WCAP 13587
Revision 1

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

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PREFACE

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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 margins of safety can be demonstrated by using the proposed criteria and requirements of the ASME Code, Section XI, Appendix X.

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-two vessels were included in this evaluation.

The bounding analysis utilized unirradiated J-R curve data (i.e., material resistance data) for WOG reactor vessel upper, intermediate and lower shell course materials. These curves were adjusted to reflect the effect of expected end-of-license fluence 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-two 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 both Linear-Elastic 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 for safety margins equivalent to those in Appendix G.

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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 Code of Federal Regulations, 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 be 50 ft-lbs (68 joules) or greater 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 dated February 1991, and the pending Appendix X of Section XI. Currently, this Appendix X criteria 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 [5], "Reactor Vessel Structural Integrity," which requested information from utilities regarding compliance of their reactor vessel(s) with 10CFR50, Appendices G and H 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

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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 committed to implement the NRC's request. NUMARC and the industry then 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 Section III of the ASME Code.

This report supersedes Revision 0 of WCAP-13587 (i.e., original report) and has been prepared to further describe the basis for the upper shelf energy values of the current participating WOG plants. Section 2 of the original report has been modified to include actual upper shelf energy values of plant specific data in support of the generic bounding analysis. Furthermore, this revision incorporates a section describing the margin available based on ASME Section XI, Appendix X requirements in terms of upper shelf energy.

1.3 Overall Approach

To achieve this objective, a generic bounding evaluation using the proposed criteria of ASME Section XI, Appendix X, 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 analyses are defined. Material and mechanical properties are determined using ASME Code minimum values for conservatism. Upper shelf energy values at end of license are calculated for the beltline region material according to the guidelines specified in ASTM E-185 [17], Branch Technical Position MTEB 5-2 of the Standard Review Plan [18], and Regulatory Guide 1.99, Rev. 2 [16]. 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

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- 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 <u>Level A and B Conditions</u>

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 <u>Level C Conditions</u>

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

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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:

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

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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.

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2.0 MATERIALS

2.1 Background

As discussed in the introduction, 10 CFR Part 50 Appendix G requires minimum values of upper-shelf energy, as determined from Charpy V-notch tests. For the unirradiated preservice condition, the minimum upper-shelf energy as determined from Charpy V-notch test 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. Revision 4 of the Westinghouse Electric Corporation Vessel Equipment Specification (E-Spec), 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 unirradiated 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 requires 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, cleanliness of the material and the directionality of forming the material. The decrease in upper-shelf energy during service life is associated with radiation damage. 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 Vessel 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 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 were imposed to provide compliance with 10 CFR Part 50. Tables 2-1 and 2-2 contain reactor vessel initial and end-of-life upper shelf energy values for all participating WOG plants. The methodology used in generating the upper shelf energy values in Tables 2-1 and 2-2 is contained in Section 2.2. Based on Table 2-1, plants 7, 37, and 41 have upper shelf energy values of less than 50 ft-lbs during service life. These three 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. First, Linde 80 weldments exhibit initial upper shelf energy values less than 65 ft-lbs. Second, the copper content of these weldments is generally greater than 0.25 weight percent. Linde 80 weldments are not included in this report because all Linde 80 weldments were exempt from the WOG program. Based upon a review of surveillance capsule reports,

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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, "Plant Specific Reactor Vessel Surveillance 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 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 irradiation 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: 1) 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 2) 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. Both methodologies were used to calculate the decrease in upper-shelf energy for the "bounding analyses" of this report.

2.2 Methodology for Calculating Upper Shelf Energy

This section describes the methodology used to calculate plant specific initial and end-of-license upper shelf energy values for the upper shell course, intermediate shell course, lower shell course, and associated welds of the reactor pressure vessel for all participating WOG plants. The data used to calculate plant specific initial and end-of-life upper shelf energy values was obtained from certified material test reports, surveillance capsule programs, Generic Letter 92-01 [5] responses, and ex-vessel dosimetry programs.

2.2.1 Unirradiated Upper Shelf Energy

The unirradiated upper shelf energy values were calculated for the upper shell course, intermediate shell course, lower shell course, and surveillance welds using the guidelines specified in ASTM E-185 [17] and Branch Technical Position MTEB 5-2 of the Standard Review Plan [18], which meet the requirements in 10 CFR Part 50, Appendices G and H. The unirradiated Charpy V-notch data used to determine the initial upper shelf energy values was obtained from material certification and surveillance capsule program reports. Generic Letter 92-01 responses were also used to obtain supplemental Charpy test data. Results are tabulated in Tables 2-1 and 2-2 for all participating WOG plants.

2.2.2 End-of-License Upper Shelf Energy

The end-of-license (EOL) upper shelf energy values were calculated for the beltline region materials listed in Section 2.2.1 using the Regulatory Guide 1.99, Rev. 2 [16] methodology. The EOL upper shelf energy

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is the most limiting value determined by 1) calculating the decrease in upper shelf energy as a function of copper content and EOL fluence using Figure 2 of Reference 16, or 2) plotting reduced plant surveillance data on Figure 2 of Reference 16 and fitting the data with a line drawn parallel to the existing lines as the upper bound of all data (the decrease in upper shelf energy was calculated from the newly plotted data based on EOL fluence). For the upper shell course, intermediate shell course, lower shell course, and surveillance welds, plant specific 1/4-t EOL fluence values were used as defined in 10 CFR Part 50, Appendix G. The plant specific EOL fluence values were obtained from surveillance capsule program analyses and plant responses to Generic Letter 92-01. Results of the EOL upper shelf energy values are listed in Tables 2-1 and 2-2.

However, it should be recognized that the Regulatory Guide 1.99, Rev. 2 procedures are conservative per SECY 91-333 [19]:

> "In May 1988, the staff issued Regulatory Guide 1.99, Revision 2, to provide updated procedures for calculating the shift in the reference nilductility temperature (RTNDT) and calculating the change in Charpy USE as a function of neutron fluence. The procedure for calculating the shift in RTNDT is believed to be adequate, although the staff continues to monitor and evaluate the surveillance database to identify the need for further updates to the procedure. However, the staff found the procedure for calculating the change in Charpy USE to be inadequate, and recent results from the NRC's research program provide a much improved method for calculating this change."

2.3 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 five base materials:

- SA-302, Grade A (Plate), (See Appendix A)
- 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
- Grau Lo

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Westinghouse designed nuclear steam supply systems also had reactor pressure vessels fabricated by Babcock & Wilcox that used Linde 80 flux during welding of the shell courses. Reactor pressure vessel weldments fabricated by Babcock & Wilcox are not covered by this report because these plants were exempt from this WOG program.

2.4 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 the ASME Code [6]:

- Yield Strength (σ_Y) = 37.8 ksi for SA-302, Grade A* and 43.8 ksi for all other materials
- Ultimate Strength (σ_U) = 75 ksi for SA-302, Grade A* and 80 ksi for all other materials
- Youngs Modulus (E) = 26.4 Mpsi

2.5 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 given by Reddy and Ayres [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.

2.6 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 required in Section 1.4, these two material parameters are used to show that the ASME Section XI, Appendix X requirements are satisfied.

2.6.1 Representative Values

Correlations for J-R curve values with temperature, USE values and crack extension are given by Eason et al [8]. The model relations are as follows:

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

where:

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

The values for a_1, a_2, a_3, d_1, d_2, d_4, d_5 and C4 are constants taken from Appendix B of Eason et al [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 by Eason et al [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-license (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 lowest projected end-of-license USE 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 by Hiser and Terrell [9] and are considered to be a lower bound for all J-R data. Data given by Begley [10] also shows the effect of manganese sulfide inclusions in the steel, however the J-R curves of Reference 10 exhibit a much higher resistance to ductile tearing than the values given in Reference 9. In order to perform the most conservative bounding analysis the J-R values were recalculated using the information from Reference 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 correlation of Eason et al [8]. 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 Reference 9 can be utilized to calculate $(dJ/da)_{material}$. The bounding J-R curve values are summarized in Table 2-4.

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Table 2-1
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 1: Inter. Shell Forging	SA-508, Cl 3	2	112	83
Plant 1: Lower Shell Forging	SA-508, Cl 3	2	108	89[(6)]
Plant 1: Upper Shell	SA-508, Cl 3	2	85	65
Plant 2: Inter. Shell Forging	SA-508, Cl 3	2	146	132[(6)]
Plant 2: Lower Shell Forging	SA-508, Cl 3	2	130	96
Plant 2: Upper Shell	SA-508, Cl 3	2	84	65
Plant 3: Inter. Shell Forging	SA-508, Cl 2	2	142	140[(6)]
Plant 3: Lower Shell Forging	SA-508, Cl 2	2	149	145[(6)]
Plant 3: Upper Shell Forging	SA-508, Cl 2	2	86	53
Plant 4: Inter. Shell	SA-533, Gr B, Cl 1	3	97	72
Plant 4: Inter. Shell	SA-533, Gr B, Cl 1	3	100	75
Plant 4: Lower Shell	SA-533, Gr B, Cl 1	3	91	81[(6)]
Plant 4: Lower Shell	SA-533, Gr B, Cl 1	3	97	70
Plant 4: Upper Shell	SA-508, Cl 2	3	96	57
Plant 5: Inter. Shell	SA-533, Gr B, Cl 1	3	100	72
Plant 5: Inter. Shell	SA-533, Gr B, Cl 1	3	100	62[(6)]
Plant 5: Lower Shell	SA-533, Gr B, Cl 1	3	103	75
Plant 5: Lower Shell	SA-533, Gr B, Cl 1	3	99	71
Plant 5: Upper Shell	SA-508, Cl 2	3	97	57
Plant 6: Inter. Shell	SA-533, Gr B, Cl 1	3	81	79[(6)]
Plant 6: Inter. Shell	SA-533, Gr B, Cl 1	3	107	83
Plant 6: Lower Shell	SA-533, Gr B, Cl 1	3	106	84
Plant 6: Lower Shell	SA-533, Gr B, Cl 1	3	92	73
Plant 6: Upper Shell	SA-533, Gr B, Cl 1	3	101	83
Plant 6: Upper Shell	SA-533, Gr B, Cl 1	3	91	76
Plant 7: Inter. Shell	SA-302, Gr A	3	62	46
Plant 7: Inter. Shell	SA-302, Gr A	3	64	60[(6)]
Plant 7: Inter. Shell	SA-302, Gr A	3	74	69[(6)]

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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 7: Lower Shell	SA-302, Gr A	3	78	62
Plant 7: Lower Shell	SA-302, Gr A	3	74	56
Plant 7: Lower Shell	SA-302, Gr A	3	77	59
Plant 7: Upper Shell	SA-302, Gr B	3	54	42
Plant 7: Upper Shell	SA-302, Gr B	3	80	61
Plant 7: Upper Shell	SA-302, Gr B	3	62	50
Plant 8: Inter. Shell	SA-533, Gr B, Cl 1	3	94	68
Plant 8: Inter. Shell	SA-533, Gr B, Cl 1	3	83	60
Plant 8: Lower Shell	SA-533, Gr B, Cl 1	3	85	62
Plant 8: Lower Shell	SA-533, Gr B, Cl 1	3	83	57(6)
Plant 8: Upper Shell	SA-508, Cl 2	3	101	61
Plant 9: Inter. Shell	SA-533, Gr B, Cl 1	3	83	61
Plant 9: Inter. Shell	SA-533, Gr B, Cl 1	3	79	58
Plant 9: Lower Shell	SA-533, Gr B, Cl 1	3	82	61
Plant 9: Lower Shell	SA-533, Gr B, Cl 1	3	78	58
Plant 9: Upper Shell	SA-533, Gr B, Cl 1	3	98	80
Plant 9: Upper Shell	SA-533, Gr B, Cl 1	3	80	66
Plant 9: Upper Shell	SA-533, Gr B, Cl 1	3	98	80
Plant 10: Inter. Shell Forging	SA-508, Cl 2	3	74	50(6)
Plant 10: Lower Shell Forging	SA-508, Cl 2	3	80	58
Plant 10: Upper Shell Forging	SA-508, Cl 2	3	87	74
Plant 11: Inter. Shell	SA-533, Gr B, Cl 1	3	104	88(6)
Plant 11: Inter. Shell	SA-533, Gr B, Cl 1	3	94	67
Plant 11: Lower Shell	SA-533, Gr B, Cl 1	3	84	63
Plant 11: Lower Shell	SA-533, Gr B, Cl 1	3	83	63
Plant 11: Upper Shell Forging	SA-508, Cl 2	3	103	62
Plant 12: Inter. Shell	SA-533, Gr B, Cl 1	3	115	87
Plant 12: Inter. Shell	SA-533, Gr B, Cl 1	3	118	90

IPEC00014508

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 12: Lower Shell	SA-533, Gr B, Cl 1	3	103	95(6)
Plant 12: Lower Shell	SA-533, Gr B, Cl 1	3	125	95
Plant 12: Upper Shell Forging	SA-508, Cl 2	3	83	68
Plant 13: Inter. Shell Forging	SA-508, Cl 2	3	92	68
Plant 13: Lower Shell Forging	SA-508, Cl 2	3	85	68(6)
Plant 13: Upper Shell Forging	SA-508, Cl 2	3	75	60
Plant 14: Inter. Shell	SA-533, Gr B, Cl 1	3	83	66
Plant 14: Inter. Shell	SA-533, Gr B, Cl 1	3	71	55
Plant 14: Lower Shell	SA-533, Gr B, Cl 1	3	98	79
Plant 14: Lower Shell	SA-533, Gr B, Cl 1	3	88	71
Plant 14: Upper Shell	SA-533, Gr B, Cl 1	3	84	71
Plant 14: Upper Shell	SA-533, Gr B, Cl 1	3	90	76
Plant 15: Inter. Shell	SA-533, Gr B, Cl 1	4	80	64
Plant 15: Inter. Shell	SA-533, Gr B, Cl 1	4	81	62
Plant 15: Inter. Shell	SA-533, Gr B, Cl 1	4	98	78(6)
Plant 15: Lower Shell	SA-533, Gr B, Cl 1	4	86	67
Plant 15: Lower Shell	SA-533, Gr B, Cl 1	4	97	78
Plant 15: Lower Shell	SA-533, Gr B, Cl 1	4	90	70
Plant 15: Upper Shell	SA-533, Gr B, Cl 1	4	87	77
Plant 15: Upper Shell	SA-533, Gr B, Cl 1	4	92	81
Plant 15: Upper Shell	SA-533, Gr B, Cl 1	4	76	67
Plant 16: Inter. Shell	SA-533, Gr B, Cl 1	4	100	76
Plant 16: Inter. Shell	SA-533, Gr B, Cl 1	4	86	66
Plant 16: Lower Shell	SA-533, Gr B, Cl 1	4	110	88
Plant 16: Lower Shell	SA-533, Gr B, Cl 1	4	103	79
Plant 16: Upper Shell	SA-533, Gr B, Cl 1	4	>54(3)	>48
Plant 16: Upper Shell	SA-533, Gr B, Cl 1	4	>48(2)	>43
Plant 16: Upper Shell	SA-533, Gr B, Cl 1	4	>37(1)	>33

IPEC00014509

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 17: Inter. Shell	SA-302, Gr B	4	88	64(6)
Plant 17: Inter. Shell	SA-302, Gr B	4	82	70(6)
Plant 17: Inter. Shell	SA-302, Gr B	4	79	66(6)
Plant 17: Lower Shell	SA-302, Gr B	4	78	59
Plant 17: Lower Shell	SA-302, Gr B	4	74	53
Plant 17: Lower Shell	SA-302, Gr B	4	77	56
Plant 17: Upper Shell	SA-302, Gr B	4	105	86
Plant 17: Upper Shell	SA-302, Gr B	4	65	53
Plant 17: Upper Shell	SA-302, Gr B	4	90	74
Plant 18: Inter. Shell	SA-533, Gr B, Cl 1	4	101	99(6)
Plant 18: Inter. Shell	SA-533, Gr B, Cl 1	4	105	80
Plant 18: Inter. Shell	SA-533, Gr B, Cl 1	4	112	90
Plant 18: Lower Shell	SA-533, Gr B, Cl 1	4	95	72
Plant 18: Lower Shell	SA-533, Gr B, Cl 1	4	115	92
Plant 18: Lower Shell	SA-533, Gr B, Cl 1	4	103	82
Plant 18: Upper Shell	SA-533, Gr B, Cl 1	4	72	64
Plant 18: Upper Shell	SA-533, Gr B, Cl 1	4	68(4)	62
Plant 18: Upper Shell	SA-533, Gr B, Cl 1	4	95	85
Plant 19: Inter. Shell Forging	SA-508, Cl 2	4	94	78(6)
Plant 19: Lower Shell Forging	SA-508, Cl 2	4	141	106
Plant 19: Upper Shell Forging	SA-508, Cl 2	4	98	70
Plant 20: Inter. Shell Forging	SA-508, Cl 2	4	134	105
Plant 20: Lower Shell Forging	SA-508, Cl 2	4	134(5)	105
Plant 20: Upper Shell Forging	SA-508, Cl 2	4	101	72
Plant 21: Inter. Shell	SA-533, Gr B, Cl 1	4	96	75
Plant 21: Inter. Shell	SA-533, Gr B, Cl 1	4	82	64
Plant 21: Inter. Shell	SA-533, Gr B, Cl 1	4	92	72
Plant 21: Lower Shell	SA-533, Gr B, Cl 1	4	83	65

IPEC00014510

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 21: Lower Shell	SA-533, Gr B, Cl 1	4	102	80
Plant 21: Lower Shell	SA-533, Gr B, Cl 1	4	105	82
Plant 21: Upper Shell	SA-533, Gr B, Cl 1	4	96	86
Plant 21: Upper Shell	SA-533, Gr B, Cl 1	4	89	80
Plant 21: Upper Shell	SA-533, Gr B, Cl 1	4	65	59
Plant 22: Inter. Shell	SA-533, Gr B, Cl 1	4	90	72
Plant 22: Inter. Shell	SA-533, Gr B, Cl 1	4	100	80
Plant 22: Inter. Shell	SA-533, Gr B, Cl 1	4	107	88
Plant 22: Lower Shell	SA-533, Gr B, Cl 1	4	116	96
Plant 22: Lower Shell	SA-533, Gr B, Cl 1	4	113	94
Plant 22: Lower Shell	SA-533, Gr B, Cl 1	4	118	97
Plant 22: Upper Shell	SA-533, Gr B, Cl 1	4	94	83
Plant 22: Upper Shell	SA-533, Gr B, Cl 1	4	104	94
Plant 22: Upper Shell	SA-533, Gr B, Cl 1	4	92	81
Plant 23: Inter. Shell	SA-533, Gr B, Cl 1	4	95	74
Plant 23: Inter. Shell	SA-533, Gr B, Cl 1	4	104	81
Plant 23: Inter. Shell	SA-533, Gr B, Cl 1	4	84	66
Plant 23: Lower Shell	SA-533, Gr B, Cl 1	4	85	66
Plant 23: Lower Shell	SA-533, Gr B, Cl 1	4	83	65
Plant 23: Lower Shell	SA-533, Gr B, Cl 1	4	87	68
Plant 23: Upper Shell	SA-533, Gr B, Cl 1	4	77	65
Plant 23: Upper Shell	SA-533, Gr B, Cl 1	4	76	65
Plant 23: Upper Shell	SA-533, Gr B, Cl 1	4	84	74
Plant 24: Inter. Shell	SA-302, Gr B, Mod.	4	102	78
Plant 24: Inter. Shell	SA-302, Gr B, Mod.	4	97	72
Plant 24: Inter. Shell	SA-302, Gr B, Mod.	4	96	72
Plant 24: Lower Shell	SA-302, Gr B, Mod.	4	72	55
Plant 24: Lower Shell	SA-302, Gr B, Mod.	4	94	69

IPEC00014511

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-302, Gr B, Mod.	4	68	58(6)
Plant 24: Upper Shell	SA-302, Gr B, Mod.	4	62(3)	55
Plant 24: Upper Shell	SA-302, Gr B, Mod.	4	64(3)	57
Plant 24: Upper Shell	SA-302, Gr B, Mod.	4	85(3)	74
Plant 25: Inter. Shell	SA-302, Gr B Mod.	4	76	60(6)
Plant 25: Inter. Shell	SA-302, Gr B Mod.	4	75	59(6)
Plant 25: Inter. Shell	SA-302, Gr B Mod.	4	74	50(6)
Plant 25: Lower Shell	SA-302, Gr B Mod.	4	71	52
Plant 25: Lower Shell	SA-302, Gr B Mod.	4	88	65
Plant 25: Upper Shell	SA-302, Gr B Mod.	4	69	59
Plant 25: Upper Shell	SA-302, Gr B Mod.	4	64	57
Plant 25: Upper Shell	SA-302, Gr B Mod.	4	67	58
Plant 26: Inter. Shell	SA-533, Gr B, Cl 1	4	82	64
Plant 26: Inter. Shell	SA-533, Gr B, Cl 1	4	102	80
Plant 26: Inter. Shell	SA-533, Gr B, Cl 1	4	115	90
Plant 26: Lower Shell	SA-533, Gr B, Cl 1	4	78	61
Plant 26: Lower Shell	SA-533, Gr B, Cl 1	4	77	60
Plant 26: Lower Shell	SA-533, Gr B, Cl 1	4	81	62
Plant 26: Upper Shell	SA-533, Gr B, Cl 1	4	66	60
Plant 26: Upper Shell	SA-533, Gr B, Cl 1	4	67	61
Plant 26: Upper Shell	SA-533, Gr B, Cl 1	4	107	97
Plant 27: Inter. Shell	SA-533, Gr B, Cl 1	4	106	83
Plant 27: Inter. Shell	SA-533, Gr B, Cl 1	4	90	70
Plant 27: Inter. Shell	SA-533, Gr B, Cl 1	4	106	83
Plant 27: Lower Shell	SA-533, Gr B, Cl 1	4	77	60
Plant 27: Lower Shell	SA-533, Gr B, Cl 1	4	76	59
Plant 27: Lower Shell	SA-533, Gr B, Cl 1	4	79	62
Plant 27: Upper Shell	SA-533, Gr B, Cl 1	4	86	78

IPEC00014512

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 27: Upper Shell	SA-533, Gr B, Cl 1	4	104	94
Plant 27: Upper Shell	SA-533, Gr B, Cl 1	4	103	93
Plant 28: Inter. Shell	SA-533, Gr B, Cl 1	4	91	63
Plant 28: Inter. Shell	SA-533, Gr B, Cl 1	4	99	68
Plant 28: Inter. Shell	SA-533, Gr B, Cl 1	4	90	62
Plant 28: Lower Shell	SA-533, Gr B, Cl 1	4	112	79
Plant 28: Lower Shell	SA-533, Gr B, Cl 1	4	122	86
Plant 28: Lower Shell	SA-533, Gr B, Cl 1	4	100	76
Plant 28: Upper Shell	SA-533, Gr B, Cl 1	4	72[(4)]	66
Plant 28: Upper Shell	SA-533, Gr B, Cl 1	4	82	75
Plant 28: Upper Shell	SA-533, Gr B, Cl 1	4	86	78
Plant 29: Inter. Shell	SA-533, Gr B, Cl 1	4	116	81
Plant 29: Inter. Shell	SA-533, Gr B, Cl 1	4	114	82
Plant 29: Inter. Shell	SA-533, Gr B, Cl 1	4	77	59
Plant 29: Lower Shell	SA-533, Gr B, Cl 1	4	110	79
Plant 29: Lower Shell	SA-533, Gr B, Cl 1	4	103	75
Plant 29: Lower Shell	SA-533, Gr B, Cl 1	4	116	85
Plant 29: Upper Shell	SA-533, Gr B, Cl 1	4	76	68
Plant 29: Upper Shell	SA-533, Gr B, Cl 1	4	75	68
Plant 29: Upper Shell	SA-533, Gr B, Cl 1	4	81	72
Plant 30: Inter. Shell	SA-533, Gr B, Cl 1	4	106	84
Plant 30: Inter. Shell	SA-533, Gr B, Cl 1	4	97	83[(6)]
Plant 30: Inter. Shell	SA-533, Gr B, Cl 1	4	107	87
Plant 30: Lower Shell	SA-533, Gr B, Cl 1	4	98	78
Plant 30: Lower Shell	SA-533, Gr B, Cl 1	4	103	82
Plant 30: Lower Shell	SA-533, Gr B, Cl 1	4	121	97
Plant 30: Upper Shell	SA-533, Gr B, Cl 1	4	87	78
Plant 30: Upper Shell	SA-533, Gr B, Cl 1	4	79	70

IPEC00014513

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 30: Upper Shell	SA-533, Gr B, Cl 1	4	69	62
Plant 31: Inter. Shell	SA-533, Gr B, Cl 1	4	117	88
Plant 31: Inter. Shell	SA-533, Gr B, Cl 1	4	101	79
Plant 31: Lower Shell	SA-533, Gr B, Cl 1	4	99	73
Plant 31: Lower Shell	SA-533, Gr B, Cl 1	4	84	72(6)
Plant 31: Upper Shell	SA-533, Gr B, Cl 1	4	75(4)	68
Plant 31: Upper Shell	SA-533, Gr B, Cl 1	4	77(4)	68
Plant 31: Upper Shell	SA-533, Gr B, Cl 1	4	85	76
Plant 32: Inter. Shell	SA-533, Gr B, Cl 1	4	91	70(6)
Plant 32: Inter. Shell	SA-533, Gr B, Cl 1	4	98	83(6)
Plant 32: Inter. Shell	SA-533, Gr B, Cl 1	4	104	88(6)
Plant 32: Lower Shell	SA-533, Gr B, Cl 1	4	93	69
Plant 32: Lower Shell	SA-533, Gr B, Cl 1	4	83	61
Plant 32: Lower Shell	SA-533, Gr B, Cl 1	4	85	63
Plant 32: Upper Shell	SA-533, Gr B, Cl 1	4	74	63
Plant 32: Upper Shell	SA-533, Gr B, Cl 1	4	79	68
Plant 32: Upper Shell	SA-533, Gr B, Cl 1	4	68	57
Plant 33: Inter. Shell	SA-533, Gr B, Cl 1	4	127	100
Plant 33: Inter. Shell	SA-533, Gr B, Cl 1	4	127	100
Plant 33: Inter. Shell	SA-533, Gr B, Cl 1	4	135	107
Plant 33: Lower Shell	SA-533, Gr B, Cl 1	4	87	69
Plant 33: Lower Shell	SA-533, Gr B, Cl 1	4	100	79
Plant 33: Lower Shell	SA-533, Gr B, Cl 1	4	89	88(6)
Plant 33: Upper Shell	SA-533, Gr B, Cl 1	4	118	107
Plant 33: Upper Shell	SA-533, Gr B, Cl 1	4	121	110
Plant 33: Upper Shell	SA-533, Gr B, Cl 1	4	133	120
Plant 34: Inter. Shell	SA-533, Gr B, Cl 1	4	78	62
Plant 34: Inter. Shell	SA-533, Gr B, Cl 1	4	100	79

IPEC00014514

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 34: Inter. Shell	SA-533, Gr B, Cl 1	4	99	78
Plant 34: Lower Shell	SA-533, Gr B, Cl 1	4	104	88(6)
Plant 34: Lower Shell	SA-533, Gr B, Cl 1	4	105	83
Plant 34: Lower Shell	SA-533, Gr B, Cl 1	4	101	80
Plant 34: Upper Shell	SA-533, Gr B, Cl 1	4	103	93
Plant 34: Upper Shell	SA-533, Gr B, Cl 1	4	88	80
Plant 34: Upper Shell	SA-533, Gr B, Cl 1	4	101	91
Plant 35: Inter. Shell	SA-533, Gr B, Cl 1	4	94	73
Plant 35: Inter. Shell	SA-533, Gr B, Cl 1	4	103	80
Plant 35: Inter. Shell	SA-533, Gr B, Cl 1	4	88	69
Plant 35: Lower Shell	SA-533, Gr B, Cl 1	4	85	66
Plant 35: Lower Shell	SA-533, Gr B, Cl 1	4	78	61
Plant 35: Lower Shell	SA-533, Gr B, Cl 1	4	98	76
Plant 35: Upper Shell	SA-533, Gr B, Cl 1	4	83	76
Plant 35: Upper Shell	SA-533, Gr B, Cl 1	4	75	68
Plant 35: Upper Shell	SA-533. Gr B, Cl 1	4	108	98
Plant 36: Inter. Shell	SA-533, Gr B, Cl 1	4	111	89
Plant 36: Inter. Shell	SA-533, Gr B, Cl 1	4	101	81
Plant 36: Inter. Shell	SA-533, Gr B, Cl 1	4	105	84
Plant 36: Lower Shell	SA-533, Gr B, Cl 1	4	107	86
Plant 36: Lower Shell	SA-533, Gr B, Cl 1	4	106	85
Plant 36: Lower Shell	SA-533, Gr B, Cl 1	4	108	86
Plant 36: Upper Shell	SA-533, Gr B, Cl 1	4	76	69
Plant 36: Upper Shell	SA-533, Gr B, Cl 1	4	87	79
Plant 36: Upper Shell	SA-533, Gr B, Cl 1	4	86	78
Plant 37: Inter. Shell Forging	SA-508, Cl 2	4	83	74(6)
Plant 37: Lower Shell Forging	SA-508, Cl 2	4	85	65
Plant 37: Upper Shell Forging	SA-508, Cl 2	4	68	49

IPEC00014515

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: Inter. Shell	SA-533, Gr B, Cl 1	4	110	86
Plant 38: Inter. Shell	SA-533, Gr B, Cl 1	4	104	81
Plant 38: Inter. Shell	SA-533, Gr B, Cl 1	4	106	83
Plant 38: Lower Shell	SA-533, Gr B, Cl 1	4	111	85
Plant 38: Lower Shell	SA-533, Gr B, Cl 1	4	122	94
Plant 38: Lower Shell	SA-533, Gr B, Cl 1	4	127	98
Plant 38: Upper Shell	SA-533, Gr B, Cl 1	4	89	82
Plant 38: Upper Shell	SA-533, Gr B, Cl 1	4	85	78
Plant 38: Upper Shell	SA-533, Gr B, Cl 1	4	82	75
Plant 39: Inter. Shell	SA-533, Gr B, Cl 1	4	109	86
Plant 39: Inter. Shell	SA-533, Gr B, Cl 1	4	129	102
Plant 39: Inter. Shell	SA-533, Gr B, Cl 1	4	122	96
Plant 39: Lower Shell	SA-533, Gr B, Cl 1	4	124	95
Plant 39: Lower Shell	SA-533, Gr B, Cl 1	4	118	91
Plant 39: Lower Shell	SA-533, Gr B, Cl 1	4	123	95
Plant 39: Upper Shell	SA-533, Gr B, Cl 1	4	114	105
Plant 39: Upper Shell	SA-533, Gr B, Cl 1	4	124	114
Plant 39: Upper Shell	SA-533, Gr B, Cl 1	4	127	117
Plant 40: Inter. Shell Forging	SA-508, Cl 2	4	79	57
Plant 40: Lower Shell Forging	SA-508, Cl 2	4	72	55[6]
Plant 40: Vessel Ring Forging	SA-508, Cl 2	4	114	82
Plant 41: Inter. Shell Forging	SA-508, Cl 2	4	62	44
Plant 41: Lower Shell Forging	SA-508, Cl 2	4	111	87
Plant 41: Vessel Ring Forging	SA-508, Cl 2	4	96	85
Plant 42: Inter. Shell Forging	SA-508, Cl 2	4	110	86
Plant 42: Lower Shell Forging	SA-508, Cl 2	4	121	94
Plant 42: Vessel Ring Forging	SA-508, Cl 2	4	93	84

IPEC00014516

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)
Note: (1) Forty (40) percent shear - maximum: 65 percent longitudinal (2) Fifty (50) percent shear - maximum: 65 percent longitudinal (3) Seventy (70) percent shear - maximum: 65 percent longitudinal (4) Ninety (90) percent shear - maximum: 65 percent longitudinal (5) Ninety (90) percent shear - maximum (6) Calculated using surveillance capsule data and R.G. 1.99, Rev. 2 guidelines				

IPEC00014517

Table 2-2
WOG Reactor Vessel Surveillance Weld Data

Plant	Weld Flux	Initial USE (ft-lbs)	EOL USE (ft-lbs)
Plant 1	UM89	103	89(1)
Plant 2	UM89	78	75(1)
Plant 3	Linde 1092	126	68(1)
Plant 4	Linde 1091	149	103(1)
Plant 5	-	148	130(1)
Plant 6	Linde 124	93	86(1)
Plant 7	Linde 1092	113	53(1)
Plant 8	Linde 1092	112	71(1)
Plant 9	Linde 0091	145	100
Plant 10	Grau Lo (LW320)	111	71(1)
Plant 11	Grau Lo (LW320)	91	59(1)
Plant 12	Included as Part of B&W Owners Group Program		
Plant 13	Smit 89 (Saf 89)	102	97(1)
Plant 14	Linde 124	88	70
Plant 15	Linde 1092	110	69(1)
Plant 16	Linde 124	77	69(1)
Plant 17	ARCOS B5	106	50(1)
Plant 18	Linde 1092	112	65(1)
Plant 19	Grau Lo (LW320)	132	129(1)
Plant 20	Grau Lo (LW320)	129	101
Plant 21	Linde 0091	146	114
Plant 22	Linde 0091	152	122
Plant 23	Linde 124	92	72
Plant 24	Linde 1092	120	72(1)
Plant 25	Linde 1092	121	68(1)
Plant 26	Linde 0091	161	126
Plant 27	Linde 0091	138	108
Plant 28	Linde 1092	114	64

IPEC00014518

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

Plant	Weld Flux	Initial USE (ft-lbs)	EOL USE (ft-lbs)
Plant 29	Linde 1092	94	52
Plant 30	Linde 1092	112	69(1)
Plant 31	Linde 124	100	79(1)
Plant 32	Linde 1092	104	85(1)
Plant 33	Linde 124	96	85(1)
Plant 34	Linde 124	113	98(1)
Plant 35	Linde 0091	133	104
Plant 36	Linde 124	96	77
Plant 37	Smit 89 (Saf 89)	104	68(1)
Plant 38	Linde 124	94	73
Plant 39	Linde 124	112	88
Plant 40	Smit 89 (Saf 89)	113	63(1)
Plant 41	Grau Lo (LW320)	134	105
Plant 42	Grau Lo (LW320)	144	112

Note: (1) Calculated using surveillance capsule data and R.G. 1.99, Rev. 2 guidelines

IPEC00014519

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

Case	Lowest EOL USE (ft-lbs)
2 Loop	
SA-508, Cl 2	53
SA-508, Cl 3	65
2 Loop Bounding Value	***50***(1)
3 Loop	
SA-302, Gr A	46
SA-302, Gr B	42
SA-508, Cl 2	50
SA-533, Gr B, Cl 1	55
3 Loop Bounding Value	***42***
4 Loop	
SA-302, Gr B	53
SA-302, Gr B Mod.	50
SA-508, Cl 2	44
SA-533, Gr B, Cl 1	48
4 Loop Bounding Value	***44***

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

IPEC00014520

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

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

IPEC00014521

IPEC00014521

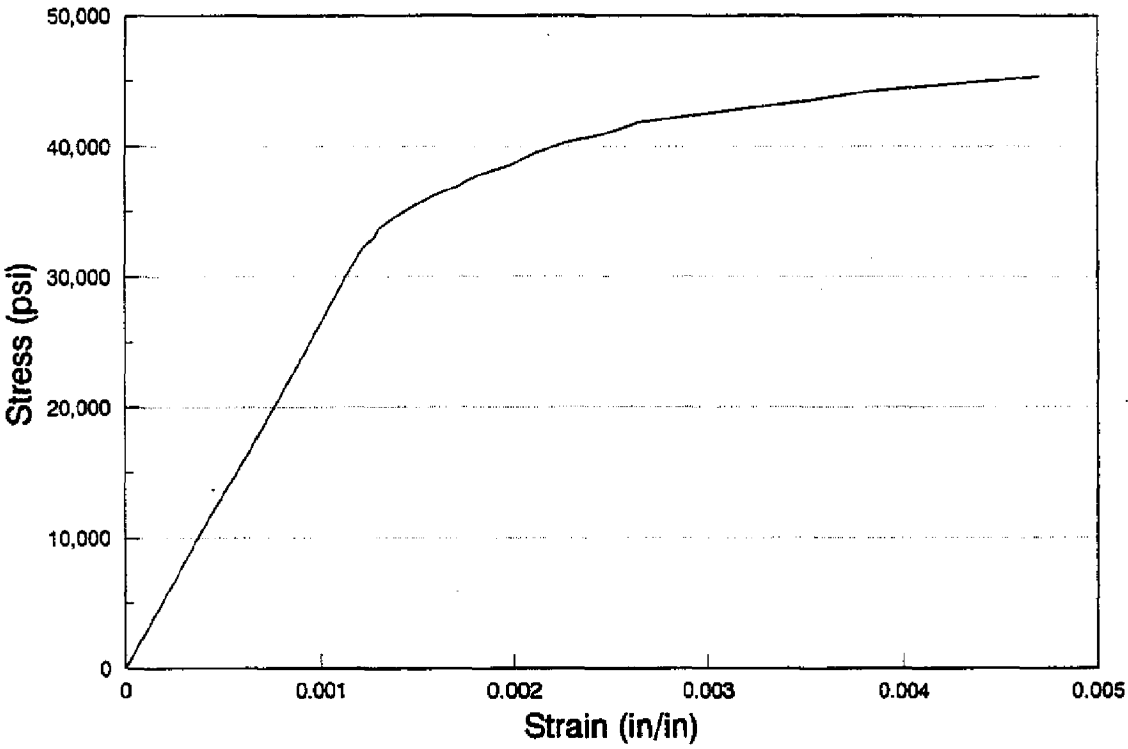

Figure 2-1
Representative Stress - Strain Curve
For Code Minimum Values

IPEC00014522

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 7 and 17 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 Reference 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 Reference 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 $0.20 \le a/t \le 0.50$, and includes the effect of pressure acting on the flaw faces.

IPEC00014523

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:

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

$$F_3 = 0.690 + 3.127\ (a/t) - 7.435\ (a/t)^2 + 3.532\ (a/t)^3$$

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:

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

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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 because it 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 Reference 11 approach. The through-wall stress distribution was first determined using the appropriate material, loading conditions, 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 stress intensity factors.

Based on this evaluation, it was concluded that the approach identified in Reference 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.

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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. 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 elastic stress distributions are given in Figure 3-4 for the small steam line break, Figure 3-5 for the large steam line break, and Figure 3-6 for the large LOCA. The elastic-plastic stress distributions are also shown for comparison. From Figures 3-5 and 3-6, 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

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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 procedure for obtaining $J_{applied}$ from the PCFAD output has been prepared by J. Bloom, the author of the PCFAD code, and is contained in Appendix B. The steps as described in the Appendix B procedure are as follows:

1. Run PCFAD using plant specific material and geometry parameters with Pressure and Fracture Toughness set to unity ($K_{Ic} = 1$, $P = 1$).

2. Take KR' from the output and multiply it by the actual pressure of the transient to obtain K_I.

3. Calculate $J_{elastic}$ using the following relationship:

$$J_{elastic} = \frac{1000 K_I^2 \, (1-\nu^2)}{E}$$

4. $J_{applied}$ is then obtained by dividing $J_{elastic}$ by KR^2 where KR is the value of the failure assessment diagram at SR' x pressure.

Instead of calculating K_I by multiplying KR' by P, K_I can alternatively be obtained from the actual elastic stress distribution. The elastic stress distribution resulting from thermal and pressure loadings are then used to calculate K_I. The stress intensity factor has been calculated using the Raju-Newman model as described in Reference 15. K_I values were calculated for each case. Values of the stress intensity factor for a 1.05" flaw were also determined. These values are necessary to calculate dJ/da. $J_{applied}$ is evaluated at both 1" and 1.05" the difference in these values is divided by 0.05 to determine the slope of the curve. Note only a portion of the Appendix B method is used. The portion relates $J_{elastic}$ to $J_{applied}$. As described in Appendix B, use of this methodology is justified as results are benchmarked against finite element results.

The pressure used in determining KR for the conversion from $J_{elastic}$ to $J_{applied}$ is the peak transient pressure for the Large and Small SLB. These values are:

Small SLB: 2335 psi
Large SLB: 2344 psi

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The PCFAD output for each case is contained in Table 3-3 for Level C transients and Table 3-4 for Level D transients. It should be noted that only the stability conditions are evaluated for the 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.

As an example of this procedure, the plant 7 (plant 5 in Rev. 0) PCFAD output as contained in Attachment C of Appendix C is utilized to calculate the $J_{applied}$ value. The stress intensity factor at the crack tip for a 0.25t flaw for the level C transient has been calculated to be 93.65 ksi $(inches)^{0.5}$ using the technique in Reference 15. $J_{elastic}$ can then be calculated as follows:

$$J_{elastic} = \frac{(1000)\ (93.65\ ksi\sqrt{in})^2\ (1 - 0.3^2)}{26400\ ksi} = 302.31\ in\text{-}lbs/in^2$$

Based on the PCFAD output contained in Appendix C, SR' = 0.151. The peak pressure of the small steam line break transient is 2.335 ksi as listed previously. The SR' value multiplied by this peak pressure is 2.335 x 0.151 = 0.353. The KR value corresponding to SR = 0.353 is 0.99. $J_{applied}$ is then calculated as follows:

$$J_{applied} = \frac{J_{elastic}}{KR^2} = \frac{302.31}{(0.99)^2} = 308\ in\text{-}lbs/in^2$$

3.4 Available Margin

The margins available as shown in Tables 3-2 through 3-4 are in terms of J. The margins can be transformed into terms of upper shelf energies by solving for the variable CVN from the equations shown in Section 2.6.1. Upper shelf energy values have been calculated for each case which equate $J_{applied}$ and $J_{material}$. These values represent the lowest upper shelf energy which would satisfy the Appendix X requirements as stated in Section 1.4. These values have been tabulated in Table 3-5.

IPEC00014528

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 7	9.875 *	77.97 *
Plant 17	10.625	77.00

* These are dimensions for the upper shell of Plant 7 as this base metal exhibited the lowest EOL Charpy energy.

IPEC00014529

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	384	318	702	2925	*Yes*
3 Loop	500	321	585 (527)*	2140 (599)*	*Yes*
4 Loop	590	345	614	2330	*Yes*
Plant 7 Plant 17	525 548	222 197	527 614	599 2330	*Yes* *Yes*
* Calculations based on J-R curves in Reference 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	311	225	702	2925	*Yes*
3 Loop	310	252	585 (527)*	2140 (599)*	*Yes*
4 Loop	311	225	614	2330	*Yes*
Plant 7 Plant 17	308 319	250 240	527 614	599 2330	*Yes* *Yes*
* Calculations based on J-R curves in Reference 9.					

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Table 3-4
Level D Conditions

Case	Applied $dJ_{applied}/da$	Material $dJ_{material}/da$	Met Criteria?
2 Loop	447	2925	*Yes*
3 Loop	447	2140 (599)*	*Yes*
4 Loop	447	2330	*Yes*
Plant 7 Plant 17	443 468	599 2330	*Yes* *Yes*

* Calculations based on J-R curves in Reference 9.

Table 3-5
Minimal Acceptable USE Values

Case	Lowest USE (ft-lbs)
2 Loop	29
3 Loop	42
4 Loop	43

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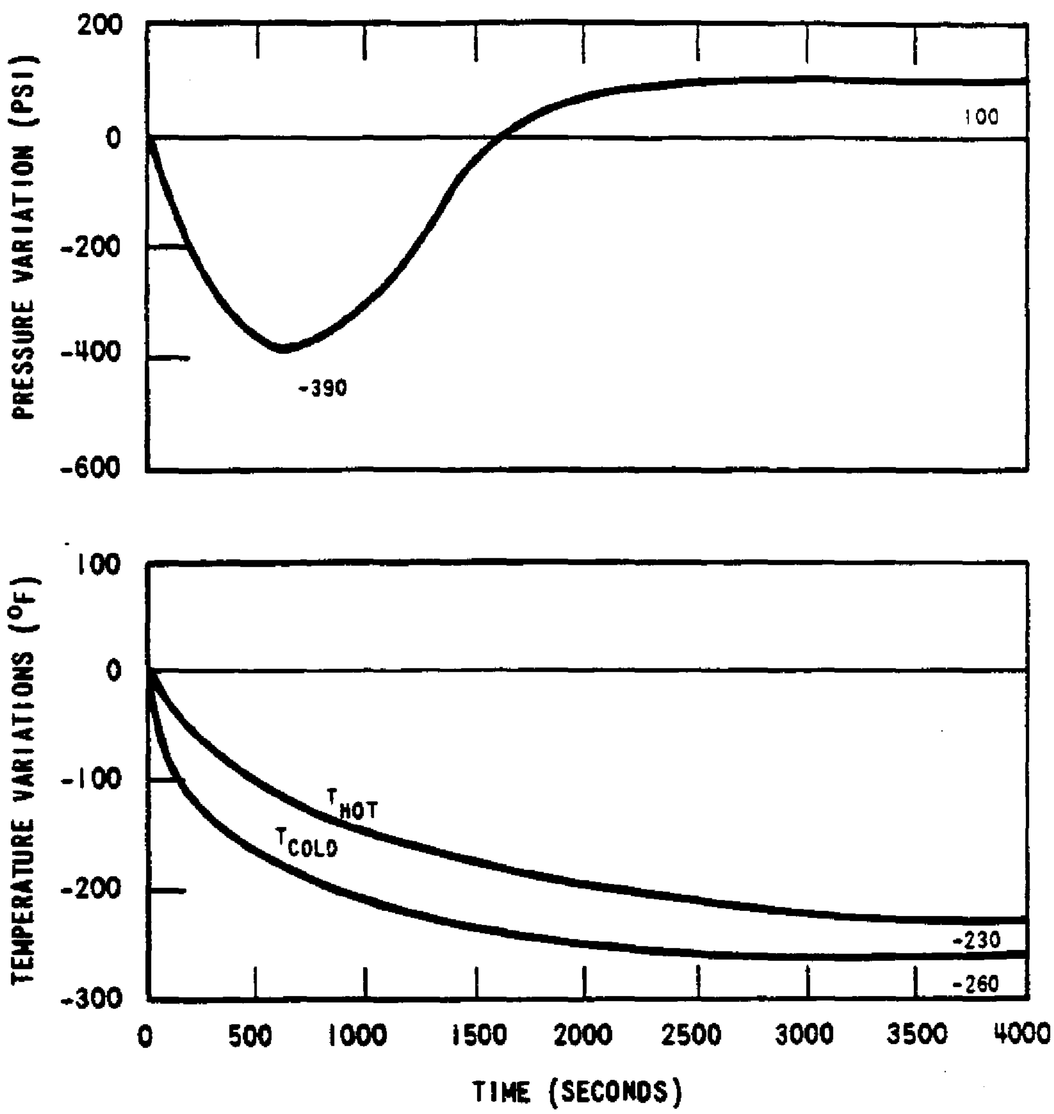

Figure 3-1
Temperature and Pressure History for Small Steam Line Break Transient

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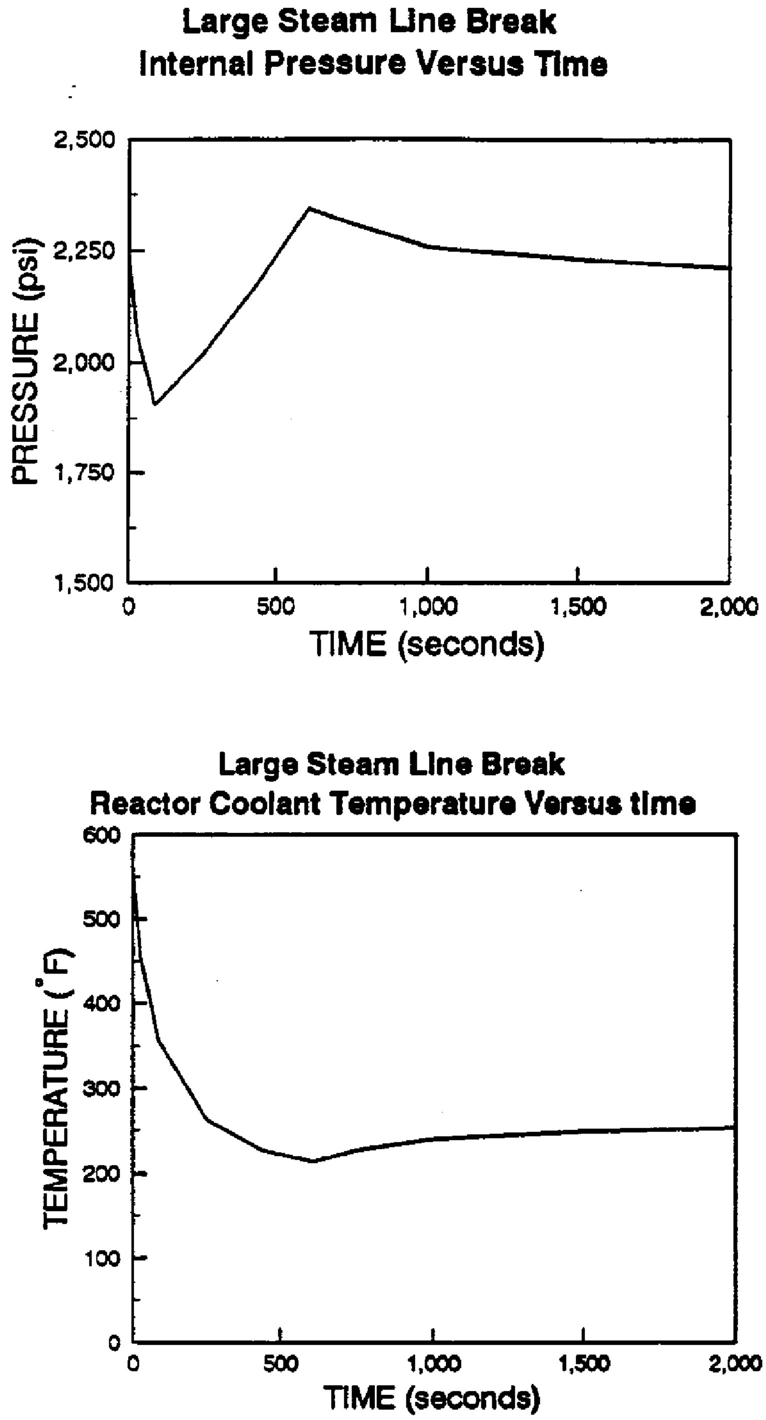

Figure 3-2
Temperature and Pressure History for Large Steam Line Break Transient

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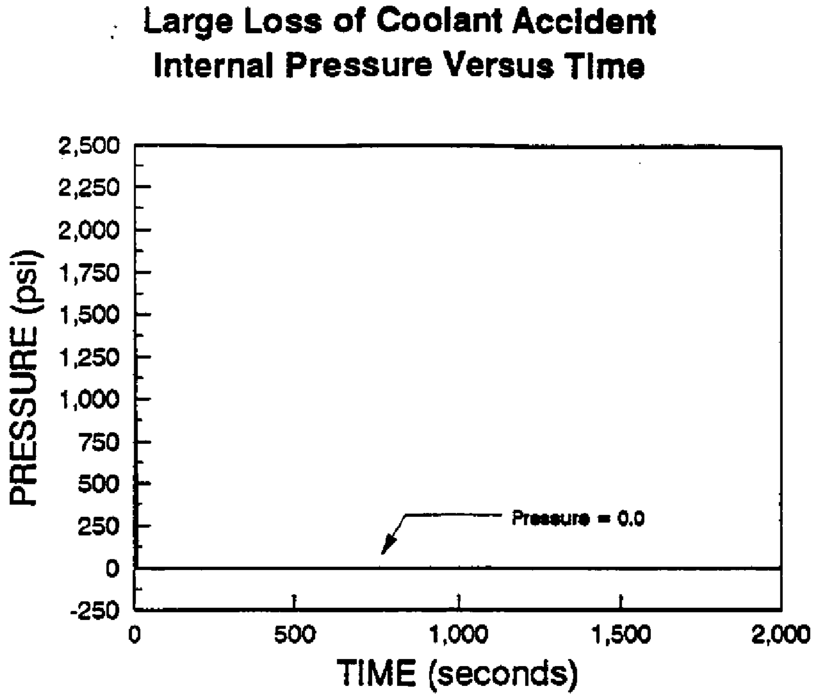

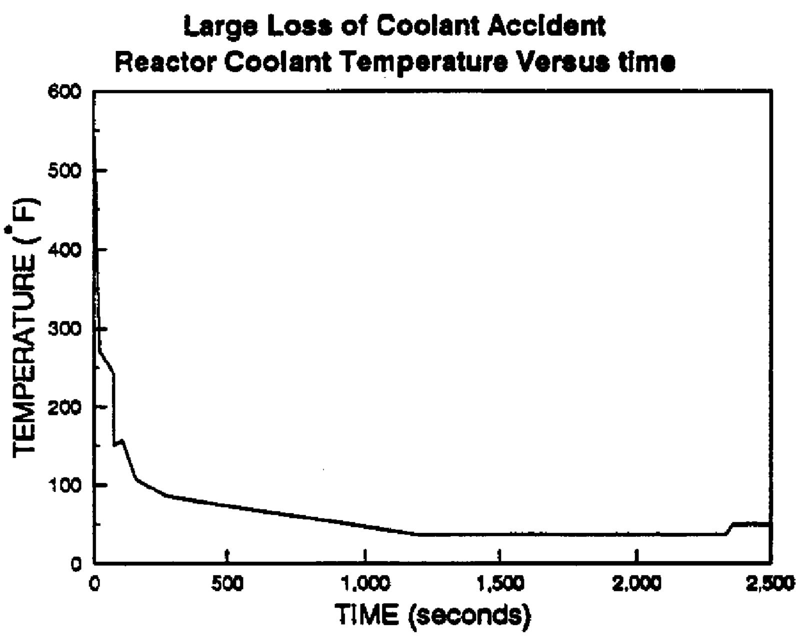

Figure 3-3
Temperature and Pressure History for Large LOCA Transient

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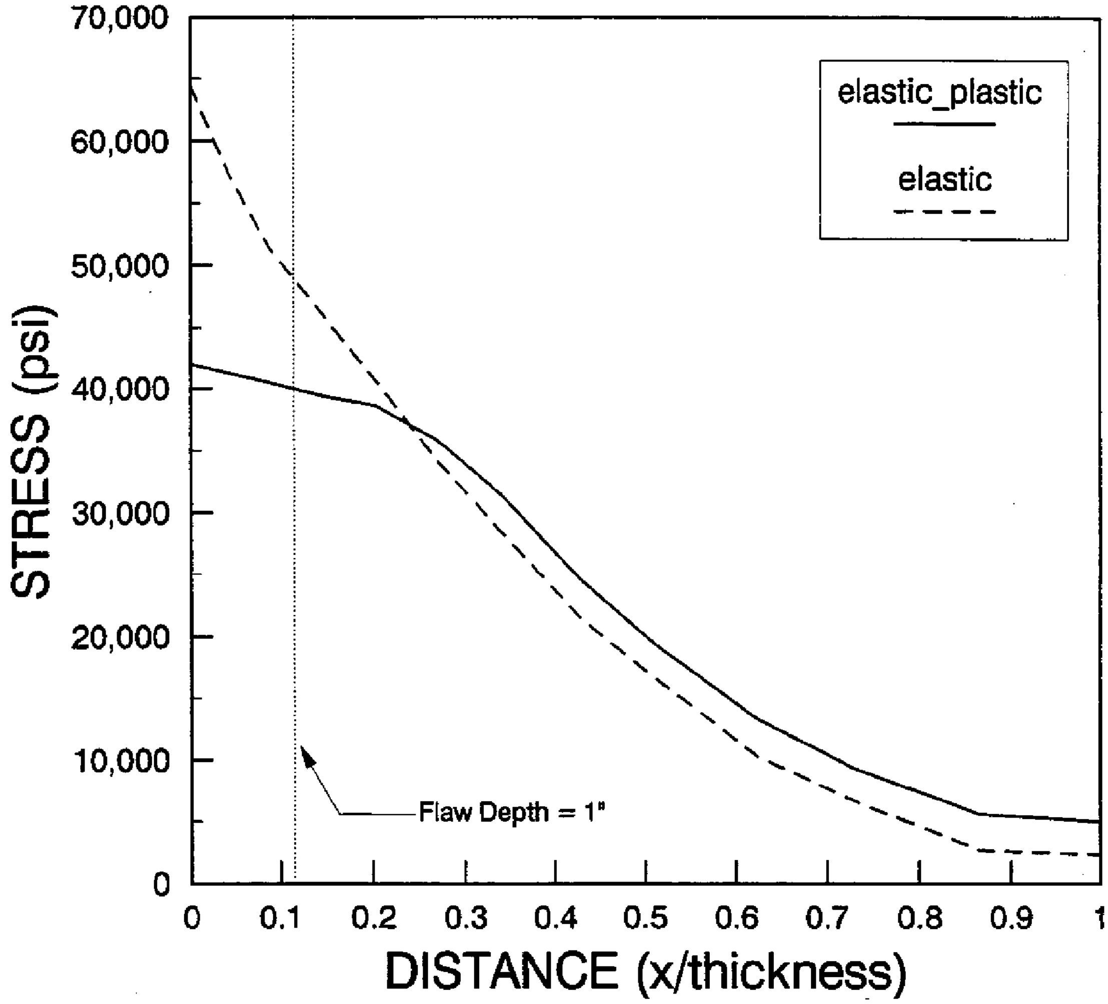

Figure 3-4
Through-Wall Elastic and Elastic-Plastic Stress Distribution for Small Steam Line Break Transient

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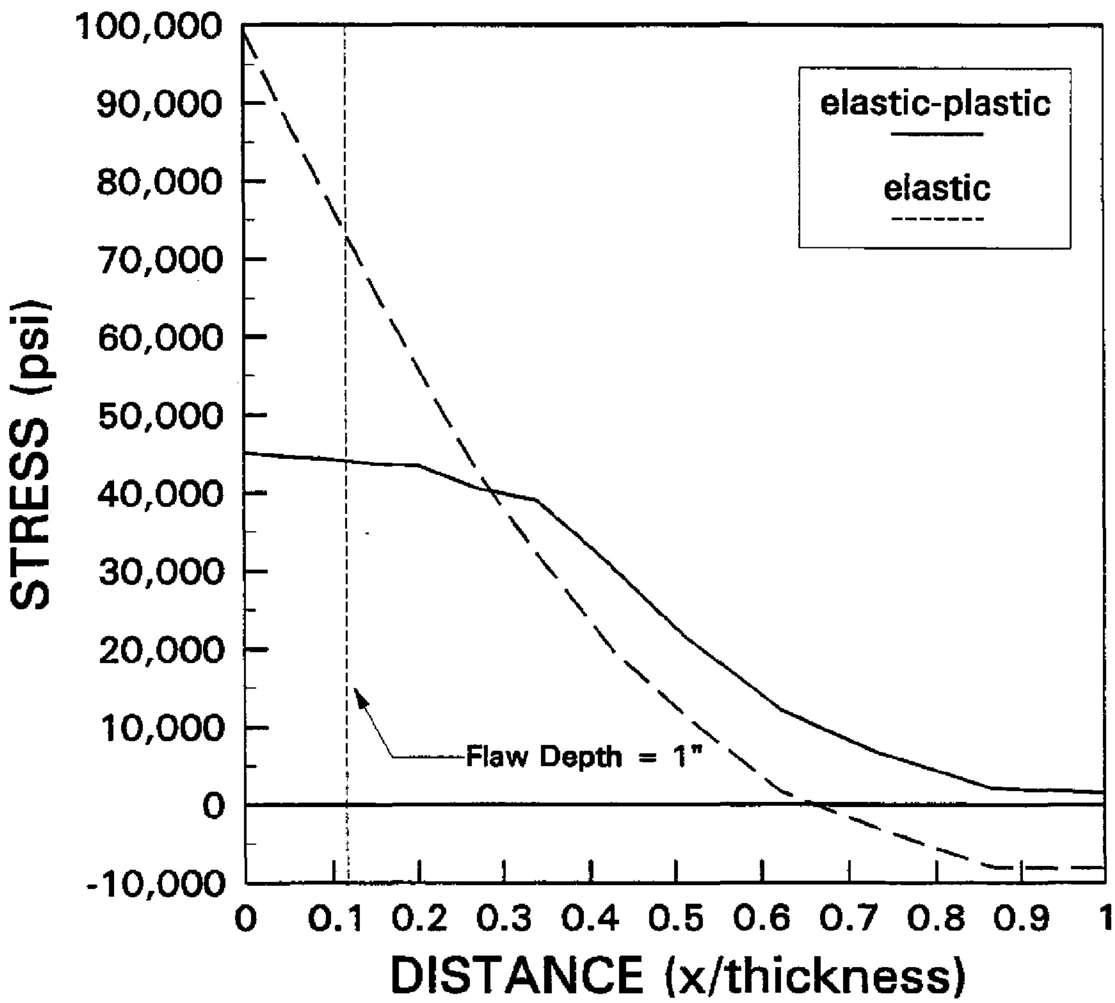

Figure 3-5
Through-Wall Elastic and Elastic-Plastic Stress Distribution for Large Steam Line Break Transient

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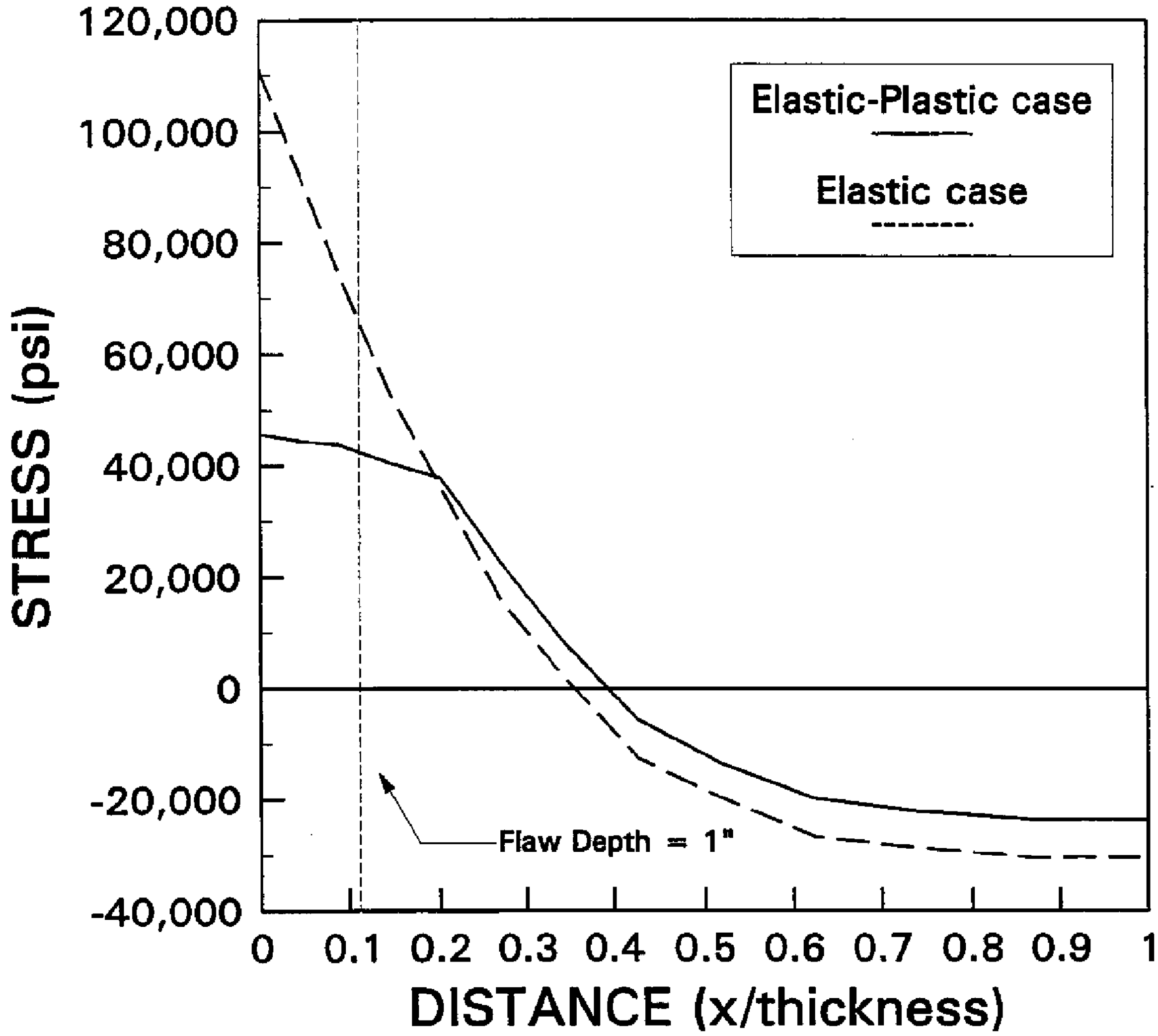

Figure 3-6
Through-Wall Elastic and Elastic-Plastic Stress Distribution for Large LOCA

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4.0 RESULTS AND 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 end of license life. This was accomplished by demonstrating that the reactor vessel beltline 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 bounding 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.

In order to obtain a better knowledge of the maximum available margin that can exist, upper shelf energy values were calculated for 2, 3, and 4 loop plants which equate $J_{applied}$ and $J_{material}$. These values are listed in Table 3-5 and represent the lowest upper shelf energies that satisfy the Appendix X requirements.

IPEC00014538

5.0 REFERENCES

1. Code of Federal Register 10 CFR Part 50, Appendix G "Fracture Toughness Requirements."

2. ASME Boiler and Pressure Vessel Code, Section III, July 1989.

3. ASME Boiler and Pressure Vessel Code, Section XI, July 1989.

4. R. Johnson, "Resolution of the Reactor Vessel Materials Toughness Safety Issue, Volumes 1 and 2," Nuclear Regulatory Commission Report NUREG 0744, 1982.

5. USNRC Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54 [4]" Dated March 6, 1991.

6. ASME Boiler and Pressure Vessel Code, Section III - Division I Appendices, June 1983.

7. Reddy G. B., Ayres D. J. "High-Temperature Elastic-Plastic and Creep Properties for SA533 Grade B and SA508 Materials," Combustion Engineering Research Project 2055-8, Paper NP-2763, December 1982.

8. Eason E. D., et al, NUREG/CR-5729, MCS 910401, "Multivariable Modeling of Pressure Vessel and Piping J-R Data," May 1991.

9. Hiser A. L., Terrell J. B., NUREG/CR-5265, MEA-2320, "Size Effects on J-R Curves for A 302-B Plate," dated January 1989.

10. Begley J. A., WCAP-13554, "Effects of Section Size and Cleanliness on the Upper Shelf and Transition Range Toughness of Three Nuclear Pressure Vessel Steels," August 1992.

11. "Background to Evaluation Procedures in ASME Section XI for Assessment Vessels with Low Upper Shelf Charpy Energy Levels", Revision 13, November 25, 1992.

12. Westinghouse Electric Computer ANalysis (WECAN) Code, File 87-1J7-WESAD-R1, dated December 1987.

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

14. Bloom J. M., and Lee D. R., "Users Guide for the Failure Assessment Diagram Computer Code FAD, "Babcock and Wilcox, Rev. 4, April 1990.

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15. Newman J.C. Jr., and Raju. I.S., "Stress Intensity Factors for Internal Surface Cracks in Cylindrical Pressure Vessels", ASME Trans., Journal of Pressure Vessel Technology, Vol. 102, 1980, pp 342-346.

16. Regulatory Guide 1.99, Proposed Revision 2, "Radiation Damage to Reactor Vessel Materials", U.S. Nuclear Regulatory Commission, February, 1986.

17. American Society For Testing And Materials (ASTM), Designation: E-185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706(IF)".

18. Branch Technical Position - MTEB 5-2, "Fracture Toughness Requirements", included in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition", USNRC.

19. USNRC SECY-91-333, "Additional Requirements for Yankee Rowe Pressure Vessel Issues", October 22, 1991.

IPEC00014540

APPENDIX A

SA 302 GRADE A ASSESSMENT

Plant 7 was procured to SA 302 Grade B specifications which required a minimum yield strength of 50 ksi and minimum ultimate strength of 80 ksi at room temperature per ASME Section III [2]. Two intermediate core region plates of Plant 7 fell below the yield and ultimate strengths as specified for SA 302 Grade B and were reclassified as SA 302 Grade A. At room temperature, SA 302 Grade A minimum yield strength is 45 ksi and minimum ultimate strength is 75 ksi per [2]. At a temperature of 600°F, SA 302 Grade A minimum yield strength is 37.8 ksi and minimum ultimate strength is 75 ksi per [2].

As shown in Table 2-1, the upper shell of Plant 7 demonstrates the lowest EOL USE. The lowest EOL USE for SA 302 Grade B is 42 ft-lbs whereas the lowest EOL USE for SA 302 Grade A is 46 ft-lbs. For this bounding analysis, it was judged that the decrease in margin between material and applied J-integral values due to the lower mechanical properties of the SA 302 Grade A material would be more than offset by the increase in margin between the material and applied J-integral values due to the higher EOL USE of the intermediate shell. Consequently, the mechanical properties of the upper shell material, SA 302 Grade B, were used in this analysis along with the lower EOL USE for this material.

IPEC00014541

APPENDIX B

Procedure for Obtaining J-Applied from PCFAD

First the dimensions of the particular flawed geometry (structure) are input into PCFAD along with the relevant material properties, Ramberg-Osgood constants, α,n, Poisson's ratio, ν, Young's modulus, E, and yield strength, σ_{ys}. Since a Ramberg-Osgood material is discussed, the ultimate strength, σ_{ult}, can be set arbitrarily (it is suggested that a value of σ_{ult} of at least 1.2 times σ_{ys} be used). Remember the program, PCFAD, calculates α,n based on the inputs of σ_{ys} and σ_{ult}. Therefore, these values must be replaced with actual α,n values from a fit of the actual stress-strain tensile data used in the finite element analysis.

Next, the load, stress, or pressure is set equal to unity and toughness is input as K_{IC} and set to unity as well.

The attached output file "J-APPLIED" is given in USA customary units for a semi-elliptical interior axial flaw in a pressurized vessel with inside radius of 90 inches. The flaw is a quarter of the 9-inch wall and has an aspect ratio, length to depth of 6:1. Note that the output first lists the coordinates of the FAD curve SR, KR followed by the assessment points SR', KR'.

The limit load, P_L, is obtained from the inverse of SR' (P_L = P/0.1609) while the stress intensity factor, K_I, is KR' times the load, stress, or pressure in units of ksi (inches)$^{0.5}$ (K_I = 30.5237P).

For the example chosen, the J-APPLIED value in in-lb/in^2 units for a pressure of 1 ksi is obtained as follows:

At P = 1 ksi, SR = SR' = 0.161

At that value of SR, the corresponding KR value is approximately 0.997 (taken from the table of SR, KR values). The stress intensity factor is 30.5237 ksi (inch)$^{0.5}$ which corresponds to a $J_{elastic}$ = 28.3 in-lb/in^2 from the relationship

$$\frac{1000K_I^2\ (1-\nu^2)}{E} = J_{elastic} \qquad (1)$$

where E is of the units of KSI.

J-APPLIED is then taken from the expression

$$J = J_{elastic} / (KR)^2 \qquad (2)$$

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For the example chosen,

$$J = 28.3/(.997)^2 = 28.4 \text{ in-lb/in}^2$$

For P = 2.5 KSI, SR = 2.5 x .161 = .4025 and KR = .984 from the SR, KR table. The stress intensity factor = 2.5 x 30.5237 = 76.31 KSI $(\text{inch})^{0.5}$. The corresponding $J_{elastic}$ value is 176.63 in-lb/in^2 using expression (1) and J-APPLIED equals $176.63/(.984)^2 = 182.4$ in-lb/in^2.

The resulting table of J-APPLIED values is given along with a comparison of results found from a finite element analysis by W. W. Wikening ASME PVP Vol. 106, August 1984.

Pressure (KSI)	J-APPLIED (PCFAD) (in-lb/in^2)	J-APPLIED (FEA) (in-lb/in^2)
2.5	182	186
4.5	708	738
6.0	2,484	2,638
7.5	14,600	13,466

The differences in the table are due primarily to interpolation of the output of PCFAD. While values of SR are calculated per .005 interval, the program only prints values every .05 interval.

The key expressions for converting output from PCFAD into J-APPLIED values are (1) and (2). Also, remember that SR, KR, SR', KR' are all linear in load/pressure/stress.

J. M. Bloom
Babcock & Wilcox
April 30, 1992

IPEC00014543

```
Listing of file J-APPLIED.lst

      PROGRAM FAD, VERSION 4A,  JUL 30, 1990
       7/26/1991
      PRESSURIZED CYLINDER
      AXIAL PART THRU FLAW


      LISTING OF INPUT DATA :

      PRESSURIZED CYLINDER
      AXIAL PART THRU FLAW

      CASE NUMBER =  2

      INPUT/OUTPUT IN CUSTOMARY USA UNITS
            n (RAMBERG-OSGOOD CONST) ........          8.6000
            ALPHA (RAMBERG-OSGOOD CONST.) ...          1.4000
            POISSON RATIO ...................          0.3000
            E (YOUNGS MODULUS) ..........MPSI         30.0000
            YS (YIELD STRENGTH) ..........KSI         60.0000
            US (ULTIMATE STRENGTH) .......KSI         75.0000
            A (CRACK DEPTH) ...........INCHES          2.2500
            L (CRACK LENGTH) ..........INCHES         13.5000
            T (THICKNESS) .............INCHES          9.0000
            RI (INSIDE RADIUS) ........INCHES         90.0000
            H1 (CALIBRATION FUNCTION) .......          6.8753
            P (PRESSURE) .................KSI          1.0000

      INPUT KIC
              VALUE =          1.0000(KSI-IN^.5)

         PLANE STRAIN                      FORMULATION.

            F A D CURVE COORDINATES

                 SR          KR

               0.0500      1.000
               0.1000      0.999
               0.1500      0.998
               0.2000      0.996
               0.2500      0.993
               0.3000      0.991
               0.3500      0.988
               0.4000      0.984
               0.4500      0.979
               0.5000      0.974
               0.5500      0.966
               0.6000      0.955
```


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0.6500	0.938
0.7000	0.915
0.7500	0.883
0.8000	0.840
0.8500	0.787
0.9000	0.725
0.9500	0.657
1.0000	0.587
1.0500	0.519
1.1000	0.456
1.1500	0.398
1.2000	0.348

FAILURE ASSESSMENT POINTS

P (PRESSURE) ....KSI 1.00

DELTA-A (IN)	KIC (KSI-IN^.5)	SR'	KR'	S.F.
0.0000	1.000	0.1609	30.5237	0.0328

IPEC00014545

APPENDIX C

Response to NRC Request for Additional Information on WCAP-13587 Rev. 0

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION ON WCAP-13587, "REACTOR VESSEL UPPER SHELF ENERGY BOUNDING EVALUATION FOR WESTINGHOUSE PRESSURIZED WATER REACTORS"

(1) **Executive Summary, Page ii - The summary states that a total of 43 vessels were included in the evaluation. The letter from D.J. Modeen (NUMARC) to J.E. Richardson dated January 21, 1993 lists only 42 vessels covered by the WOG. Which is the correct number?**

The correct number of vessels is 43.

(2) **Criteria Synopsis, Page 1-2 - For level A and B conditions it is stated, "If the base meterial is governing, the postulated flaw must be axially oriented". The ASME Code Case states ".... for the base metal, postulate both interior axial and circumferential flaws ...." Although the axial flaw case should be limiting, were circumferential flaws considered as required by the Code Case?**

$J_{applied}$ and $dJ_{applied}/da$ were calculated for both axial and circumferential flaws as required by the Code Case. For each case given in Table 3-2 of WCAP-13587, the applied J parameters were greater for axial flaws than for circumferential flaws (see Attachment B). Consequently these values were used as a limiting case.

(3) **2.1 Background, Page 2-1 - The text states that ".... plants 5,14, 15 and 17 have upper shelf energy values of less than 50 ft-lbs during service life." Table 2-1 shows that the EOL USE values for plant 17 are all above 50 ft-lbs while 49 ft-lbs is shown for plant 16. Are the values tabulated correctly?**

This is a typographical error in the text of WCAP-13587. Section 2.1 Background, Page 2-1, the text should state that "....plants 5, 14, 15 and 16 have upper shelf energy values of less than 50 ft-lbs during service life." This correction will be made in the next revision of this document.

4) **2.3 and 2.4 Mechanical Properties and Stress-Strain Curve, Page 2-3 - Minimum mechanical properties for RPV materials at 600°F were used as per the ASME Code. The intent was to bound all of the participating plants. However, no mechanical property data was provided for the individual plants to determine if this is a "bounding" approach. Provide values of yield strength, ultimate strength and ductility for the materials for each vessel.**

The ASME Boiler and Pressure Vessel Code, Section III - Division I Appendices provides lower bound mechanical properties to be used in the design of reactor vessels. The Reactor Vessel equipment specification requires that all vessel materials meet this requirement. Since the purpose of this analysis is to provide a bounding evaluation, use of the ASME properties is judged to be appropriate.

IPEC00014546

(5) **2.5 J-R Curves, Page 2-4 - The text states that the NUREG/CR-5729 correlations are "material independent." This statement is not entirely true as A302 Gr B data was specifically excluded from database used to develop the correlations. Hence, the correlations are not meant to apply to A302 Gr B.**

As described in Section 2.5.1, Representative Values, of WCAP-13587, the J-R values based on the model described in NUREG/CR-5729 were directly utilized for all materials other than ASTM-A302B. This material was considered to be unique; therefore, actual ASTM-A302B data given in NUREG/CR-5265 was considered appropriate in determining $J_{material}$ and $dJ_{material}/da$ for this material. The material of NUREG/CR-5265 was generated specifically for testing and deliberately had an initial upper shelf energy value of 50 ft-lbs. The assumption that an operating plant with a reactor vessel constructed from ASTM-A302B would have such a low initial USE is unrealistic and considered to be extremely conservative for use in this analysis. As not much J-R data for ASTM-A302B with low USE values is available, this data was considered to be bounding. The only use of the NUREG/CR-5729 model for ASTM-A302B was to estimate a decrease in $J_{material}$ due to a drop in USE for carbon steel materials for bounding the end of life condition.

(6) **2.5 J-R Curves, Page 2-5 - The text states that the J-R curves were developed from the NUREG/CR-5729 correlations using a temperature of 390.5°F which "represents the greatest temperature at the crack tip for a 1/4t flaw" for level A and B conditions. As this results in a very large temperature differential across the vessel wall and is non-conservative for the J-R curve correlation, additional justification for use of this temperature is required. Why are the J-R curves not determined using 600° F as was done with the mechanical properties?**

A two-dimensional finite element model was constructed for typical reactor vessel beltline geometries. The 100°F/hour cooldown rate transient representing the Level A and B Condition loading was applied to this model. Based on the results of this analysis, the temperature at a location 25% of the way through the wall for the most severe stress distribution was determined to be 390.5°F. The use of this calculated temperature at the crack tip is considered to produce both realistic and representative results.

Although it was recognized that the required safety margins were incorporated into the analysis in other areas (factors on accumulation pressure, etc) to ensure confidence in the results of the analysis, an additional factor of safety was incorporated by using minimum mechanical properties at 600°F as opposed to 390.5°F. The equations used in WCAP-13587 to calculate $J_{applied}$ for Level A and B conditions that include mechanical properties are:

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

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

Based on Reference[6] of WCAP-13587, the pertinent mechanical properties at 600°F and 400°F (400°F = 390.5°F) for WOG reactor vessel materials, with the exception of SA-302, Grade B* are as follows:

	σ_y (ksi)	E (Mpsi)
600°F	43.8	26.4
400°F	45.1	27.4

IPEC00014547

Use of mechanical properties at 400°F as opposed to 600°F in the equations listed above would result in a smaller value of J and consequently a larger margin between J_{app} and J_{mat}. Consequently, use of 600°F in determining minimum mechanical properties is conservative.

(7) **2.5 J-R curves, Page 2-5 - The text describes a procedure for adjusting the A302 Gr B J-R Curve data [Ref. A] for EOL using a variation of the NUREG/CR-5729 methodology. However, as stated previously, this methodology was not meant to apply to A302 Gr.B. Also the $J_{material}$ values determined were not conservative representations (mean - two sigma) and were not temperature adjusted (the Ref. A data was for 180°F). Procedures for adjusting the A302 Gr B data are described in References B, and C.**

Use of the A302-B J_R curve given in Reference A, NUREG/CR-5729 is valid because the temperatures of 180°F and 390°F are both on the upper shelf temperature range (100 percent shear). Reference C states that J_R curves generally exhibits lower J levels (at a constant crack growth level) as the temperature increases. A difference in crack growth resistance between 390°F and 550°F is likely due to strain aging and consequently lower J levels at 550°F. Begley reported similar results in WCAP-13554, "Effects of Section Size and Cleanliness On the Upper Shelf and Transition Range Toughness of Three Nuclear Pressure Vessel Steels". However, the J level would only decrease slightly, if any decrease at all from 180°F to 390°F.

The correlation for J with Charpy impact energy given in Reference C (Figure 3) may have been meaningful for the evaluation of the Yankee Rowe vessel but it is meaningless for our bounding analysis. For example, the lower bound J_R value for a 6T-CT specimen at 52 ft-lbs. is approximately 750 in-lbs/in^2. EPRI NP-4224, September 1985, gives a value of 2800 in lbs/in^2 at 350°F after a fluence of 4 x 10^{19} n/cm^2 for H.B. Robinson A302-B material and 4150 in-lbs/in^2 for Connecticut Yankee A302-B material after a fluence of 2.22 x 10^{19} n/cm^2. The transverse Charpy values of these two heats of A302 are not known at this time, however, the longitudinal oriented specimens exhibited Charpy impact energy values of 102.5 and 106.5 ft-lbs. Further, for a heat of A302-B material with an upper shelf impact energy of 36 ft-lbs, unirradiated and 32 ft-lbs. irradiated, Westinghouse obtained 1550, 1850, and 1250 in lbs/in^2 at 350°F in the unirradiated condition and 1450, 1400, and 1150 in lbs/in^2 at 350°F for the irradiation condition (~2 x 10^{19} n/cm^2).

The data given in Reference A, NUREG/CR-5729, is not representative of commercial nuclear power reactor vessels because of the degree of straight rolling the J_R curve can be used as a lower bound cruve for A302-B material. The data in Reference A shows that the J_R curves are even lower than a 32 ft-lb A302-B material.

(8) **3.2 Analysis for Levels A and B, Page 3-3 - Show calculations resulting in applied J values in table 3-2.**

The calculations resulting in the Table 3-2 applied J values are contained in Attachment A of this letter.

IPEC00014548

(9) **3.3 Analysis for Levels C and D, Pages 3-4 and 3-5 - The text cites that an "assessment was conducted to determine the limiting level C and D transients." Show the assessment. Also, show the pressure and temperature histories for all of the significant transients and the calculations for the PCFAD analyses.**

Attachment B of this letter contains the Level C/D transient assessment and Attachment C describes the PCFAD calculations.

IPEC00014549

Attachment A
Level A and B Calculations

The Level A and B calculations have been performed in accordance with the methodology given in Section 3.2 of WCAP-13587. The Plant 5 Case input parameters are utilized in performing a sample calculation typical of all cases. This calculation uses a technique described in the draft version of ASME Code Case E512. As the methodology of Code Case E512 has changed slightly, another sample calculation is shown using the current code case methodology. Revision 1 of WCAP 13587 will contain the methodology described in Code Case E512 that is current at the time of publication of the WCAP revision (minor adjustments are currently under consideration).

Nomenclature

p	Pressure
R_i	Vessel Inner Radius
t	Vessel Thickness
SF	Safety Factor
CR	Cooldown Rate
σ_y	Yield Strength
E	Modulus of Elasticity
E_c	Corrected Modulus of Elasticity (E' in WCAP 13587)
ν	Poisson's Ratio
K_{Ipa}	Stress Intensity Due to Internal Pressure for Axial Flaw
K_{Ipc}	Stress Intensity Due to Internal Pressure for Circumferential Flaw
K_{It}	Stress Intensity Due to Temperature
a_0	Flaw Depth for Quarter Thickness Flaw with 0.1" Flaw Extension
a_{effa}	Flaw Depth for Quarter Thickness Axial Flaw with 0.1" Flaw Extension Corrected for Plastic Zone Size
a_{effc}	Flaw Depth for Quarter Thickness Circumferential Flaw with 0.1" Flaw Extension Corrected for Plastic Zone Size
J_a	Applied J for Axial Flaw
J_c	Applied J for Circumferential Flaw
a_{effao}	Flaw Depth for Quarter Thickness Axial Flaw with 0.1" Flaw Extension Corrected for Plastic Zone Size Using a Safety Factor of 1.25 on Pressure - Used in Calculating dJ/da
a_{effco}	Flaw Depth for Quarter Thickness Circumferential Flaw with 0.1" Flaw Extension Corrected for Plastic Zone Size Using a Safety Factor of 1.25 on Pressure - Used in Calculating dJ/da
J_{ao}	Applied J for Axial Flaw at 0.1" Crack Extension Using a Safety Factor of 1.25 on Pressure - Used in Calculating dJ/da
J_{co}	Applied J for Circumferential Flaw at 0.1" Crack Extension Using a Safety Factor of 1.25 on Pressure - Used in Calculating dJ/da
a_{effad}	Flaw Depth for Quarter Thickness Axial Flaw with No Flaw Extension Corrected for Plastic Zone Size Using a Safety Factor of 1.25 on Pressure - Used in Calculating dJ/da
a_{effcd}	Flaw Depth for Quarter Thickness Circumferential Flaw with No Flaw Extension Corrected for Plastic Zone Size Using a Safety Factor of 1.25 on Pressure - Used in Calculating dJ/da
J_{ad}	Applied J for Axial Flaw with No Crack Extension Using a Safety Factor of 1.25 on Pressure - Used in Calculating dJ/da
J_{cd}	Applied J for Circumferential Flaw with No Crack Extension Using a Safety Factor of 1.25 on Pressure - Used in Calculating dJ/da

IPEC00014550

Plant 5 (Rev 0, Plant 7 in Rev 1)- Draft Code Case E512 Methodology

$p := 2.485 \quad R_i := 77.97 \quad CR := 100$

$t := 9.875 \quad SF := 1.15 \cdot 1.1 \quad s := \frac{1.25}{1.15}$ (safety factor adjustment for dJ/da)

$\sigma_y := 43.8 \quad E := 2.64 \cdot 10^4 \quad \nu := 0.3 \quad E_c := \frac{E}{(1-\nu^2)}$

$$F_1(a) := 0.982 + 1.006 \cdot \left(\frac{a}{t}\right)^2 \qquad K_{Ipa}(a) := SF \cdot p \cdot \left[1 + \left(\frac{R_i}{t}\right)\right] \cdot \sqrt{\pi \cdot a} \cdot F_1(a)$$

$$F_2(a) := 0.885 + 0.233 \cdot \left(\frac{a}{t}\right) + 0.345 \cdot \left(\frac{a}{t}\right)^2 \qquad K_{Ipc}(a) := SF \cdot p \cdot \left(1 + \frac{R_i}{2 \cdot t}\right) \cdot \sqrt{\pi \cdot a} \cdot F_2(a)$$

$$F_3(a) := \left[\left[0.584 + 2.647 \cdot \left(\frac{a}{t}\right)\right] - 6.294 \cdot \left(\frac{a}{t}\right)^2\right] + 2.990 \cdot \left(\frac{a}{t}\right)^3$$

$$K_{It}(a) := \left(\frac{CR}{1000}\right) \cdot t^{2.5} \cdot F_3(a) \qquad a_o := \frac{t}{4} + 0.1 \qquad a_d := \frac{t}{4}$$

$$a_{effa} := (a_o) + \left(\frac{1}{6 \cdot \pi}\right) \cdot \left(\frac{K_{Ipa}(a_o) + K_{It}(a_o)}{\sigma_y}\right)^2 \qquad a_{effc} := (a_o) + \left(\frac{1}{6 \cdot \pi}\right) \cdot \left(\frac{K_{Ipc}(a_o) + K_{It}(a_o)}{\sigma_y}\right)^2$$

$$J_a := \frac{1000 \cdot \left(K_{Ipa}(a_{effa}) + K_{It}(a_{effa})\right)^2}{E_c} \qquad J_c := \frac{1000 \cdot \left(K_{Ipc}(a_{effc}) + K_{It}(a_{effc})\right)^2}{E_c}$$

$$a_{effao} := (a_o) + \left(\frac{1}{6 \cdot \pi}\right) \cdot \left(\frac{K_{Ipa}(a_o) \cdot s + K_{It}(a_o)}{\sigma_y}\right)^2 \qquad a_{effco} := (a_o) + \left(\frac{1}{6 \cdot \pi}\right) \cdot \left(\frac{K_{Ipc}(a_o) \cdot s + K_{It}(a_o)}{\sigma_y}\right)^2$$

$$J_{ao} := \frac{1000 \cdot \left(K_{Ipa}(a_{effao}) \cdot s + K_{It}(a_{effao})\right)^2}{E_c} \qquad J_{co} := \frac{1000 \cdot \left(K_{Ipc}(a_{effco}) \cdot s + K_{It}(a_{effco})\right)^2}{E_c}$$

$$a_{effad} := (a_d) + \left(\frac{1}{6 \cdot \pi}\right) \cdot \left(\frac{K_{Ipa}(a_d) \cdot s + K_{It}(a_d)}{\sigma_y}\right)^2 \qquad a_{effcd} := (a_d) + \left(\frac{1}{6 \cdot \pi}\right) \cdot \left(\frac{K_{Ipc}(a_d) \cdot s + K_{It}(a_d)}{\sigma_y}\right)^2$$

$$J_{ad} := \frac{1000 \cdot \left(K_{Ipa}(a_{effad}) \cdot s + K_{It}(a_{effad})\right)^2}{E_c} \qquad J_{cd} := \frac{1000 \cdot \left(K_{Ipc}(a_{effcd}) \cdot s + K_{It}(a_{effcd})\right)^2}{E_c}$$

$$dJ_a := J_{ao} - J_{ad} \qquad da := a_o - a_d \qquad dJ_c := J_{co} - J_{cd}$$

Results:

$J_a = 478.304 \qquad J_c = 177.341$

$\frac{dJ_a}{da} = 213.14 \qquad \frac{dJ_c}{da} = 57.949$

IPEC00014551

Plant 5 (Rev 0, Plant 7 in Rev 1) - Current Code Case E512 Methodology

$p := 2.485 \quad R_i := 77.97 \quad CR := 100$

$t := 9.875 \quad SF := 1.15 \cdot 1.1 \qquad s := \frac{1.25}{1.15}$ (safety factor adjustment for dJ/da)

$\sigma_y := 43.8 \quad E := 2.64 \cdot 10^4 \quad \nu := 0.3 \qquad E_c := \frac{E}{\left(1-\nu^2\right)}$

$F_1(a) := 0.982 + 1.006 \cdot \left(\frac{a}{t}\right)^2 \qquad K_{Ipa}(a) := SF \cdot p \cdot \left[1 + \left(\frac{R_i}{t}\right)\right] \cdot \sqrt{\pi \cdot a} \cdot F_1(a)$

$F_2(a) := 0.885 + 0.233 \cdot \left(\frac{a}{t}\right) + 0.345 \cdot \left(\frac{a}{t}\right)^2 \qquad K_{Ipc}(a) := SF \cdot p \cdot \left(1 + \frac{R_i}{2 \cdot t}\right) \cdot \sqrt{\pi \cdot a} \cdot F_2(a)$

$F_3(a) := \left[\left[0.690 + 3.127 \cdot \left(\frac{a}{t}\right)\right] - 7.435 \cdot \left(\frac{a}{t}\right)^2\right] + 3.532 \cdot \left(\frac{a}{t}\right)^3$

$K_{It}(a) := \left(\frac{CR}{1000}\right) \cdot t^{2.5} \cdot F_3(a) \qquad a_o := \frac{t}{4} + 0.1 \qquad a_d := \frac{t}{4}$

$a_{effa} := \left(a_o\right) + \left(\frac{1}{6 \cdot \pi}\right) \cdot \left(\frac{K_{Ipa}\left(a_o\right) + K_{It}\left(a_o\right)}{\sigma_y}\right)^2 \qquad a_{effc} := \left(a_o\right) + \left(\frac{1}{6 \cdot \pi}\right) \cdot \left(\frac{K_{Ipc}\left(a_o\right) + K_{It}\left(a_o\right)}{\sigma_y}\right)^2$

$J_a := \frac{1000 \cdot \left(K_{Ipa}\left(a_{effa}\right) + K_{It}\left(a_{effa}\right)\right)^2}{E_c} \qquad J_c := \frac{1000 \cdot \left(K_{Ipc}\left(a_{effc}\right) + K_{It}\left(a_{effc}\right)\right)^2}{E_c}$

$a_{effao} := \left(a_o\right) + \left(\frac{1}{6 \cdot \pi}\right) \cdot \left(\frac{K_{Ipa}\left(a_o\right) \cdot s + K_{It}\left(a_o\right)}{\sigma_y}\right)^2 \qquad a_{effco} := \left(a_o\right) + \left(\frac{1}{6 \cdot \pi}\right) \cdot \left(\frac{K_{Ipc}\left(a_o\right) \cdot s + K_{It}\left(a_o\right)}{\sigma_y}\right)^2$

$J_{ao} := \frac{1000 \cdot \left(K_{Ipa}\left(a_{effao}\right) \cdot s + K_{It}\left(a_{effao}\right)\right)^2}{E_c} \qquad J_{co} := \frac{1000 \cdot \left(K_{Ipc}\left(a_{effco}\right) \cdot s + K_{It}\left(a_{effco}\right)\right)^2}{E_c}$

$a_{effad} := \left(a_d\right) + \left(\frac{1}{6 \cdot \pi}\right) \cdot \left(\frac{K_{Ipa}\left(a_d\right) \cdot s + K_{It}\left(a_d\right)}{\sigma_y}\right)^2 \qquad a_{effcd} := \left(a_d\right) + \left(\frac{1}{6 \cdot \pi}\right) \cdot \left(\frac{K_{Ipc}\left(a_d\right) \cdot s + K_{It}\left(a_d\right)}{\sigma_y}\right)^2$

$J_{ad} := \frac{1000 \cdot \left(K_{Ipa}\left(a_{effad}\right) \cdot s + K_{It}\left(a_{effad}\right)\right)^2}{E_c} \qquad J_{cd} := \frac{1000 \cdot \left(K_{Ipc}\left(a_{effcd}\right) \cdot s + K_{It}\left(a_{effcd}\right)\right)^2}{E_c}$

$dJ_a := J_{ao} - J_{ad} \qquad da := a_o - a_d \qquad dJ_c := J_{co} - J_{cd}$

Results:

$J_a = 524.734 \qquad J_c = 203.926$

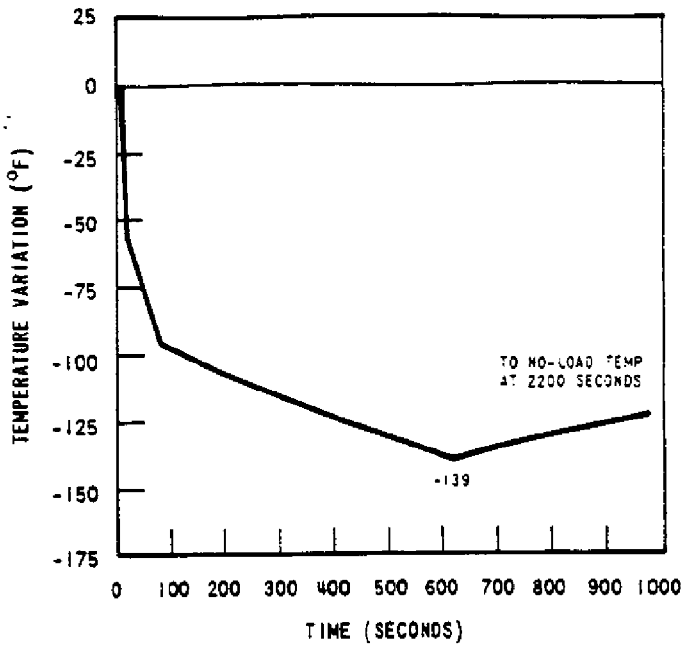
$\frac{dJ_a}{da} = 221.783 \qquad \frac{dJ_c}{da} = 61.421$

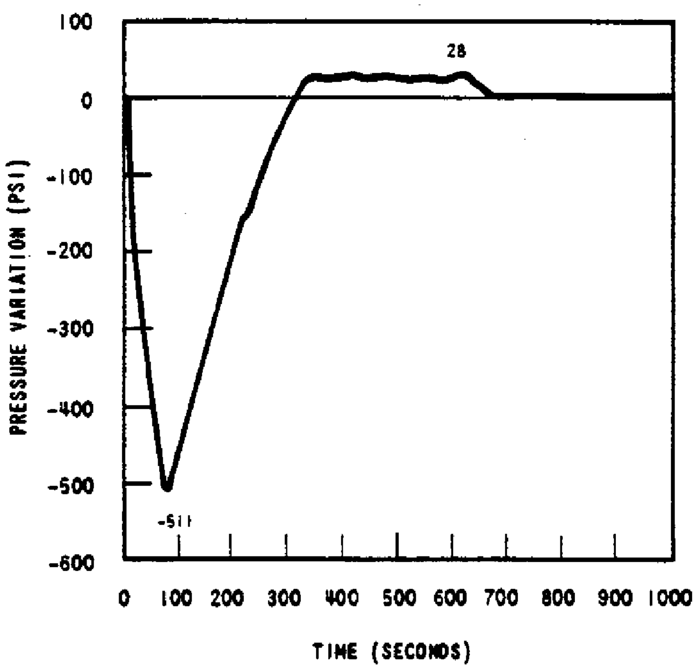
IPEC00014552

Attachment B
Transient Assessment

The relative magnitude of the through wall stresses based on transient loading is a function of the pressure and rate of change of temperature during the transient. All the transients listed in section 3.3.1 were compared on the basis of pressure and change in temperature per unit time. The temperature variation and pressure variation histories of these transients are attached. Given that larger pressures excursions and temperature change rates cause larger through wall stresses, by inspection of the attached transients a conclusion regarding the limiting transient based on engineering judgement may be reached. If a detailed stress analysis were to be performed, the small steam line break would be the limiting Level C transient and the Large LOCA and steam line break would be the limiting level D transients. Based on the through-wall stress distribution for the Level D limiting transients given in Figure 3-5 of WCAP-13587, the large steam line break has been chosen to represent the Level D condition. It created greater stresses at the crack tip for a 1" flaw.

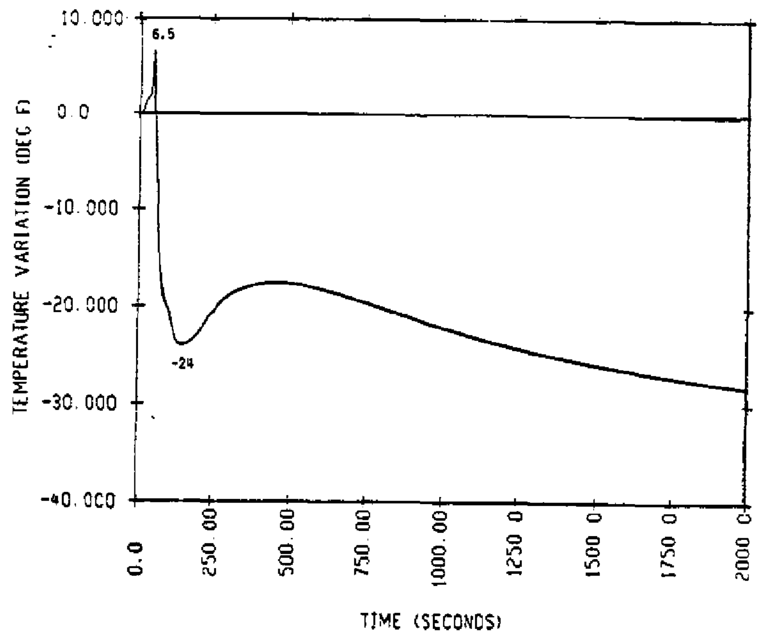
IPEC00014553

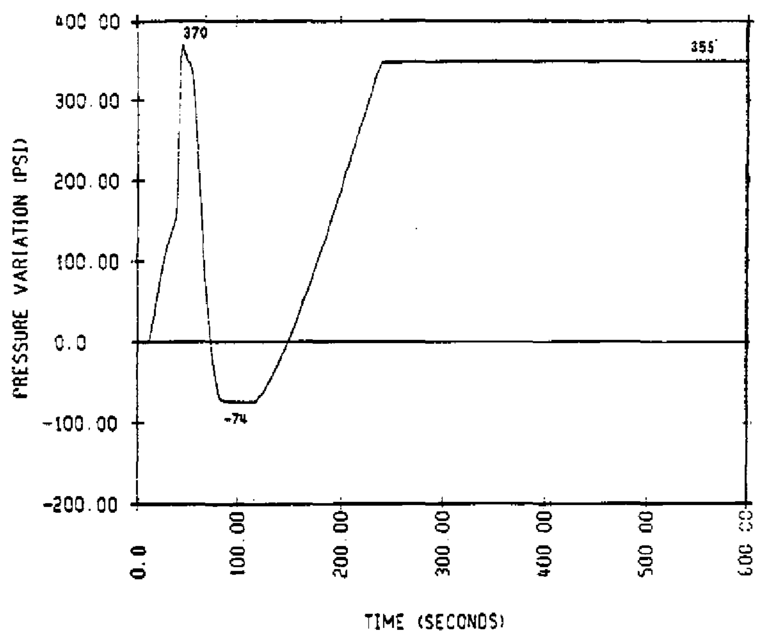




Temperature and Pressure History for Steam Generator Tube Rupture Transient (Bounded by Reactor Trip with Cooldown and Safety Injection)

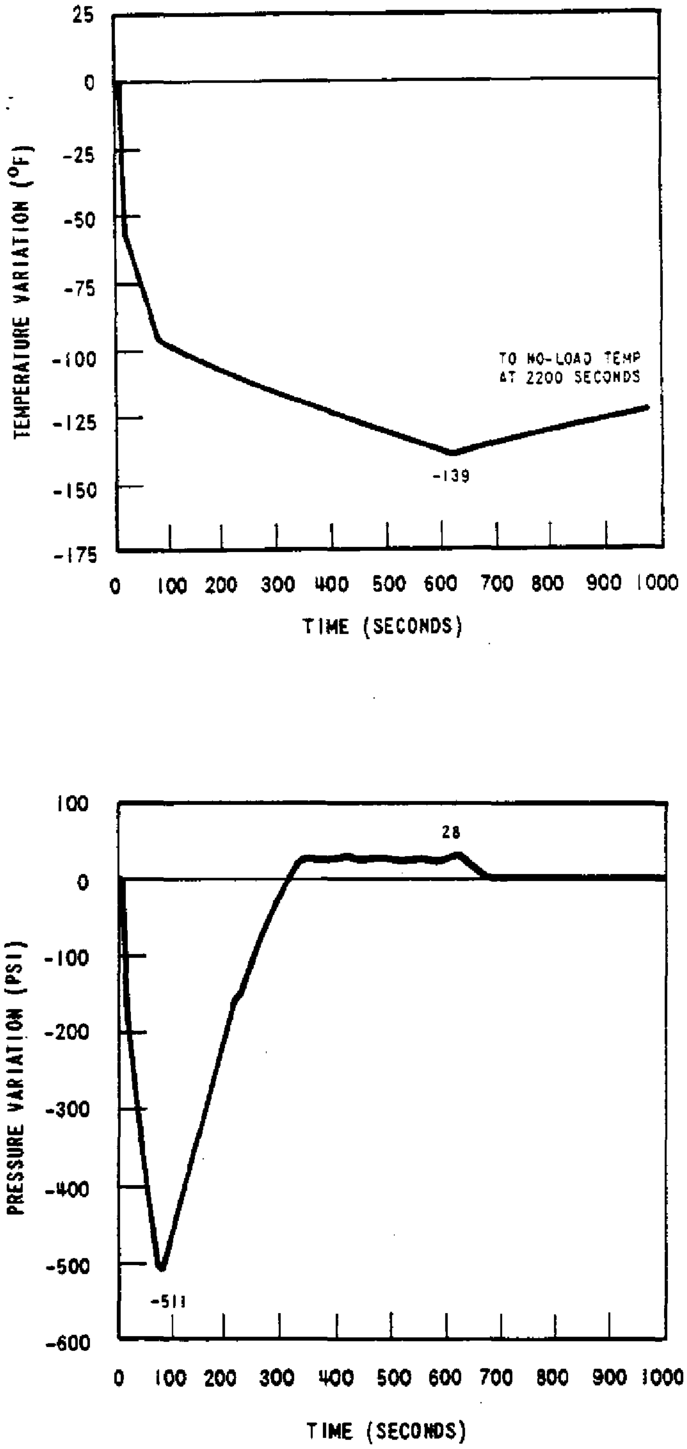
IPEC00014554





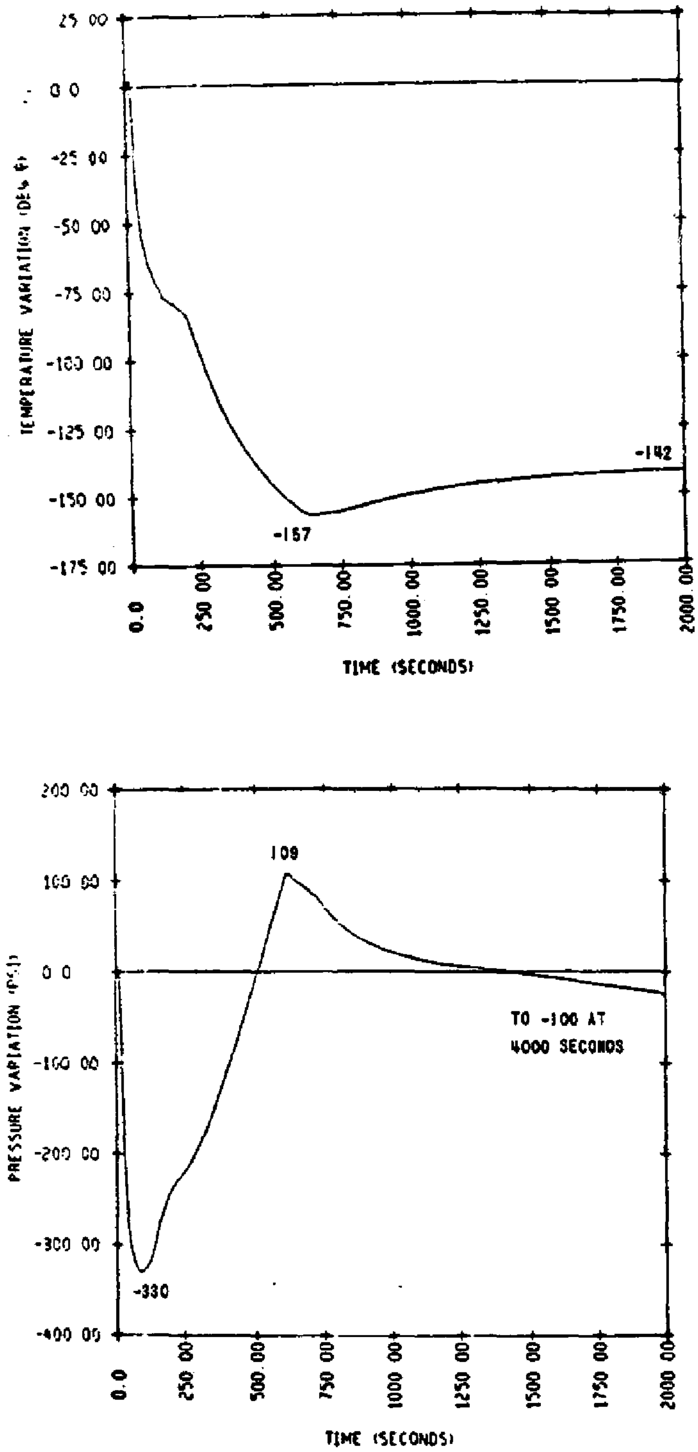
Temperature and Pressure History for Feedwater Line Break Transient

IPEC00014555



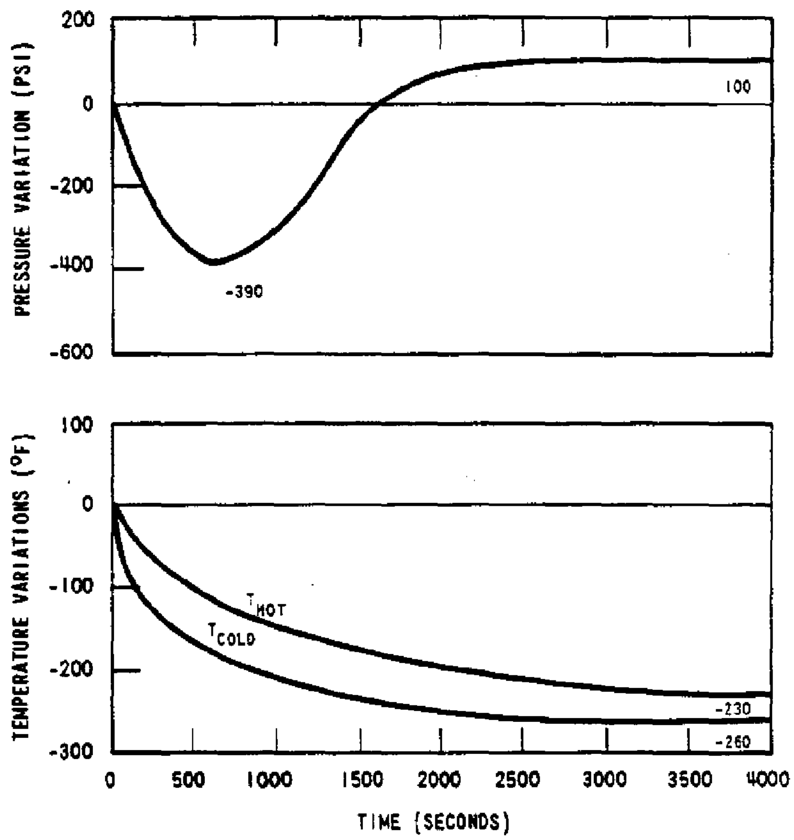
Temperature and Pressure History for Reactor Trip with Cooldown and SI Transient

IPEC00014556



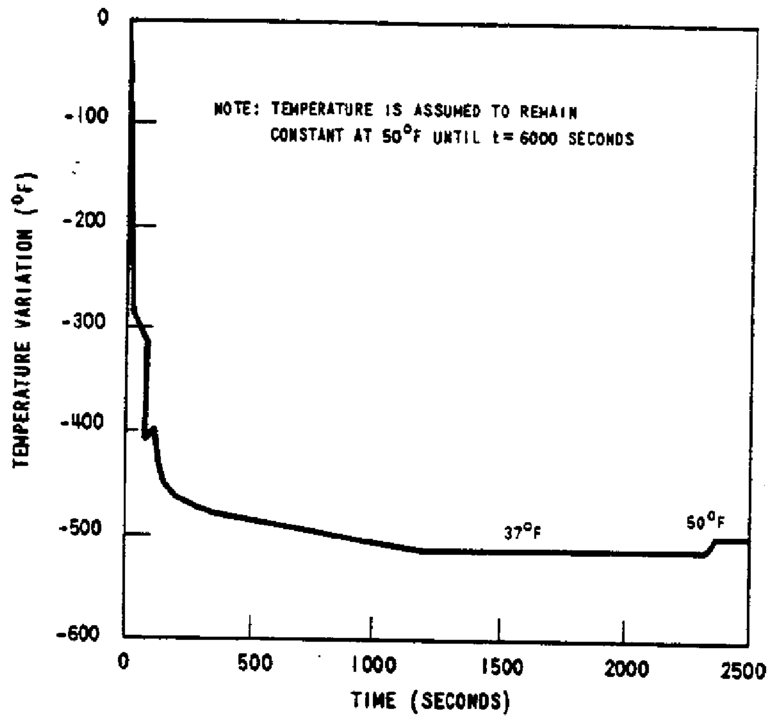
Temperature and Pressure History for Large Steam Line Break Transient

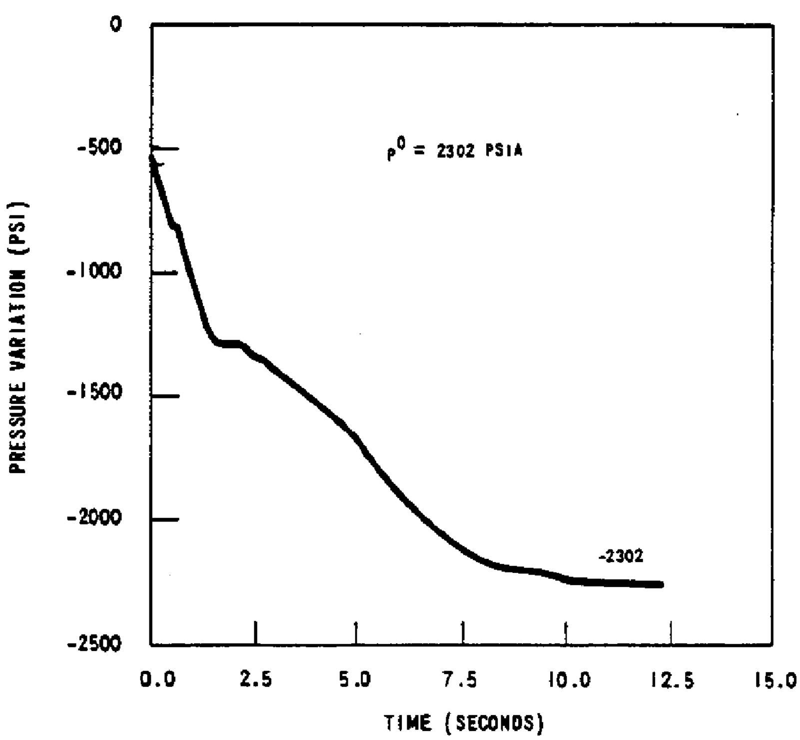
IPEC00014557



Temperature and Pressure History for Small Steam Line Break Transient

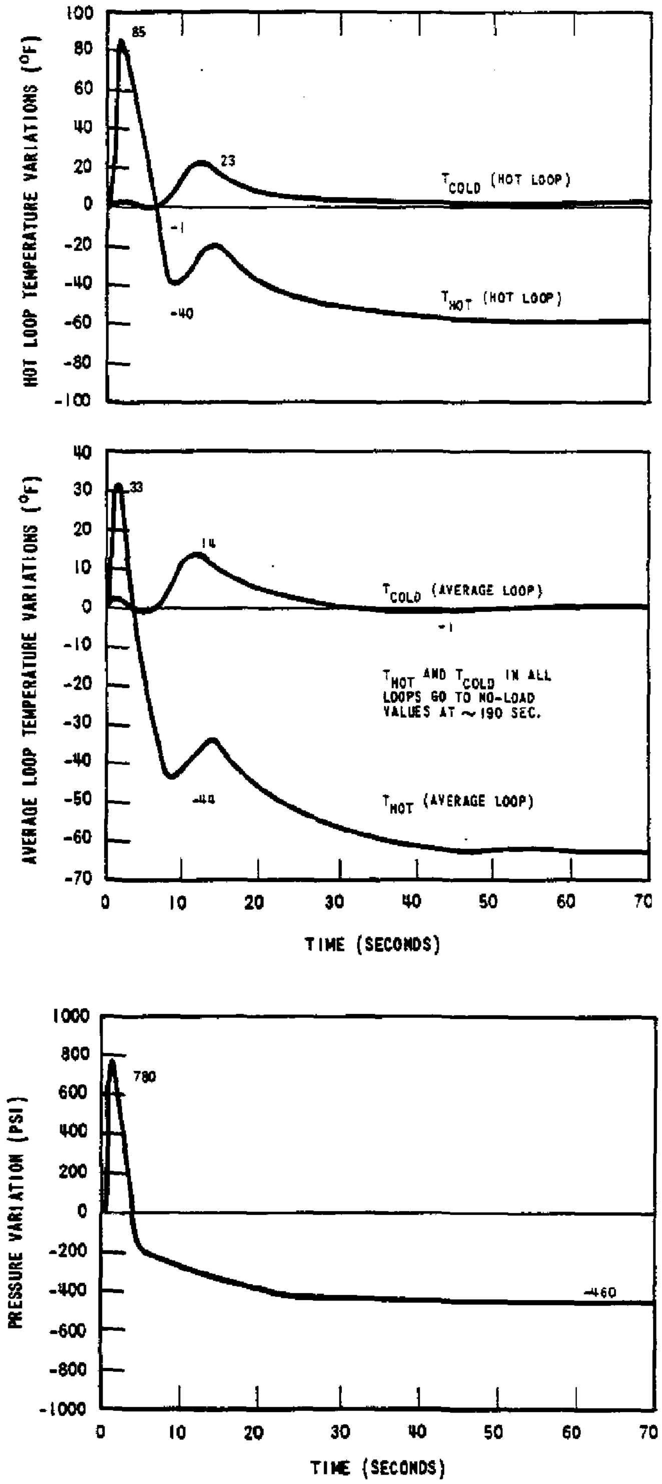
IPEC00014558





Temperature and Pressure History for Large Loss of Coolant Accident Transient

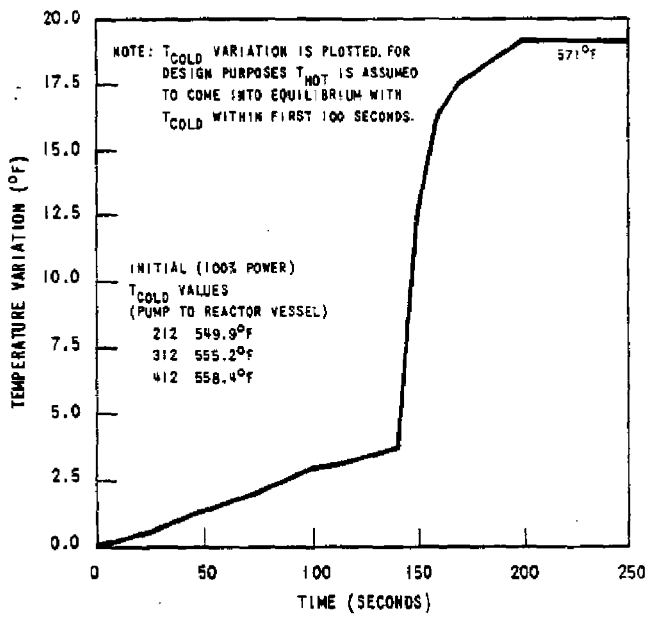
IPEC00014559

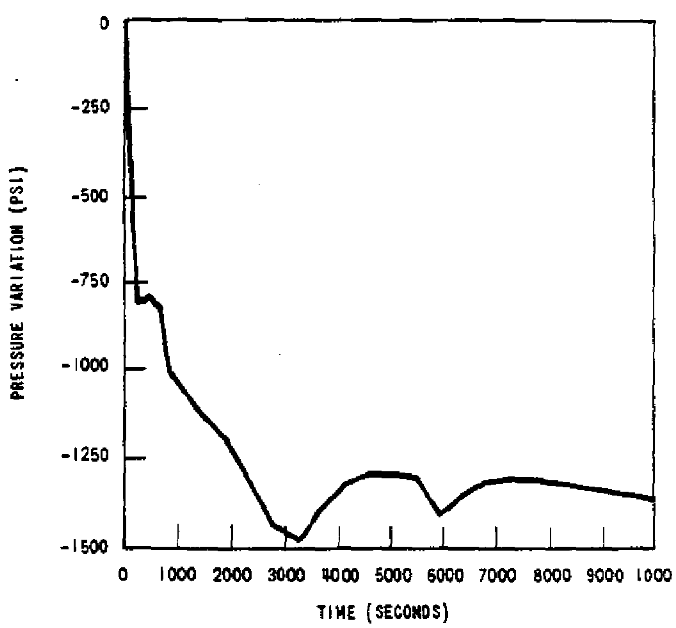


Temperature and Pressure History for Control Rod Ejection Transient

IPEC00014560

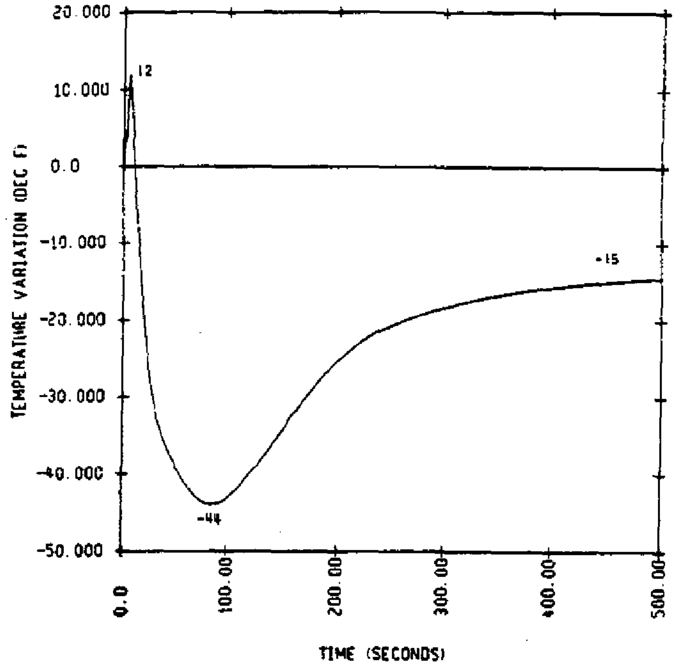
IPEC00014560

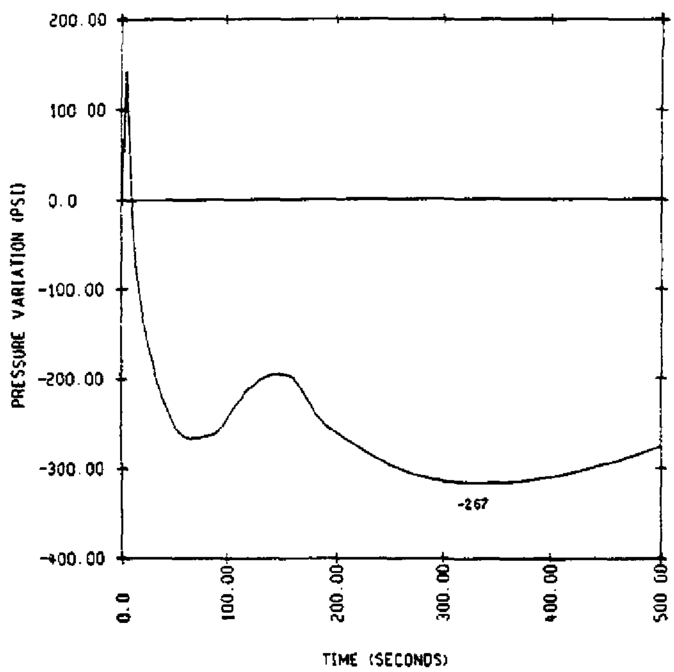




Temperature and Pressure History for Small Loss-of-Coolant Transient

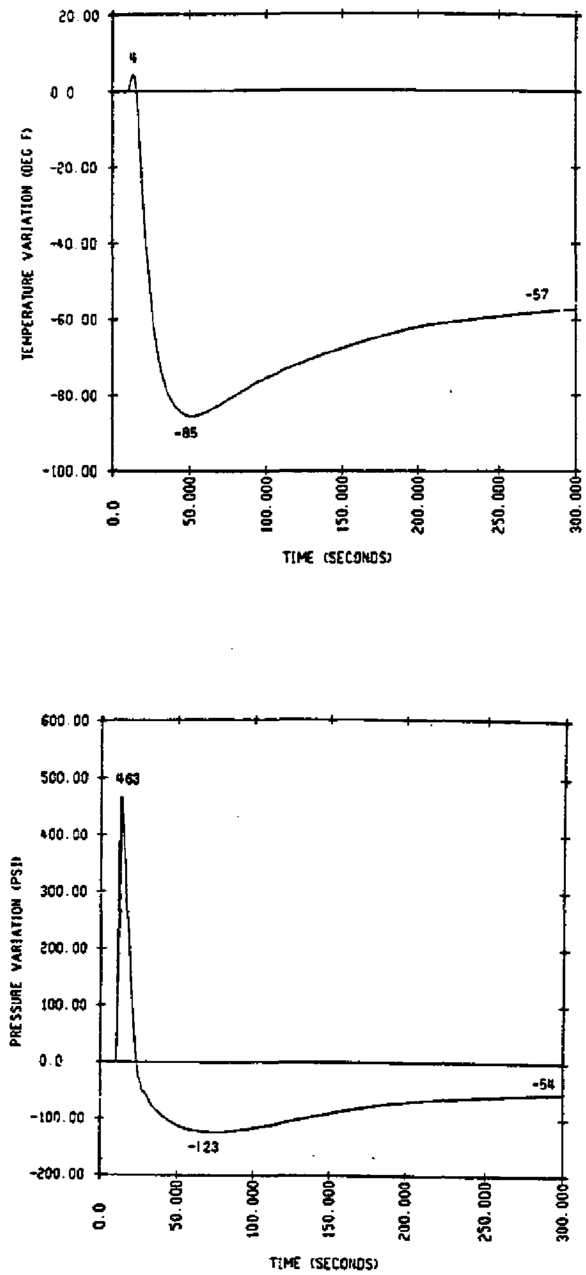
IPEC00014561





Temperature and Pressure History for Complete Loss of Flow Transient

IPEC00014562



Temperature and Pressure History for Reactor Coolant Locked Rotor Transient

IPEC00014563

Attachment C
PCFAD Calculations

The Level C/D analysis methodology is based the Failure Assessment Diagram technique as implemented in the PCFAD program and analysis technique (Reference 14 of WCAP-13587). The procedure for obtaining $J_{applied}$ from PCFAD output is attached. This procedure has been developed by J.M. Bloom of Babcock and Wilcox. A typical PCFAD output is also attached as an example. This sample represents the PCFAD output from the Plant 5 Case of WCAP-13587.

IPEC00014564

Procedure for Obtaining J-Applied from PCFAD

First the dimensions of the particular flawed geometry (structure) are input into PCFAD along with the relevant material properties, Ramberg-Osgood constants, α,n, Poisson's ratio, ν, Young's modulus, E, and yield strength, σ_{ys}. Since a Ramberg-Osgood material is discussed, the ultimate strength, σ_{ult}, can be set arbitrarily (it is suggested that a value of σ_{us} of at least 1.2 times σ_{ys} be used). Remember the program, PCFAD, calculates α,n based on the inputs of σ_{ys} and σ_{us}. Therefore, these values must be replaced with actual α,n values from a fit of the actual stress-strain tensile data used in the finite element analysis.

Next, the load, stress, or pressure is set equal to unity and toughness is input as K_{Ic} and set to unity as well.

The attached output file "J-APPLIED" is given in USA customary units for a semi-elliptical interior axial flaw in a pressurized vessel with inside radius of 90 inches. The flaw is a quarter of the 9-inch wall and has an aspect ratio, length to depth of 6:1. Note that the output first lists the coordinates of the FAD curve SR, KR followed by the assessment points SR', KR'.

The limit load, P_L, is obtained from the inverse of SR' (P_L = P/0.1609) while the stress intensity factor, K_I, is KR' times the load, stress, or pressure in units of ksi (inches)$^{0.5}$ (K_I = 30.5237P).

For the example chosen, the J-APPLIED value in in-lb/in^2 units for a pressure of 1 ksi is obtained as follows:

At P = 1 ksi, SR = SR' = 0.161

At that value of SR, the corresponding KR value is approximately 0.997 (taken from the table of SR, KR values). The stress intensity factor is 30.5237 ksi (inch)$^{0.5}$ which corresponds to a $J_{elastic}$ = 28.3 in-lb/in^2 from the relationship

$$\frac{1000\ K_I^2\ (1-\nu^2)}{E} = J_{elastic} \tag{1}$$

where E is of the units of KSI.

J-APPLIED is then taken from the expression

$$J = J_{elastic}/\,(KR)^2 \tag{2}$$

For the example chosen,

$$J = 28.3/(.997)^2 = 28.4\ \text{in-lb/in}^2$$

For P = 2.5 KSI, SR = 2.5 x .161 = .4025 and KR = .984 from the SR, KR table. The stress intensity factor = 2.5 x 30.5237 = 76.31 KSI (inch)$^{0.5}$. The corresponding $J_{elastic}$ value is 176.63 in-lb/in^2 using expression (1) and J-APPLIED equals 176.63/(.984)^2 = 182.4 in-lb/in^2.

IPEC00014565

The resulting table of J-APPLIED values is giver comparison of results found from a finite element ar Wikening ASME PVP Vol. 106, August 1984.

Pressure (KSI)	J-APPLIED (PCFAD) (in-lb/in²)	J-APPLIED (FEA) (in-lb/in²)
2.5	182	186
4.5	708	738
6.0	2,484	2,638
7.5	14,600	13,466

The differences in the table are due primarily to interpolation of the output of PCFAD. While values of SR are calculated per .005 interval, the program only prints values every .05 interval.

The key expressions for converting output from PCFAD into J-APPLIED values are (1) and (2). Also, remember that SR, KR, SR', KR' are all linear in load/pressure/stress.

J. M. BLOOM
BABCOCK & WILCOX
APRIL 30, 1992

IPEC00014566

Listing of file J-APPLIED.lst

PROGRAM FAD, VERSION 4A, JUL 30, 1990
7/26/1991
PRESSURIZED CYLINDER
AXIAL PART THRU FLAW

LISTING OF INPUT DATA :

PRESSURIZED CYLINDER
AXIAL PART THRU FLAW

CASE NUMBER = 2

INPUT/OUTPUT IN CUSTOMARY USA UNITS

n (RAMBERG-OSGOOD CONSTANT).....	8.6000
ALPHA (RAMBERG-OSGOOD CONST.)...	1.4000
POISSON RATIO...................	0.3000
E (YOUNGS MODULUS)...........MPSI	30.0000
YS (YIELD STRENGTH)..........KSI	60.0000
US (ULTIMATE STRENGTH).......KSI	75.0000
A (CRACK DEPTH)...........INCHES	2.2500
L (CRACK LENGTH)..........INCHES	13.5000
T (THICKNESS).............INCHES	9.0000
RI (INSIDE RADIUS)........INCHES	90.0000
H1 (CALIBRATION FUNCTION).......	6.8753
P (PRESSURE).................KSI	1.0000

INPUT KIC
VALUE = 1.0000(KSI-IN^.5)

PLANE STRAIN FORMULATION.

F A D CURVE COORDINDATES

SR	KR
0.0500	1.000
0.1000	0.999
0.1500	0.998
0.2000	0.996
0.2500	0.993
0.3000	0.991
0.3500	0.988
0.4000	0.984
0.4500	0.979
0.5000	0.974
0.5500	0.966
0.6000	0.955
0.6500	0.938
0.7000	0.915
0.7500	0.883
0.8000	0.840
0.8500	0.787
0.9000	0.725
0.9500	0.657
1.0000	0.587
1.0500	0.519
1.1000	0.456
1.1500	0.398
1.2000	0.348

IPEC00014567

FAILURE ASSESSMENT POINTS

P (PRESSURE)................KSI 1.00

DELTA-A (IN)	KIC (KSI-IN^.5)	SR'	KR'	S.F.
0.0000	1.000	0.1609	30.5237	0.0328

IPEC00014568

W3G4D

A14

Sn 1-15-93

GJ 2/19/93

PROGRAM FAD, VERSION 4A, JUL 30, 1990
2/17/1993

FINAL RUN

LISTING OF INPUT DATA :

FINAL RUN

CASE NUMBER = 16

INPUT/OUTPUT IN CUSTOMARY USA UNITS

n (RAMBERG-OSGOOD CONSTANT)..... 9.5606
ALPHA (RAMBERG-OSGOOD CONST.)... 0.0842
POISSON RATIO.................... 0.3000
E (YOUNGS MODULUS)..........MPSI 26.4000
YS (YIELD STRENGTH)..........KSI 32.7500
US (ULTIMATE STRENGTH).......KSI 80.0000
A (CRACK DEPTH)...........INCHES 1.0000
B (HALF CRACK LENGTH).....INCHES 3.0000
T (THICKNESS).............INCHES 9.8750
RI (INSIDE RADIUS)........INCHES 77.9690
GAMMA.....................DEGREES 2.2046
H1 (CALIBRATION FUNCTION)....... 11.1691
P (PRESSURE).................KSI 1.0000

INPUT KIC
VALUE = 1.0000(KSI-IN^.5)

PLANE STRAIN FORMULATION.

F A D CURVE COORDINDATES

SR	KR
0.0500	1.000
0.1000	0.999
0.1500	0.998
0.2000	0.997
0.2500	0.995
0.3000	0.993
0.3500	0.990
0.4000	0.988
0.4500	0.985
0.5000	0.982
0.5500	0.979
0.6000	0.976

IPEC00014569

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0.6500	0.972
0.7000	0.967
0.7500	0.960
0.8000	0.952
0.8500	0.940
0.9000	0.923
0.9500	0.901
1.0000	0.871
1.0500	0.833
1.1000	0.787
1.1500	0.734
1.2000	0.675
1.2500	0.614
1.3000	0.553
1.3500	0.494
1.4000	0.439
1.4500	0.389
1.5000	0.344
1.5500	0.304
1.6000	0.268
1.6500	0.237
1.7000	0.210
1.7500	0.187
1.8000	0.166
1.8500	0.148
1.9000	0.133
1.9500	0.119
2.0000	0.107
2.0500	0.096
2.1000	0.087
2.1500	0.079
2.2000	0.071
2.2500	0.065
2.3000	0.059
2.3500	0.054
2.4000	0.049

FAILURE ASSESSMENT POINTS

P (PRESSURE)..................KSI 1.00

DELTA-A (IN)	KIC (KSI-IN^.5)	SR'	KR'	S.F.
0.0000	1.000	0.1513	9.8144	0.1019

IPEC00014570