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Input to Items A and C of NRC's Questions on Relief Request for Inspection of Transition Piece to Bottom Head Weld at Crystal River Unit 3

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ITEM A

1. EVALUATION OF POTENTIAL DEGRADATION MECHANISMS

The transition piece to bottom head weld is subjected to operation at an elevated temperature of approximately 575°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 Crystal River Unit 3. 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, very little hydrogen will diffuse through the stainless steel to the low alloy steel.

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, for the head transition piece, a fluence of 4.79E16 n/cm^2 has been reported [5]. For purposes of this evaluation, this fluence has also been conservatively assumed for the lower shell to head transition weld (WF-154). Using the methodology provided in Reg. Guide 1.99, Rev. 2, this corresponds to a fluence factor of 0.052, resulting in a reference shift of 6.9°F and adjusted reference temperature (ART) of 16.9F°. This reduction in fluence results in a shift which is significantly less than what is reported for the beltline materials, as can be seen in Table 1.

Thermal Embrittlement

Ferritic materials, including pressure vessel steels such as those used for the Crystal River Unit 3 reactor pressure vessel, can be subject to a loss of toughness as a result of long term exposure to elevated temperatures. Temper embrittlement, produced by the segregation of impurities such as arsenic, antimony, and tin to grain boundaries is a strong function of temperature. The effect is observed in the range 750°F to 900°F. The exposure temperature for the Crystal River Unit 3 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 following discussion of stresses demonstrate 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 will be contributed mainly by startup/shutdown transients. Crack growth is expected to be very small for the remaining 167 startup/shutdown cycles at Crystal River Unit 3 [6].

Fabrication Defects

With respect to Crystal River Unit 3, both preservice, as well as the first 10-year ISI inspections, did not identify any reported flaws. As such, in the unlikely event that fabrication-related defects are found, they 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 that weld is due to fabrication. Prior vessel inspections did not identify any flaws and as such the existence of fabrication related defects in the bottom head weld is unlikely. Relief from inspection of the transition piece to bottom head weld in the Crystal River Unit 3 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

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

Pressure

An analysis due to internal pressure of 2240 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 through-wall pressure stresses in the transition piece to bottom head weld location are shown in Table 2.

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. These stresses are relatively small compared to the pressure stresses. Due to the adjacent thickness change of the transition piece, the thermal stress distributions are altered from those expected for a shell remote of 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 Crystal River Unit 3 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 the operating temperature of approximately 575°F [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 distribution at room temperature is shown in Figure 3 for the 5" thick bottom head. Because of this stress distribution, the associated stress intensity factor 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 normal operating temperature (575°F), the effects of cladding stresses diminish due to the relative coefficient of thermal expansion. It is predicted that the cladding-induced tensile stress reduces to a value of approximately 5 ksi at the normal operating temperature, with a similar percent reduction in the base metal.

Loads Associated with Welded Attachments (Flow Stabilizers)

In the original design, there are 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 and leaving only 1 inch of the original 10 inch width 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 1 inch below the centerline of the weld.

Loads from Bottom Head Penetrations

The loads on the bottom head due to the fifty-six (56) 3/4-inch penetrations were analyzed in the original Stress Report [10]. 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.

ITEM C

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 smaller than the beltline region, the stresses due to internal pressure are curtailed in the bottom head 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 appurtenances immediately adjacent to the weld that would necessitate the use of a stress concentration factor in determining the stresses. Thus, with lower stresses and a higher available toughness, the flaw tolerance capabilities of the bottom head region are expected to be significantly higher than the beltline region.

To demonstrate the increased flaw tolerance of the bottom head region, 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. These analyses utilized the limiting reference temperature for all plate and weld material in the bottom head region (RT_{NDT} = 16.9°F). Three flaw sizes were evaluated: 1/4T, 3/8T, and 1/2T. Various fluid temperature heatup and cooldown rates ranging from that identified on the current P-T Tech.

Spec. curve up to 100°F/hr were evaluated with respect to both outside surface and inside surface flaws. The limiting 100°F/hr results 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.

The results shown in Figure 4 demonstrate that the current 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 208°F (using the fluence specified on the Tech. Spec. P-T curve) for the limiting beltline material (upper shell plate) is compared to the 16.9°F value described above for the limiting bottom head material (head transition piece). 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. Stress Analysis Report #13, "Thermal/Mechanical Analysis of the Reactor Vessel Shell for Florida Power Corporation, Crystal River Unit 3", Rev. 1, January 1974, B&W Contract No. 620-0007-51.

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. B&W-2108, Rev. 1, "Fluence Tracking System", B&W Nuclear Service Company, May 1992.

6. Procedure SP-296, "Documentation of Allowable Operating Transient Cycles", Rev. 11.

7. SI Calculation Package, FPC-01Q-301, "Reactor Pressure Vessel Finite Element 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. Design Report #6 "Thermal Mechanical Stress Analysis of Reactor Lower Head & Support Skirt", Vol. 2 of 2.

Table 1

Initial and Adjusted RT_{NDT} of Selected Plates and Welds at Crystal River Unit 3

Part Name & Material	MK No.	Heat No.	Slab No.	Estimated Initial RT_{NDT} °F	Chemistry Cu wt %	Chemistry Ni wt %	Chemistry Factor °F	Adjustments For 1/4 T: ΔRT_{NDT} °F	Margin: σ_{Δ}, °F	Margin: σ_I, °F	ART_{NDT} °F
Upper Shell SA-533-65 Grade B, Class 1 *(Limiting Beltline)*	A1-207-1	C4344	2	20	0.20	0.54	141.80	154.5	17.0	0.0	208.5
Head Transition Piece SA-508-64, Class 2 (Code Case 1332-3) *(Limiting Bottom Head)*	---	124W295VA1	---	10	0.10	0.80	67.00	3.5	1.7	0.0	**16.9**

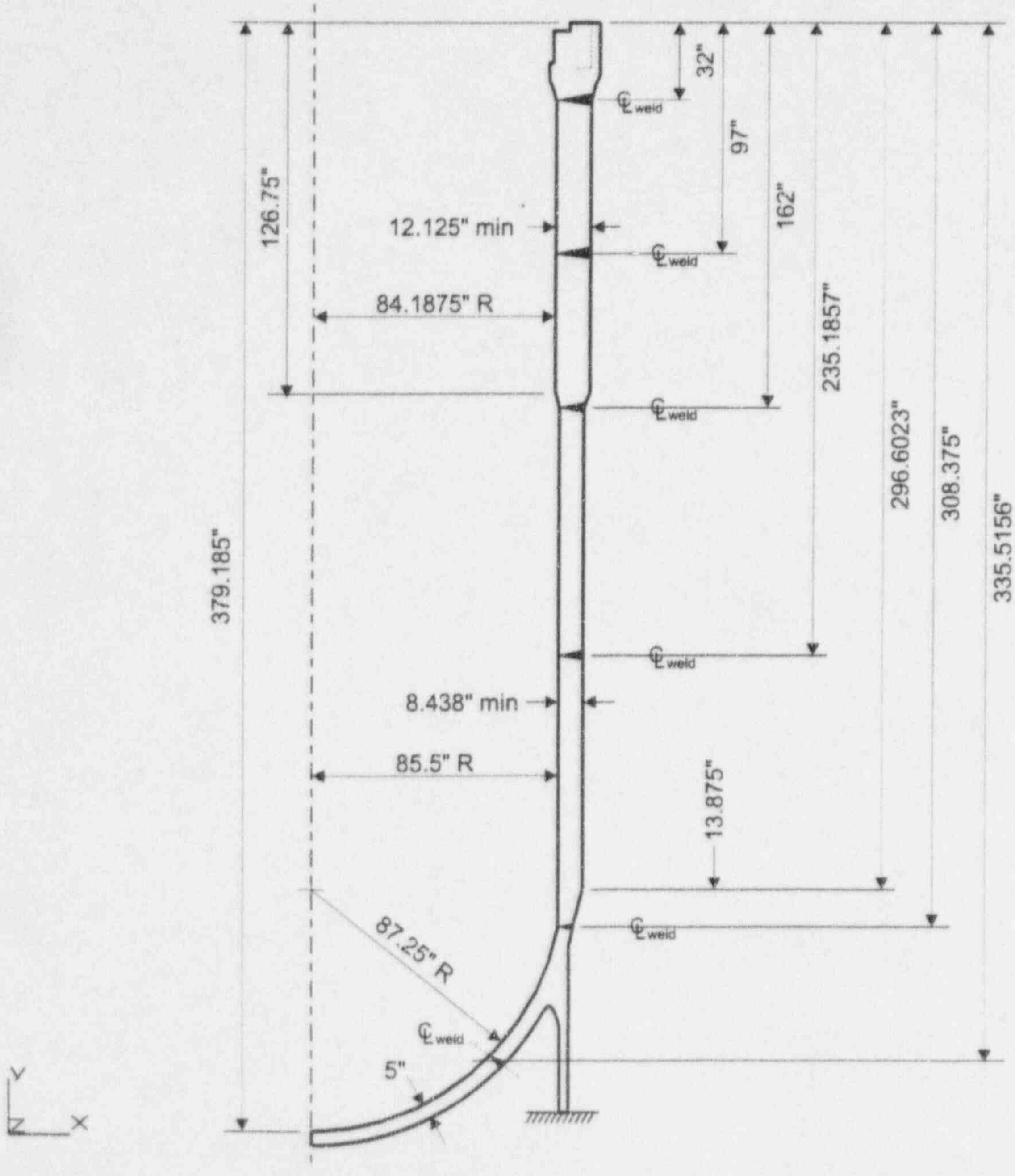
	Wall Thickness (inches) Full	Wall Thickness (inches) 1/4T	Fluence at ID	Attenuation, 1/4T $e^{-0.24x}$	Fluence 1/4 T	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
Upper Shell	8.44	2.11	2.29E+19 n/cm^2	0.603	1.38E+19 n/cm^2	1.090
Head Transition Piece	5.00	1.25	4.79E+16 n/cm^2	0.741	3.55E+16 n/cm^2	0.052

Table 2

Stresses for Bottom Head-To-Transition Piece Weld

Distance, in	Pressure (2240 psi)		End of Heatup 100°F/hr			End of Cooldown 100°F/hr		
	Stress, ksi		Stress, ksi		Temp., °F	Stress, ksi		Temp., °F
	Axial	Hoop	Axial	Hoop	Note 1	Axial	Hoop	Note 2
0.00	19.670	15.080	0.152	-13.220	572.6	-2.536	13.730	301.1
0.50	19.500	15.030	0.423	-11.780	592.7	-2.227	12.530	306.3
1.00	19.290	15.000	0.573	-10.420	587.7	-1.803	11.390	311.0
1.50	19.080	14.970	0.575	-9.235	583.3	-1.277	10.390	315.1
2.00	18.860	14.930	0.405	-8.246	579.6	-0.616	9.573	318.7
2.50	18.650	14.900	0.056	-7.458	576.4	0.194	8.936	321.7
3.00	18.470	14.870	-0.460	-6.862	573.9	1.146	8.472	324.1
3.50	18.330	14.850	-1.132	-6.461	572.0	2.229	8.188	325.9
4.00	18.250	14.840	-1.936	-6.249	570.7	3.423	8.081	327.1
4.50	18.220	14.860	-2.857	-6.243	569.9	4.706	8.172	327.7
5.00	18.190	14.850	-3.804	-6.371	569.7	5.968	8.375	327.9

1. RCS Temperature at 604°F
2. RCS Temperature at 280°F



Crystal River Unit 3 - Reactor Pressure Vessel.

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Figure 1. Vessel Geometry at CR-3 - Beltline and Bottom Head Regions

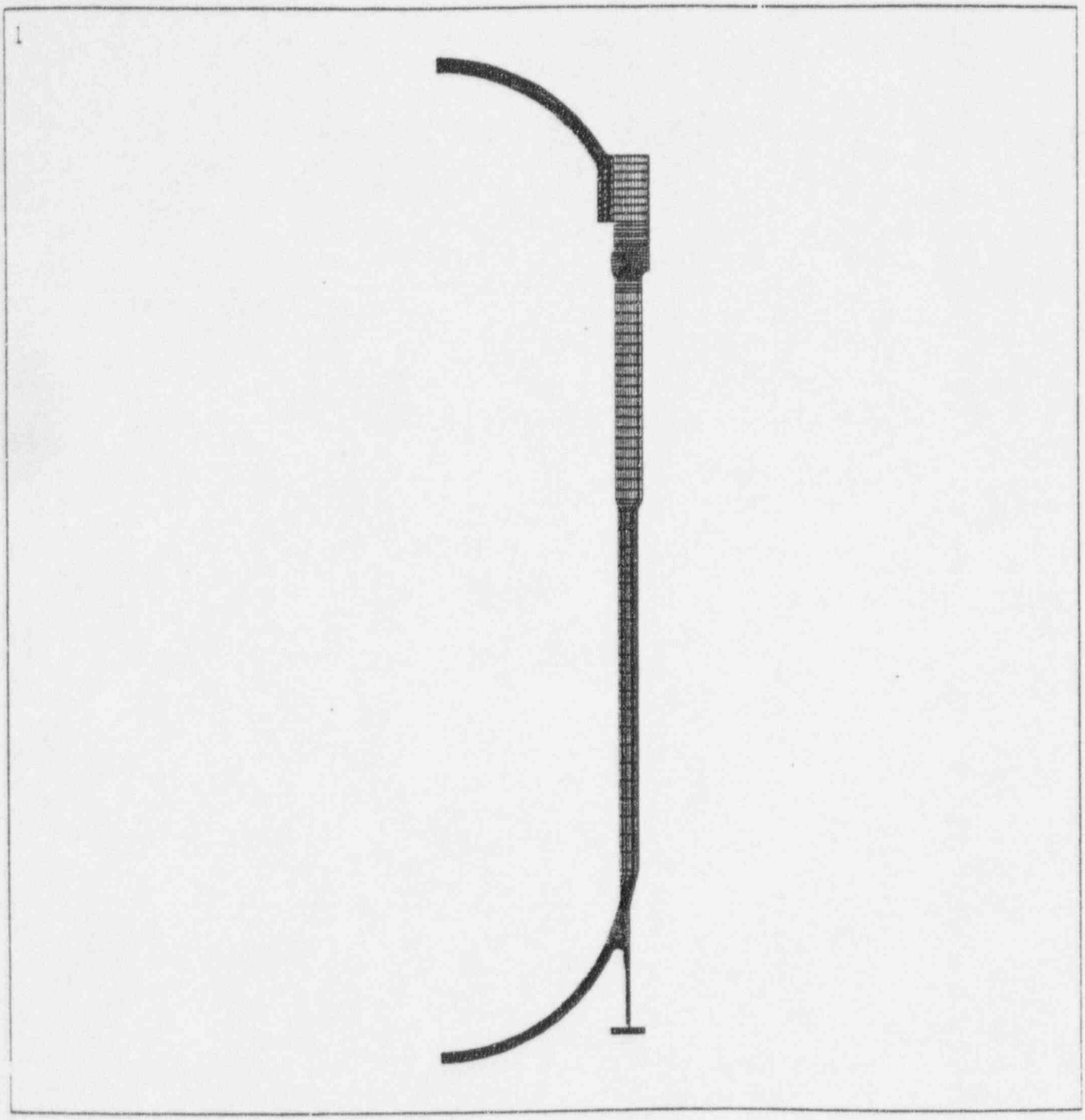

Figure 2. Reactor Vessel Axisymmetric Finite Element Model

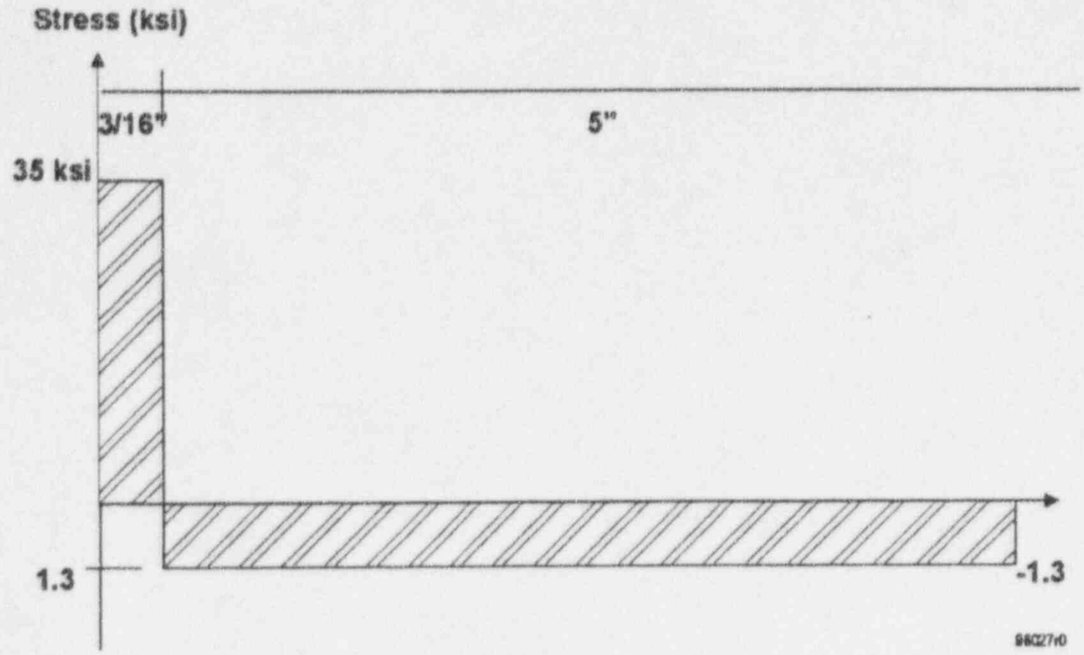

Figure 3. Clad Residual Stress

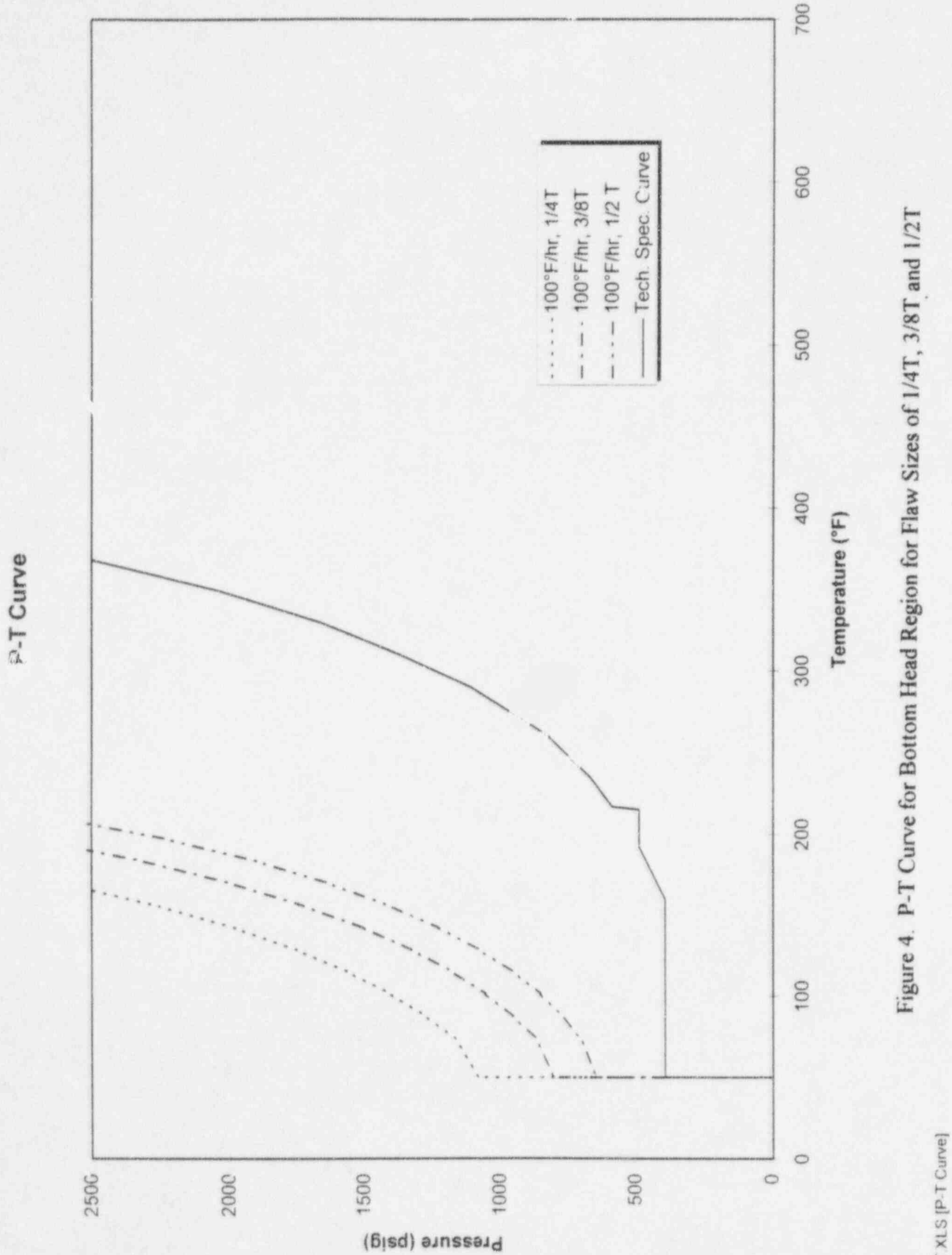

Figure 4. P-T Curve for Bottom Head Region for Flaw Sizes of 1/4T, 3/8T and 1/2T