OG-01-038 NRC Project Number 686

June 15, 2001

To: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: R.K. Anand, Project Manager
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Subject: Westinghouse Owners Group
Response to NRC Request for Additional Information on WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," (MUHP6110)

References: 1. Request For Additional Information, Letter from R. K. Anand to R. A. Newton, April 12, 2001

2. OG-01-033, Planned Date for Completion of Responses to Requests for Additional Information on WCAP-15338, "A review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," May 9, 2001, to Document Control Desk, NRC

Attached are the Westinghouse Owners Group responses to the NRC's Request for Additional Information on the WOG Report WCAP-15388, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," March 2000. The responses are formatted as an appendix to the original WCAP. Please distribute these responses to the appropriate people in your organization for their review.

Reference 2 requested that a final safety evaluation be issued following acceptance of our responses and also described the procedure that will be used by WOG to issue the approved version of the WCAP.

If you have any questions regarding these responses, please contact Warren Bamford, Westinghouse, at (412) 374-6515, or Charlie Meyer, Westinghouse, at (412) 374-5027.

Very truly yours,

Robert H Bryan

Robert H. Bryan, Chairman
Westinghouse Owners Group

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C.I. Grimes, Director, USNRC License Renewal and Standardization Branch (1L, 1A)
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NRC Request for Additional Information Regarding the Review of WCAP-15338

Questions Regarding Vessel Integrity Assessment

Underclad cracks were first discovered in October 1970 during examination of the Atucha reactor vessel. They have been reported to exist only in SA-508 Class 2 reactor vessel forgings manufactured to a coarse grain practice and clad by high-heat-input submerged arc process. The analysis documented in WCAP-15338 evaluates the fatigue crack growth of underclad cracks during 60 years of operation. The analysis documented in WCAP-15338 assumes that fabrication cracks beneath the clad will not penetrate the clad and that the fatigue crack growth could be projected using the ASME Code Section XI reference crack growth law for air environment.

1) Since it can not be ensured that the cracks will not penetrate the clad, the fatigue crack growth evaluation should be performed using the ASME Code Section XI reference crack growth law for water reactor environment. The postulated surface flaw should have an aspect ratio of 6:1 and its depth should include the clad thickness and bound the size of flaws observed during fabrication. Does the 0.295 inch crack depth discussed in the report represent the bounding size of flaw observed during fabrication or does it include the clad thickness?

2) To evaluate reactor pressure vessel integrity:

   a) The projected flaw length at the end of the license renewal period should be evaluated to the criteria in ASME Code Section XI, Paragraph IWB-3600. The fracture mechanics evaluation should include: (1) all forging materials that were susceptible to the under clad cracking (i.e. beltline, nozzle belt, vessel flange etc.), (2) embrittlement of beltline forgings at the end of the license renewal term, (3) cladding effects, (4) axial and circumferential flaw configurations, and (5)normal/upset and emergency/faulted conditions.

   b) The projected flaw length at the end of the license renewal period should be evaluated to demonstrate that the beltline forgings are not susceptible to pressurized thermal shock (PTS) during the license renewal term. The fracture mechanics analysis should be performed using the worst transient from the PTS study of 1982 (the extended HPI transient) a pressure of 2250 psi, and the embrittlement projected for the limiting beltline forging at the end of the license renewal period.

WOG Responses to NRC Questions Regarding Vessel Integrity Assessment

NOTE: The responses have been written in the form of an appendix to the original report, WCAP-15338.

Appendix A
ASME Code Section XI Flaw Evaluation

A-1 Introduction

To ensure that any underclad cracks are acceptable for service, an ASME Code Section XI flaw evaluation was carried out.

This section begins with the acceptance criteria per paragraph IWB-3600 of ASME Code Section XI, which is followed by the fatigue crack growth and allowable flaw size calculations.

A-2 Acceptance Criteria

There are two alternative sets of flaw acceptance criteria for ferritic components, for continued service without repair in paragraph IWB-3600 of ASME Code Section XI. Either of the criteria below may be used, at the convenience of the user.

1. Acceptance Criteria Based on Flaw Size (IWB-3611)
2. Acceptance Criteria Based on Stress Intensity Factor (IWB-3612)

Both criteria are comparable for thick sections, and the acceptance criteria based on stress intensity factor have been determined by past experience to be less restrictive for thin sections, and for outside surface flaws in many cases. In all cases, the most beneficial criteria have been used in the evaluation to be presented here.

A-2.1 Criteria Based on Flaw Size

The code acceptance criteria stated in IWB-3611 of Section XI for ferritic steel components 4 inch and greater in wall thickness are:

$a_f < .1\ a_c$ For Normal Conditions (Upset & Test Conditions inclusive)

and $a_f < .5\ a_i$ For Faulted Conditions (Emergency Condition inclusive)

where

a_f = The maximum size to which the detected flaw is calculated to grow until the next inspection. 15, 30, 45, and 60 year periods have been considered.

a_c = The minimum critical flaw size under normal operating conditions (upset and test conditions inclusive).

a_i = The minimum critical flaw size for initiation of nonarresting growth under postulated faulted conditions (emergency conditions inclusive).

A-2.2 Criteria Based on Applied Stress Intensity Factors

Alternatively, the code acceptance criteria stated in IWB-3612 of Section XI for ferritic steel components criteria based on applied stress intensity factors may be used:

$K_I < \frac{K_{Ia}}{\sqrt{10}}$ For normal conditions (upset & test conditions inclusive)

$K_I < \frac{K_{Ic}}{\sqrt{2}}$ For faulted conditions (emergency conditions inclusive)

where

K_I = The maximum applied stress intensity factor for the final flaw size after crack growth, using 15, 30, 45, and 60 year periods.

K_{Ia} = Fracture toughness based on crack arrest for the corresponding crack tip temperature.

K_{Ic} = Fracture toughness based on fracture initiation for the corresponding crack tip temperature.

A-3 Fatigue Crack Growth

A series of fatigue crack growth calculations was performed to provide a prediction of future growth of unclad cracks for service periods up through 60 years. This is similar to the work carried out in Section 5.4 of this report, however, the crack growth law for water environment was used here for conservatism. The crack growth rate curves used in the analyses were taken directly from Appendix A of Section XI of the ASME Code.

For water environments the reference crack growth curves are shown in Figure 5-1, and growth rate is a function of both the applied stress intensity factor range, and the R ratio (K_{min}/K_{max}) for the transient.

For $R \leq 0.25$

$$\left(\Delta K_I < 19 \text{ ksi} \sqrt{\text{in}}\right), \; \frac{da}{dN} = \left(1.02 \text{ x } 10^{-6}\right) \Delta K_I^{5.95}$$

$$\left(\Delta K_I > 19 \text{ ksi} \sqrt{\text{in}}\right), \; \frac{da}{dN} = \left(1.01 \text{ x } 10^{-3}\right) \Delta K_I^{1.95}$$

where $\frac{da}{dN}$ = Crack Growth rate, micro-inches/cycle.

For $R \geq 0.65$

$$\left(\Delta K_I < 12 \text{ ksi} \sqrt{\text{in}}\right), \ \frac{da}{dN} = \left(1.20 \times 10^{-5}\right) \Delta K_I^{5.95}$$

$$\left(\Delta K_I > 12 \text{ ksi} \sqrt{\text{in}}\right), \ \frac{da}{dN} = \left(2.52 \times 10^{-1}\right) \Delta K_I^{1.95}$$

For R ratio between these two extremes, interpolation is recommended.

The crack growth evaluation was performed for surface flaws in the beltline region of a generic Westinghouse 3-loop reactor vessel. The results are shown in Tables A-3.1 and A-3.2 for postulated axial and circumferential flaw depths ranging from 0.05 inch (1.3mm) to 0.30 inch (7.6mm), which is beyond the 0.295 inch (7.5mm) maximum depth of an underclad cold crack as discussed in Section 2 of the main body of this report. Note that the final flaw depths (a_f's) for the water environment are slightly higher than those presented in Tables 5-1 and 5-2 for air environment.

Table A-3.1: Crack Growth Results for Beltline Region, Axial Flaw (Water Environment)

Initial Flaw Depth (in.)	Depth after 15 years	Depth after 30 years	Depth after 45 years	Depth after 60 years
Flaw shape l/a = 2				
0.050	0.0500	0.0500	0.0500	0.0501
0.125	0.1252	0.1255	0.1257	0.1259
0.200	0.2009	0.2018	0.2027	0.2036
0.250	0.2517	0.2534	0.2550	0.2568
0.300	0.3028	0.3056	0.3084	0.3113
Flaw shape l/a = 6				
0.050	0.0502	0.0504	0.0506	0.0508
0.125	0.128	0.1311	0.1343	0.1379
0.200	0.2071	0.2143	0.2219	0.2302
0.250	0.26	0.2699	0.2799	0.2907
0.300	0.3122	0.3244	0.337	0.3505
Continuous Flaw				
0.050	0.0505	0.0509	0.0514	0.0519
0.125	0.1303	0.1357	0.1415	0.1479
0.200	0.2114	0.2231	0.2354	0.249
0.250	0.2656	0.2817	0.2989	0.3181
0.300	0.3202	0.3413	0.3639	0.3891

Note: Aspect Ratio l/a = flaw length / flaw depth

Table A-3.2: Crack Growth Results for Beltline Region, Circumferential Flaw (Water Environment)

Initial Flaw Depth (in.)	Depth after 15 years	Depth after 30 years	Depth after 45 years	Depth after 60 years
Flaw shape l/a = 2				
0.050	0.0500	0.0500	0.0500	0.0500
0.125	0.1250	0.1250	0.1251	0.1251
0.200	0.2001	0.2001	0.2002	0.2003
0.250	0.2501	0.2502	0.2504	0.2505
0.300	0.3002	0.3004	0.3006	0.3008
Flaw shape l/a = 6				
0.050	0.05	0.05	0.0501	0.0501
0.125	0.1253	0.1255	0.1258	0.126
0.200	0.201	0.2019	0.2028	0.2037
0.250	0.2517	0.2533	0.255	0.2567
0.300	0.3027	0.3053	0.3079	0.3107
Continuous Flaw				
0.050	0.05	0.0501	0.0501	0.0502
0.125	0.1256	0.1262	0.1268	0.1274
0.200	0.2022	0.2044	0.2066	0.2089
0.250	0.2539	0.2577	0.2616	0.2658
0.300	0.3062	0.3123	0.3186	0.3252

Note: Aspect Ratio l/a = flaw length / flaw depth

A-4 Allowable Flaw Size Determination – Normal, Upset & Test Conditions

The allowable flaw size for normal, upset and test conditions was calculated using the criteria in Section A-2.2. The fracture toughness for ferritic steels has been taken directly from the reference curves of Appendix A, Section XI. In the transition temperature region, these curves can be represented by the following equations:

$$K_{Ic} = 33.2 + 2.806 \exp. [0.02 (T\text{-}RT_{NDT} + 100°F)]$$

$$K_{Ia} = 26.8 + 1.233 \exp. [0.0145 (T\text{-}RT_{NDT} + 160°F)]$$

where K_{Ic} and K_{Ia} are in $ksi\sqrt{in}$.

The upper shelf temperature regime requires utilization of a shelf toughness, which is not specified in the ASME Code. A value of 200 $ksi\sqrt{in}$ has been used here. This value is consistent with general practice in such evaluations.

The minimum allowable flaw size was calculated using the most governing transients under normal operating conditions for the location of interest. To select the most governing transients for the beltline region, the stress intensity factors (K_I) for several normal, upset and test conditions were calculated for axial flaws in the beltline region. The axial flaws were chosen since the hoop stresses are higher than the axial stresses in the beltline region, as can be evident from the crack growth results in Tables A-3.1 and A-3.2. The stress intensity factors are plotted in Figures A-4.1 through A-4.3 for three different flaw shapes. Note that several transients were considered for each flaw shape, to ensure that the most governing transient would be chosen. The allowable flaw depth was chosen as the intersection of the stress intensity factor curve with the allowable fracture toughness, which is $200/\sqrt{10} = 63.2\, ksi\sqrt{in}$. The minimum allowable flaw size results for normal, upset and test conditions are provided below:

Table A-4.1: Allowable Flaw Size Summary For Beltline Region – Normal, Upset & Test Conditions

Flaw Shape	Governing Transient	Allowable Flaw Size	
		inches	(a/t)
Aspect Ratio 2:1	Inadvertent Safety Injection	4.07	(0.525)
Aspect Ratio 6:1	Reactor Trip with Cooldown and S.I.	1.34	(0.173)
Continuous Flaw	Excessive Feedwater Flow	0.67	(0.086)

Note: Wall Thickness = 7.75" is used here.

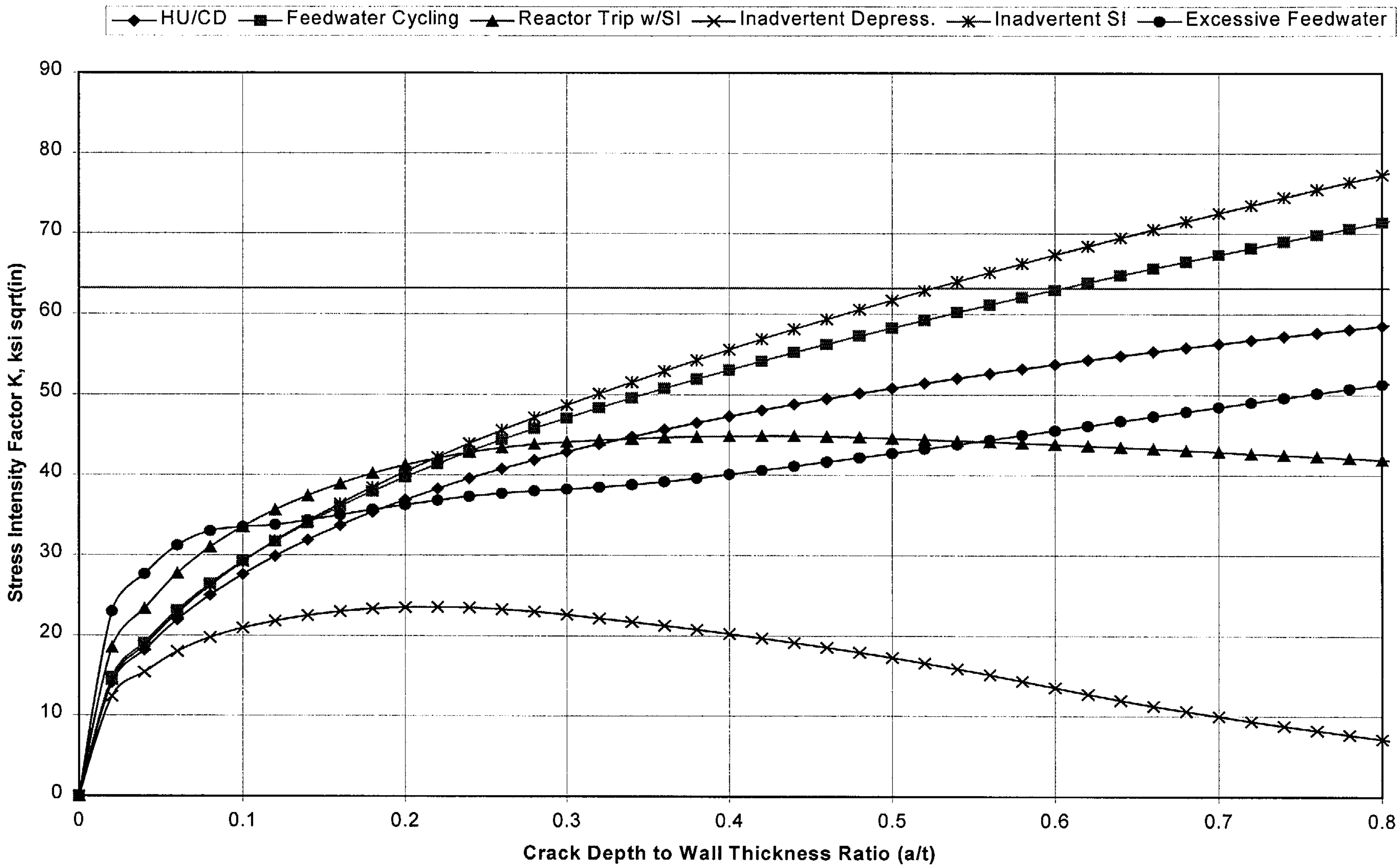

Figure A-4.1 Allowable Flaw Size Determination – Beltline Region, Axial Flaw, AR 2:1

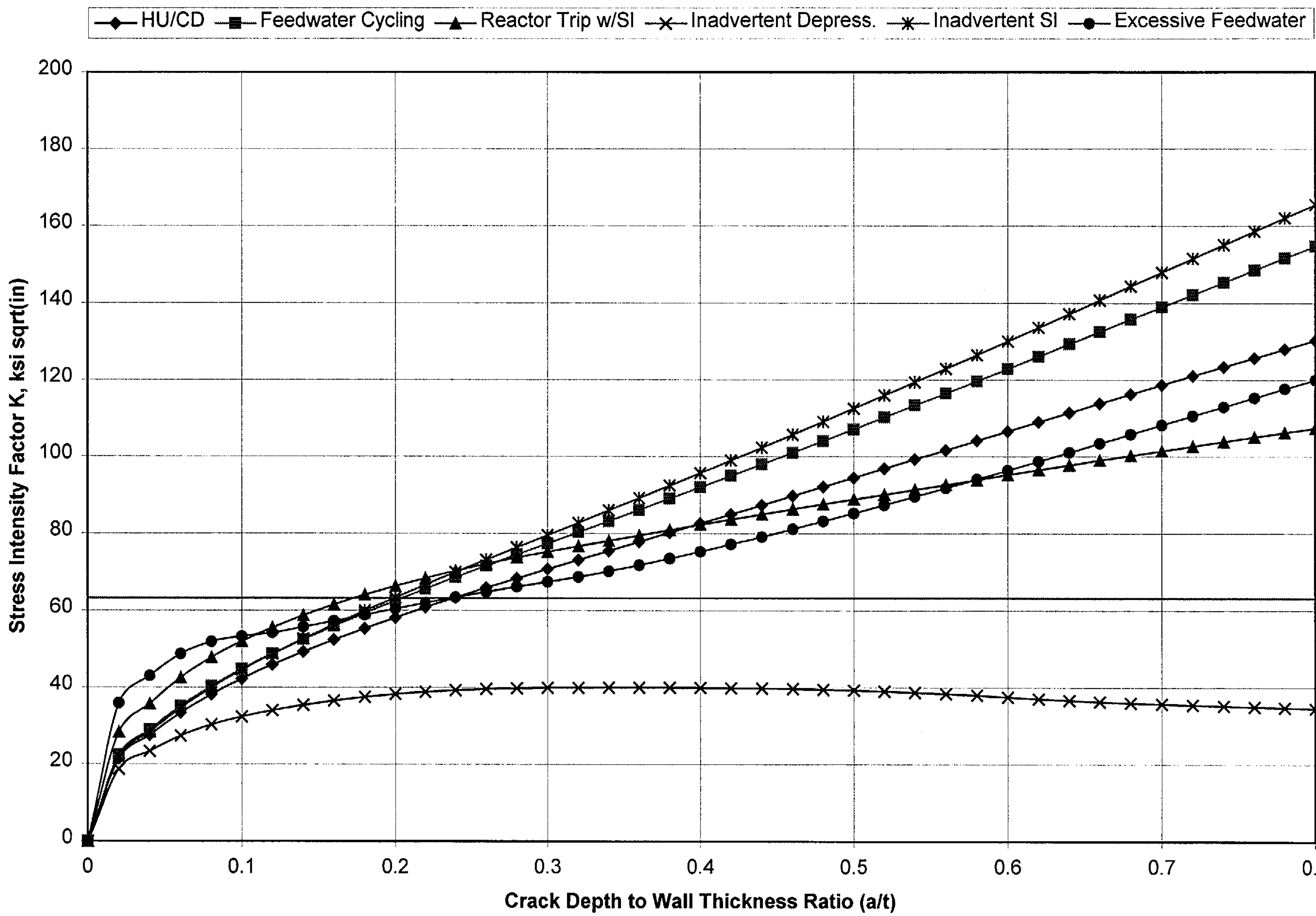

Figure A-4.2 Allowable Flaw Size Determination – Beltline Region, Axial Flaw, AR 6:1

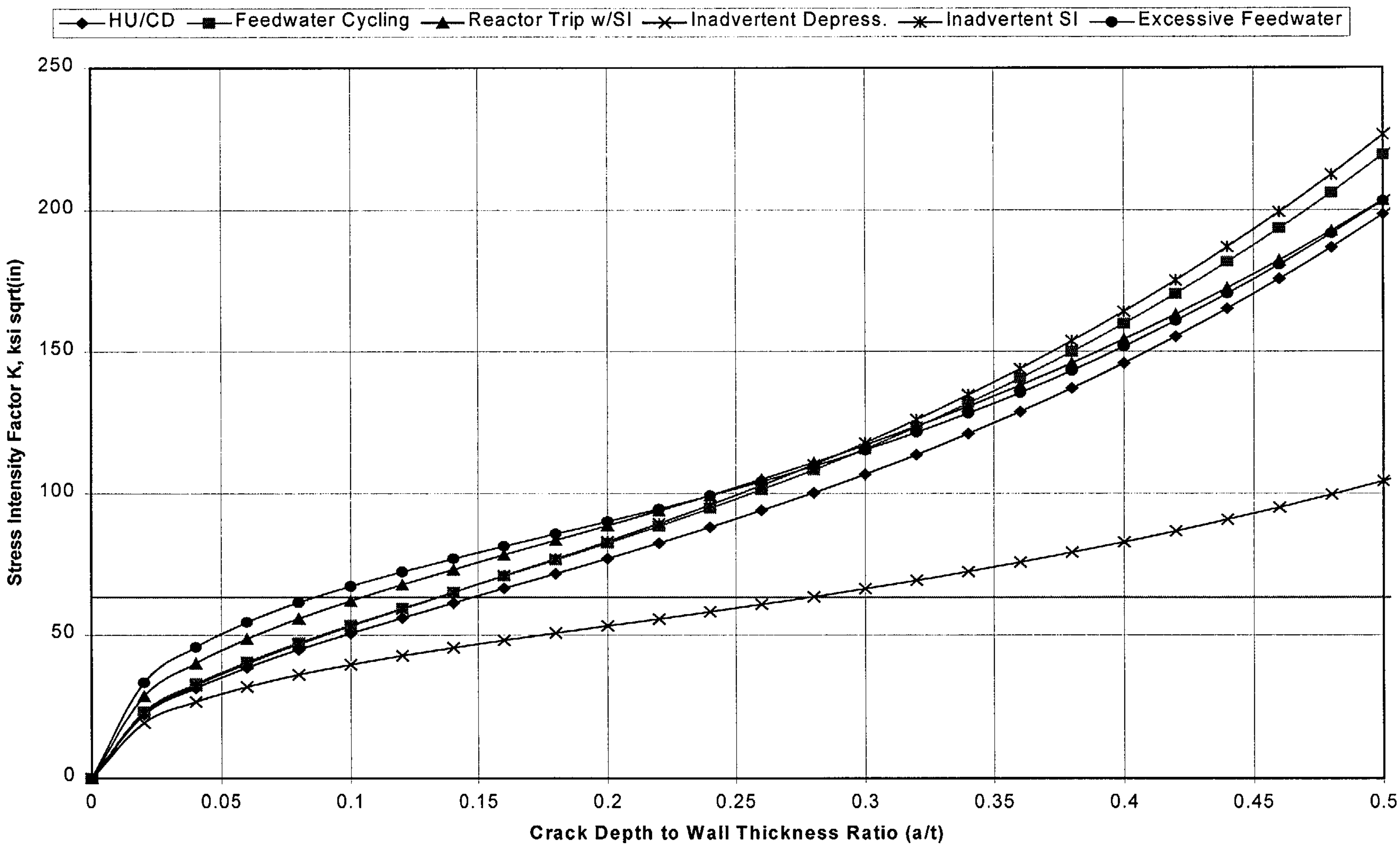

Figure A-4.3 Allowable Flaw Size Determination – Beltline Region, Axial Flaw, Continuous

A-5 Allowable Flaw Size Determination –Emergency & Faulted Conditions

The selection of the governing transient for emergency and faulted conditions was not as straightforward as the selection for normal, upset and test conditions, primarily due to the pressurized thermal shock (PTS) issue. This issue had previously resulted in an extensive probabilistic risk assessment study by Westinghouse Owners' Group (WOG) to identify the overall risk from PTS on a typical Westinghouse plant. The study included all transients that could potentially result in a pressurized thermal shock of the reactor vessel. The summary of the WOG risk assessment for PTS showed that the key contributors to the total risk occur from small LOCA and steam generator tube rupture (SGTR), because of the combination of severe transient characteristics with relatively high frequencies of transient occurrence.

The ASME Code in its present form, however, does not take transient frequencies into consideration and requires an evaluation of flaw indications using the most limiting emergency/faulted condition transient. Therefore, the WOG PTS risk analysis results could not be used directly, but they were used to guide the determination of the key transients to be considered here.

To determine the governing emergency and faulted conditions for a generic Westinghouse 3-loop reactor vessel, a series of transients were studied. These transients included the large LOCA and large steamline break (LSB) and the dominating transients from the WOG pressurized thermal shock studies. This work led to the conclusion that the following transients should be considered in the deterministic assessments for the beltline region:

- Steam Generator Tube Rupture (SGTR)
- Small LOCA
- Large LOCA
- Large Steamline Break (LSB)

Thermal stress and fracture analyses were performed for the beltline region, utilizing the characteristics of the above four transients. The limiting circumferential and axial flaws were used in performing the fracture analyses. The resulting critical flaw depths for a range of shapes are shown in Table A-5.1.

From Table A-5.1, it may be seen that the large steamline break transient is the governing transient for the beltline region. The detailed assessments performed for the tube rupture and small LOCA transients serve to verify this conclusion. Also, from the standpoint of total risk, it is worthy of note that these latter two transients are the dominant ones. Section XI of the ASME Code presently requires that only the most severe transient be evaluated, regardless of its probability of occurrence, so the large steamline break is the governing transient for the beltline region. Therefore, using the criteria in Section A-2.1, the minimum allowable flaw size for emergency and faulted conditions is summarized in Table A-5.2, for axial flaws.

Table A-5.1: Critical Flaw Size Summary For Beltline Region – Emergency & Faulted Conditions

Emergency/Faulted Condition	Flaw Orientation	Continuous Flaw (inches)	Continuous Flaw (a/t)	Apect Ratio 6:1 (inches)	Apect Ratio 6:1 (a/t)	Aspect Ratio 2:1 (inches)	Aspect Ratio 2:1 (a/t)
Steam Generator Tube Rupture	Longitudinal	$a_i = 2.50$	0.323	$a_i = 5.51$	0.711	$a_i = 7.75$	1.000
	Circumferential	$a_i = 7.75$	1.000	$a_i = 7.75$	1.000	ai = 7.75	1.000
Large Steamline Break	Longitudinal	$a_i = 2.50$	0.323	$a_i = 3.39$	0.437	ai = 7.75	1.000
	Circumferential	$a_i = 2.21$	0.285	$a_i = 7.75$	1.000	ai = 7.75	1.000
Small LOCA	Longitudinal	$a_i = 2.56$	0.330	$a_i = 5.74$	0.741	ai = 7.75	1.000
	Circumferential	$a_i = 7.75$	1.000	$a_i = 7.75$	1.000	ai = 7.75	1.000
Large LOCA	Longitudinal	$a_i = 7.75$	1.000	$a_i = 7.75$	1.000	ai = 7.75	1.000
	Circumferential	$a_i = 7.75$	1.000	$a_i = 7.75$	1.000	ai = 7.75	1.000

Note: Wall Thickness = 7.75” is used here.

Table A-5.2: Allowable Axial Flaw Size Summary For Beltline Region – Emergency and Faulted Conditions

Flaw Shape	Allowable Flaw Size	
	inches	(a/t)
Aspect Ratio 2:1	3.88	(0.501)
Aspect Ratio 6:1	1.70	(0.219)
Continuous Flaw	1.25	(0.162)

Note: Wall Thickness = 7.75" is used here.

A-6 Summary and Conclusions

Underclad cracks found during pre-service and inservice inspections have been evaluated in accordance with the acceptance criteria of the ASME Code Section XI. Underclad cracks are very shallow, confined in depth to less than 0.295 inch and have lengths up to 2.0 inches. The fatigue crack growth assessment for these small cracks shows very little extension over 60 years, even if they were exposed to the reactor water. For the worst case scenario, a 0.30 inch deep continuous axial flaw in the beltline region would grow to 0.39 inch after 60 years. The minimum allowable axial flaw size for normal, upset and test conditions is 0.67 inch and for emergency and faulted conditions is 1.25 inches. Since the allowable flaw depths far exceed the maximum flaw depth after 60 years of fatigue crack growth, we may conclude that underclad cracks of any shape are acceptable for service for 60 years, regardless of the size or orientation of the flaws. Therefore, it may be concluded that undeclad cracks are of no concern relative to structural integrity of the reactor vessel for a period of 60 years.