PRESSURIZED WATER REACTOR OWNERS GROUP

PWROG-17031-NP
Revision 1

WESTINGHOUSE NON-PROPRIETARY CLASS 3

Update for Subsequent License Renewal: WCAP-15338-A, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants"

Materials Committee

PA-MSC-1497

May 2018

Westinghouse

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May 2018

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FirstEnergy Nuclear Operating Co.	Beaver Valley 1 & 2 (W)		X
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Wolf Creek Nuclear Operating Co.	Wolf Creek (W)		X
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EDF Energy	Sizewell B (W)		X
Electrabel	Doel 1, 2 & 4 (W)		X
	Tihange 1 & 3 (W)		X
Electricite de France	58 Units		X
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	Hanbit 3, 4, 5 & 6 (CE)		X
	Hanul 3, 4 , 5 & 6 (CE)		X
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1 Background and Introduction

As discussed in WCAP-15338-A [1], underclad cracking was initially detected at the Rotterdam Dockyard Manufacturing (RDM) Company during magnetic particle inspections of a reactor vessel in January 1971. These inspections were performed as part of an investigation initiated by RDM as a result of industry observations reported in December 1970. Subsequent evaluations by Westinghouse in the 1970s concluded that these underclad cracks would not have an impact on the integrity of reactor vessels for a full 40 years of operation. The evaluation was submitted to the Atomic Energy Commission in 1972, and the AEC review concurred. This type of underclad cracking is now commonly referred to as reheat cracking.

In late 1979, underclad cracking in reactor vessels resurfaced in the form of "cold cracking". Supplemental inspections confirmed that such cracking existed in a select group of reactor vessels. Fracture evaluations of the detected flaw indications confirmed their acceptability for a 60 year design life [1].

The purpose of this Topical Report (TR) is to update the 60 year fatigue crack growth analysis in [1] and confirm that the analysis is applicable to subsequent license renewal (SLR), up to 80 years of operation. The fracture toughness values used in Appendix A of [1] will be confirmed for 80 years of operation. Operating experience that is contained in Sections 2 and 3 of [1] will also be updated.

This TR is applicable to all Westinghouse Nuclear Steam Supply System (NSSS) plants.

Revision 1 of this TR removes unnecessary contents that are duplicates in WCAP-15338-A [1]. All evaluation results and conclusions are unchanged from Revision 0 of this TR.

2 Mechanisms of Cracking Associated with Weld Deposited Cladding

As discussed in WCAP-15338-A [1] and repeated here, underclad cracking was initially detected in 1970, and has been extensively investigated by Westinghouse and others over the past 30 years. This type of cracking in reactor vessels has also been identified in France and Japan, in addition to the United States.

The cracking has occurred in the low alloy steel base metal heat-affected zone (HAZ) beneath the austenitic stainless steel weld overlay that is deposited to protect the ferritic material from corrosion. Two types of underclad cracking have been identified.

Reheat cracking has occurred as a result of post weld heat treatment of single-layer austenitic stainless steel cladding applied using high-heat-input welding processes on ASME SA-508, Class 2 forgings. The high-heat-input welding processes effecting reheat cracking, based upon tests of both laboratory samples and clad nozzle cutouts, include: strip clad, six-wire clad and manual inert gas (MIG) cladding processes. Testing also confirmed that reheat cracking did not occur with one-wire and two-wire submerged arc cladding processes. The cracks are often numerous and are located in the base metal region directly beneath the cladding. They are confined to a region approximately 0.125 inch deep and 0.4 inch long.

Cold cracking has occurred in ASME SA-508, Class 3 forgings after deposition of the second and third layers of cladding, where no pre-heating or post-heating was applied during the cladding procedure. The cold cracking was determined to be attributable to residual stresses near the yield strength in the weld metal and base metal interface after cladding deposition, combined with a crack-sensitive microstructure in the HAZ and high levels of diffusible hydrogen in the austenitic stainless steel or Inconel weld metals. The hydrogen diffused into the HAZ and caused cold (hydrogen-induced) cracking as the HAZ cooled. Destructive analyses have demonstrated that these cracks vary in depth from 0.007 inch to 0.295 inch and in length from 0.078 inch to 2.0 inches. Typical cold crack dimensions were 0.078 inch to 0.157 inch in depth, and 0.196 inch to 0.59 inch in length. As with the reheat cracks, these cracks initiate at or near the clad/base metal fusion line and penetrate into the base metal.

3 Plant Experience with Defects in and under the Weld-deposited Cladding

In Section 3 of WCAP-15338-A [1], the historic operating experiences were discussed in detail. Additional operating experiences since the publication of [1] are discussed in this section.

3.1 PWR Service Experience Since 1999

A review of the recent service experience resulting from degraded cladding was performed and very few new instances were identified. The three cases discussed below are the only known new cases [3] and [4]. Plants cited in WCAP-15338-A [1] which are still in operation continue to experience no detrimental effects of the missing cladding. Therefore, it has been shown to be acceptable even if underclad cracks become a surface crack exposing the base metal to reactor coolant system (RCS) fluid.

1. Callaway Reactor Vessel Bottom Head Region

   An indication in the cladding region at the bottom of the reactor vessel was identified visually, due to a rust stain that was indicative of exposed low alloy steel. The indication was determined to encompass an area of 1.5 inch x 0.75 inch. The location was characterized as 302.94 degrees from the vessel "0" location, and 384.89 degrees from the flange surface. The plant has operated since 2004 with no issues, as verified by three separate inspections, each of which involved removing the core barrel.

2. Diablo Canyon Unit 1 Reactor Vessel Inlet Nozzle

   During the 2005 inspection of the Diablo Canyon Unit 1 inlet nozzle inner radius, a visual examination identified an area of approximately 1.025 inch x 0.53 inch of clad scraping (spall) at 10 degrees from the bottom dead center of the nozzle. This particular region was re-examined visually in 2014, and it was determined that there was no noticeable change in the past 9 years. No degradation was identified, nor was it expected, as the PWR RCS is de-oxygenated by the hydrogen overpressure which is present during operation.

3. Qinshan Reactor Vessel Bottom Head Region

   Indications were discovered in the bottom head region of Qinshan Phase 1 reactor vessel when it was examined in 1999. As discussed in [4], it was unclear whether the base metal was exposed. Due to the irregularity of the surface in the vicinity of the indication, a replication was made of the area and the shape of the degradation scar was determined by a laser scan. Since the original examination, the region has been examined three times, and no change has been observed.

   The evaluation in [4] concluded that Qinshan is safe to operate until 2041 as requested, a total of 50 years (end of design life).

4 Effects of Cladding on Fracture Analysis

The effects of cladding on the fracture analysis were discussed in detail in Section 4 of WCAP-15338-A [1]. Experiments were performed and measurements were taken. Fracture analyses of reactor pressure vessels subjected to thermal shock have included various assumptions regarding the behavior of the cladding and its influence on the fracture resistance of the vessel. The effect of cladding is also important because of its relevance to underclad cracks. For the most part, it was assumed that the welded clad layer, being lower in strength and higher in ductility than the low-alloy pressure vessel steel, would produce no observable effect on the strength or apparent fracture toughness of the pressure vessel. The clad layer is assumed to have a sufficient strength to reduce the stress intensity factor, or crack driving force.

As discussed in Section 4 of [1], bend bar tests were conducted to study the effect of cladding on the structural behavior in the operating reactor vessels. The residual stress measurements were discussed in [1] in detail. The residual stress measurement confirmed the bend bar test results. It was concluded in [1] that the effects of cladding will be more important at lower temperatures, where the stresses are higher. At temperatures greater than 180°F (82°C) the cladding has virtually no impact on fracture behavior, and this is the very lower end of the temperature range of plant operation. The effects of the cladding are considered for flaws that penetrate the cladding into the base metal. The actual impact of the cladding residual stress on the fracture evaluation is negligible, even for irradiated materials.

5 Vessel Integrity Assessment

This section discusses the reactor vessel integrity evaluation and assessment.

5.1 Potential for Inservice Exposure of the Vessel Base Metal To Reactor Coolant Water

As discussed in Section 5.1 of WCAP-15338-A [1], the occurrence of wastage or wall thinning of the carbon steel vessel base metal requires the breaching of the complete thickness of the cladding so that the base metal is exposed to the RCS environment. This process consists of two sequential stages:

1. Cracking and separation of a portion of the clad weld metal resulting in the exposure of the base metal to the primary water, and
2. Corrosive attack and wastage of the carbon steel base metal due to its exposure to the RCS water

Delamination and separation of the complete clad thickness can occur either by mechanical distress or by micro-cracking induced by metallurgical degradation mechanisms. Examples of mechanical distress are denting and overload (overloads can result in metal plasticity and cracking) cracking caused by mechanical impact loads such as those caused by a loose part. Metallurgical mechanisms include intergranular stress corrosion cracking (IGSCC) and transgranular stress corrosion cracking (TGSCC) mechanisms.

IGSCC of the clad metal can occur if the weld is sensitized (chromium depleted grain boundaries) and is exposed to oxygenated water. TGSCC can occur in the cladding only in the presence of a chloride environment. The typical PWR operating and shut down RCS chemistry contains oxygen and chloride levels that are significantly below the threshold levels required to initiate either IGSCC or TGSCC.

Thus there is no degradation mechanism that can contribute to additional breaching of the clad thickness and result in any exposure of the vessel base metal. Even if the base metal were exposed, the degree of corrosive attack and wastage due to operation is insignificant based on operating experience and analyses based on corrosion tests.

5.2 Fatigue Usage

As reported in WCAP-15338-A [1], the maximum cumulative fatigue usage factor for the reactor vessel is 0.04 or less for 60 years of operation. Assuming transient cycles linearly scale from 60 to 80 years, the maximum usage factor would be 0.053. This shows that the likelihood of fatigue cracks initiating during service is very low for 80 years of operation.

5.3 Acceptance Criteria

5.3.1 ASME Section XI – IWB-3500

The underclad cracks which have been identified over the years are very shallow, with a maximum depth of 0.295 inch (7.5mm). The flaw indications indicative of underclad cracks that have been identified during pre-service and inservice inspections are all within the flaw acceptance standard of the ASME Code Section XI, Paragraph IWB-3500. However, the USNRC RAI [1, Section 8] stated that the ASME Section XI IWB-3600 criteria should be used as evaluation criteria. Westinghouse provided a response to this RAI question and the USNRC accepted the response in a Safety Evaluation Report (SER) issued on September 25, 2002. The accepted response is included in Appendix A of WCAP-15338-A [1].

5.3.2 ASME Section XI – IWB-3600

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 can be used as discussed in Appendix A of WCAP-15338-A [1].

(1) Acceptance criteria based on flaw size (IWB-3611)

(2) Acceptance criteria based on stress intensity factor, K_I (IWB-3612)

Both criteria are comparable for thick sections, and the acceptance criteria based on the 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 discussed below.

5.3.2.1 Criteria Based on Flaw Size

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

$a_f < 0.1\, a_c$ for normal conditions (including upset and test conditions) and,

$a_f < 0.5\, a_i$ for faulted conditions (including emergency conditions)

where,

a_f = The maximum size to which the detected flaw is calculated to grow until the next inspection. An 80 year period is considered in the calculation herein.

a_c = The minimum critical flaw size under normal operating conditions.

a_i = The minimum critical flaw size for initiation of non-arresting growth under postulated faulted conditions.

5.3.2.2 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 can be used:

$K_I < \frac{K_{Ia}}{\sqrt{10}}$ for normal conditions (including upset and test conditions)

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

where,

K_I = the maximum applied stress intensity factor for the final flaw size after crack growth.

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.

5.4 Fatigue crack growth

A series of fatigue crack growth (FCG) calculations were performed to provide a prediction of future growth of underclad cracks for service periods up to 60 years in [1]. The 60-year FCG calculation was revised and updated for the 80-year SLR application in this TR.

To complete the fatigue crack growth analysis, the methodology of Section XI of the ASME Code was used with the entire set of design transients applied over an 80 year period. The cycles applicable to 40 years of operation were conservatively multiplied by a factor of 2 to account for 80 years of operation. The analysis assumes a flaw of a specified size and shape, considers each design transient, and calculates the crack growth, adding the crack growth increment to the original flaw size, and then repeating the process until all transient cycles have been accounted for.

The crack growth was conservatively calculated using the ASME Section XI, Appendix A, A-4300, crack growth rate for water environments [2]. This is the most current crack growth rate and is comparable to the rate used in the original analysis in [1], which dates back to the ASME Code in 1979. This crack growth rate is shown in Figure 5-1. Even though the underclad cracks are not exposed to the PWR water environment, the water crack growth rate was used for conservatism.

A series of flaw types were postulated to address the various possible shapes for the underclad cracks. Specifically, the postulated flaw depths ranged 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. The shape of the flaws analyzed (flaw depth/flaw length) ranged from 0.01 through 0.5. The results are shown in Table 5-1 through Table 5-3. The maximum flaw size of 0.4267 inch at the end of 80 years is less than the minimum allowable flaw size of 0.67 inch, presented in Section 5.5.

Therefore, it can be concluded that the crack growth is insignificant for any type of flaw which might exist at the clad/base metal interface and into the base metal for both nozzle bore and vessel shell regions.

Table 5-1: Fatigue Crack Growth Result for Beltline Region, Axial Flaw (Water Environment)

Initial Flaw Depth	Depth after 20 years	Depth after 40 years	Depth after 60 years	Depth after 80 years
Flaw Shape AR = l/a = 2				
0.050	0.0504	0.0504	0.0504	0.0504
0.125	0.1256	0.1263	0.1263	0.1271
0.200	0.2023	0.2038	0.2054	0.2077
0.250	0.2534	0.2573	0.2612	0.2651
0.300	0.3046	0.3092	0.3147	0.3193
Flaw Shape AR = l/a = 6				
0.050	0.0504	0.0512	0.0512	0.0519
0.125	0.1302	0.1349	0.1403	0.1465
0.200	0.2108	0.2224	0.2341	0.2472
0.250	0.2643	0.2790	0.2945	0.3116
0.300	0.3178	0.3364	0.3557	0.3767
Continuous Flaw (l/a = 100)				
0.050	0.0507	0.0513	0.0520	0.0527
0.125	0.1323	0.1399	0.1481	0.1578
0.200	0.2156	0.2318	0.2495	0.2693
0.250	0.2713	0.2937	0.3187	0.3469
0.300	0.3277	0.3569	0.3895	0.4267

Note: Aspect Ratio l/a = flaw length / flaw depth. Depths are in inches.

Table 5-2: FCG Results for Beltline Region, Circumferential Flaw in Water

Initial Flaw Depth	Depth after 20 years	Depth after 40 years	Depth after 60 years	Depth after 80 years
Flaw Shape AR = l/a = 2				
0.050	0.0504	0.0504	0.0504	0.0504
0.125	0.1250	0.1256	0.1256	0.1256
0.200	0.2000	0.2007	0.2007	0.2015
0.250	0.2503	0.2511	0.2519	0.2519
0.300	0.3007	0.3015	0.3023	0.3030
Flaw Shape AR = l/a = 6				
0.050	0.0504	0.0504	0.0504	0.0504
0.125	0.1263	0.1271	0.1279	0.1287
0.200	0.2031	0.2062	0.2093	0.2124
0.250	0.2550	0.2604	0.2658	0.2720
0.300	0.3077	0.3147	0.3216	0.3294
Continuous Flaw (l/a = 100)				
0.050	0.0501	0.0502	0.0503	0.0504
0.125	0.1265	0.1278	0.1291	0.1305
0.200	0.2043	0.2083	0.2124	0.2167
0.250	0.2573	0.2646	0.2721	0.2801
0.300	0.3106	0.3208	0.3315	0.3429

Note: Aspect Ratio l/a = flaw length / flaw depth. Depths are in inches.

Table 5-3: FCG Results for Inlet Nozzle to Shell Weld, Axial Flaw in Water

Initial Flaw Depth	Depth after 20 years	Depth after 40 years	Depth after 60 years	Depth after 80 years
Flaw Shape AR = l/a = 2				
0.050	0.0500	0.0500	0.0500	0.0505
0.125	0.1253	0.1253	0.1253	0.1253
0.200	0.2001	0.2011	0.2011	0.2011
0.250	0.2506	0.2506	0.2517	0.2517
0.300	0.3012	0.3022	0.3022	0.3033
Flaw Shape AR = l/a = 6				
0.050	0.0505	0.0505	0.0505	0.0505
0.125	0.1264	0.1274	0.1274	0.1285
0.200	0.2032	0.2064	0.2095	0.2127
0.250	0.2559	0.2611	0.2664	0.2717
0.300	0.3085	0.3159	0.3243	0.3327
Continuous Flaw (l/a = 100)				
0.0500	0.0502	0.0503	0.0505	0.0506
0.1250	0.1271	0.1287	0.1303	0.1321
0.2000	0.2059	0.2111	0.2164	0.2222
0.2500	0.2597	0.2693	0.2796	0.2908
0.3000	0.3141	0.3276	0.3419	0.3576

Note: Aspect Ratio l/a = flaw length / flaw depth. Depths are in inches.

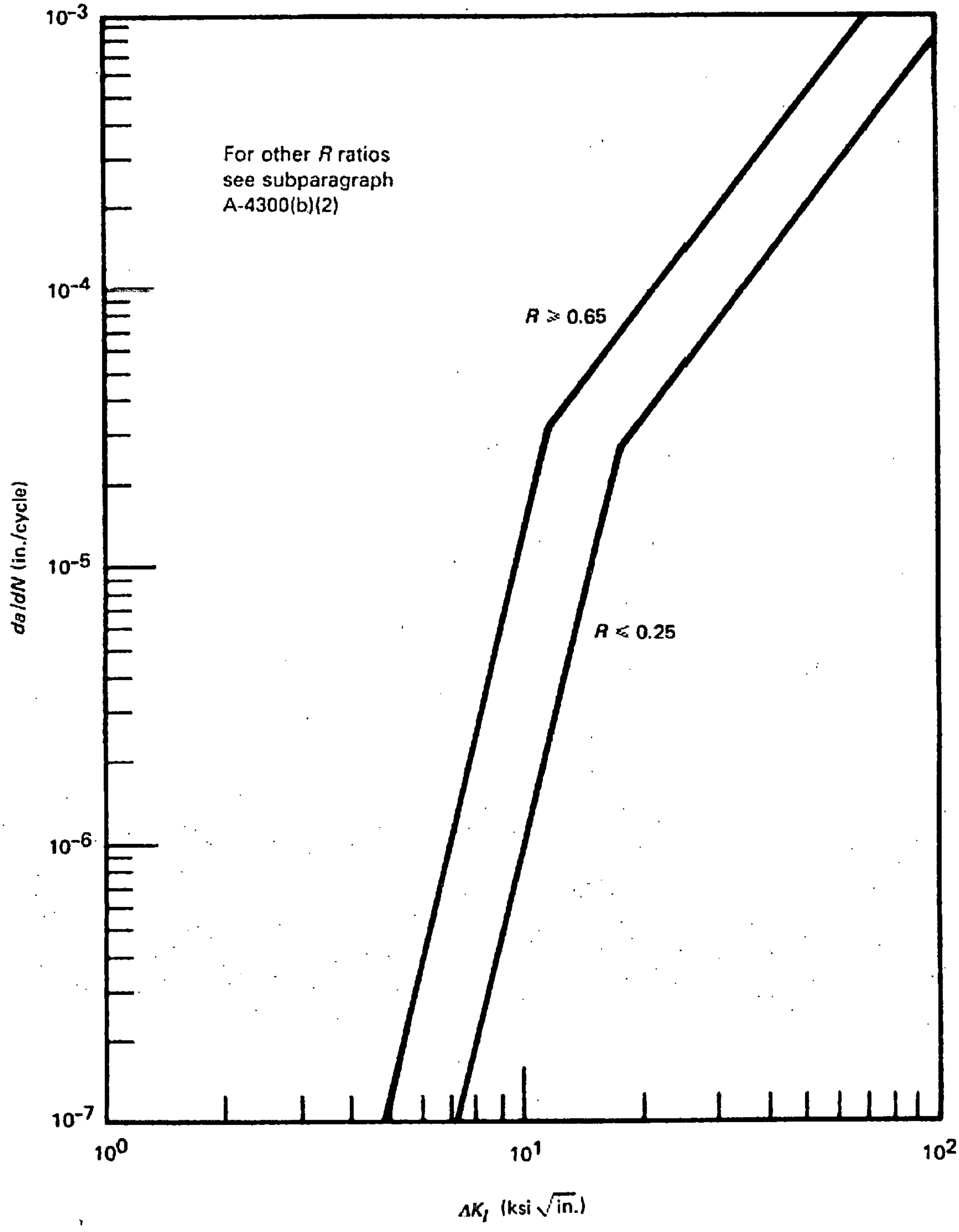

Figure 5-1: Reference Fatigue Crack Growth Curves for Carbon and Low Alloy Ferritic Steels Exposed to Water Environment [2, Fig. A-4300-2]

5.5 Allowable Flaw Size – Normal, Upset & Test Conditions

The allowable flaw size for normal, upset and test conditions was calculated and documented in Appendix A of WCAP-15338-A [1], using the criteria in Section 5.3.2.2. The fracture toughness for ferritic steels has been taken directly from the reference curves of Appendix A, ASME Section XI. In the transition temperature region, these curves can be represented by the following equations:

$$K_{Ic} = 33.2 + 20.734 \exp [0.02 (T - RT_{NDT})]$$

$$K_{Ia} = 26.8 + 12.445 \exp. [0.0145 (T - RT_{NDT})]$$

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

While these equations are the simplified form in the current ASME Section XI, they are mathematically identical to those presented in [1]; therefore, there is no impact on the results.

The upper shelf temperature regime requires utilization of a shelf toughness, which is not specified in the ASME Code. A value of 200 ksi$\sqrt{in}$ was used for upper shelf fracture toughness, as test data shows this to be a conservative value as discussed in WCAP-15338-A [1]. As shown in Table 5-4, the limiting transients are in the upper temperature range. Fracture toughness K_{IC} per ASME Section XI, A-4200 would yield values higher than 200 ksi$\sqrt{in}$. Lower temperature transients are protected by the pressure-temperature (P-T) limits per ASME Section XI, Appendix G, assuming a 1/4T flaw, which is much larger than those flaws evaluated in this TR. This remains applicable for extension of plant operations from 60 to 80 years.

The upper shelf toughness of 200 ksi$\sqrt{in}$ is used to evaluate the normal operating, upset, and test condition transients. Portions of the heatup and cooldown transients that drop to temperatures below the upper shelf region are governed by plant-specific P-T limit curves, which provide adequate margins of safety to prevent brittle fracture concerns of the reactor vessel. Therefore, the allowable flaw size determined in Appendix A of [1] remains applicable for the 80-year SLR application.

The allowable flaw size results for normal, upset and test conditions are provided in Table A-4.1 of WCAP-15338-A [1] and repeated in Table 5-4. The minimum allowable flaw size is 0.67 inch.

Table 5-4: 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: A wall thickness of 7.75 inches was used.

5.6 Allowable Flaw Size – Emergency & Faulted Conditions

The allowable flaw sizes for emergency and faulted conditions were also documented in Section A-5 of WCAP-15338-A [1] and shown in Table 5-5.

Table 5-5: Allowable Axial Flaw Sizes for Beltline Region – Emergency and Faulted Conditions

Flaw Shape	Allowable Flaw Size	
	Depth (inches)	Through-wall Ratio (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: A wall thickness of 7.75 inches was used.

As discussed in Section A-1 of WCAP-15338-A [1], the emergency and faulted conditions are ultimately governed by plant-specific treatment of pressurized thermal shock (PTS). The PTS events are covered through each plant's compliance with the screening criteria of 10CFR50.61. This screening criteria is independent of the plant operating period (whether 60 or 80 years).

The assumed upper shelf value of 200 ksi$\sqrt{in}$ was used to determine the allowables, and the temperatures of the emergency and faulted transients considered correspond to the upper shelf for the material. The RT_{NDT} is not expected to change significantly from 60 to 80 years as the rate of material embrittlement from sustained exposure decreases at higher fluence levels, and it does not impact the evaluations summarized herein since the normal operating, upset, and test condition transients result in the most limiting allowable flaw size (0.67 inch) using a conservative upper shelf toughness of 200 ksi$\sqrt{in}$. There are also several conservatisms included in the analysis. Underclad cracks are assumed to be surface cracks, which results in a conservative K_I. Conservatively assuming the flaw is exposed to water, the crack growth rate for a water environment is used. This results in a higher growth rate than assuming an air environment. The full flaw depth is assumed to be in the base material, and linear elastic fracture mechanics is used to determine the allowable flaw sizes. Therefore, the calculation of allowable flaw size for 60 years in [1] remains applicable for 80 years. Note that the largest flaw size of 0.4267 inch at the end of 80 years shown in Table 5-1 is less than the minimum allowable flaw size of 0.67 inch Table 5-4.

6 Summary and Conclusions

The purpose of this report is to update the 60 year FCG analysis in WCAP-15338-A [1] and confirm that the rest of the evaluation in [1] remains applicable to 80 years of operation.

As summarized in [1], there are many levels of defense in depth relative to the underclad cracks. There is no known mechanism for the creation of additional flaws in this region; therefore, the only potential concern is the potential propagation of the existing flaws.

Flaw indications indicative of underclad cracks have been evaluated in accordance with the acceptance criteria in the ASME Code, Section XI. These indications have been identified during pre-service and inservice inspections in those plants that were considered to have cladding conditions which have the potential for underclad cracking. These flaw indications were dispositioned as being acceptable for further service without repair or detailed evaluation, because they meet the conservative requirements of the ASME Code Section XI, Paragraph IWB-3500. Fracture evaluations have also been performed to evaluate underclad cracks, and the results also concluded that the flaws are acceptable.

A number of previous operation experience summarized in [1] involved cladding cracks, as well as exposure of the base metal due to cladding removal. These cladding cracks were postulated to extend into the base metal in the analysis. In these cases the cracks were postulated to be exposed to the water environment, and successive monitoring inspections were performed. No changes of the indications were identified due to propagation or further deterioration of any type. Based on these observations, these inspections were terminated.

Finally, underclad cracks identified during pre-service and inservice inspections have been evaluated in accordance with the acceptance criteria in the ASME Code, Section XI. The observed underclad cracks are very shallow, confined in depth to less than 0.295 inch and have lengths up to 2.0 inches. The FCG assessment for these small cracks concluded that there would be very little growth for 80 years of operation, even if they were exposed to the RCS water and with a crack tip pressure of 2,500 psi. For the worst case scenario, a 0.30-inch deep continuous axial flaw in the beltline region would grow to 0.43 inch after 80 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 maximum flaw depth of 0.4267 inch after 80 years of FCG is less than the minimum allowable flaw size of 0.67 inch, underclad cracks of any shape are acceptable for 80 years, regardless of the size or orientation of the flaws. Therefore, it can be concluded that underclad cracks are acceptable relative to the structural integrity of the reactor vessel for 80 years.

7 References

1. Westinghouse Report, WCAP-15338-A, Rev. 0, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," October 2002.
2. ASME Boiler & Pressure Vessel Code Section XI, 2001 Edition through 2003 Addenda.
3. Westinghouse Document, LTR-PSDR-TAM-14-003, Rev. 0, "Reactor Vessel Inlet Nozzle Cladding Damage Assessment for Diablo Canyon Unit 1," February 2014.
4. Westinghouse Report, WCAP-18158-P, Rev. 0, "Qinshan Phase I Reactor Vessel Cladding Wear Evaluation for Operating Life Extension up to 50 years," October 2016.