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Reactor Vessel Upper Shelf Energy Bounding Evaluation For Westinghouse Pressurized Water Reactors

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Executive Summary

Appendix G, "Fracture Toughness Requirements" to 10 CFR Part 50 requires that materials of the reactor vessel beltline must have Charpy upper-shelf energy no less than 50 ft-lbs, "Unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by the Appendix G of the ASME Code." Such margin of safety can be demonstrated by using the proposed ASME Code Section XI, Appendix X criteria and requirements.

The objective of this report is to demonstrate that all participating Westinghouse Owner's Group (WOG) plant reactor vessels have a margin of safety equivalent to that required by Appendix G of the ASME Code. This is accomplished by performing generic bounding evaluations as per the proposed ASME Section XI, Appendix X. The evaluations cover all the WOG plant reactor vessels, except those vessels fabricated by Babcock & Wilcox because those vessels were exempt from this WOG program. A total of forty-three vessels were included in this evaluation.

The bounding analysis utilized unirradiated J-R curve data (i.e., material resistance data) for WOG reactor vessel intermediate and lower shell course materials. These curves were adjusted to reflect expected end-of-life values. As reactor vessels for different size plants have different geometries, representative geometry parameters were chosen for 2, 3 and 4 loop plants. Generic bounding analyses were performed for these cases. Two of the forty-three reactor vessels were evaluated using plant specific parameters. $J_{applied}$ values were then calculated using Linear-Elastic Fracture Mechanics (LEFM) techniques for Level A and B conditions and Elastic-Plastic Fracture Mechanics (EPFM) techniques for Level C and D conditions.

Based upon the results of the bounding analysis, all participating WOG plants meet the ASME Section XI Appendix X criteria.

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1.0 INTRODUCTION

1.1 Background

The question of adequacy of upper shelf fracture toughness for operation of nuclear reactor vessels has been considered and studied for a number of years. The Federal Register, in 10 CFR Part 50, Appendix G [1] contains a minimum upper shelf energy (USE) requirement in terms of Charpy V-notch impact energy. Per Appendix G, the USE must remain above 50 ft-lbs (68 joules) unless it is demonstrated, in a manner approved by the Directorate of Nuclear Reactor Regulation, that lower values of energy will provide margins of safety against fracture equivalent to those required by the ASME Boiler and Pressure Vessel Code, Section III, [2] Appendix G. If this requirement is not met, three actions are required:

1. Perform a volumetric examination of the beltline region, covering the material of concern, and characterize any indications found, per ASME Section XI. [3]

2. Obtain additional evidence of the material fracture toughness.

3. Perform a revised safety analysis using the above information demonstrating adequate margin for continued operation.

In 1982, the Nuclear Regulatory Commission (NRC) published proposed procedures for the analysis required by 10CFR50 for operating reactor vessels as NUREG 0744 [4]. Revision 1 of this NUREG was subsequently issued addressing industry concerns. At the time of publication of this document, the NRC officially requested the ASME Code to recommend criteria for evaluation of reactor vessels which do not meet the Charpy USE requirement of 50 ft-lbs. These criteria have been documented in a letter to the NRC (2-91) [5] and the pending Appendix X of Section XI. Currently, this Code Case is nearing acceptance by the ASME Code, and is being reviewed by the NRC.

More recently, the NRC has issued Generic Letter (GL) 92-01, "Reactor Vessel Structural Integrity," which requested information from utilities regarding compliance of their reactor vessel(s) with 10CFR50, Appendix G requirements through end of license life. The NRC has performed an initial review of all GL 92-01 submittals and placed each plant into one of three categories with respect to compliance with USE requirements:

- Category A: Known Non-compliance
- Category B: Indeterminate Compliance Status
- Category C: Known Compliance

The NRC has informed the industry that two plants (both BWRs) have been identified as being in Category A, six to eight plants have been identified to be in Category C and the majority of the plants fall under Category B. The NRC has requested that the industry demonstrate equivalent margins of safety

by performing a generic bounding analysis in accordance with ASME Section XI, Appendix X. As this issue covers all domestic PWRs and BWRs, the Nuclear Management and Resource Council (NUMARC) has become involved to provide a unified approach in determining a resolution. NUMARC has committed to implement the NRC's request. NUMARC and the industry have decided that the most effective manner to perform the generic bounding analysis would be through the NSSS Owners Groups.

1.2 Purpose

The objective of this report is to demonstrate that all participating Westinghouse Owner's Group (WOG) plant reactor vessels demonstrate a margin of safety equivalent to that required by Appendix G of the ASME Code.

1.3 Overall Approach

To achieve this objective, a generic bounding evaluation using the proposed ASME Section XI, Appendix X criteria has been performed. The evaluation covers participating WOG plant reactor vessels except those fabricated by Babcock and Wilcox because these are exempt from this WOG program.

The approach is as follows: In the first step, Appendix X criteria are identified. Next, all the input to the analysis are defined. Material and mechanical properties are determined using ASME Code minimum values for conservatism. J-R data (i.e., material resistance data) is compiled from available literature. Geometry data are based on the representative values from 2, 3, and 4 loop plants. Additionally, two plant specific cases are evaluated. Next, limiting transients are selected for each condition evaluated. An analysis in accordance with Appendix X requirements is then performed. Finally, applied J integral data is compared to the material allowable J-R data to determine Appendix X compliance.

1.4 Criteria Synopsis

ASME Section XI, Appendix X contains acceptance criteria for three levels of service load conditions:

- Level A and B conditions corresponding to Normal and Upset operational conditions.
- Level C conditions corresponding to Emergency operational conditions
- Level D conditions corresponding to Faulted operational conditions

The Appendix X criteria for these service load conditions are given in the following sections.

1.4.1 Level A and B Conditions

For a postulated semi-elliptical surface flaw with flaw depth to wall thickness ratio a/t = 0.25 and with an aspect ratio, surface length to flaw depth of 6 to 1, and oriented along the weld of concern, two criteria must be satisfied as described below. If the base metal is governing, the postulated flaw must be axially oriented. Smaller flaw sizes may be used on an individual case basis if a smaller size of the above

postulated flaw can be justified. The expected accumulation pressure to be discussed below is the maximum pressure which satisfies the requirement of ASME Section III, NB-7311(b). The two criteria are:

1. The crack driving force must be shown to be less than the material toughness as given below:

$$J_{applied} < J_{0.1}$$

where $J_{applied}$ is the J-integral value calculated for the postulated flaw under pressure and thermal loading where the assumed pressure is 1.15 times expected accumulation pressure, and with thermal loading using the plant specified heatup and cooldown conditions. The parameter $J_{0.1}$ is the J-integral characteristic of the material resistance to ductile tearing ($J_{material}$), as usually denoted by a J-R curve, at a crack extension of 0.1 inch.

2. The flaw must be stable under ductile crack growth as given below:

$$\frac{dJ_{applied}}{da} < \frac{dJ_{material}}{da}$$

at $J_{applied} = J_{material}$

where $J_{applied}$ is calculated for the postulated flaw under pressure and thermal loading for all service Level A and B conditions where the assumed pressure is 1.25 times expected accumulation pressure, with thermal loading as is defined in Section 3.0.

The J-integral resistance versus crack growth curve used should reflect a conservative bound representative of the vessel material under evaluation.

1.4.2 Level C Conditions

When the upper shelf Charpy energy of any weld material is less than 50 ft-lb, postulate interior semi-elliptic surface flaws with their major axis oriented along the weld of concern and the flaw plane oriented in the radial direction. Postulate both interior axial and circumferential flaws and use the toughness properties for the corresponding orientation. Consider postulated surface flaws with depths up to one tenth the base metal wall thickness, plus the clad, but with total depth not to exceed 1.0 inch and with aspect ratios of 6 to 1 surface length to flaw depth. Smaller flaw sizes may be used on an individual case basis if a smaller size can be justified. For these evaluations, two criteria must be satisfied, as described below:

1. The crack driving force must be shown to be less than the material toughness as given below:

$$J_{applied} < J_{0.1}$$

where $J_{applied}$ is the J-integral value calculated for the postulated flaw in the beltline region of the reactor vessel under the governing level C condition. $J_{0.1}$ is the J-integral characteristic of the material resistance to ductile tearing ($J_{material}$), as usually denoted by a J-R curve test, at a crack extension of 0.1 inch.

2. The flaw must also be stable under ductile crack growth as given below:

$$\frac{dJ_{applied}}{da} < \frac{dJ_{material}}{da}$$

at $J_{applied} = J_{material}$

where $J_{applied}$ is calculated for the postulated flaw under the governing level C condition. The J-integral resistance versus crack growth curve shall be a conservative representation of the vessel material under evaluation.

1.4.3 Level D Conditions

When the upper shelf Charpy energy of any weld material is less than 50 ft-lb, postulate interior semi-elliptic surface flaws with their major axis oriented along the weld of concern and the flaw plane oriented in the radial direction with aspect ratio of 6 to 1. Postulate both interior axial and circumferential flaws and use the toughness properties for the corresponding orientation. Consider postulated surface flaws with depths up to one tenth the base metal wall thickness, plus the clad, but with total depth not to exceed 1.0 inch and with aspect ratios of 6 to 1 surface length to depth. Smaller flaw sizes may be used on an individual case basis if a smaller size can be justified. For these evaluations, the following criterion must be met.

The postulated flaw must be stable under ductile crack growth as given below:

$$\frac{dJ_{applied}}{da} < \frac{dJ_{material}}{da}$$

at $J_{applied} = J_{material}$.

where $J_{applied}$ is calculated for the postulated flaw under the governing level D condition. The material property to be used for this assessment is the best estimate J-R curve.

1.4.4 Safety Margins

Margins of safety have been incorporated in the analysis in a number of ways. First, a flaw having depth of 1/4 the wall thickness (1/4 t) is postulated to exist. Second, conservatism is introduced to level A and B transients by incorporating a safety factor on pressure. Finally, the probability of occurrence of level C and D condition transients is relatively low, so the assumption that this type of transient occurs represents a margin of safety.

2.0 MATERIALS

2.1 Background

As discussed in the introduction, 10 CFR Part 50 Appendix G requires minimum upper-shelf energy, as determined from Charpy V-notch tests. For the unirradiated condition, preservice, the minimum upper-shelf energy as determined from Charpy V-notch tests specimens in accordance with paragraph NB-2322.2 of the ASME Code is 75 ft-lbs unless it is demonstrated to the NRC by appropriate data and analyses that lower values of upper-shelf fracture energy will provide margins of safety against fracture equivalent to those required by Appendix G, ASME Code. The minimum value of upper-shelf energy of 75 ft-lbs was added to 10 CFR Part 50 Appendix G on July 17, 1973. The Westinghouse Electric Corporation Equipment Specification (E-Spec) Revision 4 imposed a minimum Charpy V-notch upper-shelf energy requirement for beltline region materials of 75 ft-lbs for all cases in May 1972, without distinction as to the predicted amount of irradiation damage. Thus, it can be concluded that all reactor pressure vessels fabricated in accordance with Westinghouse E-Spec, Revision 4, dated May 1972, exhibited a minimum upper-shelf Charpy energy of 75 ft-lbs for all beltline region materials unless a deviation notice was issued for a given vessel.

10 CFR Part 50 Appendix G required either implicitly or explicitly that during service life the upper-shelf energy of all reactor vessel materials must not be less than 50 ft-lbs. The initial or unirradiated upper-shelf energy is generally dependent upon the inclusion content, cleanness of the material and/or the directionality of forming the material. The decrease in upper-shelf energy during service life is associated with radiation. Residual copper has been identified as the most important chemical element in promoting the decrease in the upper-shelf energy during service life. The Westinghouse Electric Corporation Equipment Specification, Revision 3, dated July 1971, limited the copper content to 0.12 weight percent for base material (plates/forgings) and to 0.10 for weldments. Prior to July 1971, Westinghouse Equipment Specifications did not specify a maximum copper content in the procurement of the reactor pressure vessel. Therefore, for reactor pressure vessels fabricated to Westinghouse Equipment Specifications Revisions 0, 1 and 2, the possibility exists that the upper shelf energy could be less than 50 ft-lbs during service life. Revisions 3 and 4 to the Equipment Specification was meant to ensure compliance with 10CFR Part 50. Tables 2-1 and 2-2 contain reactor vessel material data for all participating WOG plants. For this non-proprietary report the plant names are not included. An anonymous number is assigned for reference. Based on Table 2-1, plants 5, 14, 15, and 17 have upper shelf energy values of less than 50 ft-lbs during service life. These four plants were fabricated with Equipment Specification Revision 0, 1 or 2.

It is generally accepted that weldments fabricated using Linde 80 fluxes will exhibit upper shelf energy values of less than 50 ft-lbs during service life. The reason Linde 80 weldments exhibit upper shelf energy of less than 50 ft-lbs during service life is two fold. Namely, Linde 80 weldments exhibit initial upper shelf energy values less than 65 ft-lbs. The copper content of these weldments is generally greater than 0.25 weight percent. Linde 80 weldments are not included in this analysis because all Linde 80 weldments were exempt from the WOG program. Based upon reviews of surveillance capsules reports,

it is not expected that weldments fabricated with other fluxes will exhibit upper shelf energy values less than 50 ft-lbs during service life. A review of Table 2-2, "WOG Reactor Vessel Weld Data," confirms the observation that weldments fabricated with fluxes other than Linde 80 will not exhibit less than 50 ft-lbs during service.

As stated above, 10 CFR Part 50 requires that the reactor pressure vessel "beltline" materials upper-shelf energy be no less than above 50 ft-lbs during service life. The "beltline" is defined as the irradiated region of the reactor vessel that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material. Therefore, the upper shell course, intermediate shell course, lower shell course and all associated weldments of the reactor pressure vessel are considered to be in the beltline and were assessed as part of this report. The decrease in upper-shelf energy during the service life of the reactor pressure vessel was determined using the methodology given in Regulatory Guide 1.99, Revision 2. The Regulatory Guide identifies two methods for predicting the decrease in upper-shelf energy; by plotting the reduced plant surveillance data on Figure 2 of the guide and fitting the data with a line drawn parallel to the existing lines as the upper bound of all data; or when surveillance data are not available assume that the upper-shelf energy decreases as a function of fluence and copper content as indicated in Figure 2 of the guide. For this "bounding analysis," the latter method was utilized because not all of the beltline materials were included in the surveillance program.

2.2 Reactor Pressure Vessel Materials

As shown in Table 2-1, all Westinghouse Owners Group plant reactor pressure vessels were constructed using one or more of the following four base materials:

- SA-302, Grade B (Plate)
- SA-533, Grade B, Class 1 (Plate)
- SA-508, Class 2 (Forging)
- SA-508, Class 3 (Forging)

As shown in Table 2-2, all Westinghouse Owners Group plant reactor pressure vessels covered by this report were fabricated using one or more of the following fluxes during welding of the shell courses.

- Linde 1092
- RACO 3
- Linde 124
- Linde 0091
- SMIT 89
- Gran Lo

Westinghouse designed Nuclear Steam Supply Systems also had reactor pressure vessels fabricated by Babcock & Wilcox which used Linde 80 flux during welding of the shell courses. Reactor pressure

vessels fabricated by Babcock & Wilcox are not covered by this report because these plants were exempt from this WOG program.

2.3 Mechanical Properties

As the analysis is intended to bound all participating plants, the minimum mechanical properties at an operational temperature of 600°F are used. These values are assumed to be conservative as they represent the minimum yield strength, ultimate strength and Youngs Modulus allowed by the ASME Code [6] for vessel materials. They are taken directly from [6]:

- Yield Strength (σ_Y) = 43.8 ksi
- Ultimate Strength (σ_U) = 80 ksi
- Youngs Modulus (E) = 26.4 Mpsi

2.4 Development of Stress-Strain Curve

A representative stress-strain curve was developed for use in this analysis. This curve was generated using typical stress-strain data for carbon steel as found in [7]. This curve was adjusted to have the code minimum values as listed herein. The linear-elastic portion was developed to have a slope equal to the code minimum Youngs Modulus. The plastic portion strain values were reduced so as to have the yield stress point equal to the code minimum value. This representative curve is shown in Figure 2-1.

2.5 J-R Curve Value Selection

The objective of this section is to determine limiting values for $J_{material}$ at 0.1" crack extension and $(dJ/da)_{material}$ for each reactor vessel base material (no weld metal fell below 50 ft-lbs.) and each size plant (2, 3 or 4 loop). As shown in Section 1.3, these two material parameters furnish the limiting values for $J_{applied}$ to meet the ASME Section XI, Appendix X requirements.

2.5.1 Representative Values

Correlations for J-R curve values with temperature, USE values and crack extension are contained in [8]. The model relations are as follows:

$$\ln J = \ln C1 + C2 \ln(\Delta a) + C3(\Delta a)^{C4}$$

where:

$$\ln C1 = a_1 + a_2 \ln (CVN) + a_3 T$$
$$CVN = \text{Charpy energy (ft-lbs)}$$
$$T = \text{Temperature (°F)}$$
$$C2 = d_1 + d_2 \ln C1$$
$$C3 = d_4 + d_5 \ln C1$$
$$C4 < 0$$

The values for a_1, a_2, a_3, d_1, d_2, d_4, d_5 and C4 are taken from Appendix B of [8]. These correlations were developed using an extensive amount of test data and advanced pattern recognition tools. These correlations are material independent; however, as indicated in [8], different correlations are utilized for base material and weld fluxes.

Based on this model, J-R values were calculated for each size plant. The values obtained were considered to be limiting as the lowest end of life (EOL) USE for each size plant was utilized. These J-R values were then adjusted to reflect a 2σ (standard deviation) margin for conservatism. The values are tabulated in Table 2-3. The temperature value used in the correlation was 390.5°F. This value represents the greatest temperature at the crack tip for a 1/4t flaw. As will be discussed in the following section, Level A and B conditions are limiting. The 1/4t flaw temperature value is assumed for this flaw size in the criteria for level A and B conditions. The temperature value of 390.5° is based on the Level A and B cooldown transient. Both $J_{material}$ and $(dJ/da)_{material}$ may be calculated using this methodology.

Actual J-R data are available for SA-302 grade B material having an initial USE of 50 ft-lbs. These J-R data are given in [9] and are considered to be a lower bound for all J-R data. Data given in [10] also shows the effect of manganese sulfide inclusions in the steel, however the J-R curves of [10] exhibit a much higher resistance to ductile tearing than the values given in [9]. In order to perform the most conservative bounding analysis the J-R values were recalculated using the information from [9]. Based on [9] the lowest value for $J_{material}$ at 0.1" crack extension for SA-302B material with USE of 50 ft-lbs is 694 in-lb/in^2. This $J_{material}$ value may be adjusted for EOL using the [8] correlation. Based on this model, the percentage decrease in $J_{material}$ for 0.1" crack extension per unit decrease in USE can be calculated. It was determined that the percent decrease in $J_{material}$ per unit decrease in USE never exceeded 3%. Based on Table 2-3, the lowest EOL USE for SA-302B is 42 ft-lbs (3 loop plant). Reducing 694 in-lbs/in^2 by 24% (3% drop per ft-lb x 8 ft-lbs) yields a $J_{material}$ value of 527 in-lbs/in^2. Similarly, another point on the J-R curve given in [9] can be utilized to calculate $(dJ/da)_{material}$. The limiting J-R curve values are summarized in Table 2-4.

Table 2-1
WOG Reactor Vessel Material Data[1]

Description of Beltline Region Material	Base Material Specification	Number of Loops	Initial USE (ft-lbs)	EOL USE (ft-lbs)
PLANT 1: Inter. Shell Forging	SA-508, Cl 3	2	146	111
PLANT 1: Lower Shell Forging	SA-508, Cl 3	2	108	82
PLANT 2: Lower Shell Forging	SA-508, Cl 3	2	108	82
PLANT 2: Inter. Shell Forging	SA-508, Cl 3	2	112	85
PLANT 3: Inter. Shell Forging	SA-508, Cl 2	2	142	112
PLANT 3: Lower Shell Forging	SA-508, Cl 2	2	149	118
PLANT 4: Inter. Shell	SA-533, Gr B, Cl 1	3	104	79
PLANT 4: Inter. Shell	SA-533, Gr B, Cl 1	3	94	66
PLANT 4: Lower Shell	SA-533, Gr B, Cl 1	3	89	63
PLANT 4: Lower Shell	SA-533, Gr B, Cl 1	3	83	63
PLANT 5: Upper Shell[2]	SA-302, Gr B	3	54	42
PLANT 5: Inter. Shell[2]	SA-302, Gr A	3	62	46
PLANT 5: Inter. Shell[2]	SA-302, Gr A	3	64	49
PLANT 5: Lower Shell	SA-302, Gr A	3	78	56
PLANT 5: Inter. Shell	SA-302, Gr A	3	74	57
PLANT 5: Lower Shell	SA-302, Gr A	3	98	59
PLANT 5: Lower Shell	SA-302, Gr A	3	89	62
PLANT 5: Upper Shell[2]	SA-302, Gr B	3	80	61
PLANT 5: Upper Shell[2]	SA-302, Gr B	3	62	49
PLANT 6: Inter. Shell	SA-533, Gr B, Cl 1	3	91	66
PLANT 6: Inter. Shell	SA-533, Gr B, Cl 1	3	83	60

Note: [1] Two entries in this table for the same plant and shell represent two different heats of material.

[2] These plant EOL USE values have been derived from available records verified and utilized as part of this generic bounding analysis. The values for other plants in this table are based on an initial assessment and are subject to change.

[3] Seventy (70) percent shear - maximum

[4] Fifty (50) percent shear - maximum

[5] Ninety (90) percent shear - maximum: 65 percent longitudinal

Table 2-1 (continued)
WOG Reactor Vessel Material Data

Description of Beltline Region Material	Base Material Specification	Number of Loops	Initial USE (ft-lbs)	EOL USE (ft-lbs)
PLANT 6: Lower Shell	SA-533, Gr B, Cl 1	3	85	54
PLANT 6: Lower Shell	SA-533, Gr B, Cl 1	3	83	62
PLANT 6: Inter. Shell	SA-533, Gr B, Cl 1	3	87	65
PLANT 6: Inter. Shell	SA-533, Gr B, Cl 1	3	99	73
PLANT 6: Lower Shell	SA-533, Gr B, Cl 1	3	86	62
PLANT 6: Lower Shell	SA-533, Gr B, Cl 1	3	76	55
PLANT 7: Inter. Shell Forging	SA-508, Cl 2	3	74	58
PLANT 7: Lower Shell Forging	SA-508, Cl 2	3	80	59
PLANT 8: Inter. Shell	SA-533, Gr B, Cl 1	3	83	64
PLANT 8: Inter. Shell	SA-533, Gr B, Cl 1	3	79	61
PLANT 9: Lower Shell	SA-533, Gr B, Cl 1	3	82	62
PLANT 9: Lower Shell	SA-533, Gr B, Cl 1	3	78	60
PLANT 10: Inter. Shell Forging	SA-508, Cl 2	3	94	71
PLANT 10: Lower Shell Forging	SA-508, Cl 2	3	85	61
PLANT 11: Inter. Shell	SA-533, Gr B, Cl 1	3	81	62
PLANT 11: Inter. Shell	SA-533, Gr B, Cl 1	3	107	83
PLANT 11: Lower Shell	SA-533, Gr B, Cl 1	3	106	84
PLANT 11: Lower Shell	SA-533, Gr B, Cl 1	3	92	73
PLANT 12: Inter. Shell	SA-533, Gr B, Cl 1	3	100	72
PLANT 12: Inter. Shell	SA-533, Gr B, Cl 1	3	100	65
PLANT 12: Lower Shell	SA-533, Gr B, Cl 1	3	103	75
PLANT 12: Lower Shell	SA-533, Gr B, Cl 1	3	99	71
PLANT 13: Inter. Shell	SA-533, Gr B, Cl 1	3	115	87
PLANT 13: Inter. Shell	SA-533, Gr B, Cl 1	3	118	90
PLANT 13: Lower Shell	SA-533, Gr B, Cl 1	3	103	78
PLANT 13: Lower Shell	SA-533, Gr B, Cl 1	3	125	95

Table 2-1 (continued)
WOG Reactor Vessel Material Data

Description of Beltline Region Material	Base Material Specification	Number of Loops	Initial USE (ft-lbs)	EOL USE (ft-lbs)
PLANT 14: Inter. Shell	SA-533, Gr B, Cl 1	4	100	76
PLANT 14: Inter. Shell	SA-533, Gr B, Cl 1	4	86	66
PLANT 14: Lower Shell(2)	SA-533, Gr B, Cl 1	4	58	47
PLANT 14: Lower Shell(2)	SA-533, Gr B, Cl 1	4	69	52
PLANT 14: Upper Shell(2)	SA-533, Gr B, Cl 1	4	54(3)	49
PLANT 14: Upper Shell(2)	SA-533, Gr B, Cl 1	4	48(4)	44
PLANT 15: Inter. Shell Forging(2)	SA-508, Cl 2	4	62	44
PLANT 15: Lower Shell Forging(2)	SA-508, Cl 2	4	111	86
PLANT 16: Inter. Shell	SA-302, Gr B, Mod.	4	102	78
PLANT 16: Inter. Shell	SA-302, Gr B, Mod.	4	97	72
PLANT 16: Inter. Shell	SA-302, Gr B, Mod.	4	96	72
PLANT 16: Lower Shell(2)	SA-302, Gr B, Mod.	4	72	55
PLANT 16: Lower Shell(2)	SA-302, Gr B, Mod.	4	94	69
PLANT 16: Lower Shell	SA-302, Gr B, Mod.	4	68	49
PLANT 16: Upper Shell	SA-302, Gr B, Mod.	4	62(5)	55
PLANT 16: Upper Shell	SA-302, Gr B, Mod.	4	64(5)	57
PLANT 16: Upper Shell	SA-302, Gr B, Mod.	4	85(5)	74
PLANT 17: Inter. Shell	SA-302, Gr B	4	76	55
PLANT 17: Inter. Shell	SA-302, Gr B	4	75	59
PLANT 17: Inter. Shell	SA-302, Gr B	4	73	56
PLANT 17: Lower Shell	SA-302, Gr B	4	71	52
PLANT 17: Lower Shell	SA-302, Gr B	4	88	65
PLANT 18: Inter. Shell Forging	SA-508, Cl 2	4	79	58
PLANT 18: Lower Shell Forging	SA-508, Cl 2	4	72	54
PLANT 19: Inter. Shell(2)	SA-302, Gr B	4	88	63
PLANT 19: Inter. Shell(2)	SA-302, Gr B	4	82	57

Table 2-1 (continued)
WOG Reactor Vessel Material Data

Description of Beltline Region Material	Base Material Specification	Number of Loops	Initial USE (ft-lbs)	EOL USE (ft-lbs)
PLANT 19: Inter. Shell	SA-302, Gr B	4	79	55
PLANT 20: Inter. Shell	SA-533, Gr B, Cl 1	4	83	63
PLANT 20: Inter. Shell	SA-533, Gr B, Cl 1	4	74	56
PLANT 20: Lower Shell	SA-533, Gr B, Cl 1	4	98	74
PLANT 20: Lower Shell	SA-533, Gr B, Cl 1	4	89	67
PLANT 21: Inter. Shell	SA-533, Gr B, Cl 1	4	111	87
PLANT 21: Inter. Shell	SA-533, Gr B, Cl 1	4	90	70
PLANT 21: Inter. Shell	SA-533, Gr B, Cl 1	4	106	83
PLANT 21: Lower Shell	SA-533, Gr B, Cl 1	4	77	60
PLANT 21: Lower Shell	SA-533, Gr B, Cl 1	4	76	59
PLANT 21: Lower Shell	SA-533, Gr B, Cl 1	4	79	62
PLANT 22: Inter. Shell	SA-533, Gr B, Cl 1	4	91	62
PLANT 22: Inter. Shell	SA-533, Gr B, Cl 1	4	98	67
PLANT 22: Inter. Shell	SA-533, Gr B, Cl 1	4	104	73
PLANT 22: Lower Shell	SA-533, Gr B, Cl 1	4	93	68
PLANT 22: Lower Shell	SA-533, Gr B, Cl 1	4	83	61
PLANT 22: Lower Shell	SA-533, Gr B, Cl 1	4	85	62
PLANT 23: Inter. Shell	SA-533, Gr B, Cl 1	4	87	64
PLANT 23: Inter. Shell	SA-533, Gr B, Cl 1	4	86	65
PLANT 23: Inter. Shell	SA-533, Gr B, Cl 1	4	78	61
PLANT 23: Lower Shell	SA-533, Gr B, Cl 1	4	111	83
PLANT 23: Lower Shell	SA-533, Gr B, Cl 1	4	103	78
PLANT 23: Lower Shell	SA-533, Gr B, Cl 1	4	116	88
PLANT 24: Inter. Shell	SA-533, Gr B, Cl 1	4	94	73
PLANT 24: Inter. Shell	SA-533, Gr B, Cl 1	4	103	80
PLANT 24: Inter. Shell	SA-533, Gr B, Cl 1	4	88	69

Table 2-1 (continued)
WOG Reactor Vessel Material Data

Description of Beltline Region Material	Base Material Specification	Number of Loops	Initial USE (ft-lbs)	EOL USE (ft-lbs)
PLANT 24: Lower Shell	SA-533, Gr B, Cl 1	4	83	65
PLANT 24: Lower Shell	SA-533, Gr B, Cl 1	4	78	61
PLANT 24: Lower Shell	SA-533, Gr B, Cl 1	4	98	76
PLANT 25: Inter. Shell	SA-533, Gr B, Cl 1	4	82	66
PLANT 25: Inter. Shell	SA-533, Gr B, Cl 1	4	102	82
PLANT 25: Inter. Shell	SA-533, Gr B, Cl 1	4	115	92
PLANT 25: Lower Shell	SA-533, Gr B, Cl 1	4	78	62
PLANT 25: Lower Shell	SA-533, Gr B, Cl 1	4	77	62
PLANT 25: Lower Shell	SA-533, Gr B, Cl 1	4	81	65
PLANT 26: Inter. Shell	SA-533, Gr B, Cl 1	4	80	64
PLANT 26: Inter. Shell	SA-533, Gr B, Cl 1	4	81	62
PLANT 26: Inter. Shell	SA-533, Gr B, Cl 1	4	98	75
PLANT 26: Lower Shell	SA-533, Gr B, Cl 1	4	86	67
PLANT 26: Lower Shell	SA-533, Gr B, Cl 1	4	97	78
PLANT 26: Lower Shell	SA-533, Gr B, Cl 1	4	90	70
PLANT 27: Inter. Shell	SA-533, Gr B, Cl 1	4	78	62
PLANT 27: Inter. Shell	SA-533, Gr B, Cl 1	4	100	80
PLANT 27: Inter. Shell	SA-533, Gr B, Cl 1	4	99	79
PLANT 27: Lower Shell	SA-533, Gr B, Cl 1	4	104	83
PLANT 27: Lower Shell	SA-533, Gr B, Cl 1	4	105	84
PLANT 27: Lower Shell	SA-533, Gr B, Cl 1	4	101	81
PLANT 28: Inter. Shell	SA-533, Gr B, Cl 1	4	117	88
PLANT 28: Inter. Shell	SA-533, Gr B, Cl 1	4	101	79
PLANT 28: Lower Shell	SA-533, Gr B, Cl 1	4	99	73
PLANT 28: Lower Shell	SA-533, Gr B, Cl 1	4	84	63
PLANT 29: Inter. Shell	SA-533, Gr B, Cl 1	4	96	75

Table 2-1 (continued)
WOG Reactor Vessel Material Data

Description of Beltline Region Material	Base Material Specification	Number of Loops	Initial USE (ft-lbs)	EOL USE (ft-lbs)
PLANT 29: Inter. Shell	SA-533, Gr B, Cl 1	4	82	64
PLANT 29: Inter. Shell	SA-533, Gr B, Cl 1	4	92	72
PLANT 29: Lower Shell	SA-533, Gr B, Cl 1	4	83	65
PLANT 29: Lower Shell	SA-533, Gr B, Cl 1	4	102	80
PLANT 29: Lower Shell	SA-533, Gr B, Cl 1	4	105	82
PLANT 30: Inter. Shell	SA-533, Gr B, Cl 1	4	95	74
PLANT 30: Inter. Shell	SA-533, Gr B, Cl 1	4	104	81
PLANT 30: Inter. Shell	SA-533, Gr B, Cl 1	4	84	66
PLANT 30: Lower Shell	SA-533, Gr B, Cl 1	4	85	66
PLANT 30: Lower Shell	SA-533, Gr B, Cl 1	4	83	65
PLANT 30: Lower Shell	SA-533, Gr B, Cl 1	4	87	68
PLANT 31: Inter. Shell	SA-533, Gr B, Cl 1	4	127	102
PLANT 31: Inter. Shell	SA-533, Gr B, Cl 1	4	127	102
PLANT 31: Inter. Shell	SA-533, Gr B, Cl 1	4	87	70
PLANT 31: Lower Shell	SA-533, Gr B, Cl 1	4	135	107
PLANT 31: Lower Shell	SA-533, Gr B, Cl 1	4	100	80
PLANT 31: Lower Shell	SA-533, Gr B, Cl 1	4	94	75
PLANT 32: Inter. Shell	SA-533, Gr B, Cl 1	4	101	78
PLANT 32: Inter. Shell	SA-533, Gr B, Cl 1	4	105	81
PLANT 32: Inter. Shell	SA-533, Gr B, Cl 1	4	112	90
PLANT 32: Lower Shell	SA-533, Gr B, Cl 1	4	92	70
PLANT 32: Lower Shell	SA-533, Gr B, Cl 1	4	115	92
PLANT 32: Lower Shell	SA-533, Gr B, Cl 1	4	103	82
PLANT 33: Inter. Shell	SA-533, Gr B, Cl 1	4	98	72
PLANT 33: Inter. Shell	SA-533, Gr B, Cl 1	4	102	75
PLANT 33: Inter. Shell	SA-533, Gr B, Cl 1	4	97	71

Table 2-1 (continued)
WOG Reactor Vessel Material Data

Description of Beltline Region Material	Base Material Specification	Number of Loops	Initial USE (ft-lbs)	EOL USE (ft-lbs)
PLANT 33: Lower Shell	SA-533, Gr B, Cl 1	4	123	91
PLANT 33: Lower Shell	SA-533, Gr B, Cl 1	4	133	98
PLANT 33: Lower Shell	SA-533, Gr B, Cl 1	4	108	84
PLANT 34: Inter. Shell Forging	SA-508, Cl 2	4	93	72
PLANT 34: Lower Shell Forging	SA-508, Cl 2	4	100	76
PLANT 35: Inter. Shell	SA-533, Gr B, Cl 1	4	90	72
PLANT 35: Inter. Shell	SA-533, Gr B, Cl 1	4	100	80
PLANT 35: Inter. Shell	SA-533, Gr B, Cl 1	4	107	88
PLANT 35: Lower Shell	SA-533, Gr B, Cl 1	4	116	96
PLANT 35: Lower Shell	SA-533, Gr B, Cl 1	4	113	94
PLANT 35: Lower Shell	SA-533, Gr B, Cl 1	4	118	97
PLANT 36: Inter. Shell	SA-533, Gr B, Cl 1	4	106	84
PLANT 36: Inter. Shell	SA-533, Gr B, Cl 1	4	97	76
PLANT 36: Inter. Shell	SA-533, Gr B, Cl 1	4	107	87
PLANT 36: Lower Shell	SA-533, Gr B, Cl 1	4	98	77
PLANT 36: Lower Shell	SA-533, Gr B, Cl 1	4	103	81
PLANT 36: Lower Shell	SA-533, Gr B, Cl 1	4	121	96
PLANT 37: Inter. Shell	SA-533, Gr B, Cl 1	4	111	87
PLANT 37: Inter. Shell	SA-533, Gr B, Cl 1	4	101	79
PLANT 37: Inter. Shell	SA-533, Gr B, Cl 1	4	105	82
PLANT 37: Lower Shell	SA-533, Gr B, Cl 1	4	107	83
PLANT 37: Lower Shell	SA-533, Gr B, Cl 1	4	106	83
PLANT 37: Lower Shell	SA-533, Gr B, Cl 1	4	108	84
PLANT 38: Inter. Shell	SA-533, Gr B, Cl 1	4	110	88
PLANT 38: Inter. Shell	SA-533, Gr B, Cl 1	4	113	90
PLANT 38: Inter. Shell	SA-533, Gr B, Cl 1	4	106	85

Table 2-1 (continued)
WOG Reactor Vessel Material Data

Description of Beltline Region Material	Base Material Specification	Number of Loops	Initial USE (ft-lbs)	EOL USE (ft-lbs)
PLANT 38: Lower Shell	SA-533, Gr B, Cl 1	4	111	89
PLANT 38: Lower Shell	SA-533, Gr B, Cl 1	4	122	98
PLANT 38: Lower Shell	SA-533, Gr B, Cl 1	4	127	102
PLANT 39: Inter. Shell	SA-533, Gr B, Cl 1	4	129	101
PLANT 39: Inter. Shell	SA-533, Gr B, Cl 1	4	117	91
PLANT 39: Inter. Shell	SA-533, Gr B, Cl 1	4	129	101
PLANT 39: Lower Shell	SA-533, Gr B, Cl 1	4	129	101
PLANT 39: Lower Shell	SA-533, Gr B, Cl 1	4	123	96
PLANT 39: Lower Shell	SA-533, Gr B, Cl 1	4	110	86
PLANT 40: Inter. Shell	SA-533, Gr B, Cl 1	4	109	87
PLANT 40: Inter. Shell	SA-533, Gr B, Cl 1	4	129	103
PLANT 40: Inter. Shell	SA-533, Gr B, Cl 1	4	122	98
PLANT 40: Lower Shell	SA-533, Gr B, Cl 1	4	124	99
PLANT 40: Lower Shell	SA-533, Gr B, Cl 1	4	118	94
PLANT 40: Lower Shell	SA-533, Gr B, Cl 1	4	123	98
PLANT 41: Inter. Shell Forging	SA-508, Cl 2	4	110	88
PLANT 41: Lower Shell Forging	SA-508, Cl 2	4	123	98
PLANT 42: Inter. Shell Forging	SA-508, Cl 2	4	141	104
PLANT 42: Lower Shell Forging	SA-508, Cl 2	4	148	111
PLANT 43: Inter. Shell Forging	SA-508, Cl 2	4	137	107
PLANT 43: Lower Shell Forging	SA-508, Cl 2	4	135	105

Table 2-2
WOG Reactor Vessel Weld Data

Plant	Weld Flux	Initial USE (ft-lbs)[1]	EOL USE (ft-lbs)[1]
1	UM89	103	67
2	UM89	103	74
3	LINDE 1092	126	68
4	LINDE 1092	113	57
5	LINDE 1092	112	69
6	LINDE 0091	149	98
7	SMIT 89	111	81
8	LINDE 0091	145	100
9	SMIT 89	95	70
10	LINDE 124	93	72
11	LINDE 0091	148	114
12	LINDE 124	77	62
13	GRAN LO	134	105
14	LINDE 1092	119	83
15	LINDE 1092	121	83
16	SMIT 89	113	61
17	ARCOS B5	106	51
18	LINDE 124	94	70
19	LINDE 0091	143	107
20	LINDE 1092	104	74
21	LINDE 1092	94	57
22	LINDE 1091	133	104
23	LINDE 0091	161	124
24	LINDE 1092	116	69
25	LINDE 124	113	88
26	LINDE 124	100	81
27	LINDE 0091	146	114

Note: [1] Initial assessment, values are subject to change.

Table 2-2 (continued)
WOG Reactor Vessel Weld Data

Plant	Weld Flux	Initial USE (ft-lbs)[1]	EOL USE (ft-lbs)[1]
28	LINDE 124/0091	90	69
29	LINDE 0091/124	100	78
30	LINDE 0091/1092	112	69
31	LINDE 0091/124/1092	124	78
32	SMIT 89	110	78
33	LINDE 0091	134	107
34	LINDE 0091/1092	112	73
35	LINDE 124/0091	96	75
36	LINDE 124/0091	94	75
37	LINDE 124	98	76
38	LINDE 124	112	90
39	GRAN LO	144	112
40	GRAN LO	133	109
41	GRAN LO	129	101

Table 2-3
Lowest End-of-Life Upper Shelf Energy Values for Each Case

Case	Lowest EOL USE (ft-lbs)
2 Loop	
SA-508, Cl 2	112.18
SA-508, Cl 3	82.08
2 Loop Limiting Value	***50(1)***
3 Loop	
SA-A302, Gr B	42
SA-A508, Cl 2	57
SA-A533, Gr B, Cl 1	53
3 Loop Limiting Value	***42***
4 Loop	
SA-A302, Gr B	49
SA-A508, Cl 1	44
SA-A533, Gr B, Cl 1	47
4 Loop Limiting Value	***44***

(1) 50 ft-lbs is assumed for 2 loop plants as a conservative representation.

Table 2-4
J-R Data for Reactor Vessel Base Metal Materials

Case	J @ Δ a = 0.1	$dJ_{material}/da$
2 Loop Limiting Value	702	2925
3 Loop Limiting Value	585 (527)*	2140 (599)*
4 Loop Limiting Value	614	2330
* Calculations based on J-R curves in [9]		

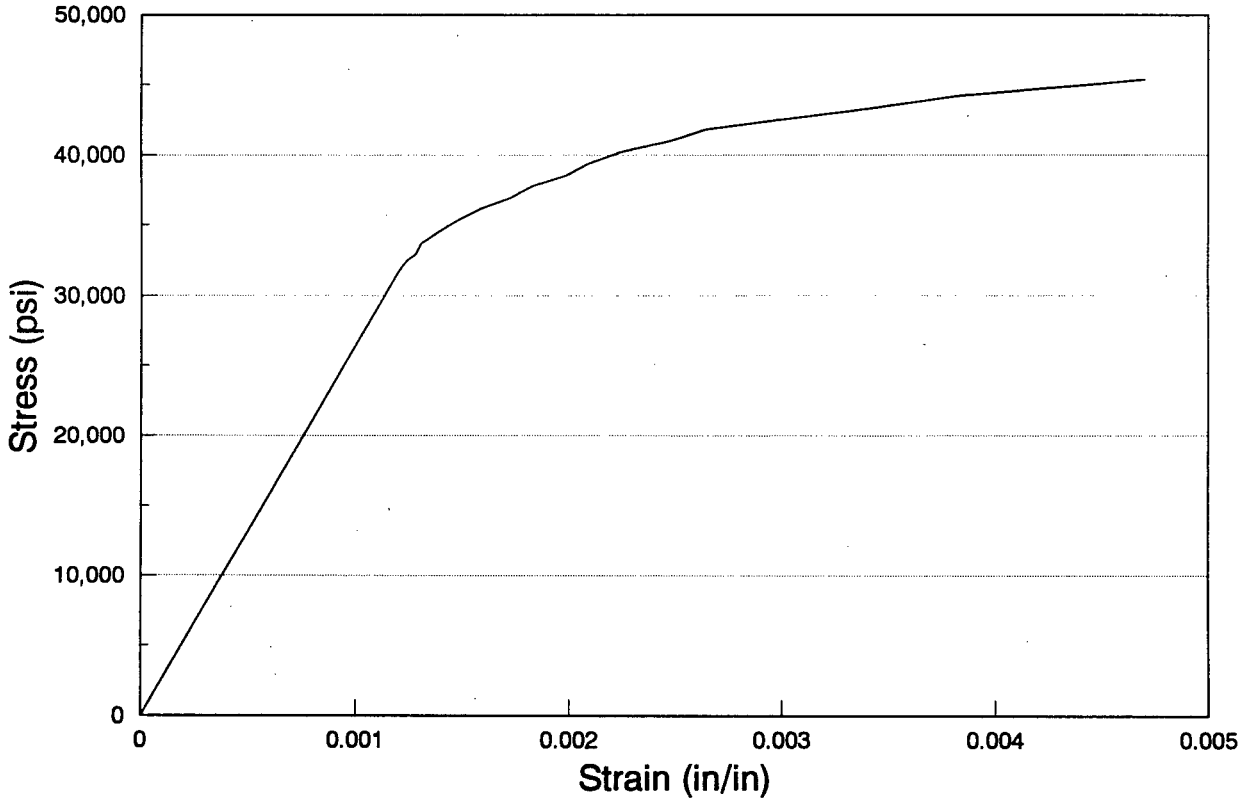


Figure 2-1
Representative Stress - Strain Curve

3.0 ANALYSIS

This section describes the analyses performed to determine the applied J-integral value required for an ASME Section XI, Appendix X evaluation. The inputs needed to perform this analysis are: Material stress-strain curve, mechanical properties, geometry and appropriate transients. The stress-strain curve and mechanical properties are described in Section 2.0. The geometry and transient inputs along with the analyses are described in this section.

3.1 Geometry

The two geometry parameters required as inputs to the analysis are reactor vessel inner radius and thickness. Representative values of these parameters were chosen for 2, 3 and 4 loop plants. Additionally, two plants judged not to be bounded by the representative parameters were also considered. These were plants 5 and 19 from Table 2-1. The geometry values are given in Table 3-1.

3.2 Analysis for Level A and B Conditions

The stresses due to Level A and B conditions are significantly lower than the material yield strength. Consequently $J_{applied}$ can be calculated using linear-elastic fracture mechanics (LEFM) techniques with a plastic-zone correction. Guidelines for performing Level A and B analyses are contained in [11]. This contains a procedure for calculating $J_{applied}$ which was developed by the ASME Code Committee that generated the ASME Section XI, Appendix X requirements. This approach was utilized to determine the applied J values for each of the cases listed in Table 3-1.

The procedure was developed specifically for this application. Consequently, it is applicable for a semi-elliptical flaw with an aspect ratio of 6 to 1. Axial and circumferential flaws may be calculated per this procedure. The methodology is as follows:

First, the stress intensity factors (K_I) due to pressure and thermal loadings are calculated. The stress intensity factor is a measure of the driving force of crack extension. K_I is a function of the size of the crack, the applied stress, and the geometry of the structure.

For an axial flaw with a length to depth aspect ratio of 6 to 1, the stress intensity factor, K_{Ip}, due to internal pressure is given by [11]:

$$K_{Ip} = (SF)\ p\ [1 + (R_i/t)]\ (\pi a)^{0.5}\ F_1$$

$$F_1 = 0.982 + 1.006\ (a/t)^2$$

where, "a" is the flaw depth, R_i is the inner radius of the vessel, t is the wall thickness, p is the internal pressure, and (SF) is the safety factor on pressure. This equation for K_{Ip} is valid for $0.20 \leq a/t \leq 0.50$, and includes the effect of pressure acting on the flaw faces.

For a circumferential flaw with a length to depth aspect ratio of 6 to 1, the stress intensity factor due to internal pressure is given by:

$$K_{Ip} = (SF)\ p\ [1 + (R_i/(2t))]\ (\pi a)^{0.5}\ F_2$$

$$F_2 = 0.885 + 0.233\ (a/t) + 0.345\ (a/t)^2$$

This equation for K_{Ip} is valid for $0.20 \leq a/t \leq 0.50$, and includes the effect of pressure acting on the flaw faces.

For both axial and circumferential flaws with aspect ratio of 6 to 1, the stress intensity factor due to radial thermal gradients is given by:

$$K_{It} = ((CR)/1000)\ t^{2.5}\ F_3$$

$$F_3 = 0.584 + 2.647\ (a/t) - 6.294\ (a/t)^2 + 2.990\ (a/t)^3$$

where, t is in inches, K_{It} is in ksi$\sqrt{\text{in.}}$ and, (CR) is the cooldown rate in °F/hour. This equation for K_{It} is valid for $0.20 \leq a/t \leq 0.50$ and $0 \leq (CR) \leq 100$°F/hour. The through-wall temperature distribution used to develop this equation is the same through-wall distribution assumed in Appendix G of Section III and in Section XI. The thermal stress distribution includes the temperature dependence of material properties.

Using these stress intensity factors, the J-integral or $J_{applied}$ can be calculated. The calculation of the J-integral due to the applied loads should account for the full elastic-plastic behavior of the stress-strain curve for the material. For a reactor vessel with a low upper shelf Charpy energy level, the J-integral can be calculated using the stress intensity factor with the plastic-zone correction for plain-strain. This procedure is as follows.

The stress intensity factors due to internal pressure, K_{Ip}, and radial thermal gradients, K_{It}, are first calculated using the actual postulated flaw depth "a". The effective flaw depth for small-scale yielding, a_{eff}, is then calculated by using:

$$a_{eff} = a + (1/(6\pi))\ [(K_{Ip} + K_{It})/\sigma_y]^2$$

where a is in inches, K_{Ip} and K_{It} are in ksi$\sqrt{\text{in.}}$ and σ_y is the yield strength for the material in ksi. The stress intensity factors for small-scale yielding due to internal pressure, $K_{Ip}(a_{eff})$, and due to radial thermal gradients, $K_{It}(a_{eff})$, are then calculated by substituting a_{eff} in place of "a" into the appropriate stress intensity factor equations given above.

The J-integral due to the applied loads for small-scale yielding is given by:

$$J = 1000\ [K_{Ip}(a_{eff}) + K_{It}(a_{eff})]^2/E'$$

where

$$E' = E/(1 - \upsilon^2)$$

and J is in in-lb/in^2, E is Youngs Modulus in ksi, and υ is Poisson's ratio.

The criteria as described in Section 1.4.1 mandate a plant-specific heatup or cooldown rate be utilized in the analysis for the thermal loading while a constant pressure of 1.15 or 1.25 times the maximum accumulation pressure is assumed. A cooldown rate of 100°F per hour was assumed in the analysis which is the maximum allowed by the plant technical specifications. A cooldown, as opposed to a heatup is utilized because a cooldown causes tensile stress on the inside surface, whereas the heatup causes compressive stresses. Additionally, the inside surface is where degradation due to irradiation is the greatest. The pressure loading will also cause stresses to be tensile, consequently the cooldown is the governing transient.

The maximum accumulation pressure which satisfies ASME Section III, NB-7311(b) is 2734 psi for all PWR systems manufactured by Westinghouse. This is used in conjunction with the safety factors as defined in Section 1.4.1.

In each case, the axial flaw yielded the greatest values. The results obtained using this evaluation with the appropriate inputs for each case described in Table 3-1 are given in Table 3-2, along with the J-R curve material values from Table 2-4. Based on Table 3-2, all participating plants meet the Level A & B Appendix X criteria.

Additionally, a test case was evaluated using LEFM techniques to validate the [11] approach. The through-wall stress distribution was first determined using the appropriate material and geometry inputs. It was calculated using the WECAN [12] computer code. A two dimensional finite element model was generated to model the reactor vessel beltline region using the inputs as defined in the previous sections.

Subsequently, the stress intensity factor (K_I) as described previously was calculated using the peak stress distribution for a range of postulated flaw depths. Since the stresses are in the elastic range, $J_{applied}$ could then be calculated directly from the following relation:

$$J_{applied} = \frac{K_I^2}{E}$$

Based on this evaluation, it was concluded that the approach identified in [11] produced conservative results.

3.3 Analysis for Level C and D Conditions

The stress levels achieved by imposing Level C and D transients can exceed the material yield strength. Consequently, an elastic-plastic fracture mechanics analysis is required for these conditions.

3.3.1 Transient Selection

The first step in performing the reactor vessel integrity assessment is the selection of the limiting, or bounding transient to represent the emergency and faulted conditions. An assessment was conducted to determine the limiting Level C and D transients. Level C and D transients that may potentially impact the reactor vessel are as follows [13]:

Level C Transients

Small Loss-of-Coolant Accident
Small Steam Line Break
Complete Loss of Flow
Small Feedwater Line Break

Level D Transients

Reactor Coolant Pipe Break (Large LOCA)
Large Steam Line Break
Large Feedwater Line Break
Reactor Coolant Locked Rotor
Control Rod Ejection
Steam Generator Tube Rupture
Simultaneous Steam Line Feedwater Line Break

The criteria for choosing the limiting transient were peak stress as well as the overall magnitude of total through-wall stress. Based on a review of the above transient results, it was judged that the small steam line break was the limiting Level C transient and large LOCA and large steam line break were limiting Level D transients. These transients in terms of pressure and temperature histories are shown in Figures 3-1, 3-2 and 3-3. Elastic-plastic stress analyses were performed for a two-dimensional finite element model of a typical 4-loop Westinghouse reactor vessel using the WECAN [12] computer code for all of these transients. The resulting stress distributions are given in Figure 3-4 for the small steam line break and Figure 3-5 for the large steam line break and large LOCA. From Figure 3-5, it can be seen that both Level D transients generate very similar peak stresses. The large steam line break created slightly larger stresses at the crack tip for a 1" flaw. These values were utilized for the fracture mechanics analysis as described below.

3.3.2 Elastic-Plastic Analysis

Using the small and large steam line break stress distributions, $J_{applied}$ and dJ/da were calculated using the PCFAD [14] computer code. PCFAD is a fracture mechanics computer code for use in performing safety analysis for flawed structures against failure due to the application of a postulated load. The procedure used here has been referred to as the failure assessment diagram approach for prediction of instability loads. The procedure uses a diagram with the stress intensity factor/fracture toughness ratio as the ordinate and the applied stress/net section plastic collapse stress ratio as the abscissa. For a particular stress level, flaw size and geometry, the coordinates can be readily calculated. $J_{applied}$, and subsequently dJ/da can be determined using these coordinate values.

The PCFAD results for each of the cases described in Table 3-1 using the stresses determined by the finite element analysis are given in Tables 3-3 and 3-4, along with the limiting material values. It should be noted that only the stability conditions are evaluated for level D conditions (see Section 1.4.3). Based on Tables 3-3 and 3-4, all participating plants meet the level C and D criteria.

Table 3-1
Geometry of the Cases Analyzed

Case	Thickness (in.)	Inner Radius (in.)
2 loop	6.500	66.00
3 loop	7.875	78.50
4 loop	8.500	86.50
Plant 5*	9.875	77.97
Plant 19	10.625	77.00

* These are dimensions for the upper shell of Plant 5 as this base metal exhibited the lowest EOL Charpy energy.

Table 3-2
Level A and B Conditions

Case	Applied		Material		Met Criteria?
	$J_{0.1}$ (in-lbs/in^2)	$dJ_{applied}/da$	$J_{0.1}$ (in-lbs/in^2)	$dJ_{material}/da$	
2 Loop	370	310	702	2925	*Yes*
3 Loop	472	320	585 (527)*	2140 (599)*	*Yes*
4 Loop	554	330	614	2330	*Yes*
Plant 5 Plant 19	478 492	220 200	527 614	599 2330	*Yes* *Yes*
* Calculations based on J-R curves in [9].					

Table 3-3
Level C Conditions

Case	Applied		Material		Met Criteria?
	$J_{0.1}$ (in-lbs/in^2)	$dJ_{applied}/da$	$J_{0.1}$ (in-lbs/in^2)	$dJ_{material}/da$	
2 Loop	319	360	702	2925	*Yes*
3 Loop	318	320	585 (527)*	2140 (599)*	*Yes*
4 Loop	314	320	614	2330	*Yes*
Plant 5 Plant 19	341 350	360 360	527 614	599 2330	*Yes* *Yes*
* Calculations based on J-R curves in [9].					

Table 3-4
Level D Conditions

Case	Applied	Material	Met Criteria?
	$dJ_{applied}/da$	$dJ_{material}/da$	
2 Loop	400	2925	*Yes*
3 Loop	400	2140 (599)*	*Yes*
4 Loop	380	2330	*Yes*
Plant 5 Plant 19	420 513	599 2330	*Yes* *Yes*
* Calculations based on J-R curves in [9].			

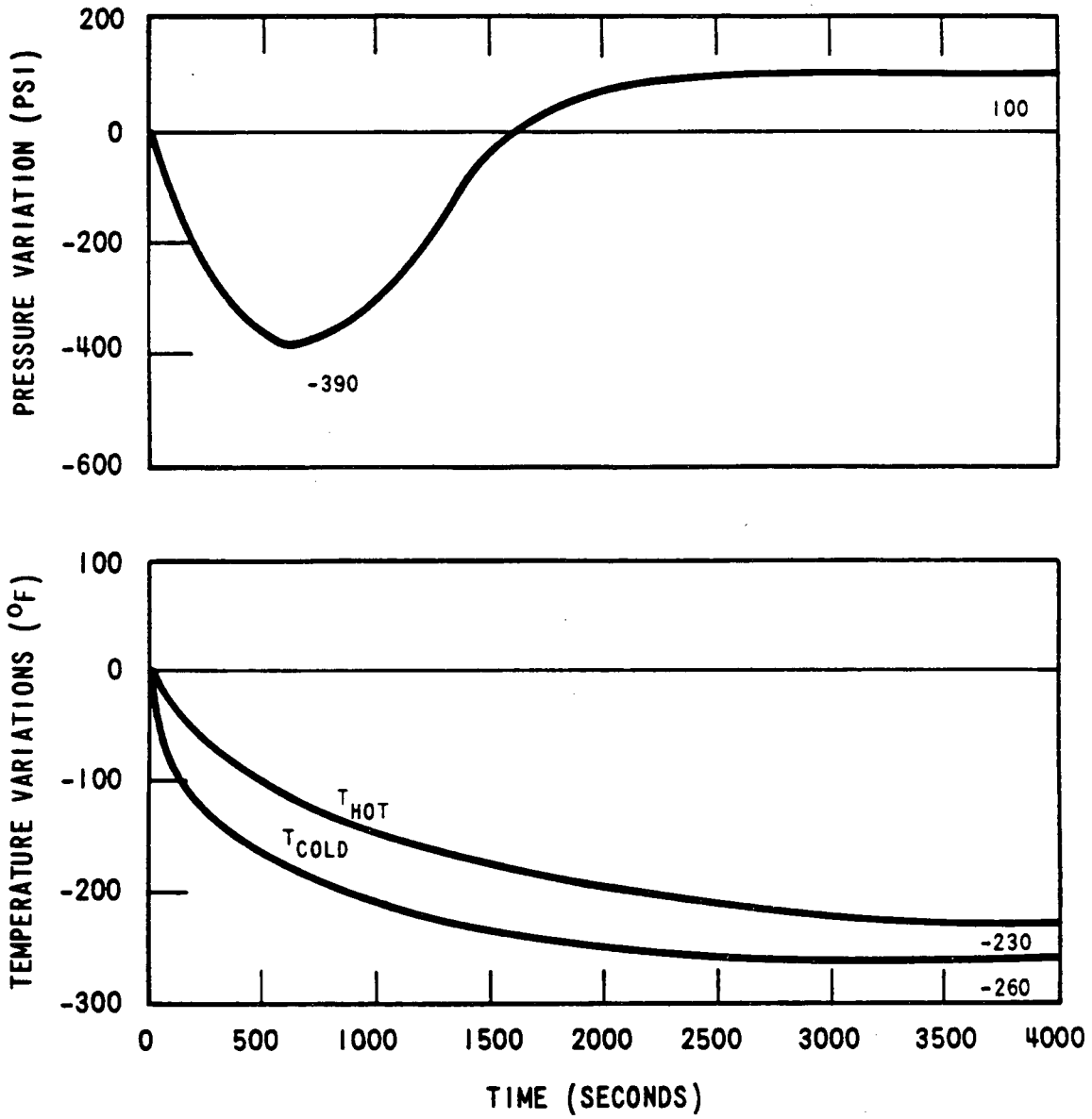
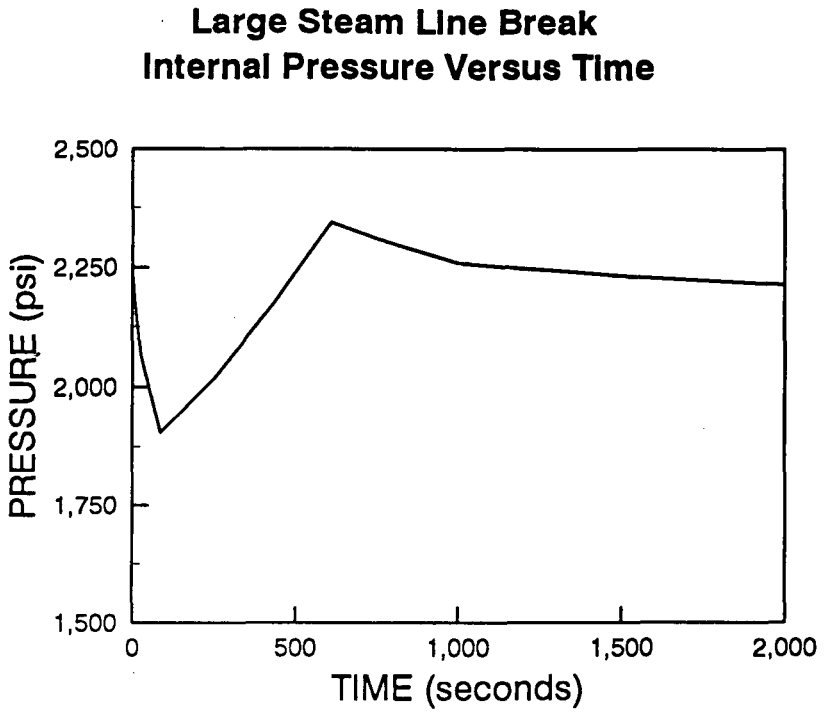


Figure 3-1
Temperature and Pressure History for Small Steam Line Break Transient



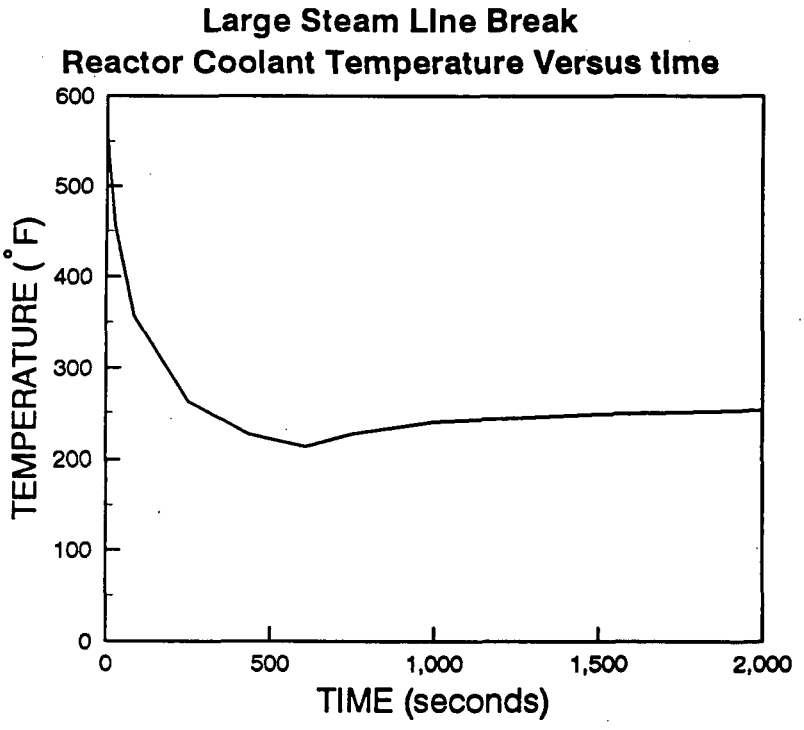
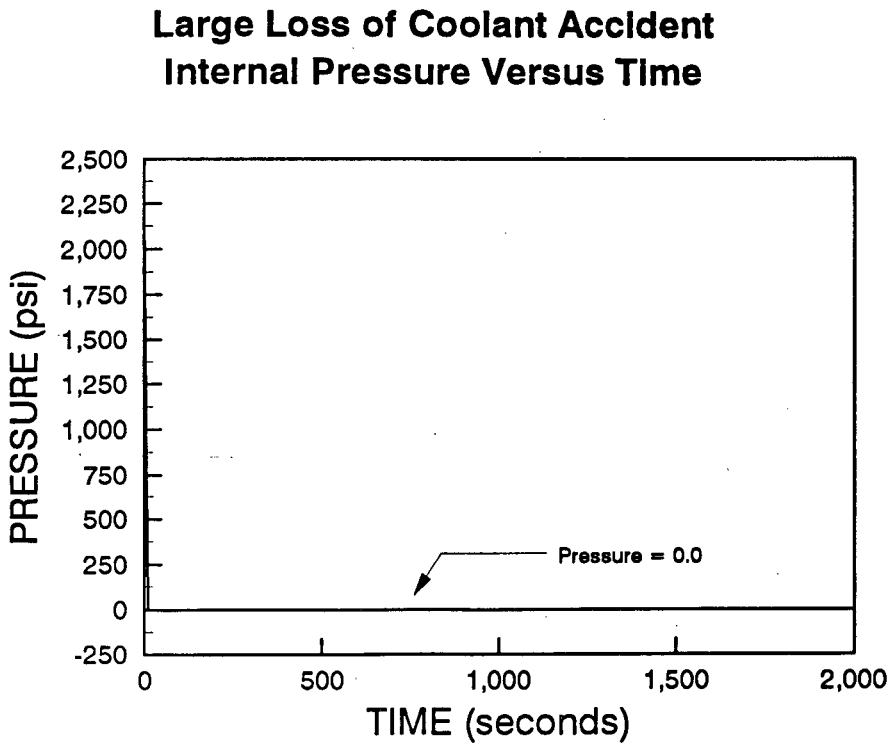


Figure 3-2
Temperature and Pressure History for Large Steam Line Break Transient



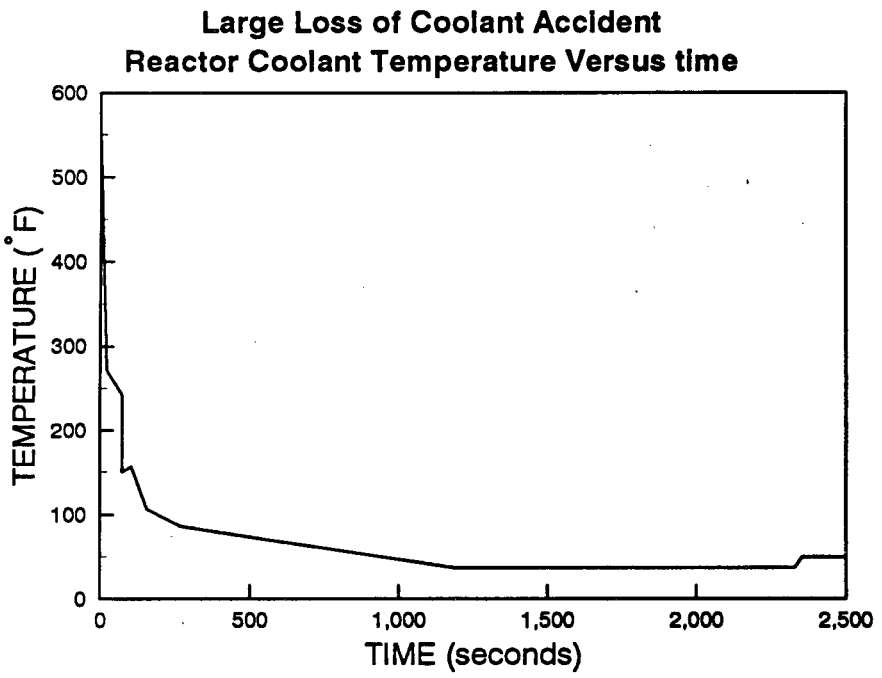


Figure 3-3
Temperature and Pressure History for Large LOCA Transient

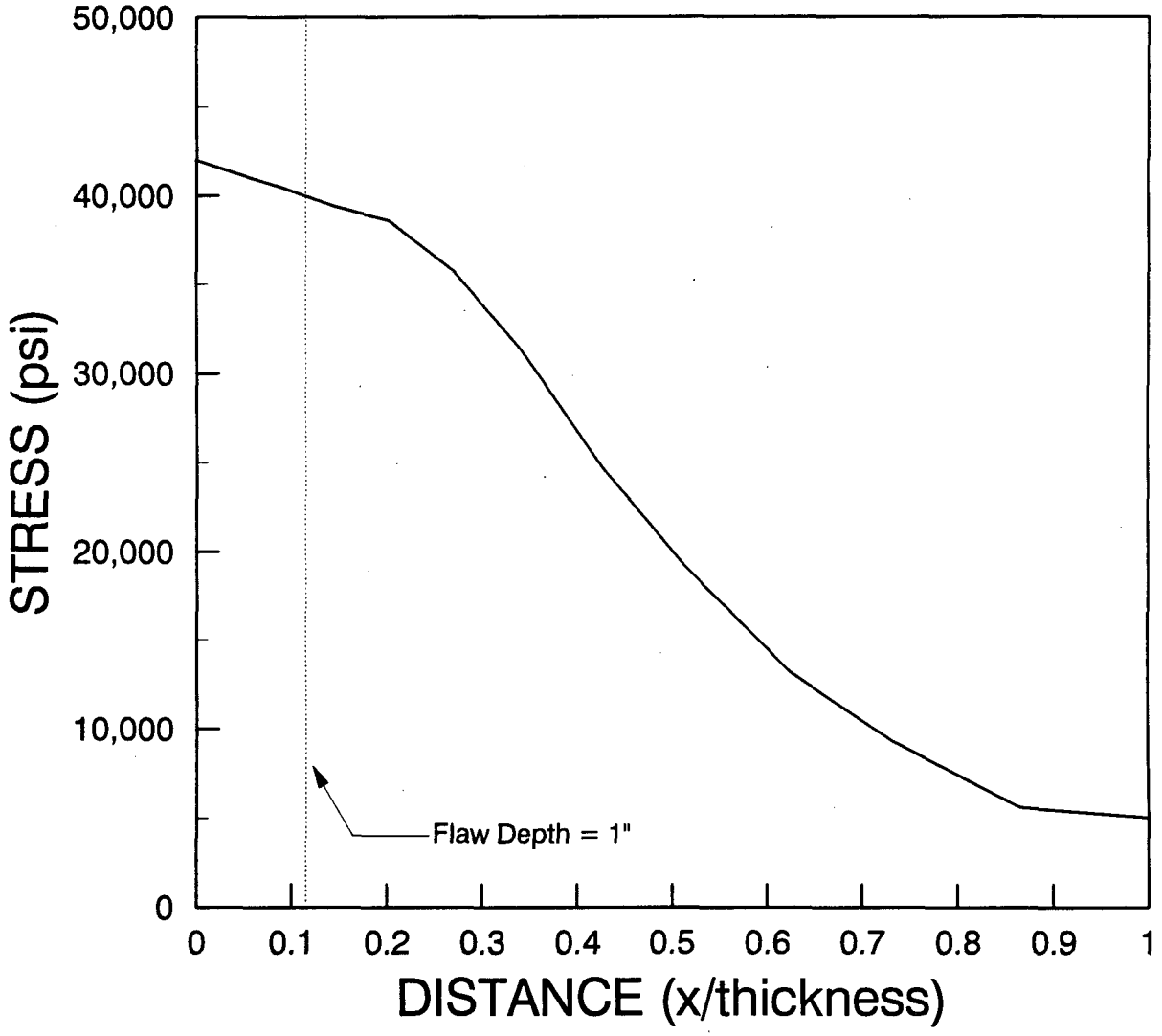


Figure 3-4
Through-Wall Elastic-Plastic Stress Distribution for Small Steam Line Break Transient

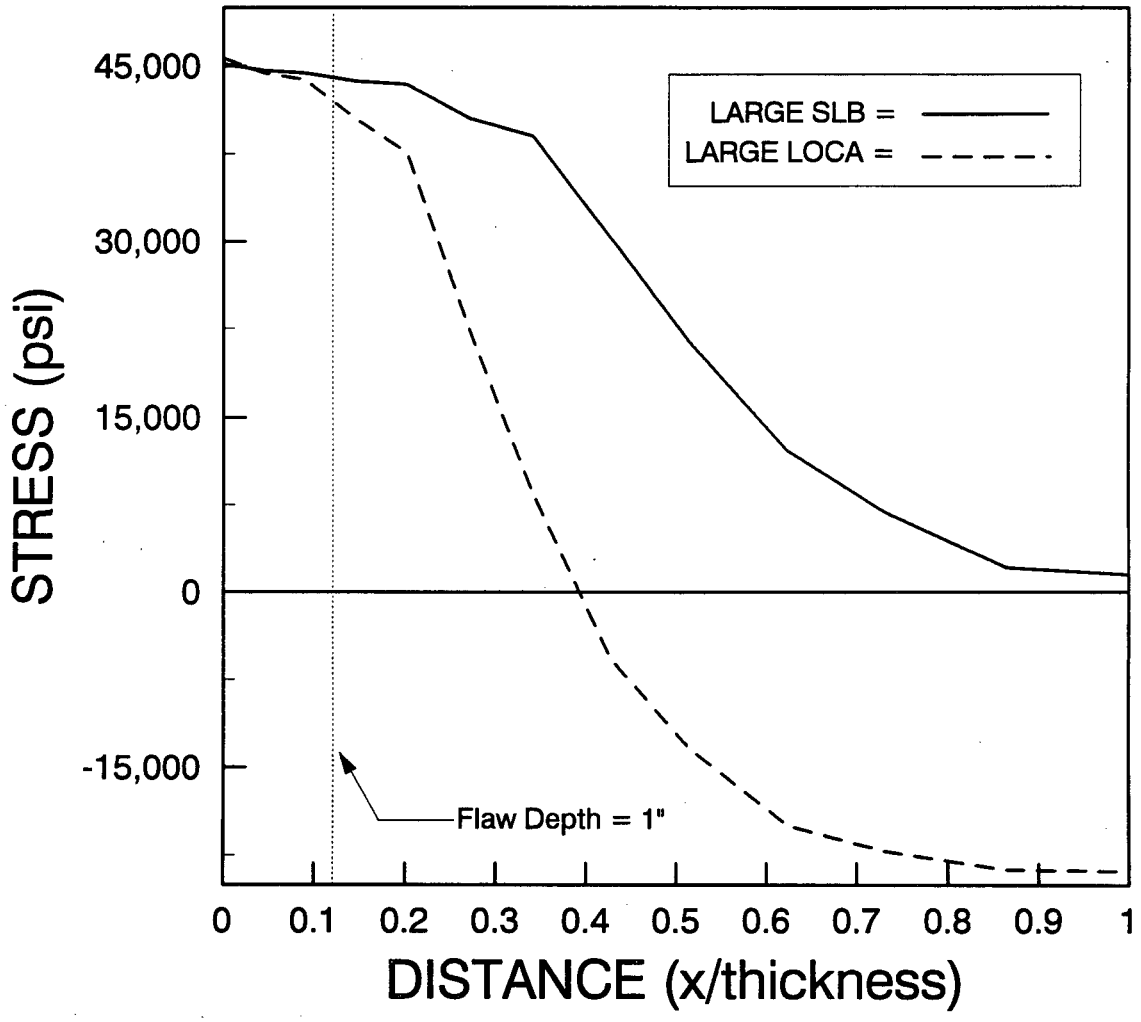


Figure 3-5
Through-Wall Elastic-Plastic Stress Distribution for Large LOCA and Large Steam Line Break Transients

4.0 CONCLUSIONS

This investigation was conducted to demonstrate that participating WOG plants' reactor vessels maintain a margin on USE equivalent to that of ASME Section III, Appendix G through license life. This was accomplished by demonstrating that the reactor vessel materials meet ASME Section XI, Appendix X criteria.

In this investigation, J-integral values were calculated for A, B, C and D level conditions using representative geometries of 2, 3, and 4-loop plants. Two plants not bounded by the representative geometries were also evaluated. Material J values representing EOL conditions were calculated based on available methodology. Comparison cases were evaluated for each material and each representative geometry.

Applied J-values for Level A and B loading conditions along with the limiting material properties are tabulated in Table 3-2. Level C and D condition results are tabulated in Tables 3-3 and 3-4. Based on the information contained in these tables, all participating WOG plants meet the ASME Section XI, Appendix X criteria.

5.0 REFERENCES

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13. Systems Standard 1.3.F, Nuclear Steam Supply System, Reactor Coolant System Design Transients for Standard Plants.

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