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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington. D.C. 20555 - 0001

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Subject: Braidwood Station Interval 2 Inservice Inspection Program: Relief Request I2R-32, Alternative Requirements for Successive Inspections of Unit 2 Reactor Vessel Weld

References:

1) Letter from T. J. Tulon (Commonwealth Edison) to U.S. NRC, "Braidwood Nuclear Power Station, Unit 2, Reactor Vessel Inspection Shell Weld Indication Evaluation," dated October 15, 1997.

2) Letter from S. N. Bailey (U.S. NRC) to O.D. Kingsley (Commonwealth Edison), "Braidwood Unit 2 Reactor Vessel Inspection Shell Weld Indication Evaluation," dated April 20, 1998.

As required by 10 CFR 50.55a (3), Braidwood is submitting, for NRC approval, a proposed alternative to existing American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," requirements.

During the inservice inspection of the Braidwood Station Unit 2 reactor vessel conducted in October 1997, an ultrasonic indication was detected in the Nozzle Shell to Intermediate Shell Weld. The indication exceeded the acceptance standards of IWB-3510, "Standards for Examination Category B-A, Pressure Retaining Welds in Reactor Vessels," but was evaluated using analytical techniques under ASME Section XI, IWB-3600, "Analytical Evaluation of Flaws," (Reference 1). The NRC reviewed this analysis and, in Reference 2, determined the Reactor Vessel shell weld to be acceptable, without repair, for continued operation for the service life of the vessel. However, the 1989 Edition, without addenda, of ASME Section XI, the applicable edition for the 2nd Interval Inservice Inspection program at Braidwood Station, requires that the indication in the vessel shell be re-examined during the next three inspection periods.

Braidwood is proposing an alternative to successive re-examinations of the Braidwood Unit 2 Reactor Vessel Weld. The alternative utilizes ASME Nuclear Code Case N-526 and is being proposed under 10 CFR 50.55a(a)(3)(ii). The discussion contained in the attached relief request, I2R-32, demonstrates that compliance with existing Code requirements would result in hardship or unusual difficulty for Braidwood Station without a compensating increase in the level of quality and safety. The proposed alternative is for Braidwood Unit 2 only but is being submitted on the Unit 1 and Unit 2 dockets because the Inservice Inspection Program is common to both Units.

Because the first period examination would be required in the upcoming Fall 2000 outage (A2R08), Braidwood is requesting resolution of the proposed alternative by April, 2000. To facilitate the NRC review, Braidwood Station is providing as Attachments 2, 3, and 4 the previously docketed information related to evaluation, analysis, and acceptance of the indication in the Unit 2 vessel weld.

Please direct any questions you may have regarding this submittal to Mr. T. W. Simpkin, Regulatory Assurance Manager, at (815) 458-2801, x2980.

Sincerely,

Timothy J. Tulon
Site Vice President
Braidwood Station

Attachments:
1) Relief Request I2R-32, Alternative Requirements for Successive Inspections of Unit 2 Reactor Vessel Weld.
2) Reactor Vessel Inspection Shell Weld Indication Evaluation.
3) Summary of Flaw Evaluation for Braidwood Unit 2.
4) Safety Evaluation by the Office of Nuclear Reactor Regulation, Commonwealth Edison Company, Reactor Vessel Shell Weld Indication Evaluation, Braidwood Nuclear Power Station, Unit 2.

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station

Attachment 1

Relief Request I2R-32

Alternative Requirements for Successive Inspections of Unit 2 Reactor Vessel Weld

RELIEF REQUEST I2R-32
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COMPONENT IDENTIFICATION

Code Class: 1
References:

1) Letter from S.N. Bailey (U.S. NRC) to O.D. Kingsley (Commonwealth Edison) dated April 20, 1998, "Braidwood, Unit 2 Reactor Vessel Inspection Shell Weld Indication Evaluation."

2) American Society of Mechanical Engineers (ASME) Nuclear Code Case N-526, "Alternative Requirements for Successive Inspections of Class 1 and 2 Vessels."

3) Letter from T. J. Tulon (Commonwealth Edison) to U.S. NRC, "Braidwood Nuclear Power Station, Unit 2, Reactor Vessel Inspection Shell Weld Indication Evaluation," dated October 15, 1997.

Examination Category: B-A, "Pressure Retaining Welds in Reactor Vessel"
Item Numbers: B1.11, "Shell Welds, Circumferential "
Description: Nozzle Shell to Intermediate Shell Weld
Component Numbers: Unit 2 Reactor Vessel (2RC01R), Weld Number 2RV-01-004
Drawing Numbers: 2RV-01

CODE REQUIREMENTS

ASME Section XI Table IWB-2500-1, Category B-A, "Pressure Retaining Welds in Reactor Vessel" requires a volumetric examination on circumferential shell welds (Item B1.11) once per 10 year inspection interval. The deferral of this examination to the end of the interval is permissible.

In addition, ASME Section XI, Subarticle IWB-2420, "Successive Inspections," paragraph (b) states:

> "If flaw indication or relevant conditions are evaluated in accordance with IWB-3132.4 or IWB-3142.4, respectively, and the component qualifies as acceptable for continuous service, the areas containing such flaw indications or relevant conditions shall be reexamined during the next three inspection periods listed in the schedules of the inspection programs of IWB-2410."

During the first 10 year inservice inspection of the Braidwood Station Unit 2 reactor vessel an ultrasonic indication was detected in the Nozzle Shell to Intermediate Shell Weld. The indication exceeded the acceptance standards of IWB-3510 but was evaluated using analytical techniques under IWB-3600 and determined to be acceptable, without repair, for continued operation (I2R-32 Reference 1).

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CODE REQUIREMENTS FROM WHICH RELIEF IS REQUESTED

Braidwood Station is requesting relief from the successive reexamination requirements of IWB-2420 (b) for the area containing the indication in the Unit 2 Reactor Vessel Nozzle Shell to Intermediate Shell Weld (2RV-01-004). Specifically, relief is requested from performing the reexaminations in the first and second periods of this interval.

BASIS FOR RELIEF

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested on the basis that conformance with the Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

In order to examine the area containing the indication in the Unit 2 Reactor Vessel Nozzle Shell to Intermediate Shell Weld (2RV-01-004) the reactor must be defueled and the lower internals and the core barrel must be removed. The schedule time for these activities prior to the examination is approximately two and one half days.

The vendor cost alone (not including site training, plant support, or potential critical path time) to perform these examinations with automated tooling in the first inspection period and second period is currently estimated at $530,000. The cost to perform this same examination at the end of the second inspection interval concurrent with the reactor vessel ten-year examination is estimated at less than $25,000. The major expense associated with the first and second inspection period reexaminations is the added equipment and personnel mobilization costs and equipment assembly and disassembly costs.

Approximately one man-rem exposure is currently expended for automated equipment assembly and disassembly in the reactor cavity area. In addition to exposure, there are approximately two to three cubic feet of solid radwaste generated during performance of automated examinations in the reactor vessel. Under current Code rules, this personnel exposure and radwaste generation would be incurred three times; once for the first inspection period examination, next for the exam conducted in the second period, and again for the reactor vessel examinations at the end of the inspection interval. Performing the examination of this shell weld area concurrent with the reactor vessel ten-year examinations will save approximately two man-rem exposure and four to six cubic feet of solid radwaste.

Therefore, in spite of the small scope of examination, the performance of this examination requires a significant expenditure of costs and dose without a compensating increase in the level of quality and safety for this subsurface, non-service induced indication.

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PROPOSED ALTERNATE PROVISIONS

Braidwood Station is proposing the alternative provisions of ASME Code Case N-526 which does not require the periodic reexaminations of IWB-2420(b) for this reactor vessel indication because all conditions stipulated in ASME Code Case N-526 are met:

(a) the flaw is characterized as subsurface in accordance with Figure 1.

> With a distance from the (outer) surface, S, of 0.46 inches and a Flaw Half Depth, a, of 0.305 inches, the indication in weld 2RV-01-004 is characterized as subsurface per the criteria of ASME Section XI, IWA-3000 as well as by the more conservative "Surface Proximity Rule" figure of Code Case N-526, (see Figure 1 on page 5).

(b) The NDE technique and evaluation that detected and characterized the flaw, with respect to both sizing and location, shall be documented in the flaw evaluation report.

> The NDE technique and evaluation that detected and characterized the indication in 2RV-01-004 is documented in the docketed flaw evaluation report contained in I2R-32 Reference 3. This NDE technique utilized a procedure that was qualified by performance demonstration in accordance with the Performance Demonstration Initiative (PDI) program for implementation of ASME Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." In addition, information related to the technique and evaluation results is maintained at Braidwood Station per the requirements of ASME Section XI, IWA-1400 (h).

(c) The vessel containing the flaw is acceptable for continued service in accordance with IWB-3600, and the flaw is demonstrated acceptable for the intended service life of the vessel.

> The results of the ASME Section XI IWB-3600 evaluation for the indication in weld 2RV-01-004 conclude that the Unit 2 Reactor Vessel is acceptable, without repair, for the current 40 year license period of the plant. This conclusion is the result of the evaluation provided in I2R-32 Reference 3 and is reaffirmed in the Safety Evaluation provided in I2R-32 Reference 1.

Also, Braidwood will be performing the examination required by ASME Section XI, Category B-A, Item B1.11, on weld 2RV-01-004, including this indication area, in the third period of this Inspection Interval.

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PERIOD FOR WHICH RELIEF IS REQUESTED

Relief is requested for the first and second periods of second ten-year inspection interval of the Inservice Inspection Program for Braidwood Unit 2. The required volumetric examination of Unit 2 Reactor Vessel shell weld 2RV-01-004, including this indication area, will be performed in the third period of this interval.

RELIEF REQUEST I2R-32
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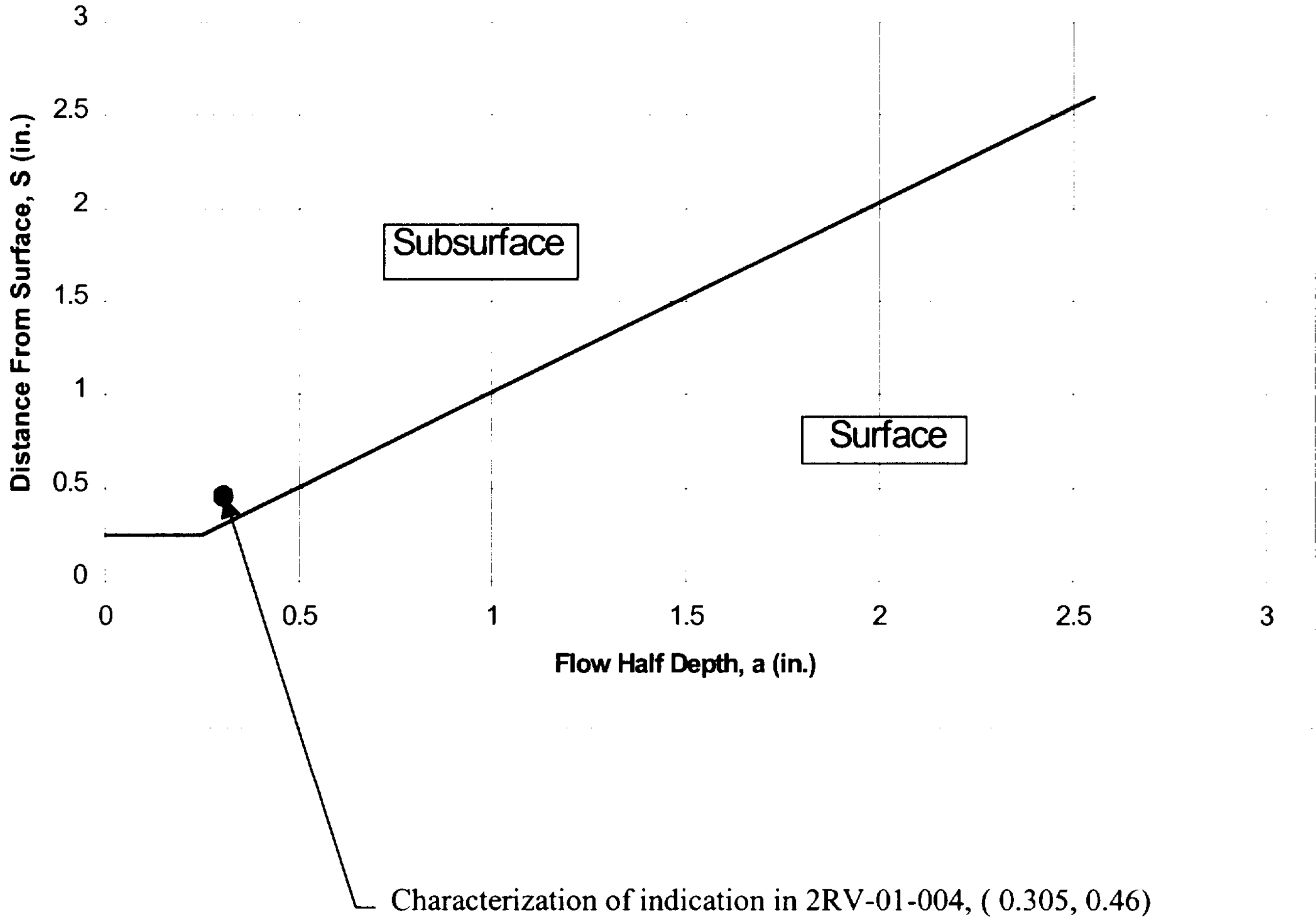
Figure 1
"Surface Proximity Rule" of Code Case N-526



Attachment 2

Reactor Vessel Inspection Shell Weld Indication Evaluation [1]

[1] Previously provided in a letter from T. J. Tulon (Commonwealth Edison) to U.S. NRC, "Braidwood Nuclear Power Station, Unit 2, Reactor Vessel Inspection Shell Weld Indication Evaluation," dated October 15, 1997.

Reactor Vessel Inspection Shell Weld Indication Evaluation

INTRODUCTION

Pursuant to the provision of ASME Section XI, 1983 Edition through Summer 1983 Addenda, paragraph IWB-3125(b), Braidwood Station submits this document to demonstrate that the indication detected in the Nozzle Shell-to-Intermediate Shell weld (Weld #2RV-01-004 in Braidwood Unit 2) during the current refueling outage (A2R06) is acceptable for service without repair or replacement in accordance with the provisions of IWB-3122.4.

The subject indication was detected by ultrasonic testing (UT) of the Reactor Pressure Vessel (RPV) Shell welds, which is required by ASME Section XI, 1983 Edition through Summer 1983 Addenda, Examination Category B-A, Item Number B1.11, and 10CFR50.55a(g)(6)(ii)(A).

UT INSPECTION TECHNIQUES AND QUALIFICATION

Braidwood Unit 2 RPV shell welds were ultrasonically examined from the inside surface of the RPV by Framatome Technologies (FTI) using a mechanized tool. The FTI procedure is a contact UT technique qualified in December 1995 to the Performance Demonstration Initiative (PDI) program at the EPRI NDE Center. This inspection technique is being performed in accordance with ISI Program Relief Request NR-29, approved by the NRC on May 13, 1997.

Three transducers, 45° shear, 45° L-wave and 70° L-wave, were used to examine the full volume of the welds. The reference sensitivity, or DAC, is established on 1/8" diameter side drilled holes and scanning is performed at noise level.

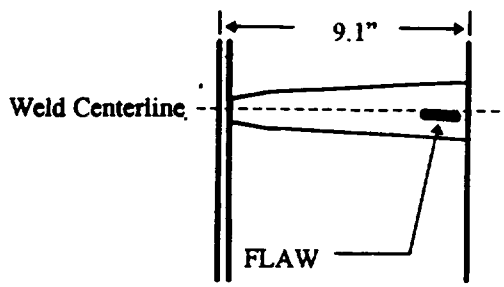
INSPECTION RESULTS

- PDI Inspection

  UT has detected one indication in weld 2RV-01-004 (See Attached Figure 3). This indication was detected at approximately 350° vessel azimuth with the 45° shear from both sides at amplitude levels of 109% and 194% of the reference. The indication length was determined by dropping the signal to noise level, and tip diffraction techniques were used to determine the planar depth. These sizing techniques use the same calibrations and transducers as the detection examination and are also PDI qualified. The indication is circumferentially oriented and is categorized as a subsurface (or embedded) planar indication located near the outside surface (See Figures 1 and 2). The characteristics of the subject indication are:

S (distance from the indication to the nearest surface, OD)	=	0.46 in.
2a (depth of indication)	=	0.61 in.
l (length of indication)	=	5.86 in.
t (measured excluding nominal clad thickness of 0.1875 in.)	=	8.91 in.

Figure 1



- Supplemental Inspection

  Additional examinations were performed on the area of concern to better characterize the indication. These examinations consisted of a Section XI, 0° L-wave, and a 45° and 60° shear wave. At a maximum gain of 38 db over reference, the 0° L-wave detected a spot or small portion of the indication. The indication was detected from both sides with the 45° examination at recordable levels of 53% and 90% DAC amplitudes. The 60° examination did not detect the indication. These examinations have confirmed that the indication is a tight planar flaw, consistent with results of the PDI examinations. There is no conclusive evidence that slag is present.

- PSI Review

  The hard copy of the PSI NDE reports and the original videotapes of the A-scan have been reviewed. The PSI inspections were conducted using an immersion UT technique. No indications were reported in this area. Efforts are underway to view the original fabrication radiographs and repair records, if any.

FLAW EVALUATION RESULTS

- Evaluation per IWB-3500

  This indication has been evaluated to the IWB-3500 Acceptance Standards and was found to exceed the limits of Table IWB-3510-1, "Allowable Planar Indications." The indication a/t of 3.4% exceeds the maximum allowable a/t of 2.4% for an aspect ratio of a/l = 0.05.

- Evaluation per IWB-3600

  The subject indication was evaluated using the rules of IWB-3600 of ASME Section XI, 1983 Edition with Summer 1983 Addenda. This is accomplished by using the "Handbook on Flaw Evaluation For Zion, Byron, and Braidwood Reactor Pressure Vessels" (WCAP-12045 and 12046) which was developed by Westinghouse based on the rules of IWB-3600 of ASME Section XI. Although these reports were prepared in 1988, the acceptance margins and code material properties used are unchanged since that time.

Figure A-2.6 (See Attached Figure 4) in Section A-2 of the attached WCAP-12046 contains the applicable flaw evaluation chart for an embedded and circumferentially oriented indication in the Nozzle Shell-to-Intermediate Shell Weld.

The indication is located in the nozzle shell to intermediate shell weld, which is a Linde 80 weld, with heat number WF 645. The weld is located above the core region as shown in Figure 2-7 of WCAP 12045, so irradiation effects are negligible. To determine the acceptable flaw sizes, the flaw evaluation used the most limiting RT_{NDT} of the two shell courses and the weld filler metal, all of which are provided in WCAP 12045.

Basic data:

t (Measured thickness excluding nominal clad thickness of 0.1875 in.)	= 8.913in.
δ (S+a, Distance from the embedded flaw centerline to OD surface)	= 0.765in.
a (Flaw depth, defined as one-half of the minor diameter)	= 0.305in.
l (Flaw length, major diameter)	= 5.860in.

The following parameters are determined for evaluating the acceptability of an embedded flaw:

a/l	(Flaw shape parameter or aspect ratio)	=0.052in.
2a/t		=0.068in.
a/t	(Flaw depth parameter)	=0.034in.
δ	(Distance from the centerline of the embedded flaw to OD surface)	=0.765in.
δ/t	(Surface proximity parameter)	=0.086in.

The subject indication in weld 2RV-01-004 with a/t = 0.034 and δ/t = 0.086 is plotted to the right and below the "Surface/Embedded Flaw Demarcation Line" of Figure A-2.6 (See Attached Figure 4). The basis calculation for the acceptable embedded flaw sizes in Figure A-2.6 was limited to a maximum flaw size of $2a/t \leq 0.25$. Since 2a/t = 0.068, this limitation on the Handbook applicability is met. Therefore, the embedded flaw is acceptable for service without repair or replacement.

To demonstrate the significant margins which exist for the indication, it was also evaluated with the assumption that it was a surface flaw. This is clearly not the case; based on IWA-3310, if S is ≥ 0.4a, the indication is a subsurface flaw. In this case, since S is 0.46 and 0.4a is 0.122, the indication is a subsurface flaw (0.46 ≥ 0.122).

To evaluate the indication as a surface flaw, the depth becomes 2a + S = 1.07 inches, which gives a/t = 0.117, and a/l = 0.18. Using Figure A-2.5 (See Attached Figure 5) of WCAP 12046 for circumferential outside surface flaws, a flaw as large as a/t = 0.2 is acceptable. Therefore, there is a substantial margin on the acceptability of this indication even if it were evaluated as a surface flaw.

ADDITIONAL AND SUCCESSIVE EXAMINATIONS

Because the indication exceeds the applicable acceptance standards of IWB-3500, the additional examination requirements of IWB-2430 apply. Based on the clarification of IWB-2430(a), first published in the Winter 1983 Addenda of Section XI, these requirements have been satisfied by the initial examination scope performed during the current refueling outage of all similar components, i.e., all RPV Shell Welds in Examination Category B-A, Item Number B1.10.

Reexaminations of the subject indication will be performed at the next three Inspection Periods in accordance with the rules of IWB-2420, unless the NRC Staff approves an alternative approach, such as Code Case N-526.

CONCLUSIONS

The subject indication was evaluated using the rules of IWB-3600 of ASME Section XI, 1983 Edition with Summer 1983 Addenda, and found to be acceptable without flaw removal, based on the Reactor Vessel design transients defined for the current 40 year license period of the plant.

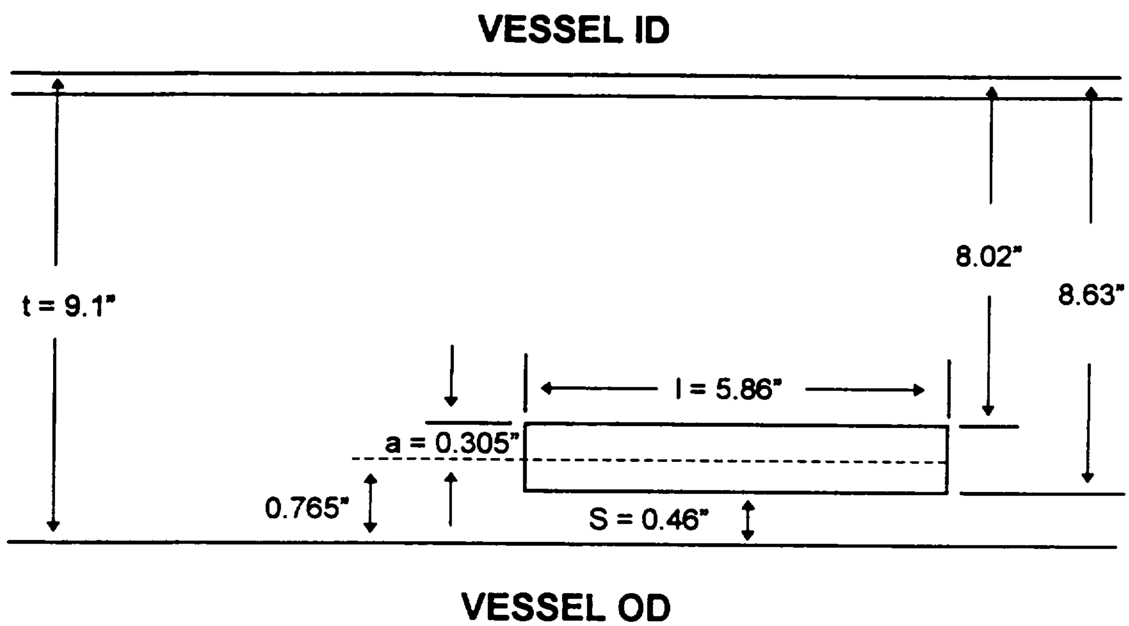


Figure 2

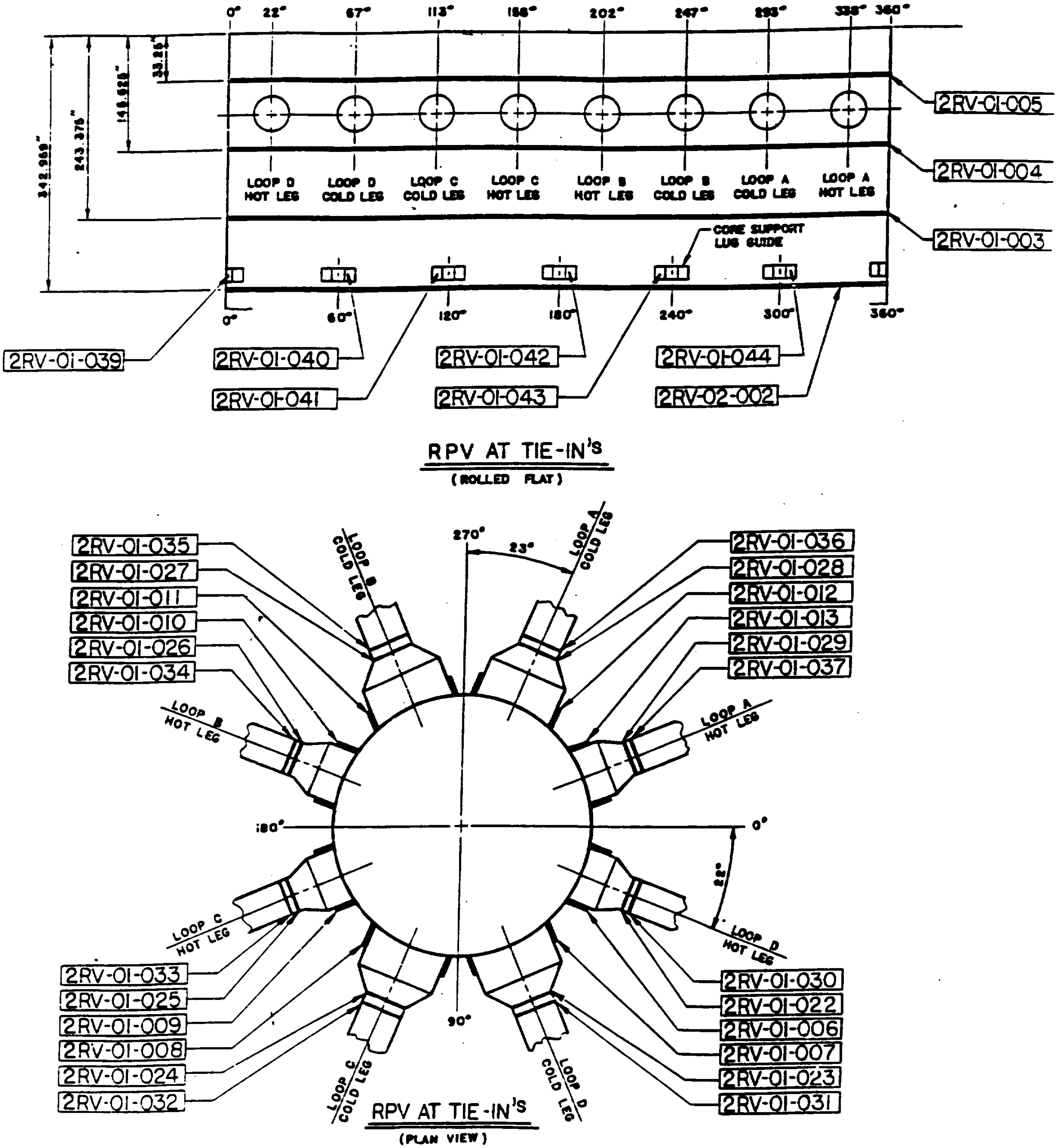

0°
22°
67°
113°
158°
202°
247°
293°
338°
360°
33.25"
148.625"
243.375"
342.969"
2RV-01-005
2RV-01-004
2RV-01-003
LOOP D HOT LEG
LOOP D COLD LEG
LOOP C COLD LEG
LOOP C HOT LEG
LOOP B HOT LEG
LOOP B COLD LEG
LOOP A COLD LEG
LOOP A HOT LEG
CORE SUPPORT LUG GUIDE
60°
120°
180°
240°
300°
2RV-01-039
2RV-01-040
2RV-01-041
2RV-01-042
2RV-01-043
2RV-01-044
2RV-02-002
RPV AT TIE-IN'S
(ROLLED FLAT)
2RV-01-035
2RV-01-027
2RV-01-011
2RV-01-010
2RV-01-026
2RV-01-034
LOOP B COLD LEG
270°
23°
LOOP A COLD LEG
2RV-01-036
2RV-01-028
2RV-01-012
2RV-01-013
2RV-01-029
2RV-01-037
LOOP B HOT LEG
LOOP A HOT LEG
180°
0°
22°
LOOP C HOT LEG
LOOP D HOT LEG
2RV-01-033
2RV-01-025
2RV-01-009
2RV-01-008
2RV-01-024
2RV-01-032
90°
2RV-01-030
2RV-01-022
2RV-01-006
2RV-01-007
2RV-01-023
2RV-01-031
LOOP C COLD LEG
LOOP D COLD LEG
RPV AT TIE-IN'S
(PLAN VIEW)

Figure 3

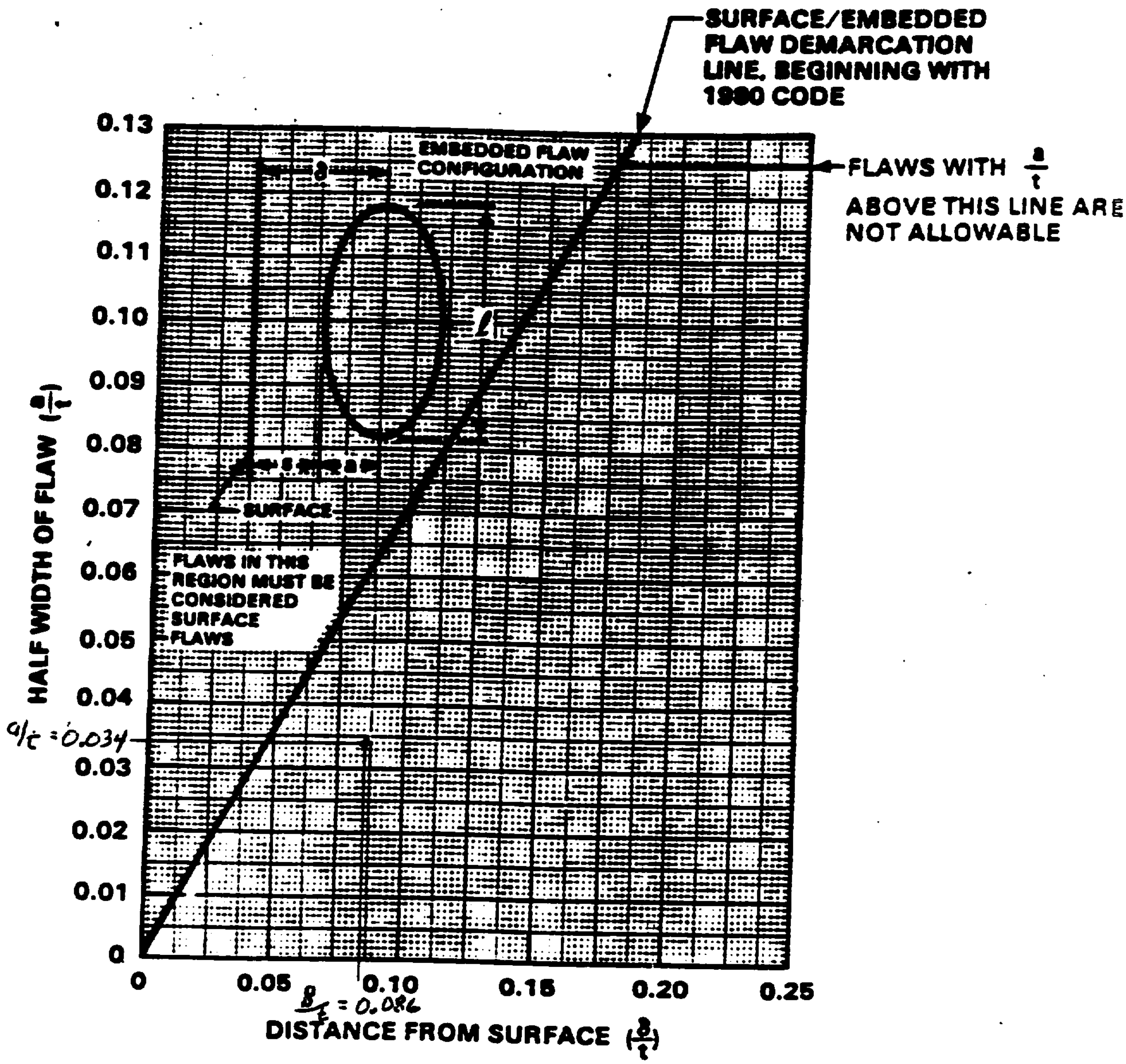


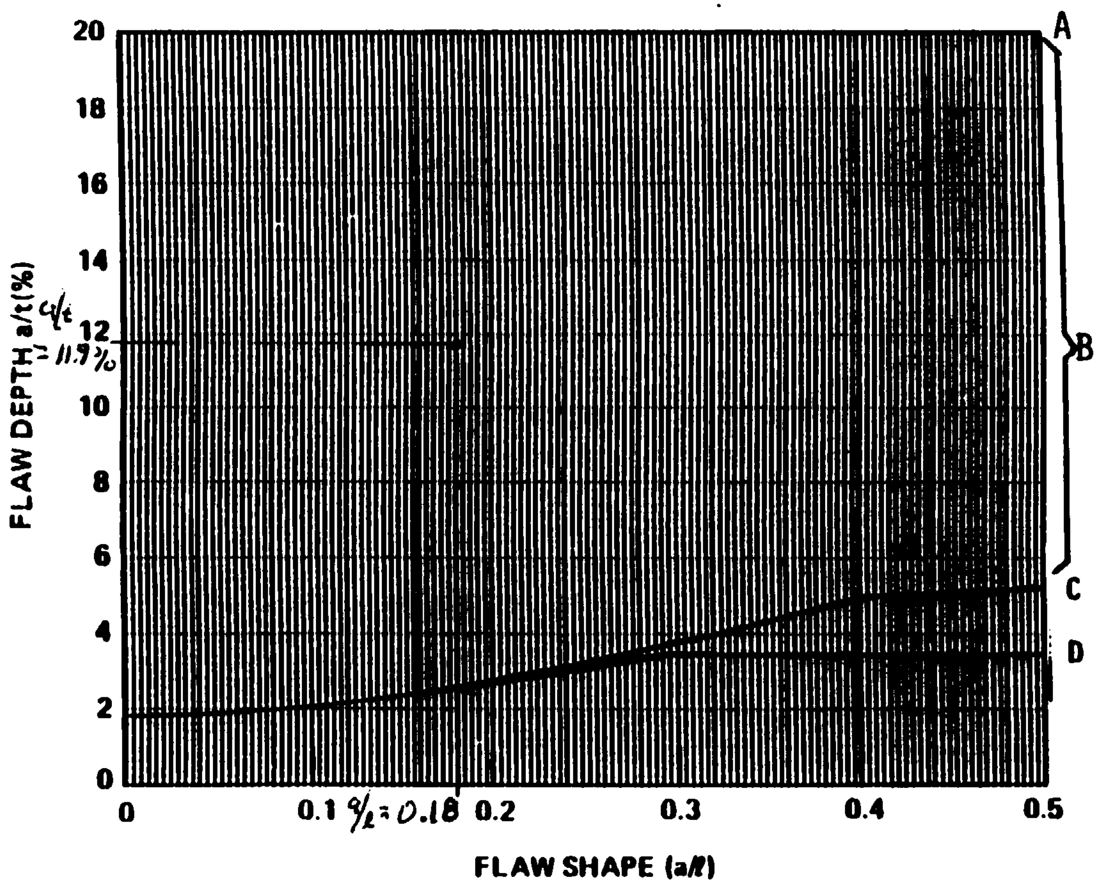
Figure A-2.6 Evaluation Chart for Nozzle Shell to Intermediate Shell Weld

X Inside Surface ___ Surface Flaw X Longitudinal Flaw

X Outside Surface X Embedded Flaw X Circumferential Flaw

Figure 4

Figure 5



LEGEND

A - The 10, 20, 30 year acceptable flaw limits.

B - Within this zone, the surface flaw is acceptable by ASME Code analytical criteria in IWB-3600.

C - ASME Code allowable since 1983 Winter Addendum.

D - ASME Code allowable prior to 1983 Winter Addendum.

Figure A-2.5 Evaluation Chart for Nozzle Shell to Intermediate Shell Weld

___	Inside Surface	X	Surface Flaw	___	Longitudinal Flaw
X	Outside Surface	___	Embedded Flaw	X	Circumferential Flaw

Attachment 3

Summary of Flaw Evaluation for Braidwood Unit 2 [2]

[2] Previously provided in a letter from T. J. Tulon (Commonwealth Edison) to U.S. NRC, "Reactor Vessel Shell Weld Indication Evaluation," dated November 25, 1997.

SUMMARY OF FLAW EVALUATION FOR BRAIDWOOD UNIT 2

This note will briefly describe the process used to develop the flaw evaluation chart used to disposition the indication recently discovered in the upper shell transition weld of Braidwood Unit 2. The indication was actually located near the outside surface of the reactor vessel, and was circumferentially orientated. Although the indication was located far enough from the surface to qualify as embedded, by the rules of Section XI, it was shown to be acceptable whether it was a surface or embedded indication. This example will be carried out to determine the largest allowable surface flaw, which was one of the results submitted.

For outside surface flaws in the upper shell transition region the most severe transients are the pressure transients, since the thermal transients result in tensile stresses near the inside surface, and compressive stresses in the region of interest here. There are no transients which result in a rapid heat-up, which would produce tensile thermal stresses in this region. The governing transients are:

Cold Hydrotest @ 3105 psi	(normal/upset)
Large Steamline Break	(emergency/faulted)

It should be mentioned that residual stresses are known to exist in this weld, but since the reactor vessel is stress relieved, the stress values are small, Measurements of stress relieved heavy section welds have shown residual stresses of about 5 ksi at each surface, with the stresses decreasing and becoming compressive in the center of the weld. These stresses are present at all times, and will have an effect on fracture at low temperatures, when the toughness is in the transition region. At higher temperatures such as these in the region of interest here, the residual stresses have no effect on the failure conditions. This has been demonstrated experimentally in the Heavy Section Steel Technology Intermediate Vessel test program. Therefore, residual stresses have not been used in the calculations discussed for this region.

It will be seen from Figure A-2.5 of WCAP 12046 (reproduced here as Figure 1) that the allowable depth for any indication, regardless of shape, is at least 20 percent of the wall thickness. The allowable depth line is across the very upper edge of the figure. This line is the result of a direct application of the Section XI acceptance criteria.

The allowable flaw depth is determined by calculation of the stress intensity factor (K) as a function of postulated flaw depth for each of the governing transients, and then determining where the K value exceeds the allowable toughness. We will follow this calculation in detail for the normal/upset case first. The stress distribution from a detailed finite element model of the reactor vessel is plotted in Figure 2. The stress intensity factor was then calculated for three different flaw shapes, and these results are shown in Figure 3. The reference used for this calculation was Raju and Newman (reference 7 of WCAP 12045, rev. 1).

The allowable toughness was determined by reducing the fracture toughness by the factor √10, as required by Section XI. The hydrotest at 3105 is only conducted before operation, but for conservatism here it is assumed to occur during service. The hydrotest temperature for Braidwood Unit 2 has been calculated as 200 F for the leak test (WACP 14970) and will be higher for the 3105 hydrotest. At 200 F for the Braidwood 2 Nozzle Shell to Intermediate Shell Weld, the fracture toughness will be on the upper shelf, since there is no irradiation effect, and the initial RT_{NDT} for this weld is -25F. The allowable toughness is then 200/√10 = 63.2 ksi -sq-rt-in, and this value is also plotted in Figure 3.

The following allowable flaw depths for normal/upset conditions result from these calculations:

Flaw Shape (a/l)	Allowable Depth (a/t)
0.01	0.269
0.1667	0.355
0.5	0.869

The flaw evaluation chart is then determined from the worst case of the results above and the results for the governing faulted condition. For the steamline break, the highest stress in the region of the flaw is early in the transient. The worst time step is at 100 secs., as may be seen from Table 1, and is plotted in Figure 4. The temperature, pressure and flow rate vs. time for this transient are provided in Appendix 1. Similar stress intensity factor calculations were done for this case, and the results were very low stress intensity factors, because of the low stresses, as seen in Figure 5. The allowable toughness is determined from the actual toughness divided by √2. The temperature exceeds 400 F in the outer region of the reactor vessel during the entire transient, so the toughness is again on the upper shelf, at 200 ksi-sq-rt-in. The calculated stress intensity factor never exceeds 46.1 for an aspect ratio of a/l = 0.1667, regardless of flaw depth. Therefore, the allowable depth for the faulted condition is a/t = 1.0. For the flaw shape a/l = 0.01, the maximum K = 71.5, so the allowable depth is also equal to the thickness. The results for the governing emergency/faulted condition:

Flaw Shape (a/l)	Allowable Depth (a/t)
0.01	1.0
0.1667	1.0

The allowable flaw depth for this location is then the more limiting result for either the normal/upset or emergency/faulted conditions. In this case the normal/upset results are governing. The allowables for the flaw evaluation chart are:

Flaw Shape (a/l)	Flaw Depth (a/t)
0.01	.269
0.1667	.355
0.5	.869

Therefore, we see that the allowable flaw depth is very large, regardless of the flaw shape for this location. For conservatism the allowable flaw depth in the chart of Figure 1 has been cut off at a/t = 0.2, since the design reference flaw is a/t = 0.25, and such a large flaw would be very unlikely.

The only other issue is the potential for fatigue crack growth during service. Since this indication is near the outside surface, and exposed to an air environment, the crack growth during service is negligible, as shown in the table below. Therefore, there is no difference in the allowable depth as a function of service time for this location, and the allowable lines for 10, 20 and 30 years of service are the same.

Initial	Crack Length After Year			
Crack Length	10	20	30	40
0.500	0.50005	0.50011	0.50016	0.50021
1.000	1.00007	1.00015	1.00022	1.00030
1.500	1.50005	1.50011	1.50016	1.50022
2.000	2.0003	2.0006	2.0009	2.00012

APPENDIX 1

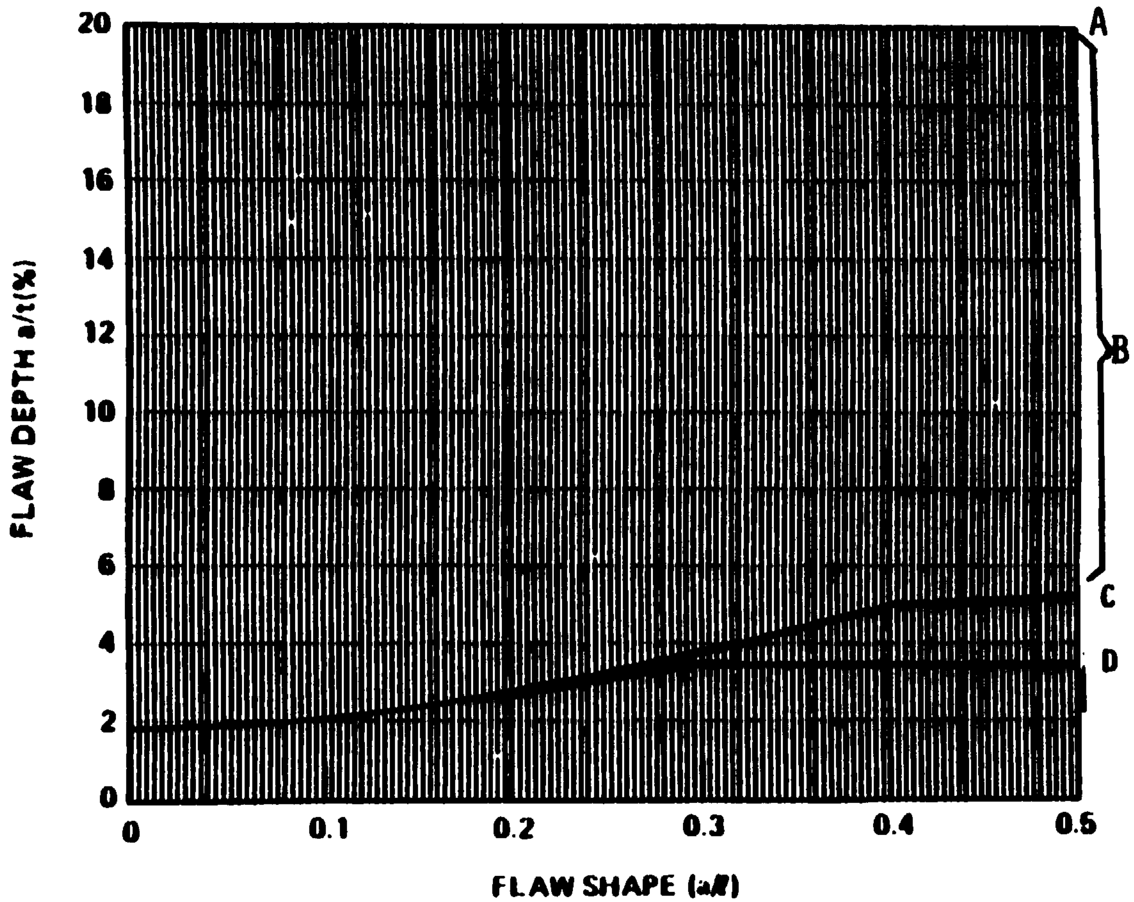
TRANSIENT DESCRIPTION FOR THE LARGE STEAM BREAK

TABLE 1

Location: Upper Shell Transition														
Transient Large Steamline Break		Temperature/Stress at Various Distances Through the Wall (x/t)												
		0.0	0.07	0.16	0.24	0.36	0.52	0.68	.0.84	1.0				
Time = 100 sec	Temp	424	481	519	540	552	556	557	557	557				
	Hoop	58.4	36.9	23.8	16.2	12.1	10.4	9.90	9.82	9.70				
	Axial	43.9	26.4	14.3	7.23	3.44	1.91	1.63	1.58	1.64				
Time = 200 sec	Temp	375	430	473	505	532	550	555	557	557				
	Hoop	66.5	48.1	33.2	22.1	12.7	6.5	4.55	4.09	4.14				
	Axial	53.2	37.1	23.7	13.3	4.5	-1.42	-3.26	-3.54	-3.02				
Time = 300 sec	Temp	337	391	436	473	509	538	550	556	556				
	Hoop	72.8	55.2	39.8	27.2	14.7	4.63	0.39	-0.99	-1.08				
	Axial	58.8	43.9	30.2	18.5	6.76	-2.85	-6.92	-8.04	-7.57				
Time = 400 sec	Temp	313	363	407	446	487	524	543	551	554				
	Hoop	75.3	59.2	44.4	31.5	17.3	4.49	-2.01	-4.87	5.13				
	Axial	60.8	47.6	34.5	22.8	9.42	-2.83	-9.06	-11.3	-10.8				
Time = 500 sec	Temp	296	342	386	422	468	509	533	546	551				
	Hoop	76.4	61.6	47.5	34.8	19.9	5.30	-3.07	-7.02	-8.02				
	Axial	61.3	49.4	37.3	25.8	11.8	-2.08	-10.1	-13.5	-13.4				
Time = 600 sec	Temp	282	328	367	403	448	494	523	539	548				
	Hoop	78.5	62.8	49.5	37.2	22.0	6.44	-3.32	-8.42	-10.0				
	Axial	61.0	50.2	39.8	27.9	13.8	-1.06	-1.04	-14.9	-15.1				

Location: Upper Shell Transition

Transient Large Steamline Break		Temperature/Stress at Various Distances Through the Wall (x/t)											
		0.0	0.07	0.16	0.24	0.38	0.52	0.68	0.84	1.0			
Time = 800 sec	Temp	278	311	343	376	417	464	498	519	529			
	Hoop	68.7	59.5	49.1	36.8	24.8	8.88	-2.48	-9.28	-11.8			
	Axial	54.3	46.9	38.4	29.3	18.4	1.42	-8.36	-15.4	-16.5			
Time = 1000 sec	Temp	278	307	335	363	402	447	481	503	515			
	Hoop	65.1	56.3	47.3	38.2	25.3	10.3	-1.03	-8.24	-11.1			
	Axial	49.9	43.8	36.5	28.6	16.9	2.77	-7.84	-14.4	-15.8			
Time = 1500 sec	Temp	277	298	320	341	371	408	437	458	469			
	Hoop	54.9	48.3	41.6	34.7	24.8	12.8	3.08	-3.60	-6.52			
	Axial	40.3	36.0	30.7	24.9	16.0	4.85	-4.22	-10.1	-11.7			
	Temp												
	Hoop												
	Axial												
	Temp												
	Hoop												
	Axial												
	Temp												
	Hoop												
	Axial												



LEGEND

A - The 10, 20, 30 year acceptable flaw limits.

B - Within this zone, the surface flaw is acceptable by ASME Code analytical criteria in IWB-3600.

C - ASME Code allowable since 1983 Winter Addendum.

D - ASME Code allowable prior to 1983 Winter Addendum.

Figure A-2.5 Evaluation Chart for Nozzle Shell to Intermediate Shell Weld

___	Inside Surface	X	Surface Flaw	___	Longitudinal Flaw
X	Outside Surface	___	Embedded Flaw	X	Circumferential Flaw

3370s/111586:10

Figure 2 Axial Stress Distribution, Braidwood 2 Upper Shell Transition
Hydrotest @ 3105 psi

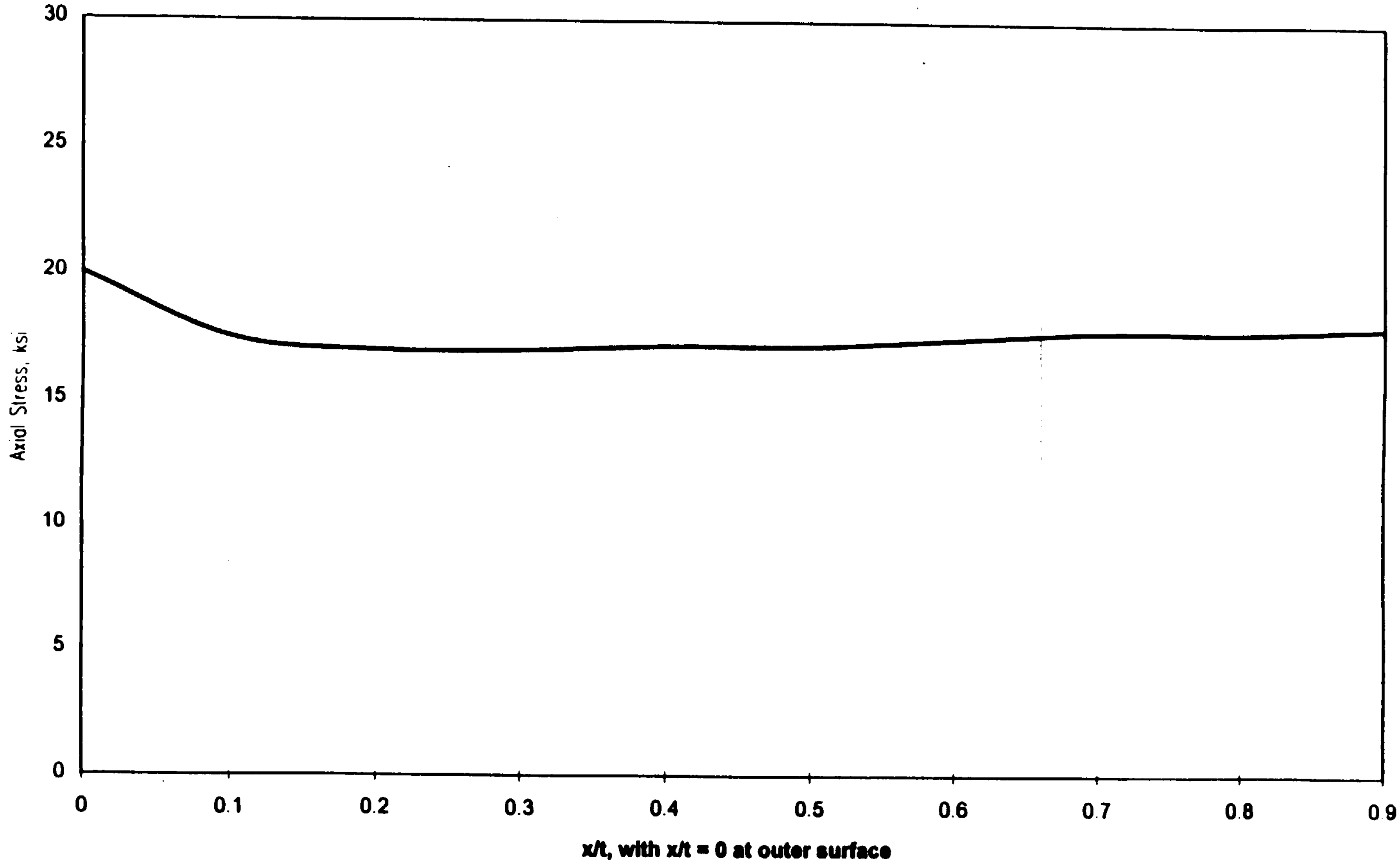


Figure 3 Stress Intensity Factors for Braidwood 2 Upper Shell Transition Weld, Circumferential Outside Surface Flaws, Hydrotest @ 3105psi

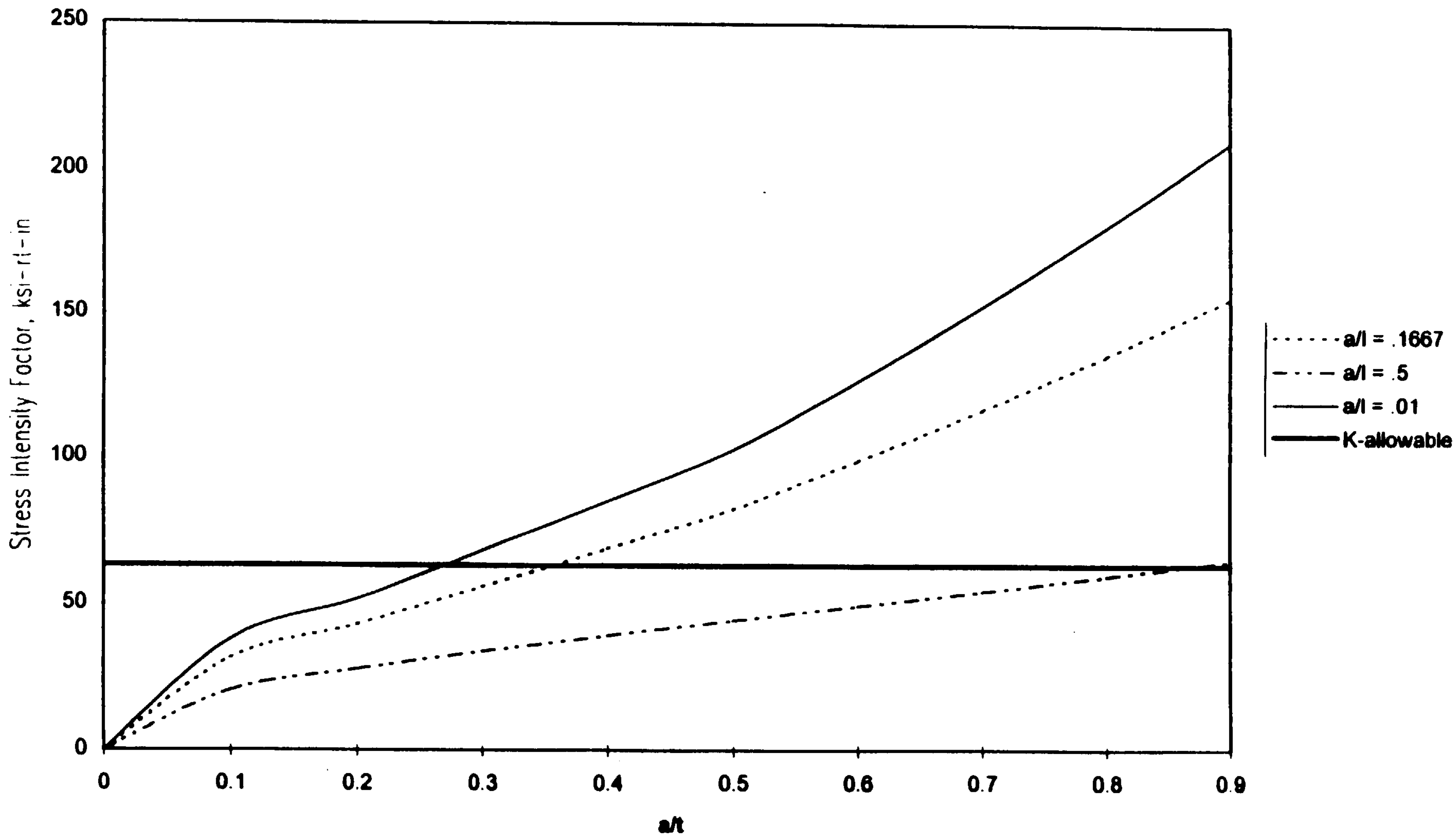


Figure 4 Axial Stress Distribution, Braidwood 2 Upper Shell Transition
Large Steam Break at 100 seconds

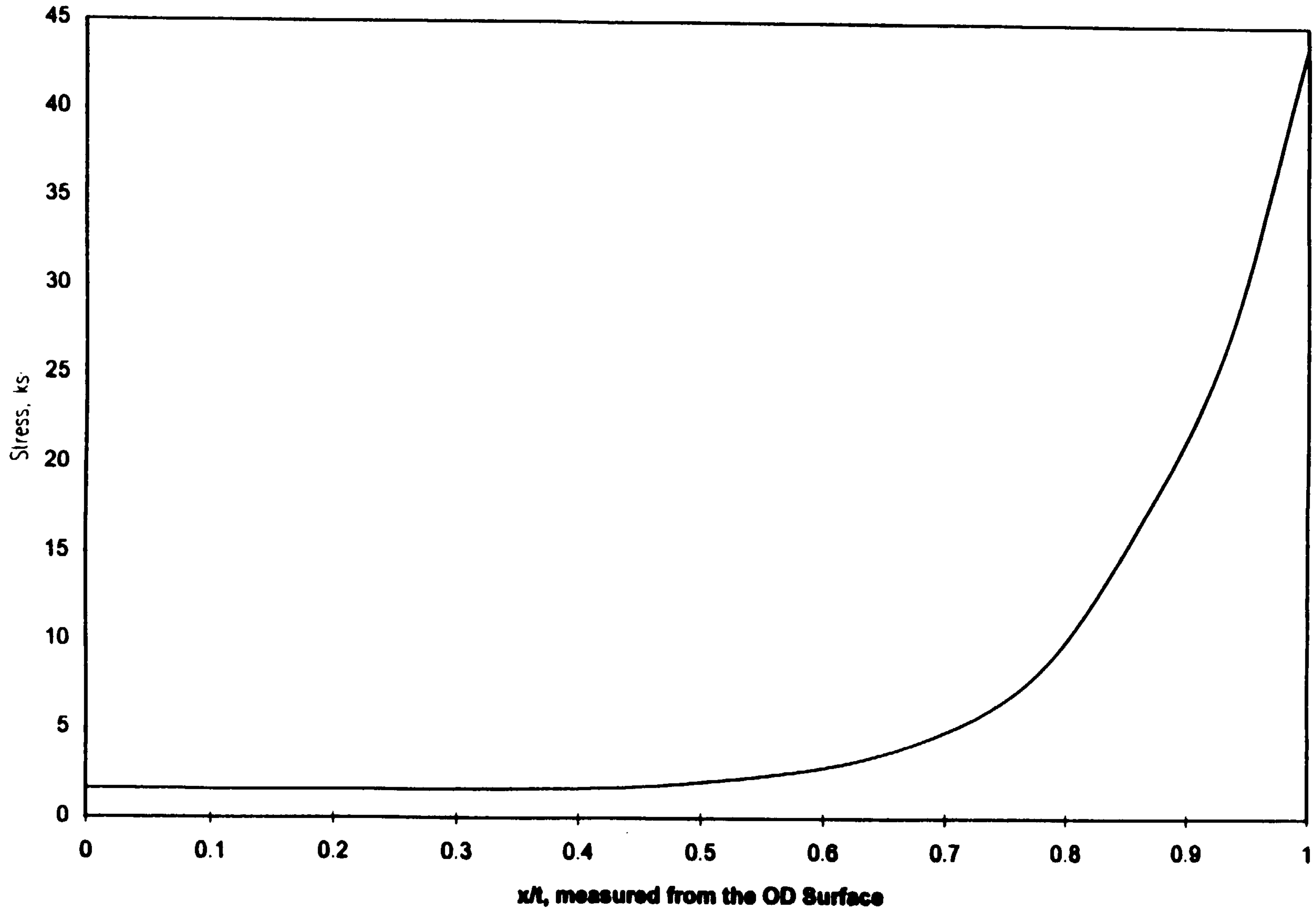
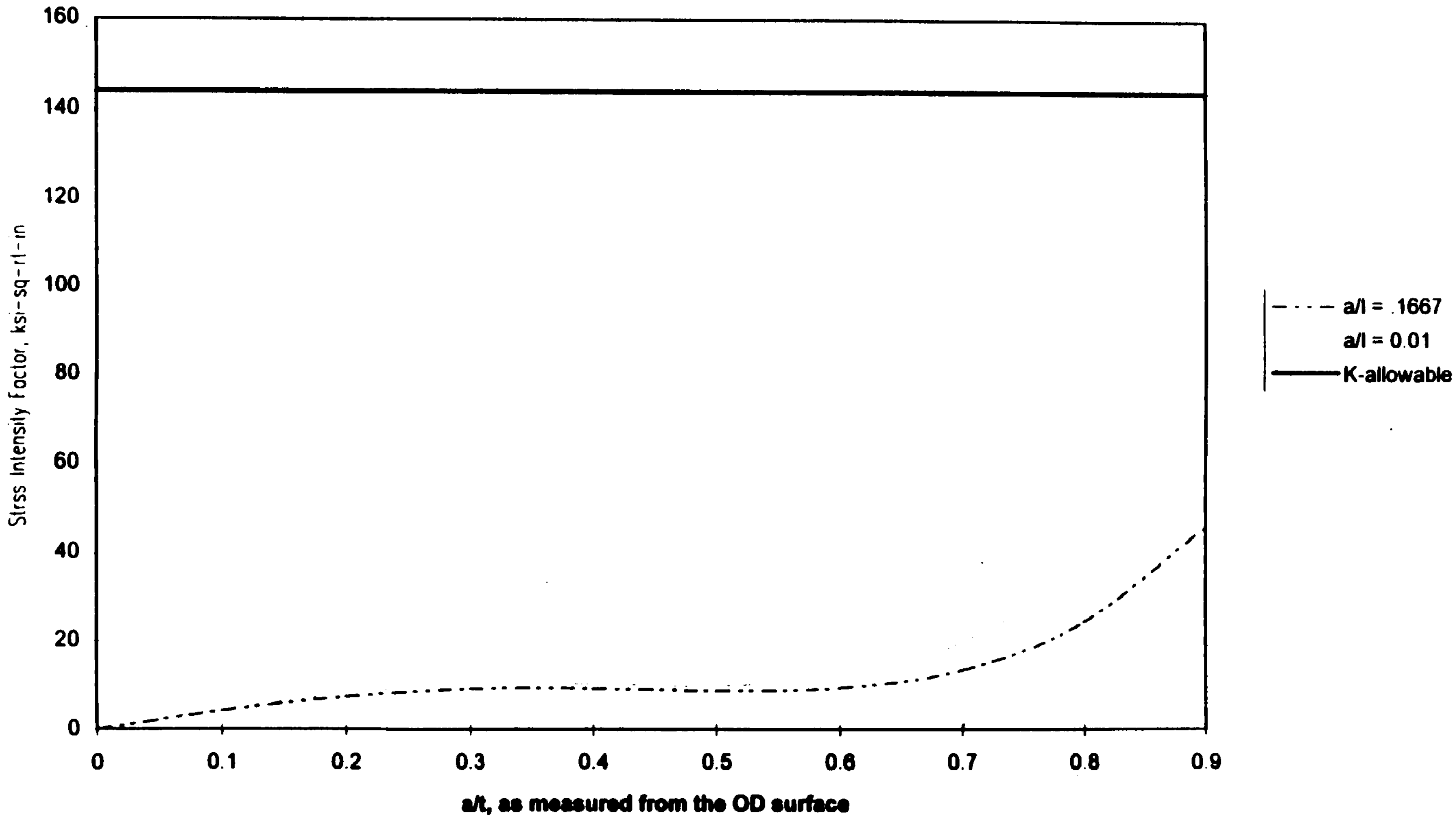


Figure 5
Stress Intensity Factors for Braidwood 2 Upper Shell Transition Weld, Circumferential Outside Surface Flaws, Large Steam Break



APPENDIX 1

TRANSIENT DESCRIPTION FOR THE LARGE STEAM BREAK

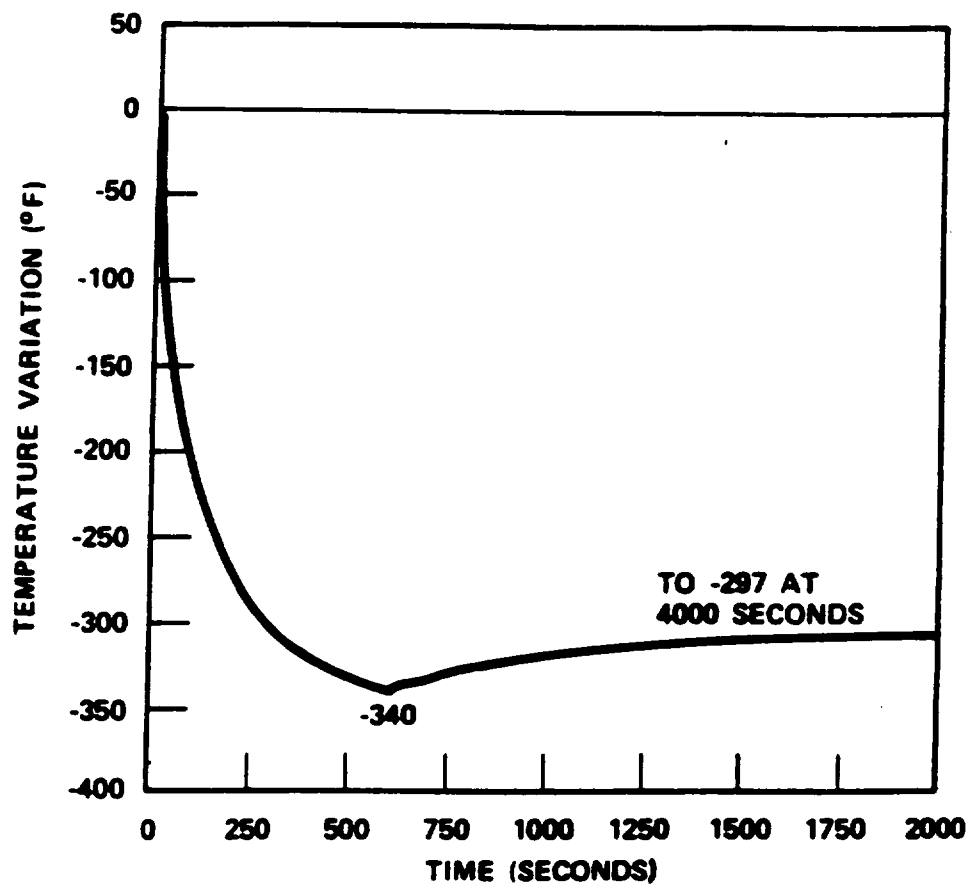


Figure 1. Cold Leg Temperature versus Time During LSB Transient, T_o = 557°F (Failed Loop)

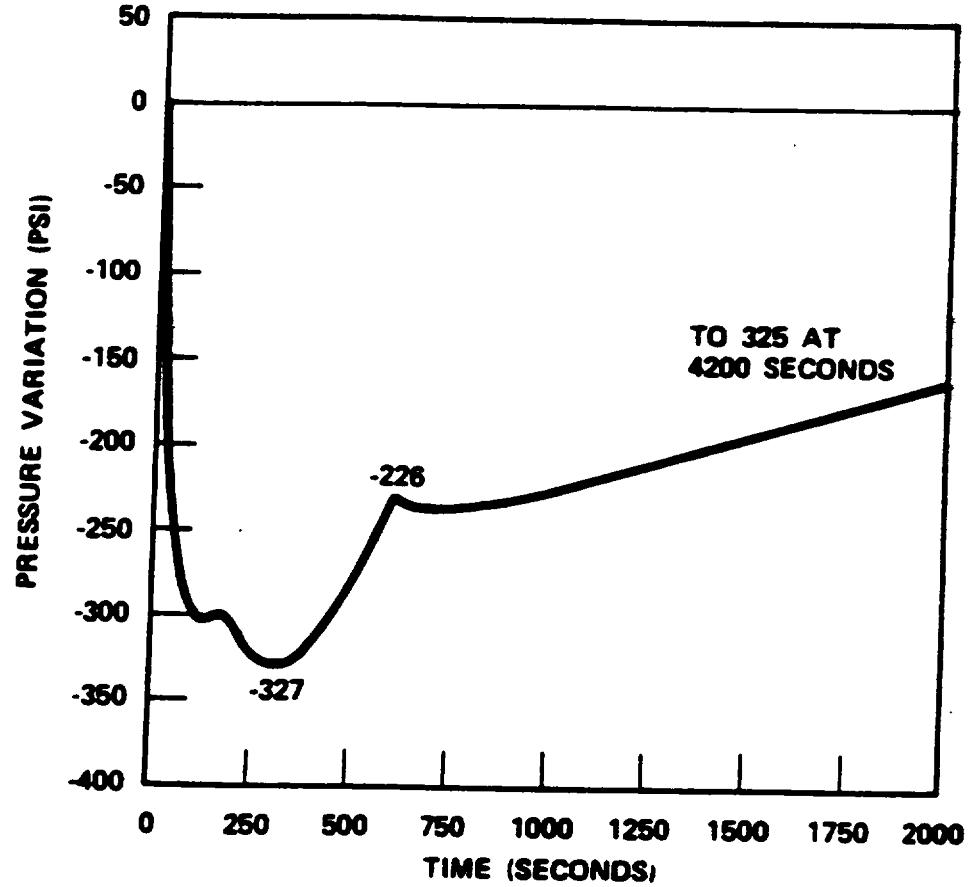


Figure 2. Reactor Coolant Pressure versus Time During LSB Transient, P_o = 2235 PSIG

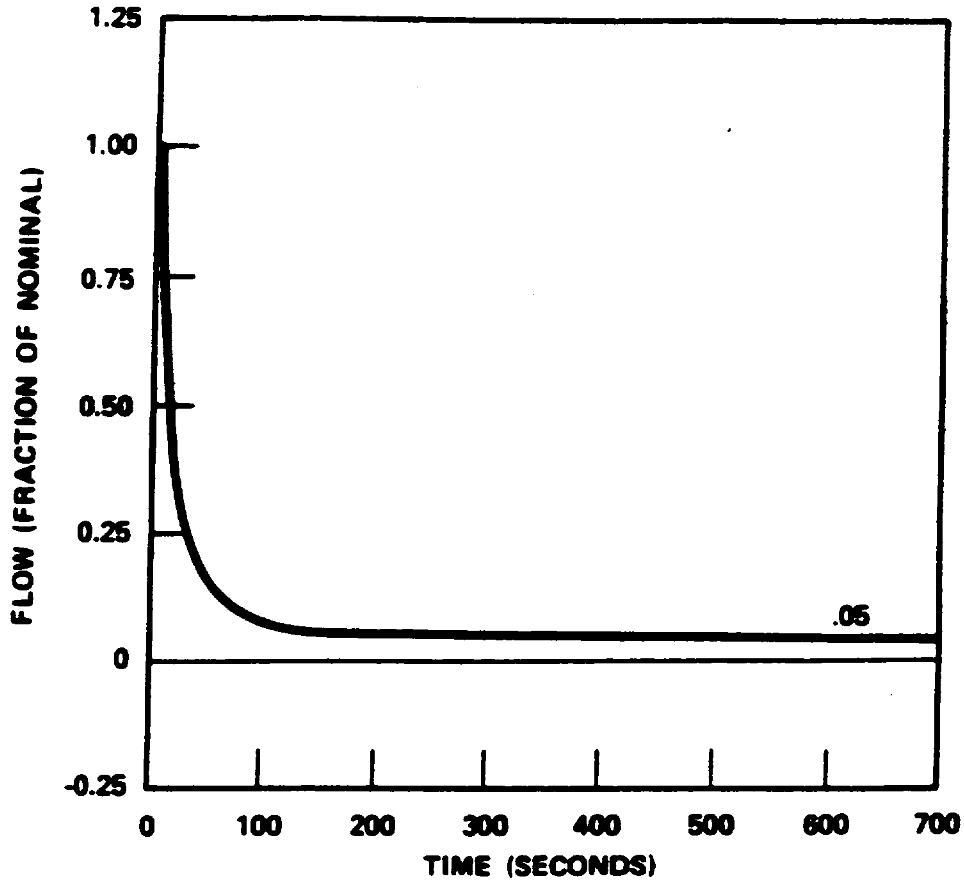


Figure 3. Reactor Coolant Flow versus Time During LSB Transient, F_o = 402400 GPM (Failed Loop)

Attachment 4

Safety Evaluation by the Office of Nuclear Reactor Regulation
Commonwealth Edison Company
Reactor Vessel Shell Weld Indication Evaluation[3]
Braidwood Nuclear Power Station, Unit 2

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[3] Previously provided in a letter from S.N. Bailey (U.S. NRC) to O.D. Kingsley (Commonwealth Edison), "Braidwood Unit 2 Reactor Vessel Inspection Shell Weld Indication Evaluation," dated April 20, 1998.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

COMMONWEALTH EDISON COMPANY

REACTOR VESSEL SHELL WELD INDICATION EVALUATION

BRAIDWOOD NUCLEAR POWER STATION, UNIT 2

DOCKET NO. STN 50-457

1.0 INTRODUCTION

By letter dated October 15, 1997, Commonwealth Edison Company (ComEd) submitted, for NRC review, its evaluation of a flaw in the nozzle shell to intermediate shell weld of the Braidwood, Unit 2, reactor pressure vessel. Additional information was provided by letter dated November 25, 1997. The flaw was found by ultrasonic testing (UT) conducted during the sixth refueling outage (A2R06) in fall 1997. The examination was performed to satisfy the requirements of the 10-year inservice inspection (ISI) in compliance with the 1983 edition of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). The submittal indicates that this flaw exceeds the allowable flaw size specified in IWB-3500 of Section XI of the ASME Code, and requires flaw evaluation using IWB-3610 of the ASME Code.

Instead of performing a plant-specific flaw evaluation, the licensee used a handbook by Westinghouse, WCAP-12046, to determine the acceptability of this flaw. This handbook provides a brief description of the methodology and evaluation charts for various welds in the main coolant system and components of Braidwood, Units 1 and 2, based on the criteria of IWB-3611 and IWB-3612. Fatigue crack growth has also been considered. Meeting the criteria of IWB-3611 or IWB-3612 is confirmed if a point, which was calculated by the user based on the flaw geometry, is within the bound of a limiting curve for certain specified service years in the evaluation chart. This handbook and its supporting document, WCAP-12045, were enclosed as supplements to this submittal.

2.0 EVALUATION

The staff has assessed the licensee's designation of this detected flaw as an embedded flaw using Figure 4 (Figure A-2.6 in WCAP-12046) and the licensee's flaw evaluation using Figure 5 (Figure A-2.5 in WCAP-12046) of the submittal. Since there is no evaluation chart for the embedded flaw at this location, the licensee conservatively used the evaluation chart for a surface flaw (Figure 5) in this application. The staff determined that the licensee applied the handbook charts adequately.

The staff also verified the validity of the handbook charts of Figures 4 and 5. The staff verified that Figure 4, which was used to determine whether an indication is a surface flaw or an

ENCLOSURE

embedded flaw, was developed in accordance with IWA-3000 of Section XI of the ASME Code, and is therefore acceptable. The staff reviewed background and technical basis in WCAP-12045, but required additional information to verify Figure 5. As a result, the licensee provided additional information by letter dated November 25, 1997, detailing the process used to develop the flaw evaluation chart of Figure 5.

The additional information revealed that the most severe transients for outside surface flaws in the upper shell transition region are the pressure transients, i.e., the cold hydro test at 3105 psi for the normal and upset conditions and the large steamline break for the emergency and faulted conditions. The stress distribution for each of the transients was obtained using a detailed finite element model of the RPV. The applied stress intensity factor (applied K) was then calculated through the influence coefficients of Raju-Newman. The allowable fracture toughness for the nozzle shell to the intermediate shell weld at 200°F with an initial RT_{NDT} of -25°F is 200 ksi$(in)^{1/2}$. This toughness value remains unchanged with the time because there is no irradiation effect. Based on the ASME safety margin of $(10)^{1/2}$ for the normal and upset conditions, the allowable depth for the assumed crack geometry of half crack depth to crack length ration (a/l) of 0.01 is 0.269T (T = the RPV wall thickness). Based on the ASME safety margin of $(2)^{1/2}$ for the emergency and faulted conditions, the allowable depth for the same crack geometry (a/l = 0.01) is 1.0T. Therefore, the normal and upset conditions are controlling, and the limiting allowable for the flaw evaluation chart is .269T. Similar calculations were performed for a/l equal to 0.1667 and 0.5, and the results confirmed that a/l of 0.01 is limiting.

The licensee also presented crack growth values corresponding to 10, 20, 30, and 40 years of service. They are negligibly small, and the limiting allowable crack depth remains essentially the same at .269T for the above specified different years of service, making the limiting curves for various service years appear outside the region of concern (crack depth of 20% T) in Figure 5. Consequently, the staff determined that the flaw evaluation methodology presented above is appropriate, and Figure 5 can be used to perform flaw evaluation in this application.

3.0 CONCLUSIONS

Based on the staff's evaluation, the staff concluded that the methodology and criteria used in generating flaw evaluation charts for the RPV of Braidwood 2 (Figures 4 and 5) are in accordance with Section XI of the ASME Code. Further, the staff confirmed that the licensee applied these charts adequately in its flaw evaluation. Hence, the reported flaw is acceptable without repair for continued operation.

Principal Contributor: S. Sheng

Dated: