UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 3 1990

MEMORANDUM FOR: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

FROM: Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

SUBJECT: GENERIC ISSUE 79, "UNANALYZED REACTOR VESSEL (PWR) THERMAL STRESS DURING NATURAL CONVECTION COOLDOWN"

Enclosed for your information and possible review is a draft of a memorandum to the EDO describing the proposed resolution of the subject generic issue. Attached to that memorandum is a draft of a memorandum to Thomas Murley on the same subject. Although the proposed resolution does not recommend any action on the part of licensees, we do make recommendations to NRR in case a plant should experience a natural convection cooldown (NCC) event which may place the reactor vessel (RV) in an unanalyzed condition. The recommendations apply to PWRs of the type analyzed which experience an NCC event that exceeds the cooldown limits of the reactor vessel analyzed, or to other PWRs that experience an NCC event which may place the RV outside of its design basis.

The ACRS agreed with the staff's proposed resolution but recommended that we notify licensees of the resolution of GI-79 and the possibility of an NRC request for confirmatory information from any licensee whose plant experiences a significant NCC event. In response to this recommendation, we prepared a draft of an information generic letter (Enclosure 3 to the Beckjord to Murley memorandum).

There has been one NCC event involving appreciable cooldown (St. Lucie 1 in 1980). Based on the amount of actual PWR operation at power, the event frequency is about 2E-3/reactor-yr.

We estimate that the cost to a licensee to perform an analysis would be about $200,000. This is based on the cost of the work performed by BNL to evaluate the reactor vessel stresses plus the work performed by the NRC staff to evaluate fracture toughness.

We do not believe the proposed generic letter necessitates CRGR review since it does not require licensee action or a response. However, we would be happy to provide a presentation to the CRGR if they wish.

Please advise us within two weeks as to whether or not the CRGR wishes to review the proposed resolution of GI-79 and the proposed information generic letter.

Eric S. Beckjord

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

Enclosure: As stated

cc: C. Heltemes
W. Minners
T. King
R. Baer
F. Cherny
J. Page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

SUBJECT: RESOLUTION OF GENERIC ISSUE 79, UNANALYZED REACTOR VESSEL (PWR) THERMAL STRESS DURING NATURAL CONVECTION COOLDOWN

The purpose of this memorandum is to formally document the resolution of the referenced generic issue.

The concern addressed under Generic Issue 79 (GI-79) was identified by the Babcock & Wilcox Co. (B&W) in 1983 as a result of its investigation into the 1980 St. Lucie Natural Convection (or Circulation) Cooldown (NCC) event.* Based on preliminary calculations B&W identified a concern that thermal stresses, beyond those considered in the original design of PWR reactor vessels (RVs), may develop in RV flanges and studs due to large axial temperature gradients across the RV closure region (i.e., potentially placing PWR RVs in a condition outside its design basis).

In July 1983 a priority ranking of "medium" was approved for GI-79.

Subsequently B&W performed a detailed stress evaluation of the closure region of their 177 Fuel Assembly (FA) RV for the NCC condition. This was submitted to the NRC staff for review as part of the GI-79 resolution process. The 177 FA RV is utilized on all operating B&W reactors. At the request of the RES staff, Brookhaven National Laboratory (BNL) reviewed the B&W analysis and performed an independent confirmatory stress analysis of the B&W 177 FA RV closure region.

Based on the results of the BNL review and analyses and a staff evaluation of the adequacy of the RV closure region fracture toughness for the NCC condition, RES has concluded that the B&W 177 FA RV closure region meets all currently applicable regulatory design criteria. RES has additionally concluded that, except for the issuance of an information generic letter (discussed below), no immediate generic or plant-specific actions are necessary for the following reasons:

1. NCC events are low in frequency of occurrence.

2. Based on the staff's evaluation of the B&W 177 FA RV, it is extremely unlikely that a single NCC transient would cause the failure of a PWR vessel.

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*St. Lucie is a Combustion Engineering designed reactor. Babcock & Wilcox postulated a similar event for reactors they designed.

3. The specific details of any actually-experienced NCC transient will determine the need for, if any, and the extent of actions required of a specific licensee to assure the adequacy of its RV for continued service.

Since a NCC transient involves rat' omplicated thermal hydraulics, and calculated stresses for B&W RV stu re near ASME Section III Code allowable stresses, and, since the stresses .he RV closure regior increase as the RV cooling rate increases, RES is making the following recommendation to NRR in the enclosed memorandum. For reportable NCC transients which may place the RV in a condition that is outside of its design basis (in accordance with 10 CFR 50.73(a)(2)(ii)(B)), the affected licensee should provide confirmation that no applicable regulatory design or fracture toughness criteria have been exceeded for the RVs listed below, which were not specifically evaluated in the resolution of this generic issue.

1. B&W 177 FA RVs that experience a NCC event involving a cooldown rate greater than 100°F/hr.

2. Westinghouse, Combustion Engineering and B&W non-177 FA RVs which experience any reportable NCC event.

Discussion of the staff and BNL evaluations are contained in Enclosures 1 and 2 to the attached memorandum to NRR.

On August 10, 1989, the RES staff presented its proposed resolution of GI-79, as described above, to the ACRS. The ACRS agreed with the resolution but recommended that the NRC advise utilities of this recommendation via formal correspondence. In response to this ACRS concern, a draft information generic letter is provided in Enclosure 3 of the attached memorandum.

Generic Issue 79 is thus considered resolved.

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

Attachment:
Memorandum for Thomas Murley from
Eric Beckjord, dated

cc: w/enclosure:
T. Murley, NRR
J. Richardson, NRR
F. Gillespie, NRR
L. Marsh, NRR
W. Minners, RES
T. King, RES
L. Shao, RES
R. Baer, RES
F. Cherny, RES
J. Page, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

MEMORANDUM FOR: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

FROM: Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

SUBJECT: RESOLUTION OF GENERIC ISSUE 79, "UNANALYZED REACTOR VESSEL (PWR) THERMAL STRESS DURING NATURAL CONVECTION COOLDOWN"

The purpose of this memorandum is to advise you of the resolution of the subject generic issue, to clearly describe the limits of the natural convection (or circulation) cooldown (NCC) analysis that was performed, and to provide RES recommendations for actions if an actual PWR NCC should occur that (1) exceeds the cooldown limits of the reactor vessel analyzed, or (2) represents an unanalyzed case. It should be noted that only the B&W 177 fuel assembly reactor vessel (177 FA RV) was analyzed. This vessel is utilized on all operating B&W reactors.

The concern addressed under Generic Issue 79 (GI-79) was identified by the Babcock & Wilcox Co. (B&W) (Reference 1) in 1983 as a result of its investigation into the 1980 St. Lucie NCC event.* Based on preliminary calculations B&W identified a concern that thermal stresses, beyond those considered in the original design of PWR reactor vessels (RVs), may develop in RV flanges and studs due to large axial temperature gradients across the RV closure region.

Subsequently B&W performed a detailed stress evaluation of the closure region of their 177 FA RV for the NCC condition (Reference 5). This was submitted to the NRC staff for review as part of the GI-79 resolution process. At the request of the RES staff Brookhaven National Laboratory (BNL) reviewed the B&W analysis and performed an independent confirmatory stress analysis of the B&W 177 FA RV closure region utilizing a maximum cooldown rate of 100°F/hr.

Based on the results of the BNL review and analyses (Reference 11) and a staff evaluation of the adequacy of the RV closure region fracture toughness for the NCC condition, RES has concluded that the B&W 177 FA RV closure region meets all currently applicable regulatory design criteria for the NCC conditions analyzed. (See Enclosures 1 and 2 which provide the details of the staff and BNL evaluations.)

*St. Lucie is a Combustion Engineering designed reactor. Babcock & Wilcox postulated a similar event for reactors they designed.

RES has additionally concluded that, except for the issuance of an information generic letter (discussed below), no immediate generic or plant-specific actions are necessary for the following reasons:

1. NCC events are low in frequency of occurrence.

2. Based on the staff's evaluation of the B&W 177 FA RV, it is extremely unlikely that a single NCC transient would cause the failure of a PWR vessel.

3. The specific details of any actually-experienced NCC transient will determine the need for, if any, and the extent of actions required of a specific licensee to assure the adequacy of its RV for continued service.

However, since a NCC transient involves rather complicated thermal hydraulics, and calculated stresses for B&W RV studs were near ASME Section III Code allowable stresses, and, since the stresses in the RV closure region increase as the RV cooling rate increases, RES recommends the following: For reportable NCC transients which may result in a condition that places the RV outside of its design basis (in accordance with 10 CFR 50.73(a)(2)(ii)(B)) the affected licensee should provide confirmation that no applicable regulatory design or fracture toughness criteria have been exceeded for the RVs listed below, which were not specifically evaluated in the resolution of this generic issue.

1. B&W 177 FA RVs that experience a NCC event involving a cooldown rate greater than 100°F/hr.

2. Westinghouse, Combustion Engineering and B&W non-177 FA RVs which experience any reportable NCC event.

The effects of NCC cycles on plant life extension were not considered by B&W or the staff as part of the resolution of this generic issue.

On August 10, 1989, the RES staff presented its proposed resolution of GI-79, as described above, to the ACRS. The ACRS approved of the resolution (Reference 12) but recommended that the NRC advise utilities of this recommendation via formal correspondence. In response to this ACRS concern, a draft information generic letter is provided in Enclosure 3.

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

Enclosures: As stated

cc: w/enclosures:
J. Taylor
J. Richardson, NRR
C. Cheng, NRR
F. Gillespie, NRR
L. Marsh, NRR
J. Bradfute, NRR

REFERENCES

1. Letter for R. C. DeYoung (NRC) from J. H. Taylor (B&W), "Unanalyzed Reactor Vessel Thermal Stress During Cooldown," March 18, 1983.

2. Memorandum for W. Minners (NRC) from R. J. Bosnak (NRC), "B&W Notification Concerning an Unanalyzed Reactor Vessel Thermal Stress During Cooldown," April 26, 1983.

3. Memorandum for R. H. Vollmer (NRC) from H. R. Denton (NRC), "Schedule for Resolving and Completing Generic Issue No. 79 - "Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown," July 25, 1983.

4. Memorandum for H. R. Denton (NRC) from R. B. Minogue (NRC), "Comments on Generic Issue 79, Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown," October 5, 1983.

5. Letter for N. P. Kadambi (NRC) from F. R. Miller (BWOG), "Transmittal of RV head Stress Evaluation Program Results," October 15, 1984.

6. Letter for W. S. Wilgus (BWOG) from D. M. Crutchfield (NRC), "Request for Meeting With the B&W Owners Group Regarding Reactor Vessel Thermal Stresses During Natural Convection Cooldown - Generic Issue No. 79," October 2, 1987.

7. Letter for J. D. Page (NRC) from J. R. Paljug (B&W/BWOG), April 8, 1988.

8. Memorandum for R. L. Baer (NRC) from J. D. Page (NRC), "Minutes of Meeting With B&W Owners Group Regarding Generic Issue 79 - Unanalyzed Reactor Vessel Thermal Stresses During Natural Convection Cooldown," May 6, 1988.

9. Letter for J. D. Page (NRC) from J. R. Paljug (B&W/BWOG), June 23, 1988.

10. Letter for J. D. Page (NRC) from J. R. Paljug (B&W/BWOG), October 26, 1988.

11. Review of Unanalyzed Reactor Vessel Thermal Stress for the B&W FA-177 Reactor Vessel, Y. S. Chung, J. A. Pires and M. Reich (BNL), May 1989.

12. Memorandum for J. M. Taylor (NRC) from F. J. Remick (ACRS), "Proposed Resolution of Generic Issue 79, Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown," August 15, 1989.

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Enclosure 1

Generic Issue-79
Stress and Fatigue Evaluation
of B&W 177 Fuel Assembly
Reactor Vessel Closure Region
for NCC Condition

The concern addressed under Generic Issue 79 (GI-79) was identified by B&W (Reference 1) in 1983 as a result of its investigation into the 1980 St. Lucie Natural Convection (or Circulation) Cooldown (NCC) event.* Based on preliminary calculations, B&W identified a concern that thermal stresses, beyond those considered in the original design of PWR vessels, may develop in the reactor vessel (RV) flanges and studs due to large axial temperature gradients across the RV closure region, i.e., a condition that was potentially outside the design basis of the PWR RVs. Initially, B&W stated that these thermal stresses could occur as a result of two different transients, (1) non-uniform cooling (coolant stagnation in the head) of the reactor coolant during a NCC or (2) after the reactor coolant pumps are secured in the normal reactor cooldown mode (i.e., the transition to Decay Heat Removal (DHR) operation).

In January 1984, the B&W Owners Group (BWOG) initiated a program to perform a detailed evaluation of the stresses induced in the 177 Fuel Assembly (FA) RV closure region for these transients; and in October 1984, a report (Reference 5) documenting the evaluation results was submitted to the NRC. The 177 FA RV is utilized in all B&W operating reactors. The NRC and its contractor (BNL) began a review of this report and BNL began to perform a conservative confirmatory stress analysis of the 177 FA RV closure region for the vessel conditions discussed herein.

In accordance with 10 CFR 50.55a, the NRC has endorsed the design criteria for Reactor Pressure Vessels as specified in Section III of the ASME Boiler and Pressure Vessel Code (hereinafter, "the Code"). The B&W and BNL evaluations are based on these criteria.

Based on the initial review, numerous questions were transmitted to BWOG by Reference 6, and a meeting between the NRC staff, its contractor, and BWOG was requested. BWOG provided draft responses to the questions (Reference 7) prior to the meeting, which was held on April 25, 1988. The results of the meeting were documented by Reference 8 and BWOG provided draft and final responses by References 9 and 10, respectively. In these responses BWOG informed the staff that the concern originally stated with respect to the normal reactor cooldown mode had been incorrectly stated. The corrected response stated that the thermal stresses of concern could occur during a NCC, which includes the subsequent transition to DHR system operation. Based on this information, the staff and BNL continued to evaluate the 177 FA vessel only for the effects of the NCC condition; that is, transient (1) described above.

*St. Lucie is a Combustion Engineering designed reactor. Babcock & Wilcox postulated a similar event for reactors they designed.

Numerous conservatisms were incorporated in the BNL analysis (Reference 12); such as utilizing a cooldown rate of 100°F/hr, maintaining the RV head fluid temperature at 600°F for the entire NCC transient (i.e., no thermal mixing between the fluid in the RV head and the fluid in the lower portion of the RV), and either a frictionless flange interface (Case 1) or an infinite friction flange interface (Case 2). As a result of these conservatisms the resultant stresses were higher than those calculated by B&W; however, all calculated stresses were less than the applicable Code allowable values. Therefore, the RES staff concludes that adequate design margins do exist in the closure region of B&W 177 FA RVs for the NCC condition analyzed.

As a result of the review conducted under GI-79, the staff has concluded that exposure of B&W 177 FA RVs to a NCC transient outside of the conditions bounded by the BNL confirmatory analysis may result in a condition that is outside the design basis of the RV. A maximum cooling rate of 100°F/hr was utilized by the RES staff and BNL for the GI-79 evaluations. B&W advised the RES staff that this cooldown rate is achievable in some B&W operating plants under NCC conditions. An NCC transient involves rather complicated thermal hydraulics, and the BNL calculated stresses in the RV studs were particularly high (98% of allowable membrane plus bending stress). Also, the stresses in the RV closure region increase as the RV cooling rate increases.

The RES staff did not specifically evaluate non-B&W 177 FA RVs. Therefore, the staff has further concluded that exposure of certain other PWR RVs to a NCC transient may result in a condition that is outside the design basis of the RV. Westinghouse (W) and Combustion Engineering (C-E) RVs can be significantly different from B&W designs, and may specific RVs are different from each other according to vintage, number of loops, vessel manufacturer, etc.

As stated above, certain NCC transients may result in conditions that are outside the design basis of PWR RVs. Therefore, RES recommends that NRR require any PWR licensee whose plant experiences a reportable NCC transient which may place the RV in an unanalyzed condition (Ref. 10 CFR 50.73(a)(2)(ii)(B)) provide confirmation that no applicable regulatory design stress criteria have been exceeded. This recommendation is limited to RVs and conditions listed below which were not specifically evaluated in the resolution of this generic issue.

1. B&W 177 FA RVs that experience a NCC event involving a cooldown rate greater than 100°F/hr.

2. Westinghouse, Combustion Engineering and B&W non-177 FA RVs which experience any reportable NCC event.

The BNL confirmatory analysis also evaluated the effects of NCC cycles on the B&W 177 FA RV closure region with respect to fatigue effects. Section III of the ASME Code requires that the total cumulative usage factor (CUF) for the RV, including the closure flanges and studs, not exceed a value of 1.0. The analysis determined that 40 NCC cycles will contribute approximately 10% of the total CUF allowed by the ASME Code for the RV studs and less than 2% to the total CUF allowed for the RV closure flanges. For the B&W 177 FA RV closure region, even with the addition of these NCC fatigue effects, the Code specified CUF is not exceeded.

Document Name:
FRACTURE TOUGHNESS EVALUATION

Requestor's ID:
BEVAN

Author's Name:
FCherny

Document Comments:
GI 79

Generic Issue-79
10CFR50 appendix G
Fracture Toughness Evaluation
of B&W 177 Fuel Assembly
Reactor Vessel Closure Region
for NCC Condition

In evaluating the fracture toughness adequacy of the B&W 177 Fuel Assembly Reactor Vessel (B&W 177 FA RV) the RES staff concluded that the RV closure region, with the exception of the closure studs and the nozzle shell course, remained at a sufficiently high temperature throughout the NCC transient to avoid brittle fracture.

For the closure studs and the nozzle shell course which can be exposed to somewhat lower temperatures, the staff performed individual evaluations. The evaluation criteria used were from Appendix G of 10 CFR 50.55a, Sections III and XI of the ASME Boiler and Pressure Vessel Code as referenced in 10 CFR 50.55a, and SRP Sections 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing," and SRP 5.3.2, "Pressure-Temperature Limits."

These staff evaluations are contained in Appendices 1 and 2 of this enclosure.

CONCLUSIONS

Compliance with the criteria utilized in the analyses discussed herein provides assurance that B&W 177 FA RVs can be safely shutdown under the conservative NCC conditions analyzed with no adverse effect on public health and safety. The staff has thus concluded that applicable regulatory requirements for RV fracture toughness are satisfied for the B&W 177 FA RV closure region for the NCC condition analyzed.

In addition, although B&W non-177 FA, Westinghouse (W), and Combustion Engineering (C-E) RVs can be significantly different dimensionally from B&W 177 FA RVs, based on the results of its analysis, the RES staff has concluded that there is a high degree of assurance that all PWR RVs can be safely exposed to at least one NCC cycle of the type analyzed and can be safely shut down with no adverse effect on the health and safety of the public.

However, as a result of the review conducted under Generic Issue 79, the staff has also concluded that exposure of PWR RVs to certain NCC transients may result in a condition that is outside the RV design basis. Therefore, RES is making the following recommendation to the Office of Nuclear Reactor Regulation (NRR). For any PWR that experiences a reportable NCC transient (in accordance with 10 CFR 50.73(a)(2)(ii)(B)), the affected licensee should confirm that no applicable regulatory fracture toughness criteria have been exceeded for the RV. This recommendation is applicable to PWR RVs and conditions listed below which were not specifically evaluated in the resolution of this generic issue.

1. B&W 177 FA RVs that experience a NCC event involving a cooldown rate greater than 100°F/hr.

2. Westinghouse, Combustion Engineering and B&W non-177 FA RVs which experience any reportable NCC event.

Reactor Vessel Shell Fracture Mechanics Evaluation

Richard E. Johnson

Introduction and Background

For a B&W 177 FA nuclear reactor pressure vessel (RPV), consider a natural circulation cooldown event from a linear-elastic fracture mechanics viewpoint using procedures recommended in the ASME Boiler and Pressure Vessel Code ("the Code").

Temperatures and stresses were calculated as functions of time into the event by the Brookhaven National Laboratory (BNL) technical staff. The relatively high temperatures at the start of the transient placed the RPV steel outside of the range of applicability of the Code procedures; therefore, the condition could be dismissed by inspection, even though the stresses at some elements were higher than those used in the following calculations. By inspection, the worst combination of parameters occurred in the RPV shell just below the closure flange (Fig. 1, element numbers 307 through 311). The stresses and temperatures listed in Table I were used in the fracture mechanics analysis.

Stresses at the finite element centroids for the locked flange interface (infinite friction) case were plotted against position across the twelve-inch shell thickness in Fig. 2. Extrapolation to the inside and outside surfaces resulted in ID and OD stresses of 17 and 6.6 ksi, respectively. A technically correct method of handling the non-linear stress distribution would be by factoring it into a uniform stress, plus a linearly distributed stress tangent to the actual distribution at the crack tip (see the dashed line on Fig. 2), plus the residual curvilinear stress. For this analysis, a less complicated and more conservative method was used. The ID and OD stresses were connected by an imaginary straight line, then that distribution was treated as if it were the result of a tensile ("membrane," in Code language) plus a linear (or "pure") bending stress. The components of stress distribution were found to be:

membrane stress = σ_m = 11.8 ksi;

bending stress = σ_b = ± 5.2 ksi.

Additional parameters used in the calculations were determined as follows.

Being generic, the analyses do not pertain to a specific RPV; therefore, the reference temperature for the material's nil-ductility transition (RT_{NDT}) is unknown. Following the guidance in the NRC Standard Review Plan Section 5.3.2, a value of 60°F was assumed.

Although of relatively small importance to the results, when the tensile yield strength, σ_{ys}, was needed, a value of about 55 ksi was used.

Whenever flaw parameters had to be selected, the following conditions applied. The crack plane was oriented normal to the maximum principal stress (i.e., an axial crack dictated by the circumferential stress), the crack front was sharp (e.g., as the leading edge of a fatigue crack), the location was at the surface carrying the higher tensile stress, and the shape was semi-elliptical with a 6-to-1 (total length to depth) aspect ratio. Postulated defects were assumed to have a depth equal to one-fourth of the RPV shell thickness as recommended in Section III of the Code.

Critical Crack Depth, a_c, Calculation

Following the procedure given in Appendix A, Section XI, of the Code, according to Article A-3300:

$$K_I = \sigma_m M_m \sqrt{\pi a/Q} + \sigma_b M_b \sqrt{\pi a/Q},$$

where: M_m can be found from Fig. 3, which is a copy of Fig. A-3300-3;

M_b can be found from Fig. 4, which is a copy of Fig. A-3300-5;

Q can be found from Fig. 5, which is a copy of Fig. A-3300-1.

For the given flaw parameters: $a/l = 0.1667$ and $a/t = 0.25$.

Enter Fig. 3 at $a/t = 0.25$, interpolate to $a/l = 0.1667$ and read out $M_m = 1.186$.

Enter Fig. 4 at $a/t = 0.25$, interpolate to $a/l = 0.1667$ and read out $M_b = 0.8$.

For the given stresses: $(\sigma_m + \sigma_b)/\sigma_{ys} = (11.8 + 5.2)/55 = 0.31$.

Enter Fig. 5 at $a/l = 0.1667$, interpolate to $(\sigma_m + \sigma_b)/\sigma_{ys} = 0.31$ and read out $Q = 1.212$.

Solve for a, substitute the fracture toughness for K_I and let $a = a_c$:

$$a_c = (Q/\pi)[K_{Ia}/(\sigma_m M_m + \sigma_b M_b)]^2,$$

where the arrest toughness, K_{Ia}, was selected rather than the less conservative initiation toughness, K_{Ic}.

From Table I, the RPV temperature near the ID is about 150°F, therefore, $T - RT_{NDT} = 150 - 60 = 90°F$. Enter Fig. 6, which is a copy of the Code Fig.

A-4200-1, at + 90°F and read out: $K_{Ia} = 72.5$ ksi $\sqrt{\text{in.}}$

Substitute values and calculate a_c:

$$a_c = (1.212/\pi)[72.5/(11.8)(1.186) + (5.2)(0.8)]^2 = 6.15 \text{ in.}$$

Since the critical crack size, based on conservative values of several parameters, is about one-half of the RPV shell thickness (12 in.), or about three times the size of a flaw which might escape detection and about ten times the size of the smallest flaw detectable by UT (about ½ in.), it was concluded that the natural circulation cooldown event is not likely to challenge the integrity of a B&W 177 FA vessel.

ASME Code, Section XI, Analysis

Use the procedure given in Appendix A, Section XI, of the Code for a postulated flaw as prescribed by Appendix G, Section III (Design Bases), in Article G-2120.

$a = t/4$; for $t = 12$ in., $a = 3$ in.

From article A-3300:

$$K_I = \sigma_m M_m \sqrt{\pi a/Q} + \sigma_b M_m \sqrt{\pi a/Q}.$$

Except for crack depth, a, values of all variables were given in the preceeding section. Substitute and solve for K_I:

$$K_I = (11.8)(1.186)\sqrt{\pi 3/1.212} + (5.2)(0.8)\sqrt{\pi 3/1.212} = 50.6 \text{ ksi } \sqrt{\text{in.}},$$

which is significantly less than the previously-determined toughness:

$$K_{Ia} = 72.5 \text{ ksi}\sqrt{\text{in.}}$$

Flaw acceptance criteria for steel components of 4 in. and greater thickness are given in IWB-3610, Section XI, ASME Code. Because the region of the RPV under examination is close to a change in shell thickness (see Fig. 1), the appropriate Code paragraph is IWB-3613, "Acceptance Criteria for Flanges and Shell Regions Near Structural Discontinuities." Furthermore, the pressure after 15 hours into the transient is 350 psi; therefore, the applicable criterion is the one given in IWB-3613(a), for conditions where the pressurization is not more than 20% of the Design Pressure, i.e.:

$$K_I < K_{Ia}/\sqrt{2}$$

From the preceeding calculation, K_{Ia} 72.5 ksi $\sqrt{\text{in.}}$, so $K_{Ia}/\sqrt{2} = 51.3$ ksi $\sqrt{\text{in.}}$ Since the toughness reduced by the safety factor was greater than K_I (= 50.6 ksi$\sqrt{\text{in.}}$), the Code criterion was met. Certainly, if the transient were treated as an emergency or faulted condition where the Code acceptance criterion is:

$$K_I < K_{Ic}/\sqrt{2},$$

the significantly larger (than K_{Ia}) value of K_{Ic}, the initiation toughness, would be enough to pass the test with a wide margin.

It was concluded that the evaluation, based on a conservatively large ¼-t flaw, demonstrated that the subject transient would not induce a failure of the RPV. Noting that a natural.circulation cooldown event is a rare transient, it can be concluded that the analysis showed that it is acceptable by ASME Code criteria.

Table I

Stresses and Temperatures Reported by BNL.

Conditions: (1) flange interface locked
(2) 15 hrs. into the transient

Element No.:	307	308	309	310	311
Hoop Stress, (psi)	15,170	12,760	10,860	9,145	7,455
Temp., °F	149.7	150.5	150.7	146.2	152.6

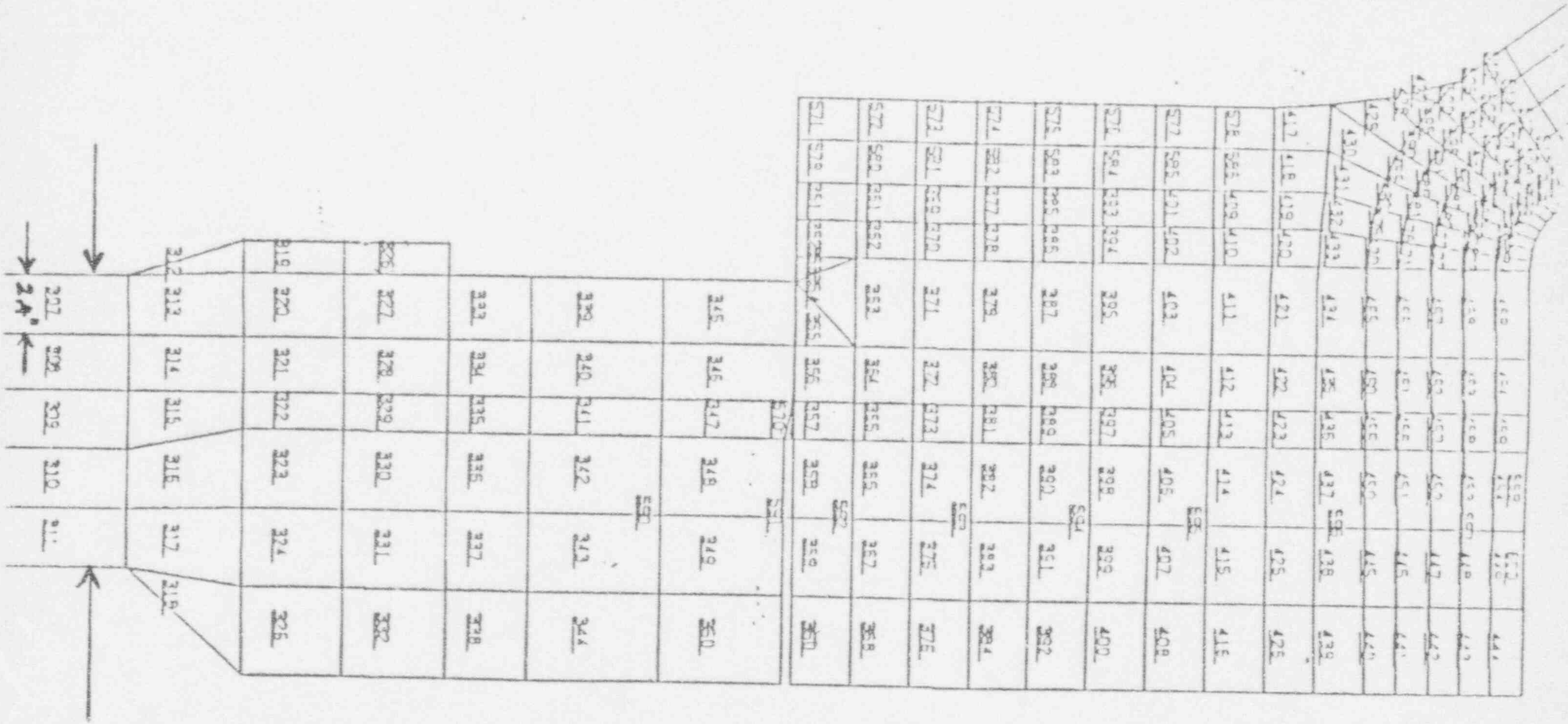


Figure 1. Finite Element Model Element Numbers (RV Head and Head Flange Juncture)

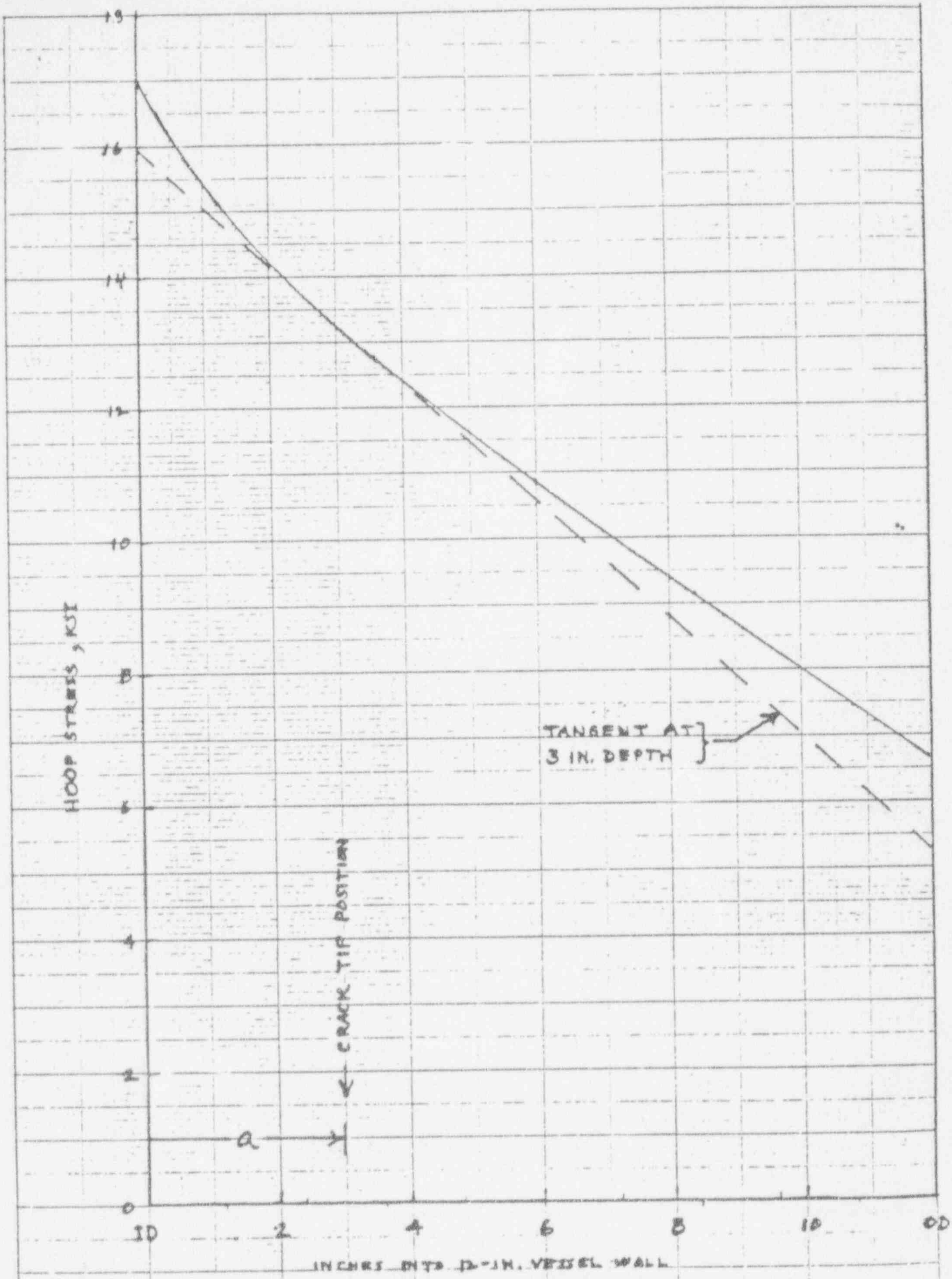
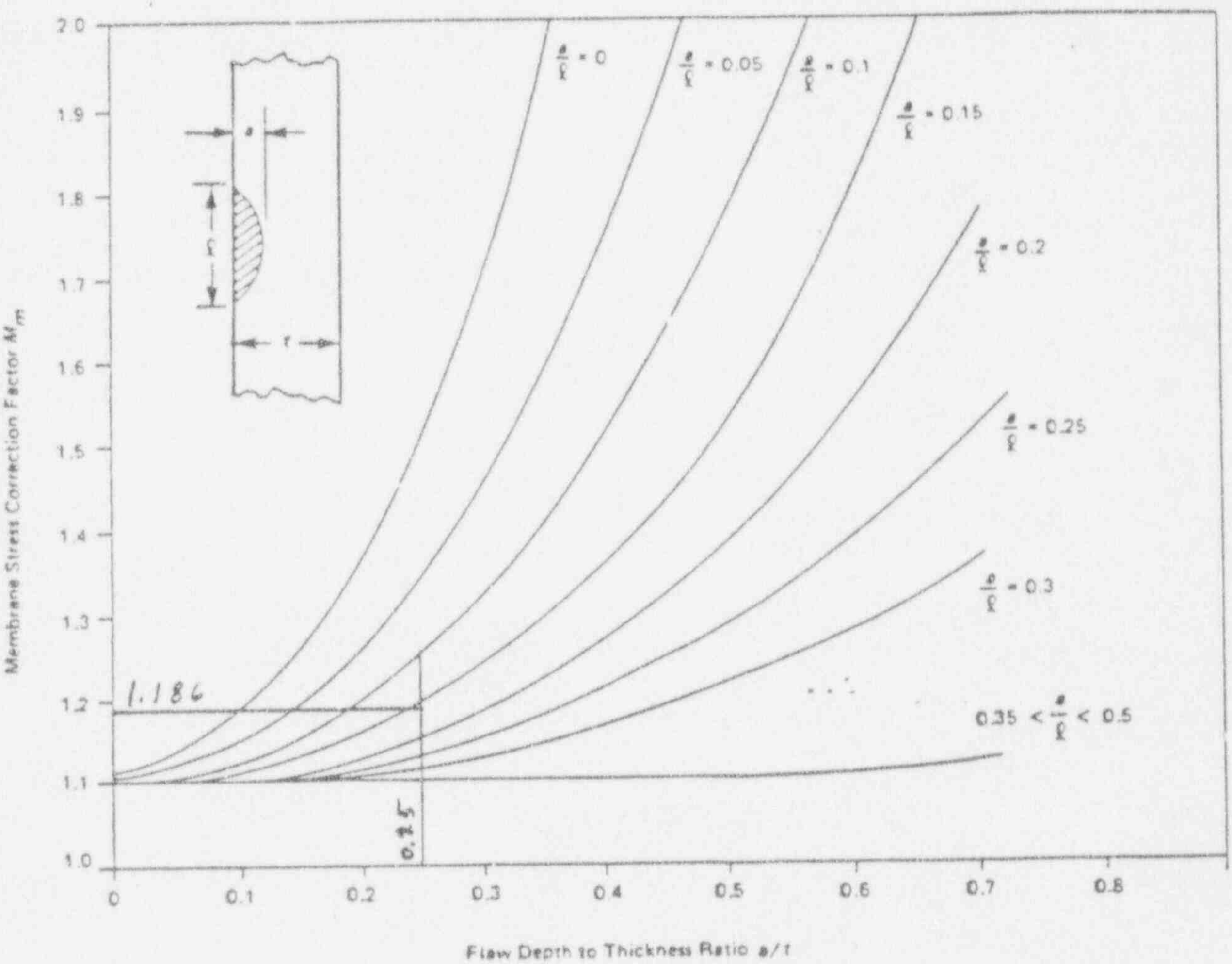


Figure 2. Shell stress distribution.

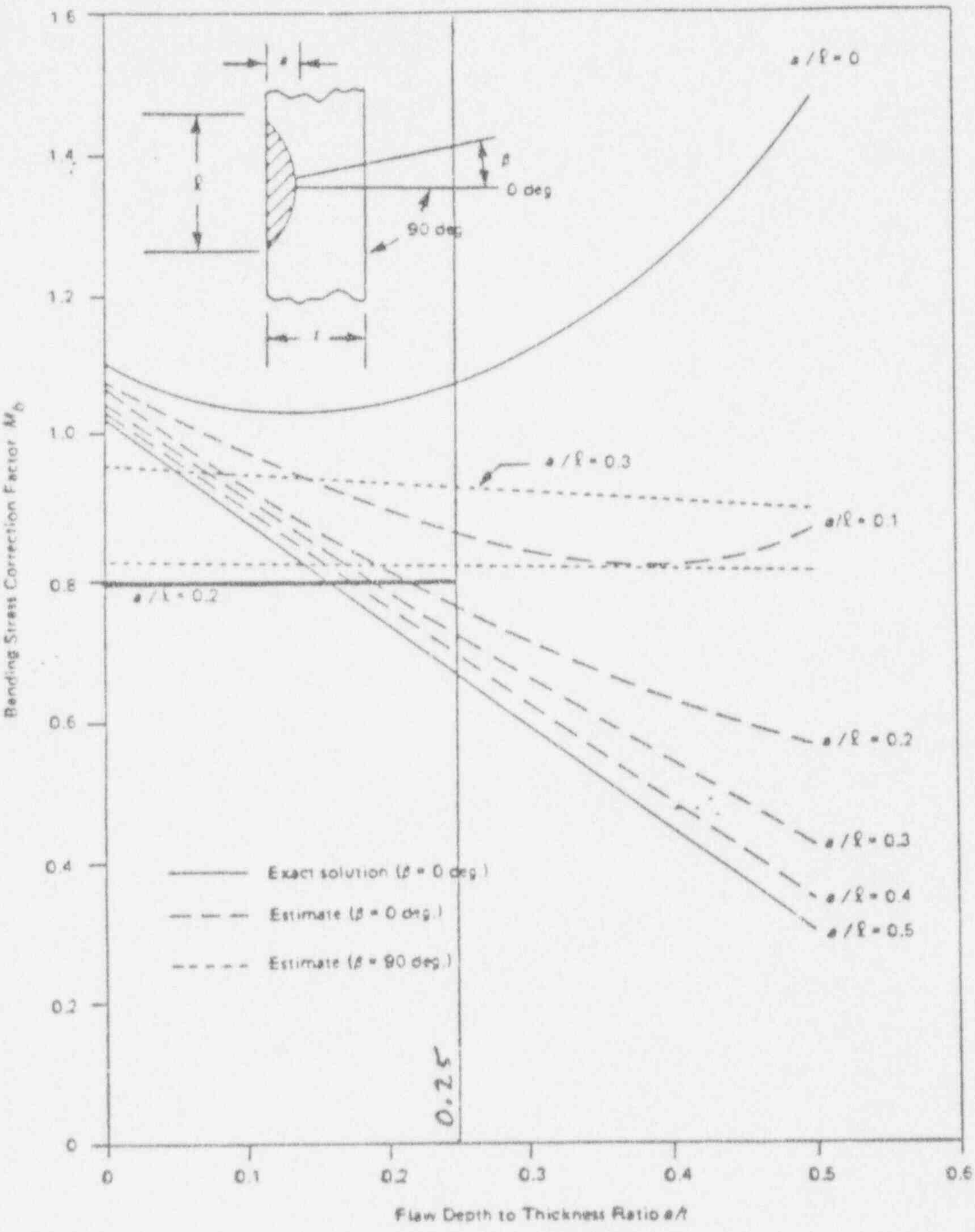


t = wall thickness

ℓ = major axis of ellipse circumscribing the flaw

FIG. A-3300-3 MEMBRANE STRESS CORRECTION FACTOR FOR SURFACE FLAWS

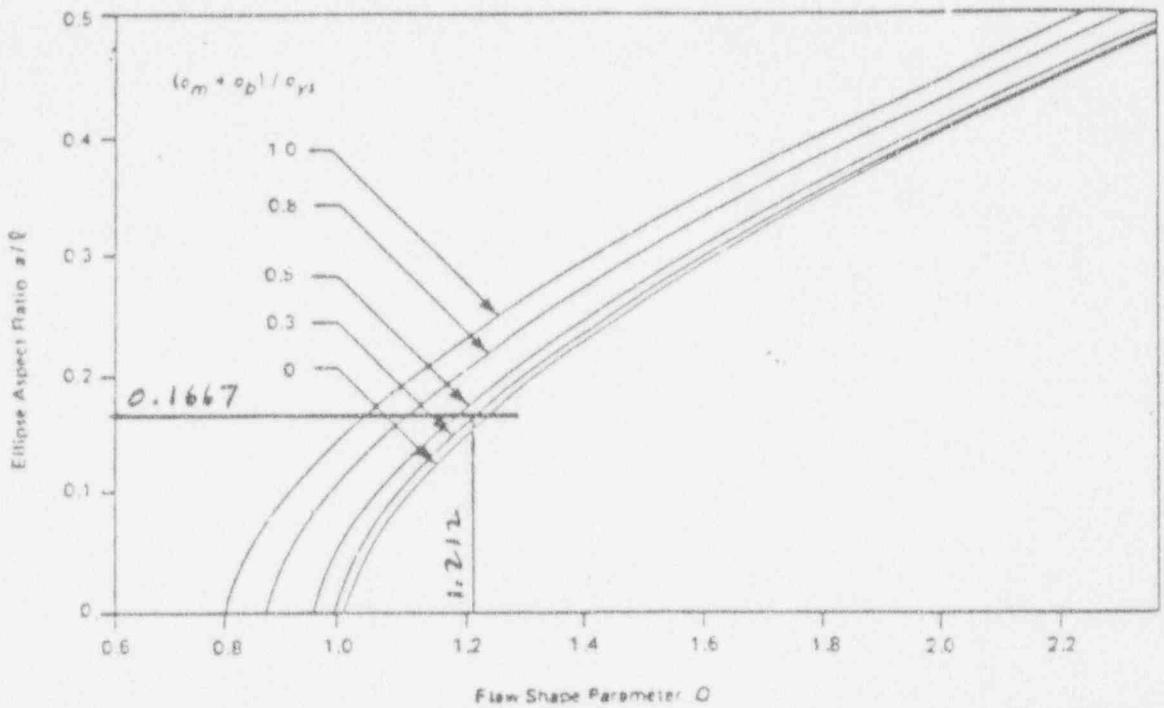
Figure 3. ASME Code method for M_m determination.



t = wall thickness
ℓ = major axis of ellipse circumscribing the flaw

FIG. A-3300-5 BENDING STRESS CORRECTION FACTOR FOR SURFACE FLAWS

Figure 4. ASME Code method for M_b determination.



σ_{ys} = specified minimum yield strength
ℓ = major axis of ellipse circumscribing the flaw

FIG. A-3300-1 SHAPE FACTORS FOR FLAW MODEL

Figure 5. ASME Code method for Q determination.

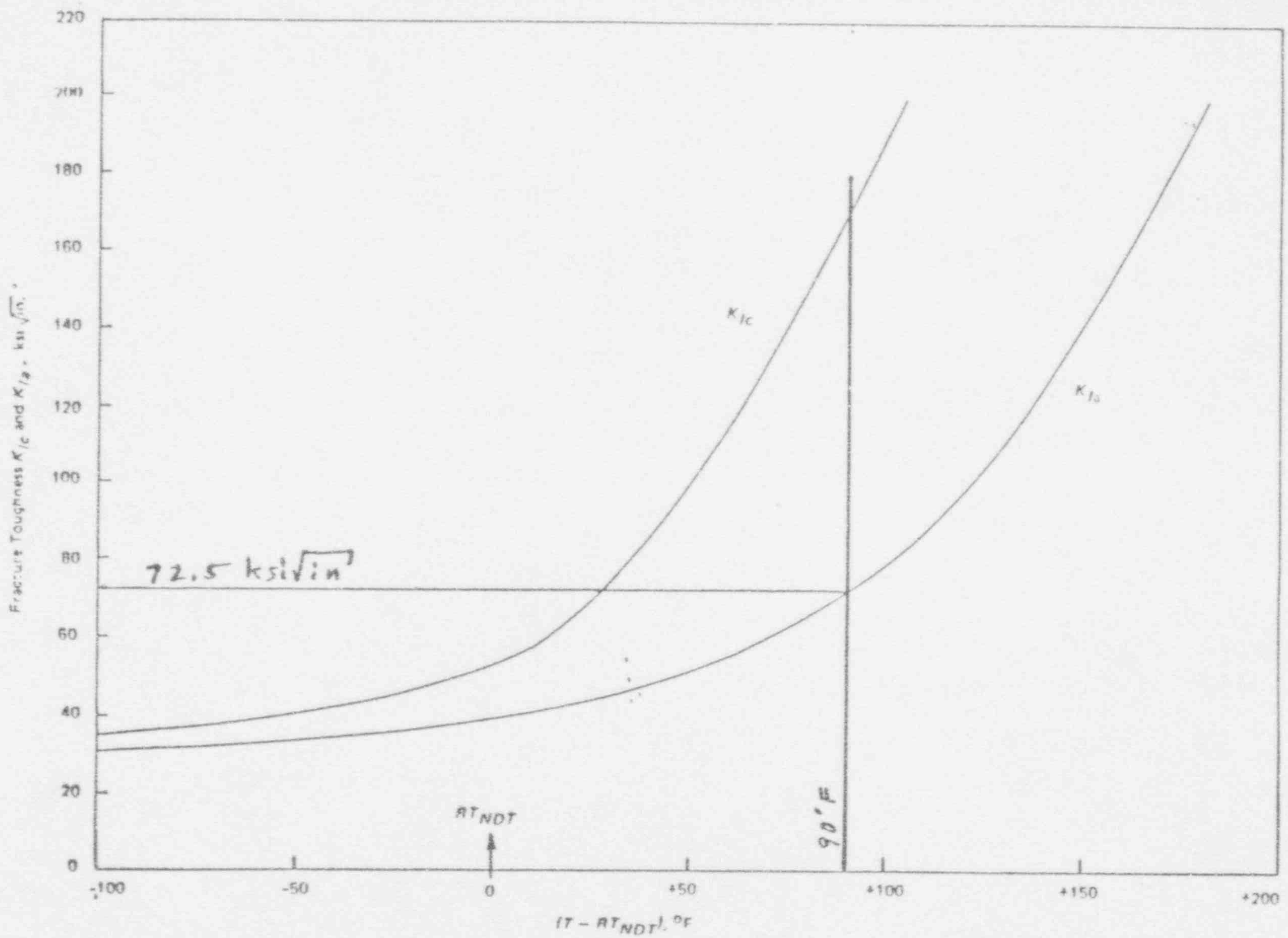


FIG. A-4200-1 LOWER BOUND K_{Ia} AND K_{Ic} TEST DATA FOR SA-533 GRADE B CLASS 1, SA-508 CLASS 2, AND SA-508 CLASS 3 STEELS

Figure 6. ASME Code curves for $T-RT_{NDT}$ determination.

Document Name:
CLOSURE STUD ANALYSIS as of 3-13-90

Requestor's ID:
BEVAN

Author's Name:
RJohnson

Document Comments:

Closure Stud Fracture Mechanics Analysis

Richard E. Johnson

Background and Summary

Reactor pressure vessel (RPV) head closure studs were analyzed using ASME Code procedures for the case of the analyzed natural circulation cooldown event in a B&W 177 FA plant. The analysis involved the calculation of the mode-one stress intensity factor, K_I, based on the stud geometry provided by B&W, the stresses calculated by the Brookhaven National Laboratory (BNL), and the reference flaw and equations prescribed by the ASME Code. The calculated value of K_I was used to enter the ASME Code curve of Plane-strain fracture toughness (crack initiation toughness), K_{Ic}, to obtain a value of relative temperature ($T - RT_{NDT}$). A conservative value of the reference temperature for the nil-ductility (fracture mode) transition, RT_{NDT}, was established based on information provided by B&W. Stated in a different way, the purpose of the exercise was to determine that temperature where the RPV stud material exhibited a fracture toughness equal to the calculated stress intensity factor. Of course, if the input parameters we-. accurately representative of the actual stud material rather than conservatively determined, as they were, the equality of $K_I = K_{Ic}$ would signal conditions for fracture instability.

To assess the safety inherent in the RPV studs, the temperature at which the stress intensity factor and toughness are equal was compared to the lowest service temperature (LST) for the studs in the given transient event. It was found that the temperature corresponding to the above equality was 108°F. According to calculations reported by BNL, the studs would not go below 200°F, approximately, during the transient; more likely the LST would be 300°F or more. Therefore, it was concluded that the temperature of 108°F was well below 200°F and the fracture toughness of the stud material would be more than enough to survive the transient without any reasonable likelihood of failure.

The details of the analysis are given in the following text.

Analysis

1. Stresses

The transient being evaluated would occur when the plant goes into shut-down with the main reactor coolant pumps tripped resulting in coolant stratification and associated thermal stresses. The transient was analyzed by BNL; stresses in the RPV closure studs from the reported BNL results are given in Table I.

2. Stress Intensity Factors

To evaluate the adequacy of the stud material fracture resistance, the procedures given in the ASME Code were followed. Appendix G, Section III, of the Code refers the analyst to WRC Bulletin 175 (Reference 1). In Reference 1, Section 7, "Toughness Requirements for Bolting," relationships are given for stress intensity factor calculations. The relationships were listed in Table II which provides the numerical, dimensionless proportionality factor relating K_I to the product of the nominal stress on the minimum cross-section of the threaded region, σ, and the square root of the crack depth, a, for values of the relative (dimensionless) crack depth. Solutions are given for two geometries: a circumfrentially-notched cylinder and a single-edge-notched (SEN) plate. For both, tension and bending loading conditions are addressed. The SEN values in Table II were used to prepare the curves in Figure 1. According to Reference 1 (see Section 7 therein), threaded fastener analyses shall use the notched cylinder values for K_I (tension) and the SEN values for K_I (bending). $K_I = K_I(T) + K_I(B)$ by the principle of superposition.

3. Flaw

In the threaded region, the total flaw depth is the crack depth plus the thread depth; B&W flange stud bolts generally use 8N threads which have a thread depth of 0.08 in. for all diameters. For nominal diameters greater than 3 in., according to Reference 1, the reference flaw (crack plus thread) depth is taken to be 0.3 in.

4. Stud Parameters

The stud geometry factors were taken from a drawing provided by the B&W Owners Group, a portion of which is shown as Attachment 1.

D (gross) = 7 in. (nominal); = $6.687 \begin{smallmatrix}+.000\\-.010\end{smallmatrix}$ (max. on dwg.); = 6.25 in. shank.
I.D. = 1. in. (axial bore).
d (at root of thread) = 6.33 in. [dwg.: $6.336 \begin{smallmatrix}+.000\\-.010\end{smallmatrix}$ in. diam.]

Thread depth = 0.08 in., so max. diam. (at threads) = 6.336 + 2(0.08) = 6.496, or 6.5 in., approximately.

Reference flaw (based on ASME Code requirements):

for $D > 3$ in.; a = 0.3 in. (a = thread depth + crack),
thus the crack at the root of a thread will be 0.22 in. deep.

The relative crack depth, $a/d = (0.3)/(6.5) = 0.046$.

5. Calculate K_I

From Table II, for tensile stresses (using the notched cylinder):

$$K_I\ (T) = 1.98\ \sigma(a)^{1/2}.$$

From Figure 1, for bending stresses (using the SEN plate):

$$K_I\ (B) = 1.89\ \sigma(a)^{1/2}.$$

The values of stress to be used in solving for K_I, according to the Code and Reference 1, are based on the minimum cross-section in the thread region (i.e., where d = 6.33 in.). Believing that BNL based the stress calculations on the minimum (thread root) diameter, the values reported in Table I can be used directly. At the onset of the transient, not only are the combined stresses relatively low but the stud temperature will be high, the fracture toughness will be high and a fracture mechanics analysis would be of no value. Therefore, the stresses reported for fifteen hours into the transient are germane. Thus:

$$K_I = K_I\ (T) + K_I\ (B)$$

$$= 1.98\ \sigma(a)^{\frac{1}{2}} + 1.89\ \sigma(a)^{\frac{1}{2}}$$

$$= (1.98)(42656)(0.3)^{\frac{1}{2}} + (1.89)(61371)(0.3)^{\frac{1}{2}}\ \text{psi(in.)}^{\frac{1}{2}}$$

$$= 46.26 + 63.53\ \text{ksi(in.)}^{\frac{1}{2}}$$

$$K_I = 109.79 \text{ or } 110\ \text{ksi(in.)}^{\frac{1}{2}}, \text{ approximately.}$$

6. Determine Temperature for $K_I = K_{Ic}$

The next step is to determine that temperature where the toughness equals K_I, because at all higher temperatures the material will have adequate toughness. The final step will be to compare the lowest service temperature (LST) to that where the toughness equals K_I and assess the margin for failure. According to the ASME Code and Reference 1, the applicable bolting toughness requirement should be based on the minimum static plane-strain fracture toughness [the ASME Code lower-bound curve for $K_{Ic} = f(T - RT_{NDT})$] rather than the arrest toughness, K_{Ia}, because of the absence of (1) dynamic loads on bolts and (2) significant strain-rate sensitivity in bolting alloys. Therefore, the applicable curve is the one for K_{Ic} in Figure 2 which is a copy of Figure A-4200-1 from Reference 2.* Entering at a value of 110 ksi(in.)$^{\frac{1}{2}}$ to the K_{Ic} curve, a value of 62.75°F, or 63°F by rounding out, was determined for $T - RT_{NDT}$. Since the reference temperature for studs will not change with time in service, the initial RT_{NDT} will suffice.

From Reference 3 (see Section 3.2, "Impact Properties of Bolting Materials," therein), it was determined that B&W RPV closure studs are manufactured from SA-540, GR.B-23 (or B-24). For the usual minimum specified yield strength of 130 ksi, the actual $\sigma_{YS} \simeq 160$ ksi. The current Code requires 45 ft-lb C_{VN}. Generally, the steels used as bolting materials will reach their Charpy V-notch impact test upper shelf energy at about + 40°F. Augmenting Reference 3 with a telephone conversation to A. L. Lowe, Jr., at B&W, Lynchburg, Virginia (one of the Reference 3 authors), it was learned that the proprietary version, BAW-10046 P, has a data bank (on page 3-27) from which one can deduce that RT_{NDT} = 45°F, or lower, as the temperature where closure stud steels meet both the energy (45 ft-lb) and lateral expansion Charpy V-notch criteria. Reference 4, in part, states:

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*Article A-1100, Appendix A of Section XI applies to ferritic materials 4 in. or more in thickness and with a specified minimum yield strength of 50.0 ksi, or less. Also, it "may be extended to other ferritic materials."

"If limited Charpy V-notch tests were performed at a single temperature to confirm that at least 30 ft-lbs was obtained, that temperature may be used as an estimate of the RT_{NDT} provided that at least 45 ft-lbs was obtained if the specimens were longitudinally oriented. If the minimum value obtained was less that 45 ft-lbs, the RT_{NDT} may be estimated as 20°F above the test temperature."

Therefore, RT_{NDT} = 45°F and, from the construction on Figure 2: $T - RT_{NDT} = 63$ becomes:

$$T = 108°F$$

as the temperature where $K_{Ic} = K_I$.

7. LST Comparison; Margin

Since the studs will be exposed to no temperature less than 200°F (more likely, no less than 300°F), there is a temperature margin of 92°F. On the same relative temperature scale of Figure 2, the vertical arrow identified as "T_{min}" shows where the LST for the subject transient would occur. The corresponding plane-strain fracture toughness, K_{Ic}, would be well above 200 ksi(in.)^{½}, i.e., the material would be fully ductile. With a toughness about twice the stress intensity factor and the direct proportionality between K and σ, the margin in stress also is at least a factor of two. Noting that the usual practice is to perform some non-destructive inspection (at least visual) of the closure studs at every refueling outage, the possibility of a crack as large as was postulated for the above analysis is unlikely, and it can be concluded that the subject transient will not induce closure stud failures.

References

1. "PVRC Recommendations on Toughness Requirements for Ferritic Materials," Welding Research Council Bulletin 175 by the PVRC Ad Hoc Task Group on Toughness Requirements," August 1972.

2. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix A.

3. H. W. Behnke, et al., "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," BAW-10046, Revision 2, Babcock and Wilcox, Nuclear Power Division, Lynchburg, Virginia, December 1984.

4. Nuclear Regulatory Commission Standard Review Plan, Section 5.3.2, "Branch Technical Position - MTEB 5-2, Fracture Toughness Requirements," page 5.3.2-14.

Table I

Stresses in the RPV closure studs for the natural circulation cooldown event based on the BNL report.

Time Into the Transient, hrs	Stresses*, psi Tension	T + B	Bending**
0	49,013	60,593	11,580
15	42,656	104,027	61,371

*From the BNL analysis of "Case 1" which is conservative, the results being higher than "Case 2."

**Bending (B) stresses by difference:

$$B = (T + B) - T;$$

where the Tension (T) and combined (T + B) values were reported by BNL.

Note: BNL included a tensile stress of 34,914 psi from stud prestress in the T value reported.

Table II

Stress intensity factor solutions as given in Reference 1.

	$K_I/\sigma\sqrt{a}$			
	Notched Cylinder		SEN	
a/d	Tens.	Bend.	Tens.	Bend.
0	1.98	1.98	1.99	1.99
0.05	1.98	1.98	2.00	1.89
0.10	2.05	2.11	2.10	1.85
0.15	2.27	2.64	2.25	1.85
0.20	2.64	3.47	2.44	1.87
0.25	3.10	5.03	2.67	1.92

Note: stress, σ, is based on the minimum cross-section in the threaded region.

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(K 89) 3-13-91

Requestor's ID:
BEVAN

Author's Name:
RJohnson

Document Comments:

Enclosure 3

GENERIC LETTER

TO: ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR PRESSURIZED WATER REACTORS (PWRs)

SUBJECT: RESOLUTION OF GI-79 AND THE POTENTIAL INADEQUACY OF PWR REACTOR VESSEL DESIGNS UNDER CERTAIN NATURAL CONVECTION COOLDOWN (NCC) TRANSIENT CONDITIONS

This letter is being provided to inform addressees of (1) the NRC resolution of Generic Issue 79 - "Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown" and (2) the possible future need for licensees whose plant experiences a reportable NCC transient which may place the RV in an unanalyzed condition to provide confirmation that no applicable regulatory design or fracture toughness criteria have been exceeded. It is expected that recipients will review the information contained herein for applicability to their facilities. For economic reasons, licensees may wish to perform an evaluation in anticipation of the potential occurrence of such an event. No new requirements are being established and no specific action or written response is required.

Background

On May 5, 1981, the NRC issued Generic Letter (GL) No. 81-21, "Natural Circulation Cooldown," in response to a natural circulation cooldown (NCC) event which occurred at St. Lucie 1 on June 11, 1980. That event resulted in the formation of a void (steam bubble) in the reactor vessel head. GL 81-21, addressed to all operating PWR power reactor licensees and applicants for operating licenses (except for St. Lucie, Unit No. 1), requested that addressees determine whether operator training and plant procedures were adequate to effect a controlled NCC from operating conditions to cold shutdown. Addressees were requested to demonstrate that capability by test and/or analysis pursuant to 10 CFR 50.54(f).

During its investigation into NCC conditions, the Babcock & Wilcox (B&W) Company identified a concern (Reference 1) that thermal stresses, beyond those considered in the original design of PWR vessels, may develop in the reactor vessel (RV) flanges and studs due to large axial temperature gradients across the RV closure region, i.e., a condition that was potentially outside the design basis of the PWR RVs. B&W subsequently notified the NRC of the concern. The concern was evaluated by the NRC as a potential generic safety issue and designated as Generic Issue No. 79 (GI-79).

Discussion

A detailed analysis of the 177 Fuel Assembly Reactor Vessel (177 FA RV) was performed by B&W and submitted to the NRC (Reference 2). The NRC utilized an independent confirmatory analysis performed by BNL (Reference 3) to evaluate the B&W submittal. The NRC concluded that the B&W 177 FA RV meets currently applicable regulatory design stress and fracture toughness criteria for the NCC conditions analyzed, (i.e., 100°F/hr maximum cooldown rate). The NRC further concluded it would be extremely unlikely that a PWR RV would fail from a single NCC event. However, due to the stated limitations of the analysis, no definitive conclusion could be made regarding compliance with applicable regulatory criteria of B&W 177 FA RVs that might experience a NCC which is outside the bounds of the analysis assumptions, or for B&W non-177 FA RVs and other PWR vessels which may experience a significant NCC event in the future.

The detailed analyses by B&W and the NRC and its contractor clearly pointed out the extremely complex nature of this type of analysis. It included numerous thermal-hydraulic and mechanical modeling assumptions which, although considered to be conservative, were not confirmed by specifically measured test data. Calculated stress results for the B&W 177 FA RV were as high as 98% of ASME Code allowable values in the RV studs. While it is recognized that this Code allowable value includes margins, variations in stresses calculated by B&W, when compared to those calculated by BNL, pointed out the possibility of a RV being in an unanalyzed condition for certain NCC events; particularly for events complicated by other factors (e.g., stuck open atmospheric dump valve, etc.).

The NRC has concluded that (1) NCC events have a low frequency of occurrence, (2) it is extremely unlikely that a single NCC transient would cause the failure of a PWR vessel, (3) B&W 177 FA RVs are considered to be analyzed for NCC events that do not exceed a cooldown rate of 100°F/hr, and (4) the actual severity of a specific NCC event will determine the need for, if any, and the extent of actions that may be required of any specific licensee following certain NCC events which may place a RV in an unanalyzed condition. Therefore, no requirement for generic or plant-specific actions was deemed necessary. However, the staff has concluded as a result of the review conducted under GI-79 that exposure of certain PWR RVs to a NCC transient may result in a condition that is outside the RV design basis.

This letter is being issued to advise PWR licensees and construction permit holders that in the event of a reportable NCC transient (in accordance with 10 CFR 50.73(a)(2)(ii)(A) and (B)) NRC may require supplemental information (in accordance with 10 CFR 75)(c)) to provide confirmation that no applicable regulatory design or fracture toughness criteria has been exceeded thereby ensuring adequate safety of the reactor vessel.

The decision to allow restart of a licensee's reactor without the need for or prior to submitting supplemental information will be made after an event. Plants for which supplemental information may be needed are those with:

1. B&W 177 FA RVs that experience a NCC event involving a cooldown rate greater than 100°F/hr.

2. Westinghouse, Combustion Engineering and B&W non-177 FA RVs which experience any reportable NCC event.

It is recognized that if the above supplemental information is required that it would be a change in NRC staff practices. This change in NRC staff practices would be justified to ensure that the design criteria, as specified by the ASME Boiler and Pressure Vessel Code, has not been exceeded when a licensee has exposed a reactor vessel to an unanalyzed condition.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

[GI 79 GENERIC LETTER] *SEE PREVIOUS CONCURRENCE

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1. Letter for R. C. DeYoung (NRC) from J. H. Taylor (B&W), "Unanalyzed Reactor Vessel Thermal Stress During Cooldown," March 18, 1983.

2. Letter for N. P. Kadambi (NRC) from F. R. Miller (BWOG), "Transmittal of RV Head Stress Evaluation Program Results," October 15, 1984.

3. Review of Unanalyzed Reactor Vessel Thermal Stress for the B&W FA-177 Reactor Vessel, Y. S. Chung, J. A. Pires and M. Reich (BNL), May 1989.

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