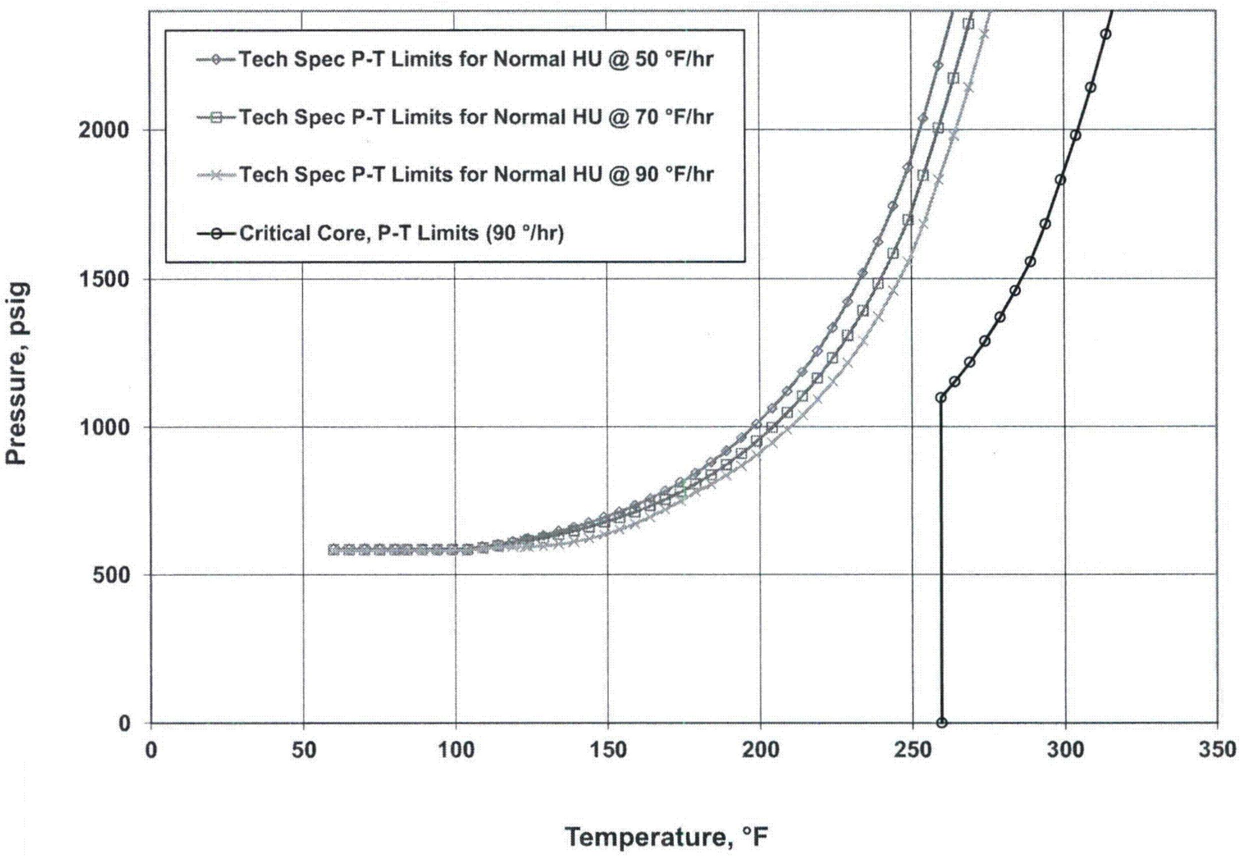
**Table 7-5: Operational Constraints for Plant Heatup**

| CONSTRAINT | RC TEMPERATURE | HEATUP RATE | RCP RESTRICTIONS |
|---|---|---|---|
| RC Temperature | T < 84°F | ≤ 15°F in any 1 hr period | NA |
| | T ≥ 84°F | ≤ 50°F, 70°F or 90°F in any 1 hr period | NA |
| RC Pumps | T ≥ 250°F | NA | None |
| | 100°F ≤ T < 250°F | NA | ≤ 3 pumps |
| | T < 100°F | NA | No pumps operating |

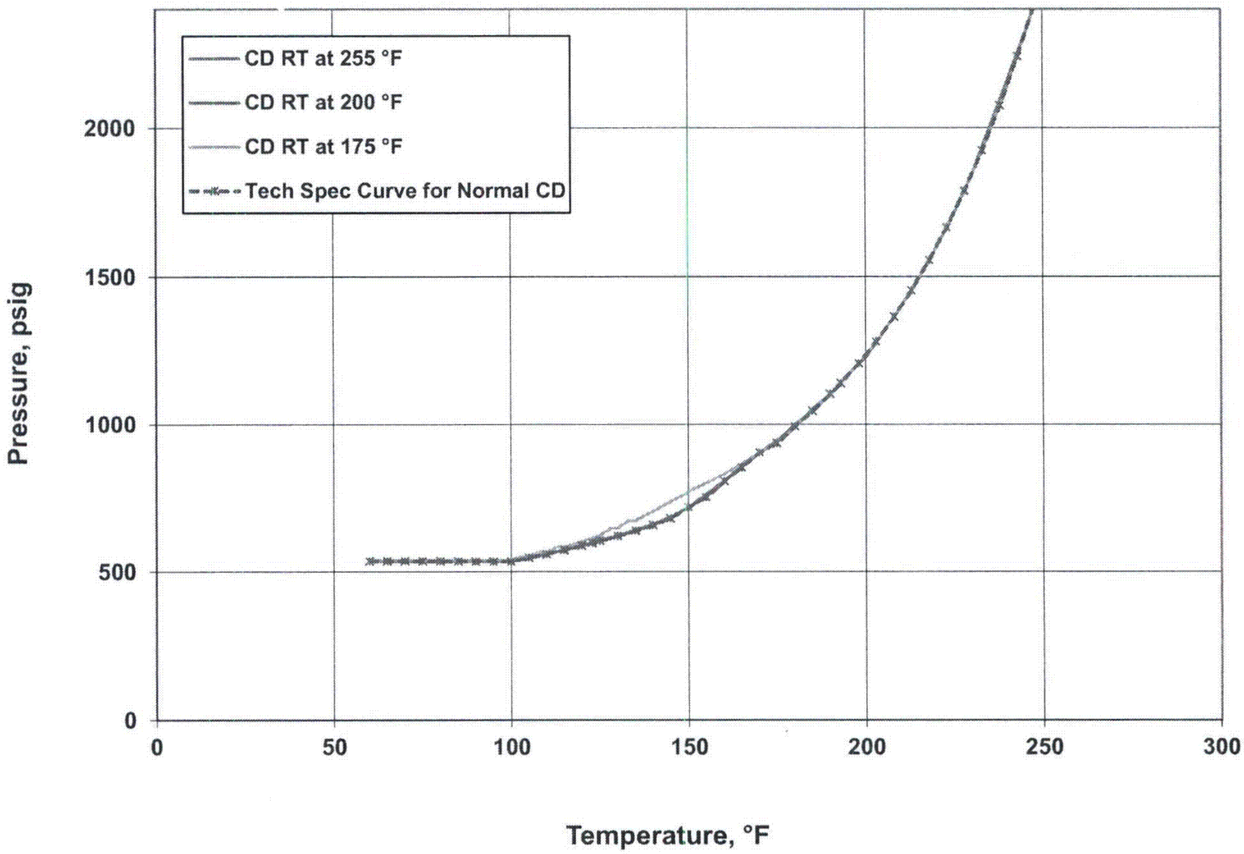
**Table 7-6: Operational Constraints for Plant Cooldown**

| CONSTRAINT | RC TEMPERATURE | COOLDOWN RATE | RCP RESTRICTIONS |
|---|---|---|---|
| RC Temperature | T ≥ 280°F | ≤ 50°F in any 1/2 hr period | NA |
| | 280°F > T ≥ 150°F | ≤ 25°F in any 1/2 hr period | NA |
| | T<150°F | ≤ 25°F in any 1 hr period | NA |
| RC Pumps | T ≥ 250°F | N/A | None |
| | 250°F > T ≥ 100°F | N/A | ≤ 3 pumps |
| | T < 100°F | N/A | No pumps operating |

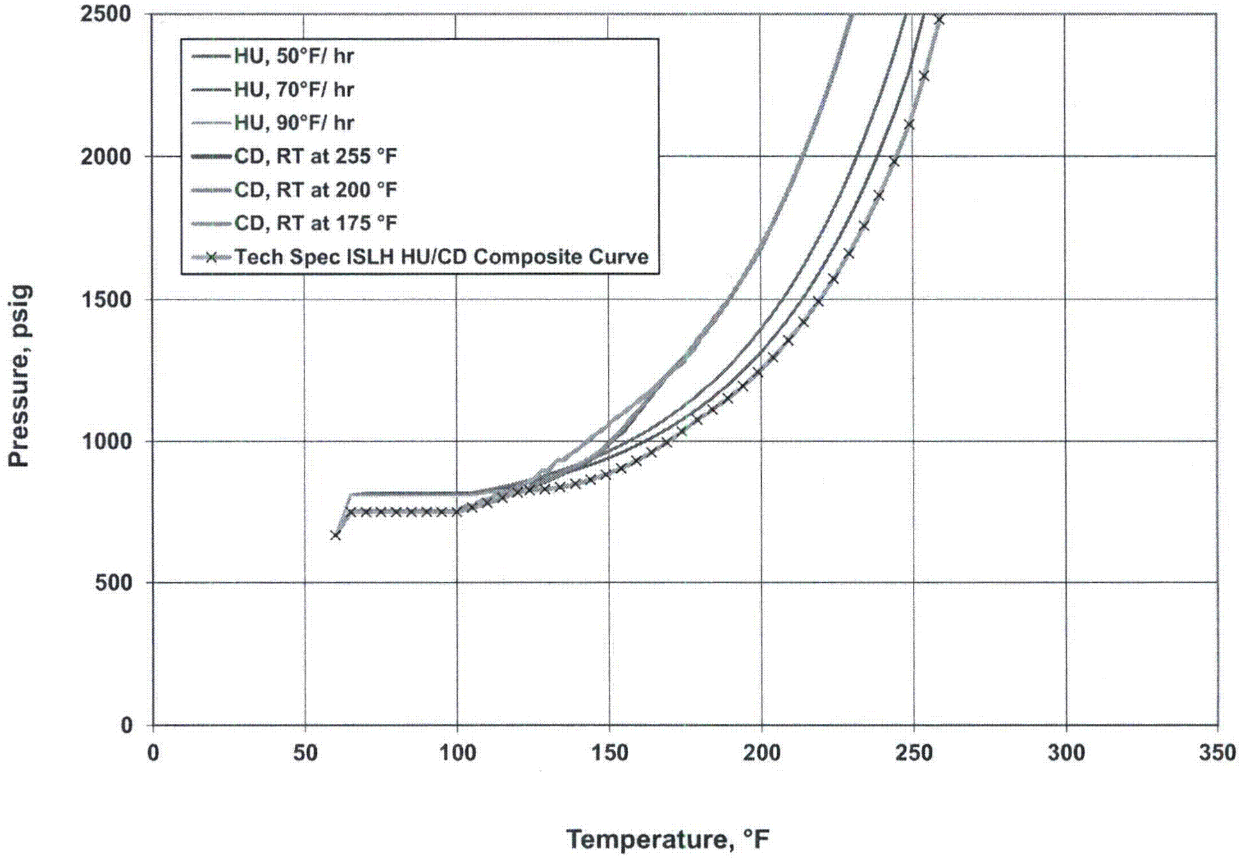
**Figure 7-1: Tech. Spec. Normal Heatup and Criticality Limit P-T Limits**



## Figure 7-2: Tech. Spec. Normal Cooldown P-T Limits



## Figure 7-3: Tech. Spec. ISLH Composite (Heatup/Cooldown) P-T Limits



## 8.0 REFERENCES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2. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, 2001 Edition with Addenda through 2002
3. ASTM Standard E 208-81, "Standard Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," American Society for Testing and Materials, Philadelphia, PA
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5. US Nuclear Regulatory Commission, "Radiation Embrittlement of Reactor Vessel Materials," Regulatory Guide 1.99, Revision 2, May 1988
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9. ANO Exemption Letter 1CAN031403.
10. Timoshenko, S.P., and Goodier, J.N., "Theory of Elasticity," Third Edition, McGraw-Hill Book Company, 1970
11. PVRC Ad Hoc Group on Toughness Requirements, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," Bulletin No. 175, Welding Research Council, August 1972

## 9.0 CERTIFICATION

Pressure-Temperature limits for the ANO-1 reactor vessel have been calculated to satisfy the requirements of 10 CFR Part 50, Appendix G using analytical methods and acceptance criteria of the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, 2001 Edition with Addenda through 2002.

6/16/14

Silvester J. Noronha, Principal Engineer
Component Analysis and Fracture Mechanics

Date

This report has been reviewed for technical content and accuracy.

6/16/14

Douglas E. Killian, Technical Consultant
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Date

Verification of independent review.

6/17/14

Timothy M. Wiger, Engineering Manager
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Date

This report is approved for release

6/16/2014

David Skulina, Project Manager

Date

**Attachment 5**

**1CAN081403**

**Pressurized Thermal Shock Assessment**

## Pressurized Thermal Shock Assessment

| RV Material Identification] | | | | Chem. Composition | | Chem. Factor | Initial $RT_{NDT}$ (°F) | 54 EFPY Peak Fluence (n/cm$^2$) | Fluence Factor | $\Delta RT_{NDT}$ (°F) | Margin (°F) | $RT_{PTS}$ (°F) | Screening Criterion (°F) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Material Location | Material Type | Material ID | Heat No. | Cu wt% | Ni wt% | | | | | | | | |
| LNBF at start of 12" thickness | ASTM A508 Cl. 2 | AYN 131 | 528360 | 0.03 | 0.70 | 20.0 | 27.5 | 1.13E+18 | 0.442 | 8.8 | 27.3 | 63.6 | 270 |
| LNBF at start of 8.44" thickness | ASTM A508 Cl. 2 | AYN 131 | 528360 | 0.03 | 0.70 | 20.0 | 27.5 | 1.42E+18 | 0.491 | 9.8 | 27.6 | 64.9 | 270 |
| LNBF at LNBF to Upper Shell Weld | ASTM A508 Cl. 2 | AYN 131 | 528360 | 0.03 | 0.70 | 20.0 | 27.5 | 1.21E+19 | 1.053 | 21.1 | 33.3 | 81.9 | 270 |
| Upper Shell Plate 1 | SA 533 Gr. B Cl. 1 | C5120-2 | C5120-2 | 0.17 | 0.55 | 122.75 | -10 | 1.34E+19 | 1.081 | 132.7 | 34.0 | 156.7 | 270 |
| Upper Shell Plate 2 | SA 533 Gr. B Cl. 1 | C5114-2 | C5114-2 | 0.15 | 0.52 | 105.6 | -10 | 1.34E+19 | 1.081 | 114.2 | 34.0 | 138.2 | 270 |
| Lower Shell Plate 1 at 8.44" thickness | SA 533 Gr. B Cl. 1 | C5120-1 | C5120-1 | 0.17 | 0.55 | 122.75 | -10 | 1.32E+19 | 1.077 | 132.2 | 34.0 | 156.2 | 270 |
| Lower Shell Plate 2 at 8.44" thickness | SA 533 Gr. B Cl. 1 | C5114-1 | C5114-1 | 0.15 | 0.52 | 105.6 | 0 | 1.32E+19 | 1.077 | 113.8 | 34.0 | 147.8 | 270 |
| LNBF to US Circ. Weld | Linde 80 | WF-182-1 | 821T44 | 0.24 | 0.63 | 177.95 | -84.2 | 1.21E+19 | 1.053 | 187.4 | 59.2 | 162.4 | 300 |
| US 1 to US 2 Axial Welds (2) | Linde 80 | WF-18 | 8T1762 | 0.19 | 0.57 | 167.0 | -48.6 | 1.06E+19 | 1.016 | 169.7 | 66.6 | 187.7 | 270 |
| US to LS Circ. Weld | Linde 80 | WF-112 | 406L44 | 0.27 | 0.59 | 182.55 | -98.0 | 1.28E+19 | 1.069 | 195.1 | 60.6 | 157.7 | 300 |
| LS 1 to LS 2 Axial Welds (2) | Linde 80 | WF-18 | 8T1762 | 0.19 | 0.57 | 167.0 | -48.6 | 1.14E+19 | 1.037 | 173.1 | 66.6 | 191.1 | 270 |

LNBF = Lower Nozzle Belt Forging, US = Upper Shell, LS = Lower Shell, Circ. = Circumferential