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Input to Items 2A and 2C of NRC's Questions on Relief Request for Inspection of Transition Piece to Bottom Head Weld of Reactor Vessel at Arkansas Nuclear One - Unit 1

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ITEM 2A

1. EVALUATION OF POTENTIAL DEGRADATION MECHANISMS

The transition piece to bottom head weld is subjected to operation at an elevated temperature of approximately 554°F for long times (on the order of 200,000 hours). The potential effects of thermal aging, irradiation embrittlement, and corrosion must be considered as potential degradation mechanisms at the bottom head weld at Arkansas Nuclear One Unit 1 (ANO-1). The change in section from the adjacent transition piece in this area may lead to increased stresses, making it susceptible to thermal fatigue. Finally, the impact of fabrication-related defects must be evaluated.

Corrosion

As noted in EPRI report NP-5461 [1], potential corrosion mechanisms include:

- General corrosion
- Stress corrosion cracking (SCC)
  - Intergranular (IGSCC)
  - Transgranular (TGSCC)
  - Irradiation assisted (IASCC)
- Erosion-corrosion
- Crevice corrosion
- Pitting
- Intergranular attack
- Hydrogen embrittlement
- Microbiologically influenced corrosion (MIC)

The entire inside surface of the reactor vessel is clad with stainless steel with nominal composition and properties equivalent to those of Type 304 [2]. The purpose of the cladding is to provide corrosion protection to the low alloy steel plate and weld metal and to assist in

maintaining reactor coolant water purity. The general corrosion rate of the stainless steel in a PWR environment is extremely low. Thus, the contributions of corrosion products to degradation of fuel rod heat transfer, to turbidity of the water, or to acting as a source of activated products that might contribute to dose rates are minimized. The stainless steel is also extremely resistant to pitting or crevice corrosion in the controlled purity PWR coolant. While stainless steels have been shown to be subject to SCC in Boiling Water Reactors, the resistance of stainless steel weld metal to SCC is much greater [3]. Further, the susceptibility of the reactor pressure vessel base material to growth of SCC is extremely low, especially under the non-oxidizing conditions in the PWR reactor vessel. In the extremely unlikely event that reactor water does contact the low alloy steel, corrosion degradation would be limited to general corrosion or galvanic corrosion in the low alloy steel at rates much less than 1 mpy [4]. The presence of stainless steel cladding, a material with extremely high resistance to velocity effects, will preclude erosion-corrosion.

Hydrogen embrittlement most often occurs when materials are charged with hydrogen as a result of processing (e.g., plating operations), exposed to a service environment with a high partial pressure of hydrogen and elevated temperature (e.g., refinery vessels), or as a result of hydrogen produced from corrosion. Environmentally assisted cracking may be the result of SCC, the active interaction between stresses and corrosion processes, or hydrogen embrittlement, where hydrogen produced from the corrosion process embrittles the structure at the crack tip producing a brittle type failure. The latter mechanism may be considered to produce the same level of degradation as that described for SCC in the paragraphs above (i.e., degradation due to SCC or hydrogen embrittlement is considered to be unlikely for this location). Since the amount of corrosion that occurs at the cladding/coolant interface is so small, the small amount of hydrogen that might diffuse through the stainless steel to the low alloy steel will also not be sufficient to cause hydrogen embrittlement.

The presence of stainless steel cladding effectively eliminates corrosion as a degradation mechanism of the low alloy steel structural components.

Neutron Embrittlement

As shown in Table 1, no fluence is reported for the transition piece to lower head weld. As such, there is no shift associated with this weld.

Thermal Embrittlement

Ferritic materials, including pressure vessel steels such as those used for the ANO-1 reactor pressure vessel, can be subjected to temper embrittlement, a loss of toughness as a result of long term exposure to elevated temperatures, observed in the range 750°F to 900°F. This form of thermal embrittlement is produced by the segregation of impurities such as arsenic, antimony, and tin to grain boundaries. The exposure temperature for the ANO-1 pressure vessel (as for other LWR reactor vessels) is far too low to produce thermal embrittlement.

Fatigue

As noted above, the change in section at the transition piece to bottom head area makes it a potential area of susceptibility for thermal fatigue. However, the discussion of stresses in the following section demonstrates that this vessel location will not be subjected to transient events, either in large numbers or of a magnitude that will produce thermal fatigue. Fatigue crack growth would be contributed mainly by startup/shutdown transients and is expected to be very small for the remaining 146 startup/shutdown cycles at ANO-1 [6]. For example, assuming an upper bound uniform through-wall stress of approximately 30 ksi results in a stress intensity (K) of approximately 100 ksi$\sqrt{in}$ for a flaw one-half the wall thickness with an aspect ratio of 1/6. The crack growth for these 146 cycles would be approximately 0.12 inches using the Appendix A fatigue crack growth curves with $R \leq 0.25$ [10]. This growth is insignificant compared to the thickness of the vessel shell.

Fabrication Defects

For the ANO-1 reactor vessel preservice and first 10-year ISI inspections, there were no reported flaws. As such, any fabrication-related defects are expected to be relatively small and insignificant.

Conclusions

This evaluation has demonstrated that service-induced degradation of the transition piece to bottom head weld as a result of corrosion, fatigue, or thermal embrittlement mechanisms is extremely unlikely. The primary contributor to the presence of flaws in this weld is due to fabrication. Prior vessel inspections did not identify any flaws such that the existence of any significant fabrication related defects in the bottom head weld is unlikely. Relief from inspection of the transition piece to bottom head weld in the ANO-1 reactor vessel appears to be justified.

2. STRESSES AND LOADS

Stresses acting on the transition piece to bottom head weld consist of the following:

- Pressure
- Thermal Transients
- Expansion/Contraction Stresses of Cladding
- Loads Associated with Welded Attachments (Flow Stabilizer Lugs)
- Stresses Resulting from Bottom Head Penetrations
- Weld Residual Stresses

The significance of these loads in relation to the weld location are discussed below.

Pressure

An analysis due to internal pressure of 2242 psig was performed in Reference 7. The geometrical details of the reactor pressure vessel shown in Figure 1 were used to construct the finite element model shown in Figure 2. The resulting typical through-wall pressure stresses in the transition piece to bottom head weld location are shown in Table 2. The stresses are less than those expected for a clean spherical shell due to the adjacent transition piece.

Thermal Transients

Several transients are described in the original Reactor Vessel Stress Reports [2]. However, it was determined that the most significant transients in the bottom head region are those associated with plant heatup and cooldown. Analyses were performed using the finite element model shown in Figure 2 to determine the stresses and temperatures associated with the heatup and cooldown transients. Maximum through-wall thermal stresses are shown in Table 2. Due to the adjacent thickness change of the transition piece, the thermal stress distributions are altered from those expected for a shell remote from a discontinuity.

Expansion and Contraction Stresses

The expansion and contraction stresses result from differences in the coefficient of thermal expansion between the stainless steel clad and the low alloy pressure vessel steel. As required by the ASME Code, during construction, post weld heat treatment (PWHT) was performed on the vessel at ANO-1 following welding operations, including the application of the cladding. PWHT is intended to reduce residual stresses arising from welding. This was achieved by subjecting the entire vessel to a temperature of about 1100° - 1200°F for 12 to 48 hours and then gradually cooling it uniformly to room temperature.

It is generally believed that due to PWHT, the vessel is stress free at normal operating temperatures [8]. However during cooling to room temperature, tensile residual stresses are generated in the stainless steel cladding because stainless steel has a higher coefficient of thermal expansion than the low alloy pressure vessel steel. It has been established that the tensile stress in the stainless steel at room temperature is on the order of yield stress (30-35 ksi for stainless steel) [9]. The cladding stress is compensated by an essentially uniform compressive force in the low alloy base metal. The postulated distribution of the clad stress at room temperature is shown in Figure 3 for the 5" thick bottom head. Because of this stress distribution, the associated cladding stress intensity factor from an inside surface crack tends to have a very sharp exponential decay through the wall of the vessel for inside surface cracks. Since the stress in the base metal is compressive, cladding induced stresses are not a factor for embedded flaws, the type of flaw most likely to be encountered during inspection. At the normal operating temperature (554°F), the effects of cladding stresses diminish due to the relative coefficient of thermal expansion.

Loads Associated with Welded Attachments (Flow Stabilizers)

In the original design, there were twelve flow stabilizer plates welded to the lower head in the vicinity of the transition piece to lower head weld with a size of approximately 37 inches by 10 inches. Before plant operation, these plates were cut-out leaving only 1 inch of the original 10 inch width lug protruding from the surface of the cladding. As such, the mechanical loads on these lugs are insignificant. The plates were made from stainless steel and, as such, there is no differential thermal expansion stress between the plate and the stainless steel cladding. The location of these plates extend from the transition piece to approximately 1 inch below the centerline of the weld.

Loads from Bottom Head Penetrations

The loads on the bottom head due to the fifty-two (52) 3/4-inch nominal pipe penetrations were analyzed in the original Stress Report [2]. These penetrations are at least 10 inches

away from the transition piece to bottom head weld and, as such, the weld location is unaffected by the loads from these penetrations.

Weld Residual Stresses

As a result of welding the transition piece to bottom head, weld residual stresses are developed. These stresses are expected to be relieved by the post weld heat treatment (PWHT) following the vessel fabrication. However, it is postulated that a small fraction of these stresses still remain after PWHT [8]. It has been shown in Reference 8 that the weld residual stress can be represented by a cosine-shaped distribution through the wall thickness with a maximum surface tensile stress of 8 ksi. The stress intensity factor associated with this stress distribution is relatively small and may be beneficial at some locations of the vessel wall in fracture mechanics assessment.

ITEM 2C

1. FRACTURE TOUGHNESS AND ASME CODE, SECTION III, APPENDIX G, FLAW SIZE

The critical flaw size in the bottom head region is typically larger than the critical flaw size in the beltline region because the bottom head, unlike the beltline, does not experience significant irradiation effects from exposure to neutron fluence. Although the thickness of the bottom head is less than the beltline region, the overall stress levels are equivalent to those in the beltline because the smaller thickness is compensated by the spherical shape (as opposed to the cylindrical shape of the beltline). For the bottom head weld in question, there are no penetrations or structural discontinuities immediately adjacent to the weld that would necessitate using pressure stresses higher than those in a clean spherical shell. Thus, with similar stresses and a higher available toughness, the flaw tolerance capability of the bottom head region is expected to be significantly better than the beltline region.

To demonstrate the increased flaw tolerance of the bottom head region, comparative analyses were performed in accordance with ASME Code, Section XI, Appendix G to establish curves of allowable pressure versus fluid temperature for various flaw sizes in the bottom head region. Since the results are comparative in nature, the hydrotest P-T curve was selected for evaluation; similar results would be expected for the normal operation P-T curves. Once determined, the bottom head curves were compared to the Tech. Spec. P-T curve to demonstrate the margin present compared to the controlling 1/4T beltline flaw.

The analyses utilized the limiting reference temperature for all plate and weld materials in the bottom head region (transition piece-to-lower head weld, with an initial RT_{NDT} of -5°F, and a σ_i of 20°F). A circumferential flaw (i.e., parallel to the weld) was assumed, so axial stresses in the bottom head region were utilized. (Evaluation of an axial flaw would produce essentially the same results.) Three flaw sizes were evaluated: 1/4T, 3/8T, and 1/2T. The 3/8T and 1/2T flaw stress intensity factors were determined by scaling the 1/4T results based on 1995 ASME Code, Section XI, Appendix A methodology. Various fluid temperature transients were evaluated with respect to both outside surface and inside surface flaws, including those identified on the current Tech. Spec. P-T curve, as well as 100°F/hr heatup and cooldown transients. The limiting 100°F/hr heatup results for an outside surface flaw are shown in Figure 4 for all three flaw sizes, and are compared against the currently established Tech. Spec. P-T curve based on the limiting beltline region, for the hydrotest condition. This is a conservative comparison, in that the maximum heatup allowed per the plant Tech. Specs. is only 50°F/hr (i.e., one-half of the 100°F/hr evaluated).

The results shown in Figure 4 demonstrate that the current Tech. Spec. P-T curve for the limiting beltline region significantly bounds that of the lower head region. For flaw sizes as large as 1/2T, significant margin remains. The primary reason for the significant difference between the bottom head and beltline regions is a result of decreased beltline toughness caused by irradiation. This effect is apparent when the adjusted reference temperature (ART) of 168.6°F (using the fluence estimated for 15 EFPY, as specified on the Tech. Spec. P-T curve) for the limiting beltline material (middle circumferential weld) is compared to the 35°F

value for the limiting bottom head material (transition piece-to-bottom head weld). Thus, the fracture toughness capability of the bottom head region is significantly bounded by the existing Tech. Spec. P-T curve for the beltline region.

REFERENCES

1. J.F. Copeland, et. al., "Component Life Estimation: LWR Structural Materials Degradation Mechanisms", EPRI report NP-5461, prepared by Structural Integrity Associates, Inc., September, 1987.

2. "Design Report - Arkansas Power & Light Company, Arkansas Power #1, Reactor Vessel and Closure Head", Including Stress Analysis Reports Numbers 1 through 11, Customer Order No. M-1-6600, B&W Contract No. 620-0008-51/52, March 1974, SI File Numbers: ANO-10Q-209 through ANO-10Q-216.

3. W. S. Hazelton, W. H. Koo, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping", NUREG-0313, Rev. 2, USNRC, January, 1988.

4. G. J. Licina, "Nuclear Power", Authorized Reprint from ASTM Manual 20, 1995.

5. SIR-95-022, "Estimation of Reactor Pressure Vessel Fracture Toughness at Arkansas Nuclear One Unit 1", March, 1995, Rev.0.

6. Telecom Report: N. Cofie (SI), E. Burns (ANO), Subject: "Cycles for RPV Analysis", Dated 2/6/95, SI File: ANO-10Q-103.

7. SI Calculation Package: ANO-10Q-302, "ANO-1 Reactor Vessel Stress Analysis", Rev. 0.

8. ASME Section XI Task Group on Reactor Vessel Integrity Requirements, "White Paper on Reactor Vessel Integrity Requirements for Level A and B Conditions" EPRI TR-100251, January 1993.

9. ASME Boiler and Pressure Vessel Code, 1989 Edition, Section III, Appendices.

10. ASME Boiler and Pressure Vessel Code, 1989 Edition, Section XI.

Table 1

Initial and Adjusted RT_{NDT} of Selected Welds at Arkansas Nuclear One Unit 1 [5]

[illegible] Part Name	Weld Metal Designation	Material Type	Filler Heat (Flux Lot)	Estimated Initial RT_{NDT} (°F)	Chemistry Cu (wt %)	Chemistry Ni (wt %)	Chemistry Factor (°F)	Adjustments For 1/4T: ΔRT_{NDT} (°F)	Margin σ_Δ (°F)	Margin σ_I (°F)	ART_{NDT} (°F)
Upper Shell to Lower Shell Weld (Middle Circumferential) 01-004 *(Limiting Beltline)*	WF-112	ASA/Linde 80	406L44 (8754)	-5	0.31	0.59	195.00	126.4	12.5	20.0	168.8
Transition Piece to Lower Head 01-006 *(Limiting Bottom Head)*	NR	Assumed Linde 80 Flux	Assumed Linde 80 Flux	-5	—	—	not applicable	not applicable	not applicable	20.0	35.0

Fluence Factor Estimates (15 EFPY):

	Wall Thickness (inches) Full	Wall Thickness (inches) 1/4T	Fluence at ID (n/cm^2)	Attenuation at 1/4T $e^{-0.24x}$	Fluence at 1/4T (n/cm^2)	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
Middle Circumferential	8.44	2.11	4.57E+18	0.603	2.75E+18	0.648
Transition Piece to Lower Head	5.00	1.25	0.00	0.741	0.00	0.000

Table 2

Typical Stresses for Bottom Head-To-Transition Piece Weld

Axial Stresses

	Stresses (ksi)					
Distance From I.D. (in)	Bolt-Up @T=70°F	Pressure 2242 psi	Heatup 50°F/hr		Cooldown 100°F/hr	
			Stress	Temp.	Stress	Temp.
0	-	17.276	-4.172	600.46	8.109	292.85
0.187	-	17.341	-4.236	600.46	7.771	292.85
0.188	-	17.210	-4.410	600.46	7.117	292.85
0.688058	-	17.560	-3.310	598.10	5.316	297.11
1.188116	-	17.910	-2.310	595.99	3.700	300.91
1.688174	-	18.260	-1.390	594.15	2.219	304.25
2.188231	-	18.610	-0.570	592.56	0.894	307.14
2.688289	-	18.940	0.150	591.23	-0.271	309.58
3.188347	-	19.270	0.770	590.14	-1.292	311.58
3.688404	-	19.600	1.310	589.30	-2.171	313.13
4.188462	-	19.940	1.760	588.71	-2.914	314.25
4.68852	-	20.320	2.110	588.36	-3.502	314.94
5.188577	-	20.700	2.410	588.24	-4.003	315.17

Hoop Stresses

	Stresses (ksi)					
Distance From I.D. (in)	Bolt-Up @T=70°F	Pressure 2242 psi	Heatup 50°F/hr		Cooldown 100°F/hr	
			Stress	Temp.	Stress	Temp.
0	-	16.629	-7.57	600.46	12.98	292.85
0.187	-	16.647	-7.59	600.46	12.67	292.85
0.188	-	16.610	-7.95	600.46	12.06	292.85
0.688058	-	16.710	-6.93	598.10	10.43	297.11
1.188116	-	16.840	-5.97	595.99	8.91	300.91
1.688174	-	16.970	-5.10	594.15	7.53	304.25
2.188231	-	17.090	-4.33	592.56	6.32	307.14
2.688289	-	17.220	-3.66	591.23	5.24	309.58
3.188347	-	17.340	-3.07	590.14	4.31	311.58
3.688404	-	17.470	-2.58	589.30	3.52	313.13
4.188462	-	17.590	-2.16	588.71	2.86	314.25
4.68852	-	17.710	-1.82	588.36	2.34	314.94
5.188577	-	17.820	-1.57	588.24	1.94	315.17

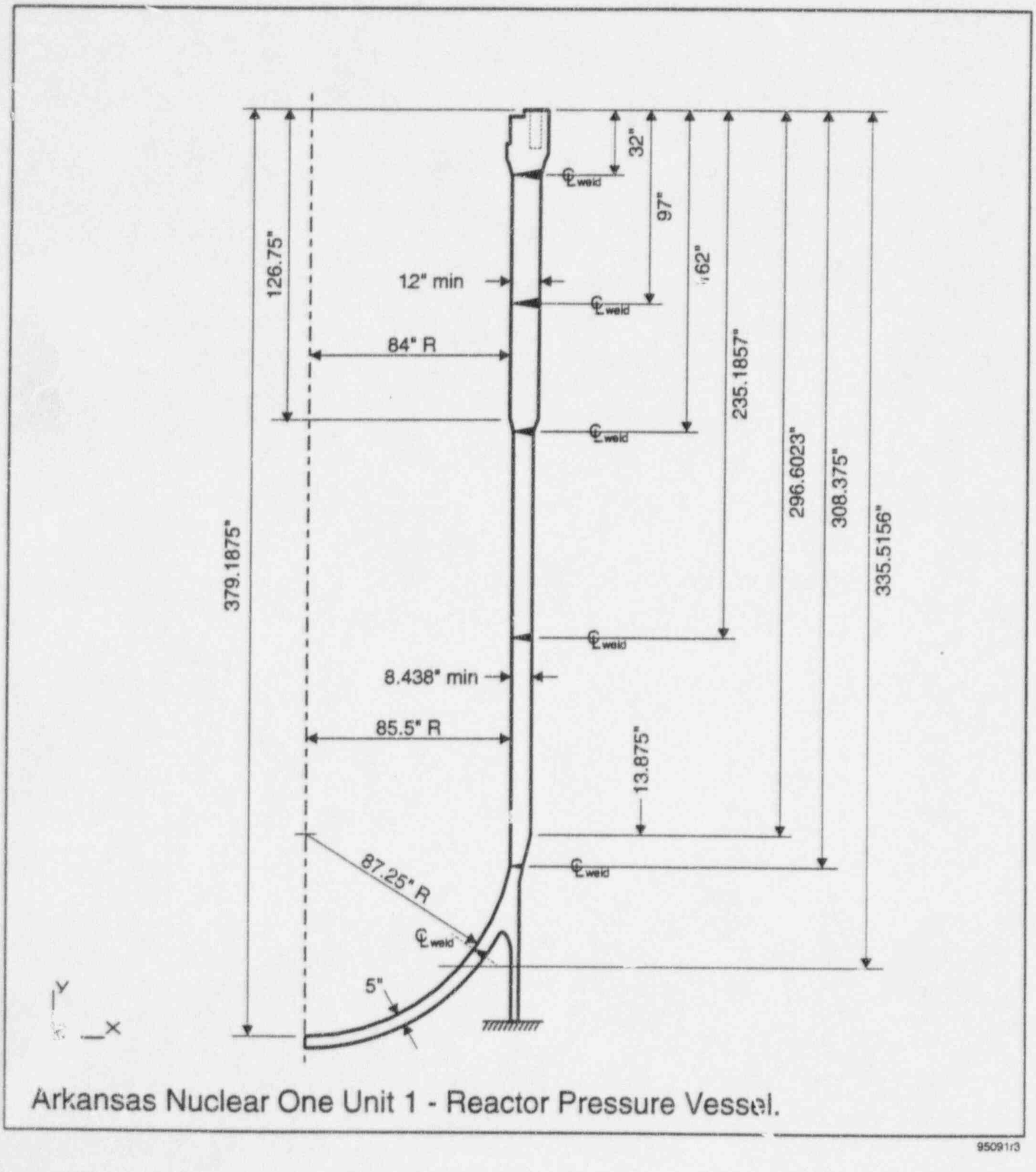

Figure 1. Vessel Geometry at ANO-1 - Beltline and Bottom Head Regions

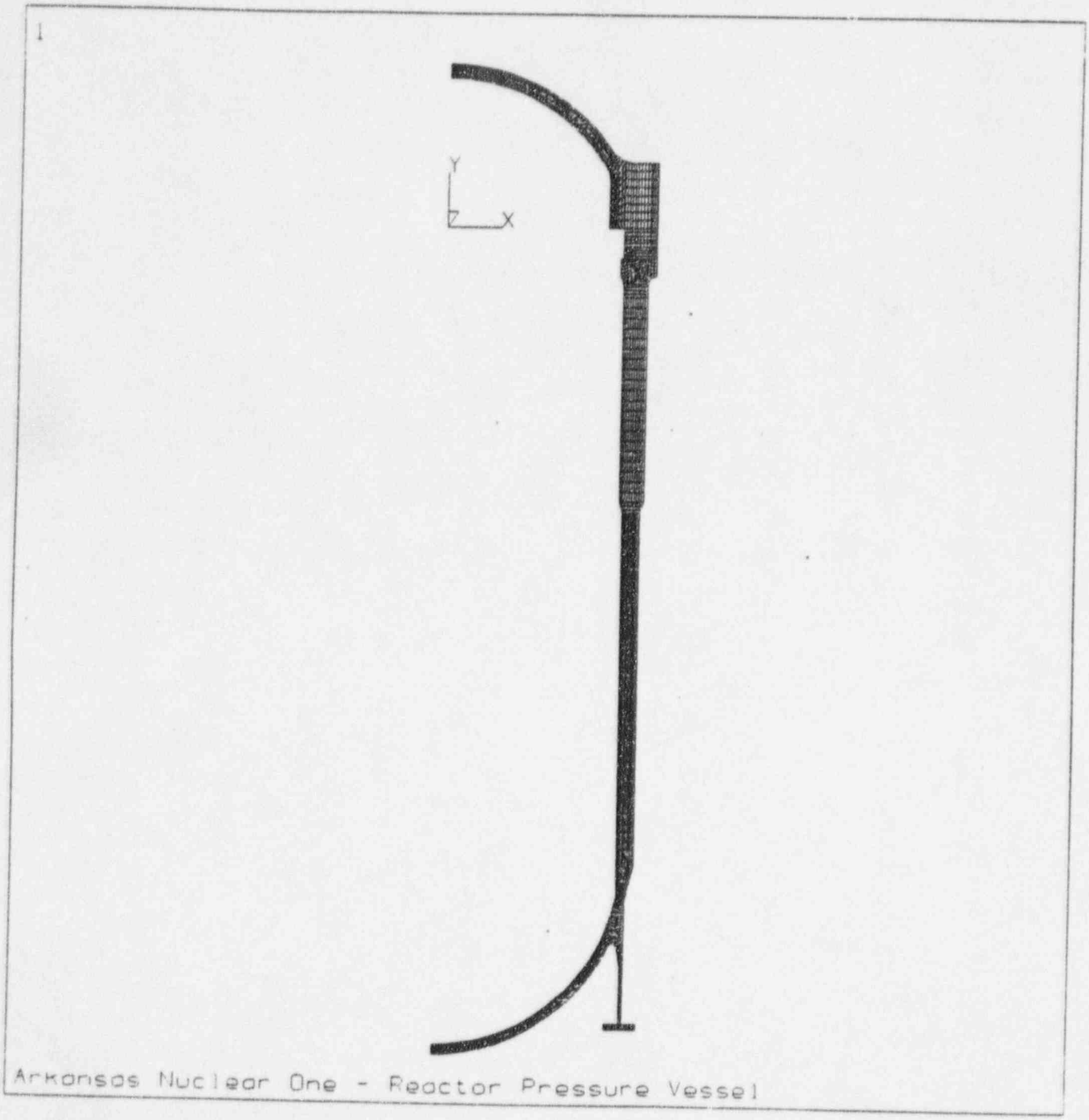

Figure 2 Reactor Vessel Axisymmetric Finite Element Model

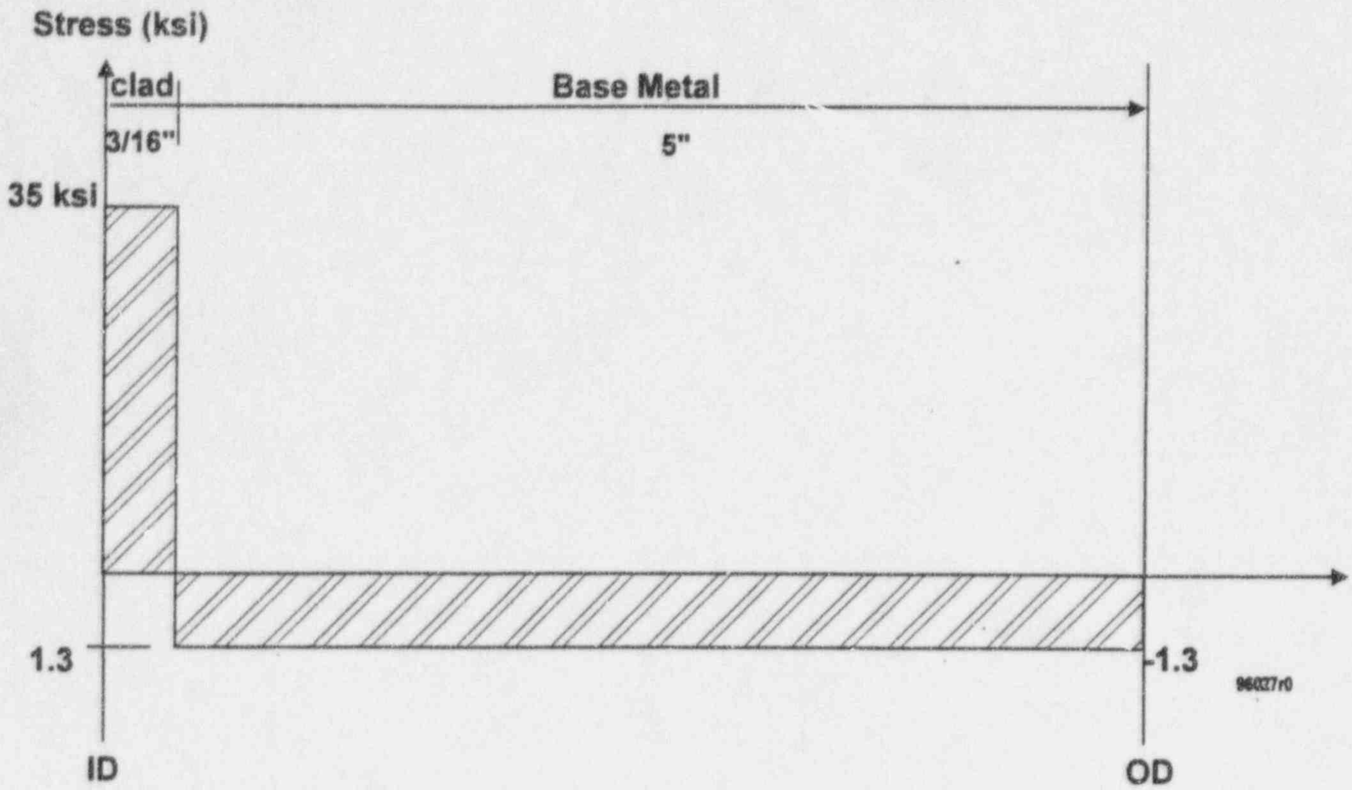

Figure 3. Through-Wall Clad Residual Stress Distribution for Bottom Head

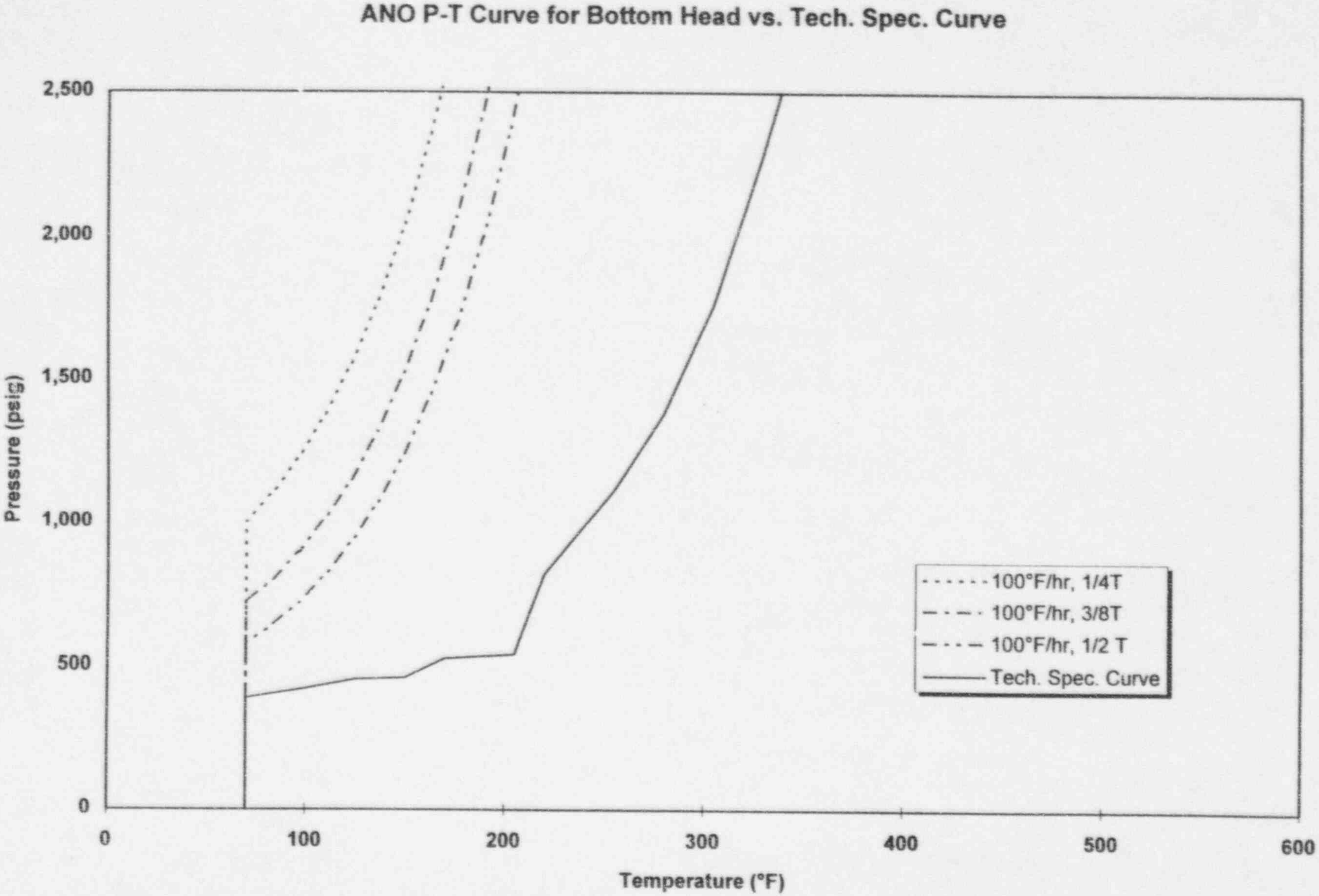

Figure 4. Comparison of Bottom Head P-T Limits for Hydrotest Conditions as Compared to 15 EFPY Technical Specification P-T Limits

P-T.XLS [P-T Curve] 2/27/96

Attachment 3

Flaw Acceptance Standards
for Arkansas Nuclear One Unit 1
Reactor Pressure Vessel Weld Inspections

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