ATTACHMENT 2

Westinghouse WCAP-18596-NP Revision 0, Braidwood Unit 1 Reactor Vessel Closure Head Dome-to-Ring Weld ASME Section XI Evaluation of As-Found Indications (Non-Proprietary)

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Revision 0

December 2020

Braidwood Unit 1 Reactor Vessel Closure Head Dome-to-Ring Weld ASME Section XI Evaluation of As-Found Indications

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December 2020

Xiaolan Song*
RV Upper Internals Design Analysis

Verifier: Reddy B. Ganta*
Structural Design and Analysis

Reviewer: Anees Udyawar*
RV/CV Design and Analysis

Approved: Lynn A. Patterson*, Manager
RV/CV Design and Analysis

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

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RECORD OF REVISIONS

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0	December 2020	Original Issue

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EXECUTIVE SUMMARY

During the Spring 2018 refueling outage (A1R20) at Braidwood Unit 1, in-service examinations (ISI) of the reactor vessel closure head dome-to-ring weld discovered several ultrasonic testing indications. A total of nineteen (19) circumferential indications were discovered during the UT examinations based on ASME Section XI Appendix VIII (PDI) qualifications and requirements. These indications were not detected in the previous ISI in 1998. Based on the nature and appearance of the nineteen (19) flaws detected during the ISI in Spring 2018 outage, the increased UT detection sensitivity for the 2018 inspection compared to the 1998 UT technology is considered the likely cause of their detection. In other words, it is likely these flaws have been present in the weld since its original installation, but they have remained undetected until 2018 ISI.

Based on comparison to ASME Section XI, 2001 Edition, 2003 Addenda [1], Paragraph IWB-3510, nine of the indications were not acceptable and require fracture mechanics evaluation to ASME Section XI IWB-3600 requirements. The nine (9) indications were previously evaluated during the Spring 2018 outage and determined to be acceptable for at least two fuel cycles of operation (36 months) in LTR-SDA-18-038-P [2]. This report performs additional plant specific IWB-3600 evaluation of the nine as-found indications to determine their acceptability for long term plant operation (i.e., 60 years).

A detailed finite element analysis for Braidwood Unit 1 reactor vessel top dome-to-ring weld and the subsequent fracture mechanics evaluation per ASME Section XI, 2013 Edition [3] (i.e., the current code Edition applicable for Braidwood Unit 1), IWB-3600 requirements are performed herein. The fatigue crack growth results show that the as-found flaws in the weld remain embedded flaws and are acceptable for 60 years of operation, which will cover the remaining design life of the plant. In addition, the primary stress evaluation shows that the reduction in cross section area due to all of the rejectable indications combined is only 1.03%; and thus a large margin for the local primary stress is maintained. Therefore, no repair or corrective actions for this weld is necessary since the flaws are acceptable for the remaining life of the plant based on the ASME Section XI IWB-3600 evaluation.

1 INTRODUCTION

During the Spring 2018 refueling outage (A1R20) at Braidwood Unit 1, in-service examinations of the reactor vessel closure head dome-to-ring weld (see Figure 1-1) discovered several ultrasonic testing indications. The UT examinations (performed from the OD-outside diameter surface) were based on ASME Section XI Appendix VIII (PDI) qualifications and requirements [1]. A total of nineteen (19) circumferential indications were discovered [9]. Based on comparison to ASME Section XI, 2001 Edition, 2003 Addenda [1], Paragraph IWB-3510, nine (9) of the indications (see Figure 1-2 and Table 3-1) were not acceptable [9] and require fracture mechanics evaluation to ASME Section XI IWB-3600 requirements. The nine (9) indications were evaluated based on ASME Section XI IWB-3600 and determined to be acceptable for at least two fuel cycles of operation (36 months) in LTR-SDA-18-038 [2]. The evaluation in [2] was based on a stress analysis done for a generic plant, which was shown applicable to Braidwood Unit 1.

The purpose of the evaluation in this report is to perform a detailed finite element analysis for Braidwood Unit 1 reactor vessel top dome-to-weld and then use the plant specific stresses to perform the fracture mechanics evaluations for the as-found flaws to determine the acceptability for 60 years of operation, which will cover the remaining design life of the plant. The fracture mechanics evaluation is performed based on IWB-3600 of the 2013 Edition ASME Section XI [3], which is the current ASME code edition of record applicable to Braidwood Unit 1.

The general methodology of the fracture mechanics evaluation is described in Section 2. The materials and geometry are provided in Section 3 of this report. Section 4 describes the loading conditions and the stresses used in the evaluation. Section 5 provides fracture mechanics evaluation results along with primary stress limit check results. Section 6 provides the conclusions. All cited references are provided in Section 7.

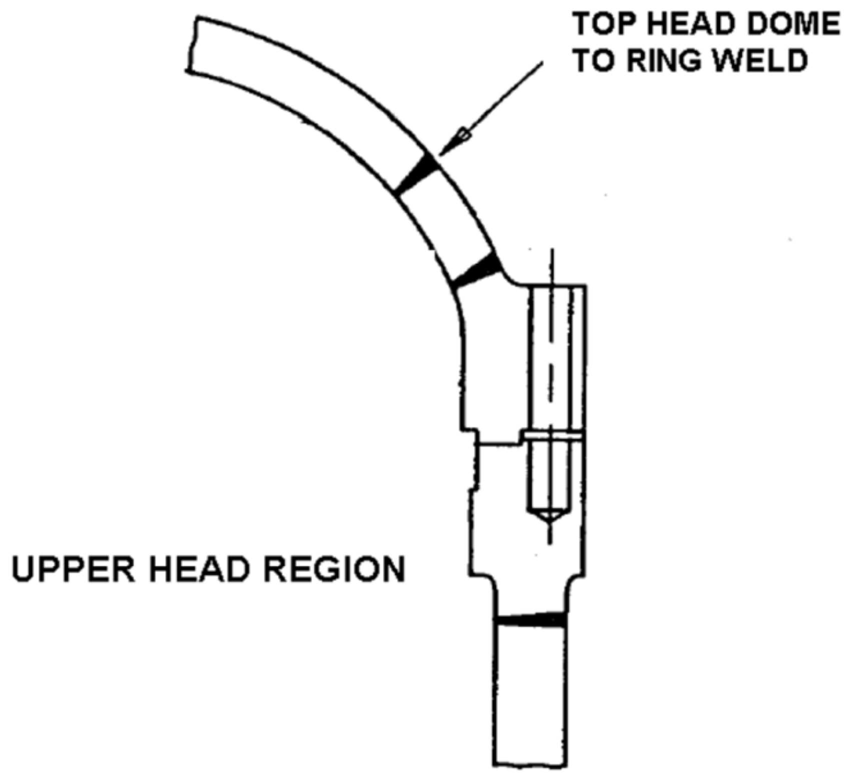

Figure 1-1: Reactor Vessel Head Dome-to-Ring Weld Location

a, c, e

Figure 1-2: General Location of Indications in the Reactor Vessel Head Dome-to-Ring Weld

2 GENERAL METHODOLOGY

2.1 ACCEPTANCE CRITERIA

Acceptance criteria for ferritic steel components 4 inch and greater in thickness is defined in IWB-3610 of ASME Code Section XI [3].

Per IWB-3610(a), the analytical procedures described in Appendix A of ASME Code Section XI is used to calculate the crack growth until the end of service lifetime of the component, see further discussion in Section 2.3.

Furthermore, IWB-3610(b) is followed to characterize the flaw as either surface or embedded. Per Figure IWB-3610-1, the flaw is considered embedded if S > 0.4a, where “S” is the distance from the crack tip to the nearest surface and “a” is the half crack depth for embedded flaws (see Figure 3-1).

Per IWB-3610(d)(1), there are two alternative sets of flaw acceptance criteria for ferritic components to assess continued service without repair for the as-found indications:

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

To determine whether a flaw is acceptable for continued service without repair, the more beneficial of the two criteria (IWB-3611 and IWB-3612) may be used.

The ASME Section XI [3] 2013 Edition, IWB-3612, “Acceptance Criteria Based on Applied Stress Intensity Factor” criteria for normal/upset/test conditions and for emergency/faulted conditions is used to determine the flaw acceptance criteria:

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

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

where:

K_I = the maximum applied stress intensity factor for the maximum size to which the detected flaw is calculated to grow after 60 years (a_f)

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

The fatigue crack growth of the as-found indications for 60 years of operation is calculated and the maximum stress intensity factors at the final crack depth is compared to the fracture toughness (K_{Ic}) considering the structural factors as discussed above to demonstrate the structural integrity of the weld with the as-found indications. The details for the methods to calculate fracture toughness, and stress intensity factor are discussed in Sections 2.2 and 2.3, respectively.

In addition, per IWB-3610(d)(2) of ASME Code Section XI [3], the primary stress limits of NB-3000 [4] shall be satisfied considering the local area reduction of the pressure retaining membrane that is equal to the area of the detected flaw(s). Positive primary stress margin shall be maintained.

2.2 FRACTURE TOUGHNESS

The material of the closure head dome is SA-533 Grade B Class 1 [5] and the material of the closure head ring is SA-508, Class 3 [6]. Lower bound K_{Ic} versus temperature curve for these materials are provided in Fig. A-4200-1, Section XI of 2013 ASME code [3]. Analytical approximation for the curve is provided in A-4200, Section XI of 2013 ASME code [3] as below:

$$K_{Ic} = 33.2 + 20.734 \exp[0.02(T-RT_{NDT})], \text{ (ksi-in}^{0.5}\text{)}$$

where:

T = Crack Tip Temperature, (°F)

RT_{NDT} = Reference Nil-Ductility Temperature, (°F)

For Braidwood Unit 1 reactor vessel dome-to-ring weld, the initial RT_{NDT} is determined to be [ ]^{a,c,e} from the weld certified material test reports (CMTRs) per Table 3-1 of WCAP-16143-P, Rev. 1 [7]. The end-of-life neutron fluence (considering 60 years of operation) at the reactor vessel dome-to-ring weld region is expected to be below 1×10^{17} n/cm^2; therefore, the neutron embrittlement effect is negligible.

Note that the value of K_{Ic} is limited to an upper shelf fracture toughness of 200 ksi-in$^{0.5}$ as a conservative and practical upper-limit on the K_{Ic} curve.

2.3 FERRITIC STEEL FATIGUE CRACK GROWTH

The nine as-found circumferential indications that were rejectable per IWB-3500 in the dome-to-ring weld are all embedded flaws. Therefore, the correlations for ferritic steel fatigue crack growth behavior of the ferritic material exposed to air environments in A-4300, Section XI of 2013 ASME Code [3] are used to calculate the fatigue crack growth of these indications, as discussed below.

The fatigue crack growth rate ($\frac{da}{dN}$) in inch/cycle, is a function of the applied crack tip stress intensity factor range (ΔK_I). The general fatigue crack growth rate equation can be expressed as follows:

$$\frac{da}{dN} = C_0(\Delta K_I)^n$$

where *n* is the slope of the log($\frac{da}{dN}$) versus log(ΔK_I) curve and is equal to 3.07 for subsurface flaws.

Parameter C_0 is a scaling constant:

$$\begin{aligned} C_0 &= 0 && for\ \Delta K_I < \Delta K_{th} \\ &= 1.99 \times 10^{-10} S && for\ \Delta K_I \geq \Delta K_{th} \end{aligned}$$

where ΔK_{th} is the threshold ΔK_I value below which the fatigue crack growth rate is negligible and S is a scaling parameter. Both ΔK_{th} and *S* are a function of the R ratio (K_{min}/K_{max}). The calculation of crack tip stress intensity factor range(ΔK_I) also changes with R ratio when $\Delta K_I \geq \Delta K_{th}$.

$$\begin{aligned} \Delta K_{th} &= 5.0 && for\ R < 0 \\ &= 5.0(1 - 0.8R) && for\ 0 \leq R < 1.0 \end{aligned}$$

The calculation of crack tip stress intensity factor range (ΔK_I) also changes with *R* ratio when $\Delta K_I \geq \Delta K_{th}$. The calculation of *S* and ΔK_I for different *R* ratio ranges is summarized below:

- For $0 \leq R \leq 1$

  $S = 25.72(2.88-R)^{-3.07}$ and $\Delta K_I = K_{max} - K_{min}$

- For $R < 0$ and $K_{max} - K_{min} > 1.12\sigma_f\sqrt{\pi a}$

  $S = 1$ and $\Delta K_I = K_{max} - K_{min}$

- For $-2 \leq R \leq 0$ and $K_{max} - K_{min} \leq 1.12\sigma_f\sqrt{\pi a}$

  $S = 1$ and $\Delta K_I = K_{max}$

- For $R < -2$ and $K_{max} - K_{min} \leq 1.12\sigma_f\sqrt{\pi a}$

  $S = 1$ and $\Delta K_I = (1-R)K_{max}/3$

  Where σ_f is the flow stress, defined as the average of the yield strength and the ultimate tensile strength of the materials per ASME Section II Part D [8].

  Note that a factor of 0.8 is applied to the limit in $K_{max} - K_{min}$ defined in A-4300 of ASME Code Section XI [3] per 10 CFR 50.55a Codes and Standards.

For the applicable transients in Table 4-1, the applied crack tip stress intensity factor ranges (ΔK_I) and the corresponding crack growth rates ($\frac{da}{dN}$) due to the cycling of the stresses are calculated. Using the number of cycles for the transients and the crack growth rates, the cumulative crack growth for all the applicable transients are calculated for 60 year life of the plant.

2.4 STRESS INTENSITY FACTOR

The methodology in Article A-3000 of 2013 ASME code, Section XI [3] is used to calculate the stress intensity factor (SIF) for embedded flaws.

Stress intensity factors for subsurface flaws is calculated using the membrane and bending stresses at the flaw location by the following equation:

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

Where

σ_m and σ_b are membrane and bending stresses; per A-3200(a) of 2013 ASME code, Section XI [3], the total stresses at the flaw location, including residual stresses and applied stresses from all forms of loading, shall be resolved into membrane and bending stresses. For nonlinear stress variations through the wall, the actual stress distribution may be approximated by a linear distribution that accurately represents or bounds the stress field over the flaw depth. The linearized stress distribution may then be characterized by the membrane stress (σ_m) and the bending stress (σ_b).

M_m and M_b are correction factors for membrane stress and bending stress, respectively. The M_m and M_b correction factors are calculated from Fig. A-3310-1 and Fig. A-3310-2 of 2013 ASME code, Section XI [3], respectively.

a is half of the crack depth for the embedded flaw

Q is the flaw shape parameter and calculated from the half crack depth (a) and crack length (l) using the following equation:

$$Q = 1 + 4.593(\frac{a}{l})^{1.65} - q_y$$

Note that q_y is the plastic zone correction factor; Per A-5200 of 2013 ASME code, Section XI [3], the plastic zone correction factor need not be taken into account in calculating the stress intensity factor and can be set to zero.

3 FLAW GEOMETRY

The dimensions and characterization of the nine as-found circumferential indications that were rejectable per IWB-3500 are provided in [9] and summarized in Table 3-1. Per IWB-3610(b) of ASME Code Section XI [3], all the flaws are subsurface (i.e., embedded) flaws meeting the flaw proximity rules of IWA-3300, with S (distance from the flaw edge to the nearest surface) larger than 0.4a (40% of half flaw depth); See Figure 3-1 for the definition of parameters “S” and “a”. Per Figure IWA-3330-1 of ASME Code Section XI [3], discontinuous embedded indications shall be considered single flaws if the distance between adjacent flaws is equal to or less than the smallest half crack depth of the adjacent flaws. However, per the weld inspection report [9], the indications are apart from each other by the distance no less than the diameter of the stud holes. Per drawing [10], the stud hole diameter is [ ]a,c,e, which is much larger than the half flaw depth of all the indications; therefore, no combination of the indications is needed. Note that Indication 18 in Table 3-1 is not analyzed since it has the smallest flaw depth and is bounded by Indication 13.

Table 3-1 Braidwood Unit 1 Reactor Vessel Closure Head Head-to-Ring Weld Indication Parameters [9]

Indication Number(1)	Flaw Length l (inch)	Upper Tip(2) (inch)	Lower Tip(2) (inch)	Head Thickness (including cladding) t (inch)	Total Flaw Depth(3) $2a$ (inch)	AR ratio l/a	a/t	S(4) (inch)	0.4a (inch)
1	2.0	5.6	6.7	7.1	1.1	3.64	0.078	0.4	0.22
2	2.0	5.5	6.6	7.1	1.1	3.64	0.078	0.5	0.22
3	2.8	5.5	6.5	7.1	1.0	5.60	0.071	0.6	0.20
6	1.5	2.4	3.2	7.1	0.8	3.75	0.057	2.4	0.16
13	4.0	5.5	6.2	7.1	0.7	11.43	0.050	0.9	0.14
16	2.8	5.1	6.5	7.1	1.4	4.00	0.099	0.6	0.28
17	2.0	2.6	3.7	7.0	1.1	3.64	0.079	2.6	0.22
18	1.5	3.1	3.5	7.0	0.4	7.50	0.029	3.1	0.08
19	1.9	2.3	4.8	7.1	2.5	1.52	0.177	2.3	0.50

Notes:

1. See Figure 1-2 for the location of the indications.
2. The UT measurements [9] are taken from the outside diameter (OD).
3. Total Flaw depth = Lower Tip – Upper Tip
4. Minimum distance from the flaw edge to the surface; For indication #6, #17, #18, and #19, the flaw is closer to the OD surface and *S* is the distance to the OD surface while for all the other indications, the flaw is closer to the inside diameter (ID) surface and *S* is the distance to the ID surface.

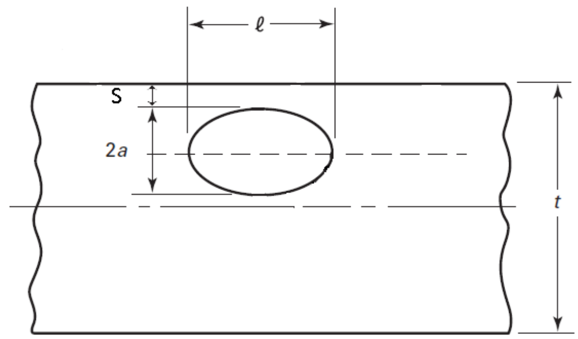

Figure 3-1 Embedded Flaw Geometry

4 LOADING CONDITIONS AND STRESS ANALYSIS

A three-dimensional finite element model of the plant specific geometry of the reactor vessel head closure regions for the Braidwood plant is used to develop the stresses for fracture mechanics evaluation. The finite element model (FEM) and fracture mechanics analysis consider two cases: 1) Braidwood Unit 1 operating with all 54 reactor vessel head studs active, and 2) Braidwood Unit 1 operating with 53 studs active; this case considers the hypothetical event if one stud becomes stuck or inactive similar to the situation for Braidwood Unit 2. Thus, the FEM and fracture mechanics evaluation considers both these cases since Braidwood Unit 1 is licensed for operation with either 54 or 53 stud configuration [7]. The most limiting results for the two cases are reported in Section 5. The latest Braidwood Unit 1 pressure and temperature reactor coolant system (RCS) transients applicable to the reactor vessel [11, 13] are considered for the stress analysis. The Braidwood RCS design transients along with the cycles of each transient are summarized in Table 4-1. Both the thermal and mechanical loadings (i.e., pressure loading and closure stud pre-load) are considered in the analyses. The combined thermal, pressure, and mechanical stresses, along with the temperature are obtained for the cut lines at the dome-to-ring weld location. An elevation view showing the locations of the cuts through the weld is provided in Figure 4-1 and the circumferential locations of the stress cuts are shown in Figure 4-2. The through-wall stresses are obtained for five cut lines, three cuts through the studs and two cuts between the studs. The highest stresses of the five cuts are used in the fracture mechanics evaluation. Since all the as-found indications are circumferential flaw, only the meridional stresses are used in the flaw evaluation. The meridional stress is the stress component perpendicular to the crack face.

The post heat treatment weld residual stresses are also considered in the fatigue crack growth and maximum stress intensity factor calculation. Post-weld heat treatment (PWHT) is performed on the RPV structural weld and the residual stresses are those remaining stresses that are not completely relaxed by the PWHT. The through-wall weld residual stress distribution from [12] is used in the flaw evaluation. The residual stress from [12] was derived in the NRC-funded Heavy Section Steel Technology (HSST) program. The weld residual stress distribution from Fig 31(a) of [12] is plotted in Figure 4-3.

Table 4-1 Braidwood Reactor Coolant System Design Transients [11, 13]

Transient #(7)	Transient	Cycles(8)	Level
1	Turbine Roll Test	20	Test
2	Unit Loading (0-15% Power)	500	A
3	Unit Unloading (15-0% Power)	500	A
4	Unit Loading (15-100% Power)	13,200	A
5	Unit Unloading (100-15% Power)	13,200	A
6	Step Load Decrease of 10% Power	2,000	A
7	Step Load Increase of 10% Power	2,000	A
8	Large Step Load Decrease	200	A
9	Loss of Load	80	B
10	Loss of Offsite Power	40	B
11	Partial Loss of RCS Flow (Case 1)	80	B
12	Partial Loss of RCS Flow (Case 2)	80	B
13	Reactor Trip No Cooldown	230	B
14	Reactor Trip w/ Cooldown and No SI	160	B
15	Reactor Trip w/ Cooldown and SI	10	B
16	Feedwater Cycling	2,000	A
17	Loop Out of Service Shutdown(1)	80	A
18	Loop Out of Service Startup(1)	70	A
19	Boron Concentration Equalization	26,400	A
20	Inadvertent RCS Depressurization	20	B
21	Inadvertent Startup of Inactive Loop (Inactive Loop – Case 1)(1)	10	B
22	Inadvertent Startup of Inactive Loop (Active Loop – Case 1)(1)	10	B
23	Inadvertent Startup of Inactive Loop (Inactive Loop – Case 2)(1)	10	B
24	Inadvertent Startup of Inactive Loop (Active Loop – Case 2)(1)	10	B
25	Control Rod Drop	80	B
26	Inadvertent Safety Injection Actuation	60	B
27	Excessive Feedwater Flow	30	B
31	Cold Overpressure Mitigation System (COMS)	240	B
32	Heatup and Cooldown	200	A
33	Small Steam Line Break	5	C
34	Small LOCA	5	C
34a	Complete Loss of Flow	5	C
35	Large LOCA(2)	1	D
36	Large Steam Line Break	1	D
37	Feedwater Line Break – Line w/ Break	1	D
38	Feedwater Line Break – Line w/o Break	1	D
39	Control Rod Ejection	1	D
40	Refueling(3)	80	A
41	Primary Side Leak Test	200	Test
42	Primary Side Hydrostatic Test(4)	10	Test

43	Reactor Coolant Pump Locked Rotor(5)	1	D
44	Steam Generator Tube Rupture(6)	1	D

Notes:

1. Loop out of service shutdown, loop out of service startup, inadvertent startup of inactive loop transients were removed from the reactor vessel transient set per [11].
2. Braidwood Unit 1 has been qualified and approved for Leak-before-Break; therefore the large LOCA transient no longer applies.
3. Refueling transient has no effect on the closure head dome-to-ring weld since the closure head is removed prior to the refueling transients.
4. Primary side hydrostatic test is replaced by the system leakage test.
5. Reactor coolant pump locked rotor transient is bounded by the control rod ejection and feedwater line break transients.
6. Steam generator tube rupture transient is bounded by the reactor trip with cooldown and safety injection transient.
7. Transient numbers are arbitrarily assigned. Transient numbers 28, 29, and 30 are not used.
8. Cycles are from WCAP-11388, Rev. 0, Volume 2 [13].

a, c, e

Figure 4-1: Elevation View of Stress Cut Locations through Dome-to-Ring Weld

a, c, e

Figure 4-2: Circumferential Planes for Stress Cut Locations

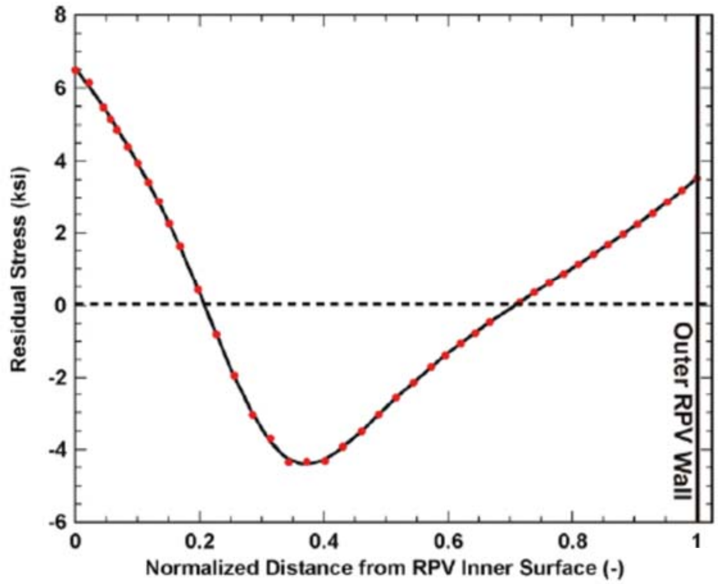

Figure 4-3: Remaining Through Wall Weld Residual Stress after PWHT [12]

5 FRACTURE MECHANICS EVALUATION

5.1 FATIGUE CRACK GROWTH

The results of the 60 year fatigue crack growth analysis for the as-found indications are summarized in Table 5-1. Note that as discussed in Section 3, Indication 18 is bounded by Indication 13 and is not analyzed. Based on the results in Table 5-1, it is observed that for all the indications, the crack growth is insignificant. The maximum flaw growth is 0.23 inch and occurs for Indication 16. At the end of the plant design life, the ligament length from the crack edge to the nearest surface (S) remains larger than 40% of the half flaw depth (0.4a) for all the indications; therefore, the flaws remain embedded.

Table 5-1 Fatigue Crack Growth Results

Indication Number	**Initial Flaw**			**Final Flaw after 60 Years of Growth**			
	Flaw Depth *2a* (inch)	***a/t*(1)**	**S(2) (inch)**	**Flaw Depth *2a* (inch)**	***a/t*(1)**	**S(2) (inch)**	**0.4*a* (inch)**
1	1.1	0.078	0.4	1.25	0.0881	0.32	0.25
2	1.1	0.078	0.5	1.25	0.0878	0.43	0.25
3	1.0	0.071	0.6	1.17	0.0822	0.52	0.23
6	0.8	0.057	2.4	0.92	0.0649	2.34	0.18
13 and 18(3)	0.7	0.050	0.9	0.81	0.0571	0.84	0.16
16	1.4	0.099	0.6	1.63	0.1145	0.48	0.33
17	1.1	0.079	2.6	1.29	0.0911	2.50	0.26
19	2.5	0.177	2.3	2.71	0.1910	2.19	0.54

Notes:

1. Ratio of half flaw depth over thickness (see Figure 3-1)
2. Distance from the crack edge to the nearest surface; For indication #6, #17, #18, and #19, the flaw is closer to the OD surface and S is the distance to the OD surface while for all the other indications, the flaw is closer to the inside diameter (ID) surface and S is the distance to the ID surface.
3. Indication 13 bounds Indication 18.

5.2 MAXIMUM STRESS INTENSITY FACTORS

The maximum stress intensity factors at the final flaw depth are calculated for all the transients. The maximum stress intensities for the Level A/B/test and Level C/D transients are compared to the corresponding fracture toughness of the weld with the structural factor considered to determine if the acceptance criteria in IWB-3612 of ASME Code, Section XI [3] is satisfied. Note that for the majority of the transients, the minimum through wall temperature during the entire transient time history is conservatively used to calculate the fracture toughness. For a few transients with a low minimum temperature during the entire transient, the lowest through-wall temperature at the corresponding time step of the maximum stress intensity factor is used as crack tip temperature. The K_I versus K_{Ic} (with structural factor) comparison results are summarized in Table 5-2. The table shows that for all the conditions, the maximum stress intensity factor is below the fracture toughness. Therefore, all nine indications meet the IWB-3600 flaw evaluation criteria for at least 60 years of plant operation and no repair is necessary for these indications.

Table 5-2 Comparison of Maximum Stress Intensity Factors and Fracture Toughness a, c, e

Notes:

1. Level A/B/Test transients 1~16, 19~20, and 25~27 are grouped together and the lowest metal temperature for these transients (i.e.,[ ]a,c,e) is used to calculate the fracture

toughness. All the Level C/D transients are grouped together and the lowest metal temperature for these transients (i.e., [ ]^{a,c,e}) is used to calculate the fracture toughness.
2. SF (structural factor) is $\sqrt{10}$ and $\sqrt{2}$ for Level A/B/Test and Level C/D, respectively.
3. The lowest through-wall temperature for all the time steps of the transient or groups of transients is conservatively used as crack tip temperature.
4. The lowest through-wall temperature at the time of the maximum stress intensity factor is conservatively used as crack tip temperature.
5. Indication 13 bounds Indication 18.

5.3 PRIMARY STRESS LIMITS

Per IWB-3610(d)(2) of ASME Code Section XI [3], the primary stress limits of NB-3000 [4] needs to be satisfied considering the local area reduction of the pressure retaining membrane that is equal to the area of the detected flaw(s). An evaluation is performed to consider the total reduction of the RV head cross section for each indication after 60 years of crack growth. The total area reduction is calculated to be 30.31 in^2. From the weld head dimensions shown on the reactor vessel head drawing [10], the head radius is [ ]^{a,c,e} inches. The minimum head thickness is [ ]^{a,c,e} inches. The weld cross section area is thus calculated to be at least [ ]^{a,c,e} in^2. The reduction in the total cross section area is thus only about [ ]^{a,c,e} due to the nine rejectable indications.

Based on the reactor vessel sizing calculation in [14], the following primary stress margins were determined:

1. To satisfy the criteria for the primary general membrane stress intensity P_m, the design minimum head thickness must be greater than the minimum required thickness due to internal pressure. From [14], the minimum required thickness for the upper head is [ ]^{a,c,e} inches while the design minimum head thickness is [ ]^{a,c,e} inches per [10]. The minimum required thickness is calculated based on the maximum allowable meridional stress assuming constant thickness through the entire weld. Since the meridional stress is inversely proportional to the thickness, the margin for P_m can be calculated using the minimum required thickness and the design thickness as [ ]^{a,c,e}.
2. The maximum primary local membrane stress (P_L) in the closure area is [ ]^{a,c,e} psi compared to the 1.5Sm limit of 40,050 psi. The margin for P_L is thus [ ]^{a,c,e}.

The [ ]^{a,c,e} reduction in the weld area due to the as-found indications is much less than the primary stress design margins. Therefore, the margins for Pm and P_L will remain positive considering the local area reduction due to the detected flaws.

6 CONCLUSIONS

Ultrasonic examinations of the Braidwood Unit 1 reactor vessel closure head dome-to-ring weld during the Spring 2018 refueling outage revealed nineteen indications. Based on ASME Section XI, 2001 Edition, 2003 Addenda [1], Paragraph IWB-3512, nine of the nineteen as-found indications were not acceptable [9] and required fracture mechanics evaluation to ASME Section XI IWB-3600 requirements.

The nine as-found indications were evaluated based on the 2013 Edition ASME Code, Section XI IWB-3600 [3], which is the ASME edition of record currently applicable to Braidwood Unit 1. It is determined that the nine indications are acceptable per IWB-3600 for at least 60 years of plant operation, which will cover the remaining design life of the plant. All flaws remain embedded after 60 years of growth. In addition, the primary stress evaluation shows that the reduction in cross section area due to all of the rejectable indications combined is only 1.03%, which is much less than the primary stress design margins (i.e., the margin of 0.50 for P_m and 0.70 for P_L). The large margin for the primary stress is thus maintained. Therefore, no repair or corrective actions for this weld is necessary.

7 REFERENCES

1. ASME Boiler and Pressure Vessel Code, 2001 Edition with 2003 Addenda, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."
2. Westinghouse Letter, LTR-SDA-18-038-P, Rev. 0, "Braidwood Unit 1 Reactor Vessel Closure Head Evaluation of As-Found Indications in the Dome to Ring Weld," April 2018.
3. ASME Boiler and Pressure Vessel Code, 2013 Edition, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."
4. ASME Boiler and Pressure Vessel Code, 2013 Edition, Section III, "Rules for Construction of Nuclear Facility Components."
5. [ ][a,c,e]
6. [ ][a,c,e]
7. [ ][a,c,e]
8. ASME Boiler and Pressure Vessel Code, 2013 Edition, Section II, Part D – Properties.
9. [ ][a,c,e]
10. [ ][a,c,e]
11. [ ][a,c,e]
12. Oak Ridge National Laboratory Report ORNL/LTR-2016/309, "Fracture Analysis of Vessels – Oak Ridge FAVOR, v16.1, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations," September 2016.
13. [ ][a,c,e]
14. [ ][a,c,e]

ATTACHMENT 3

Westinghouse Affidavit for Withholding Proprietary Information from Public Disclosure
CAW-20-5133

COMMONWEALTH OF PENNSYLVANIA:
COUNTY OF BUTLER:

(1) I, Camille T. Zozula, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of WCAP-18596-P be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

(ii) The information sought to be withheld is being transmitted to the Commission in confidence and, to Westinghouse's knowledge, is not available in public sources.

(iii) Westinghouse notes that a showing of substantial harm is no longer an applicable criterion for analyzing whether a document should be withheld from public disclosure. Nevertheless, public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower-case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower-case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 16 Dec 2020

Camille Zozula

Camille T. Zozula, Manager
Regulatory Compliance & Corporate Licensing