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Subject: PWR Owners Group
Transmittal of the Response to Request for Additional Information, RAIs 1, 2 and 3 Associated with PWROG-17031-NP, Revision 1, "Update for Subsequent License Renewal: WCAP-15338-A, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants,", PA-MSC-1497

References:

1. Letter OG-18-118, Transmittal of PWROG-17031-NP, Revision 1, "Update for Subsequent License Renewal: WCAP-15338-A, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," (PA-MSC-1497), dated May 31, 2018

2. NRC Letter of Acceptance for Review of PWROG-17031-NP, Revision 1, "Update for Subsequent License Renewal: WCAP-15338-A, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," dated June 28, 2018

3. Email from the NRC (Drake) to the PWROG (Holderbaum), Request for Additional Information, RAIs 1-3, RE: PWROG-17031-NP, Revision 1, "Update for Subsequent License Renewal: WCAP-15338-A, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," dated October 30, 2018

4. Email from the NRC (Drake) to the PWROG (Holderbaum), Additional Questions on Draft RAI-2, PWROG-17031-NP, Rev.1, dated February 21, 2019

On May 31, 2018, in accordance with the Nuclear Regulatory Commission (NRC) Topical Report (TR) program for review and acceptance, the Pressurized Water Reactor Owners Group (PWROG) requested formal NRC review and approval of PWROG-17031-NP, Revision 1 for referencing in regulatory actions (Reference 1). The report was accepted for review on June 28, 2018 (Reference

D048
NRR

2). The NRC Staff has determined that additional information is needed to complete the review per the emails dated October 30, 2018 (Reference 3) and February 21, 2019 (Reference 4).

Enclosure 1 to this letter provides formal responses to NRC RAIs 1, 2 and 3 (References 3 and 4) associated with PWROG-17031-NP, Revision 1, "Update for Subsequent License Renewal: WCAP-15338-A, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants".

Correspondence related to this transmittal should be addressed to:

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If you have any questions, please do not hesitate to contact me at (805) 545-4328 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Ken J Schrader

Ken Schrader, COO & Chairman
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JKS:am

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Enclosure 1: LTR-SDA-18-126, Revision 0, "RAIs 1, 2 and 3 Responses for PWROG-17031-NP, Revision 1 (PA-MSC-1497)

Westinghouse

To: Jim Molkenthin
cc:
From: Gordon Z. Hall
Ext: (860) 731-6114
Fax:

Date: August 21, 2019

Your ref: N/A
Our ref: LTR-SDA-18-126, Rev. 0

Subject: **Westinghouse Response to U.S. NRC Request for Additional Information for the Review of Generic Topical Report No. PWROG-17031-NP, Rev. 1**

Background and Regulatory Basis:

Pursuant to 10 CFR 54.21(c), applicants for SLR shall include an evaluation of time-limited aging analyses (TLAAs). The applicant shall demonstrate that (i) the analyses remain valid for the period of extended operation; (ii) the analyses have been projected to the end of the period of extended operation; or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

If approved by the NRC staff for generic use SLR applications (SLRAs), the generic 80-year RPV underclad cracking analysis in PWROG-17031-NP, Rev. 1 would constitute a technical basis for disposition of RPV underclad cracking in accordance with 10 CFR 54.21(c)(1)(ii).

This is because the PWROG report seeks to generically demonstrate that the analysis of postulated RPV underclad cracks has been projected to the end of the subsequent period of extended operation (SPEO, 80-year operating period), based on the following methods:

a. Section 5.4 of the TR provides generic 80-year fatigue crack growth (FCG) calculations that are based on ASME Code, Section XI, Appendix A FCG rate curves for low alloy ferritic steel in a water environment and application of the 40-year transient cycles times a factor of 2.0 to account for 80-years of operation;

b. Sections 5.5 and 5.6 of the TR address continued implementation of the same allowable flaw sizes that were previously established for 60-year applications in WCAP-15338-A (October 2002, ML083530289). The allowable flaw sizes were determined in accordance with analytical acceptance criteria for RPV flaws in the ASME Code, Section XI, IWB-3610 based on the same governing transients for normal, upset, and test conditions (Level A and B), and emergency and faulted conditions (Level C and D); and the continued use of certain assumptions for RPV beltline fracture toughness for 80-year applications.

The above background and regulatory basis is applicable to all RAIs addressed below.

RAI-1: Transient Cycles for Generic FCG Calculation

Issue:

Section 5.4 of the TR states that FCG calculations were performed in WCAP-15338-A to provide a prediction of future growth of underclad cracks for service periods up to 60-years, and the FCG calculation was updated for 80-year SLR applications. The TR also states that to complete the FCG analysis for 80-years, the methodology of the ASME Code, Section XI was used with the entire set of design transients applied over an 80-year period – specifically, the "cycles applicable to 40 years of operation were conservatively multiplied by a factor of 2.0 to account for 80-years of operation."

WCAP-15338-A, Section 9, "Attachment, WOG Letters" provides information on the types and numbers of transients that were used to calculate generic cumulative FCG for the 60-year period. Specifically, the final response to License Renewal Generic Issue No. A4 located on Page 9-10 of WCAP-15338-A provides a table of "Reactor Coolant System Transients for 40 Years," which is a generic 40-year transient set for normal, upset, and test conditions. The footnote to this table states that the "60-year number of transients is 1.5 times the 40-year number."

Request:
Please state whether the transient table shown on Page 9-10 of WCAP-15338-A still represents the generic 40-year transient set for calculating the 80-year cumulative FCG, based on the assumption of twice the 40-year cycles per Section 5.4 of the TR. If the generic 40-year transient set listed in this table has been updated since 2002, please provide the updated transient table used for the 80-year FCG calculation, or describe how the generic numbers and types of transients for normal, upset, and test conditions have changed since then.

Westinghouse Response

The transient table shown on Page 9-10 of WCAP-15338-A [1] refers to a list of transients based primarily on "Systems Standard Design Criteria 1.3". The standard set is typically modified to reflect specific steam generator types (e.g. SSDC 1.3F, SSDC 1.3X) and also consider operational experience (e.g. different number of transients). The 80-year cumulative fatigue crack growth (FCG) calculation uses twice the 40-year transient cycles specified in these standards which were for NSSS components and comprises an extensive set of transient descriptions used to represent limiting operational experience for design purposes. The transient set is meant to be representative of Westinghouse plants. This updated transient table is shown in the following page.

Reactor Coolant System Transients for 40 and 80 Years	
Transient Identification	**PWROG-17031 Number for 80 Years***
Normal Conditions	
1. Heatup and Cooldown at 100°F/hr	400
2. Load Follow Cycles (unit loading and unloading at 5% of full power/min)	26400
3. Step load increase and decrease of 10% of full power	4000
4. Large step load decrease, with steam dump	400
5. Steady state fluctuations, initial/random	3.0E5 / 6.0E6
6. Feedwater Cycling at Hot Shutdown	4000
7. Loop Out of Service, shutdown/startup	320**/140
8. Unit loading and unloading between 0% and 15% of full power	1000
9. Boron Concentration Equalization	52800
10. Refueling	160
Upset Conditions	
1. Loss of load, without immediate turbine or reactor trip	160
2. Loss of power (blackout with natural circulation in the RCS)	80
3. Loss of flow (partial loss of flow, one pump only)	160
4. Reactor trip	
-No cooldown	460
-Cooldown, no safety injection	320
-Cooldown with SI	20
5. Inadvertent RCS depressurization	40
6. Inadvertent startup of an inactive loop	20
7. Control rod drop	160
8. Inadvertent Safety Injection	120
9. Excessive Feedwater Flow	60
Test Conditions	
1. Turbine roll test	40
2. Primary side hydrostatic test	10
3. Primary Side Leakage Test	560

Notes:

* The 80-year number of transient cycles are 2 times the 40-year number.

** The Loop out of service shutdown transient was inadvertently increased by 4 times the 40-year cycle. Since it is conservative and has minimal effect on crack growth, the conservatism is allowed to be left in the analysis.

RAI-2: Fracture Toughness for Level C and D Transient Conditions

Issue:

Sections 5.5 and 5.6 of the TR address continued implementation of the same allowable flaw sizes that were previously established for 60-year applications in WCAP-15338-A (October 2002, ML083530289). This is based on consideration of the same governing transient characteristics for 3-Loop plants, as well as the continued use of time-invariant upper shelf fracture toughness (K_{Ic}) of 200 ksi√in for all transient analyses.

In order for an assumed K_{Ic} fracture toughness of 200 ksi√in to remain valid for 80-year applications, the RPV metal temperatures for all transients evaluated in the TR shall exceed the limiting adjusting RT_{NDT} values for the analyzed flaw depths by at least 104.25 °F; this is based on the K_{Ic} curve provided in the ASME Code, Section XI, Appendix A.

Request:

Considering that RPV beltline neutron embrittlement will result in significant shift in the RT_{PTS} and RT_{NDT} values for 80-year applications, please justify the continued use of 200 ksi√in as the generic RPV beltline material fracture toughness for determining the allowable flaw sizes for Level C and D service conditions in Section 5.6 of the TR.

<u>Westinghouse Response</u>

<u>Allowable Flaw Size Calculation</u>

PWROG-17031-NP [2] and WCAP-15338-A [1] calculate K_{Ic}, fracture toughness per ASME Section XI, Appendix A, A-4200. It is noted that K_{Ia} was not used in the underclad cracking evaluation. Since there is no prescribed upper limit in the ASME code, 200 ksi√in was conservatively used as a maximum value (or "upper shelf"), even if the calculated K_{Ic} is higher per the ASME Section XI, Appendix A, A-4200 formula. See Figure 1 for a visual demonstration of the 200 ksi√in value superimposed on the ASME Section XI, Appendix A K_{IC} curve.

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

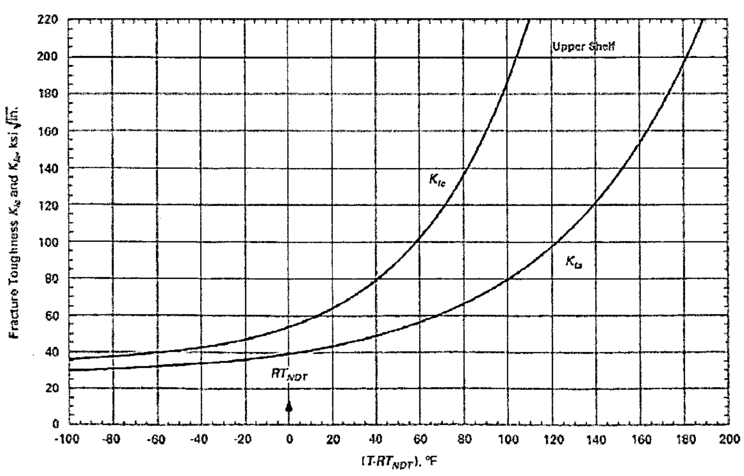

Figure 1. K_{Ic} Curve with 200 ksi√in Upper Shelf

(U.S. Customary Units)

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

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

All limiting transients for normal, upset, and test conditions have high fluid temperatures, and the calculated K_{Ic} exceeds 200 ksi$\sqrt{\text{in}}$ even if the 10CFR50.61 PTS screening criterion of 270°F is used. Therefore, K_{Ic} was limited to 200 ksi$\sqrt{\text{in}}$ to maintain conservatism and be in line with industry practices.

For transients of emergency and faulted conditions (Level C and D transients), if T- RT_{NDT} > 104.25 °F, 200 ksi$\sqrt{\text{in}}$ is used; otherwise, the K_{Ic} equation per A-4200 is used.

For the Steam Generator Tube Rupture and Small LOCA Level C and D transients, the calculated K_{Ic} exceeds 200 ksi$\sqrt{\text{in}}$ when using the 270°F 10CFR50.61 PTS screening criterion for RT_{NDT}. Typical Westinghouse plants have performed Leak Before Break (LBB) analysis and the implementation of LBB eliminates Large LOCA. Individual plants should confirm the implementation of LBB when referencing this report.

A generic Westinghouse main steam line break transient was provided to NRC in a response to NRC RAI 4.3.4-1a for Turkey Point Subsequent License Renewal [5]. This transient starts approximately at the cold leg temperature, then rapidly drops. As the transient continues, the temperature gradually decreases to approximately the boiling point of water at atmospheric conditions. The transient temperatures are not exclusively in the upper-shelf regime. Thus, K_{Ic} calculated per A-4200 is used to determine the critical flaw size. The critical flaw sizes for the Level C and D transients are based upon a typical Westinghouse Pressurized Water Reactor (PWR) for 60 years, as referenced in PWROG-17031-NP and described in WCAP-15338-A Section A-1. Consistent with the discussion in PWROG-17031-NP, Rev. 1 Section 5.6, RT_{NDT} is not expected to change significantly from 60 to 80 years as the rate of material embrittlement decreases at higher fluence levels. This "saturation" effect is evidenced by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," Figure 1.

A small increase in RT_{NDT} as a result of any additional neutron embrittlement can be accommodated given that the maximum flaw depth due to fatigue crack growth for 80 years is 0.4267 inches as shown in PWROG 17031-NP, Section 5.4. This represents a significant margin compared to the Normal/Upset/Test allowable flaw depth of 0.67 inches. As a further conservatism, underclad cracks are assumed to be surface flaws which results in a conservative K_I. The surface flaw assumption also results in a higher calculated fatigue crack growth rate as it considers a water environment.

It is important to note that the Level A/B allowable flaw size from PWROG-17031-NP [2] is 0.67", while the Level C/D allowable flaw size is 1.25". The 60-year to 80-year reduction of K_{IC} and the allowable flaw size for Level C/D due to a fluence increase would have to be more than 46% in order for the Level C/D allowable flaw size (1.25") to be smaller than the Level A/B allowable flaw size (0.67"). This reduction is unlikely given the change in fluence and radiation damage from 60 years to 80 years. Therefore, the Level A/B allowable of 0.67" in the PWROG report remains bounding.

Pressurized Thermal Shock Considerations

It is important to note that the reactor vessel must be protected from failure in two separate regions of operation, the high temperature "ductile" region and the lower temperature "brittle" region. The allowable flaw size determination demonstrates that an underclad crack will not propagate leading to a reactor vessel failure in the ductile region. Using an RT_{NDT} of 270°F (consistent with the 10CFR50.61 PTS screening criterion) ensures a K_{Ic} value of 200 ksi$\sqrt{in}$ will be used to a temperature of approximately 375°F. When using a lower RT_{NDT}, 200 ksi$\sqrt{in}$ is applicable to a lower temperature. In the lower temperature region, where brittle failure is a concern, the plant is protected by pressure-temperature limit curves (for normal heatup and cooldown operations) and 10CFR50.61 (The PTS Rule).

Regardless of the RT_{NDT} value utilized for the critical flaw size determination in WCAP-15338-A and PWROG-17031-NP, protecting the beltline region of a PWR Reactor Vessel (RV) from fracture during a large steam line break is ultimately ensured through compliance with 10 CFR 50.61. This regulation requires licensees of all operating PWRs to maintain licensed values of the reference temperature for pressurized thermal shock (RT_{PTS}) for each beltline material. These values must be below the screening values of 270°F for plates, forgings, and axial welds or below 300°F for circumferential welds. If RT_{PTS} values are projected to exceed the screening criteria, "the licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the PTS screening criterion." Additionally, licensees may subject the RV to thermal annealing or demonstrate compliance to PTS regulations via evaluation consistent with 10 CFR 50.61a. It is noted that to date, only one U.S. PWR has implemented 10 CFR 50.61a.

The NRC's original position on Pressurized Thermal Shock is summarized in Policy Issue SECY-82-465, which affirms through transient analysis and probability-weighted flaw distributions that the risk from PTS events for reactor vessels with RT_{NDT} values less than the proposed screening criterion is acceptable. It also provides, in significant detail, the basis for this conclusion, which includes an analysis of PTS transients. The PTS transients analyzed include main steam line break and small LOCA, amongst others.

A subsequent NRC study of PTS was published in NUREG-1874, which stated that "It is now widely recognized that the state of knowledge and data limitations in the early 1980s necessitated conservative treatment of several key parameters and models used in the probabilistic calculations that provided the technical basis for the current PTS Rule." NUREG-1874 confirms, through additional analysis of PTS transients, that the 10 CFR 50.61 methods and screening criteria are conservative.

NUREG-1874 provides quantitative analysis based on limiting the Through-Wall Cracking Frequency (TWCF) term for a vessel to 1×10^{-6}/ry, which is considered an acceptable risk, for multiple transients including a main steam line break. NUREG-1874 determines RT limits based on the TWCF limit. These RT limits are identical to those in 10CFR50.61a. Therefore, by mandatory compliance with 10CFR50.61a (or the more conservative 10CFR50.61), a low risk of vessel failure is ensured.

NUREG-1874 analyzed the main steam line break transient with respect to TWCF specifically, and concluded the following with regard to the main steam line break transient:

"...[E]ven though these transients produce an extremely rapid initial cooling rate of the RCS inventory (as a result of the large break area) the minimum temperature of the RCS (the boiling point of water) is generally high enough to ensure a high level of fracture toughness in the vessel wall, thereby preventing

MSLB *[Main Steam Line Break]* transients from contributing significantly to the total TWCF *[through-wall cracking frequency]* estimated for a plant."

The NRC PTS studies in SECY-82-465 and NUREG-1874 provide rigorous quantitative analysis demonstrating that PTS transients do not pose a significant risk if the mandatory requirements of 10CFR50.61 or 10CFR50.61a are met. Thus, since a main steam line break transient is considered a PTS transient, mandatory compliance with 10 CFR 50.61 or 10 CFR 50.61a inherently ensures beltline vessel integrity during this transient particularly in the low temperature region.

Elastic-Plastic Fracture Mechanics Approach

PWROG-17031-NP [2] followed the same linear elastic fracture mechanics (LEFM) methodology as is documented in WCAP-15338-A. LEFM conservatively idealizes the crack tip to be a sharp singularity and characterizes the crack tip using stress intensity factor, K, which depends on stress and crack geometry. A different approach to address the allowable flaw size is to use Elastic-Plastic Fracture Mechanics (EPFM), which removes conservatism in LEFM by considering crack tip blunting and calculates the applied J-integral around the crack tip. The calculated applied J-integral is compared to the J-material, a property that describes the material's ability to resist crack extension. ASME Section XI, Appendix K provides the EPFM analysis guidance and acceptance criteria. Areva Report, BAW-2178, Supplement 1NP-A [4], performed an Equivalent Margins Analysis (EMA) for certain reactor vessel Linde 80 welds with projected 80-year upper-shelf energy (USE) below 50 ft-lb. EMA analysis uses the EPFM approach. The Linde 80 materials are typically regarded as the most limiting group of materials for the U.S. PWR reactor vessel operating fleet considering material properties, fluences, and location. The EMA uses stresses from Surry and Turkey Point plant-specific finite element analyses and considers two steam line break transients, one of which is the Westinghouse generic large steam line break (LSB) transient from "Systems Standard Design Criteria 1.3". A very similar generic LSB transient was used in WCAP-15338-A for the allowable flaw size determination. Additionally, WCAP-15338-A considers the 3-loop configuration (such as Turkey Point and Surry) to be representative. Per ASME Section XI, Appendix K, K-2300 for Level C/D loadings, EMA postulates a flaw with depth equal to 1/10 the base metal thickness plus cladding but no larger than 1.0". The 0.67" allowable flaw size in the base metal used in the underclad cracking evaluation, PWROG-17031, is bounded by the accepted flaw depth in the base metal from the Turkey Point and Surry EMA (Level C/D), BAW-2178, Supplement 1NP-A [4]. Therefore, the EMA evaluation provides an additional level of assurance that an underclad crack would not cause a reactor vessel failure.

Summary

Through the combination of the allowable flaw size calculation, PTS considerations, and the use of EPFM, the issue of underclad cracking has been analyzed from multiple perspectives. As a result, it is concluded that the existence of underclad cracks do not pose a significant risk to plant operation to at least 80 years.

RAI-3: Allowable Flaw Depths for Large Steamline Break Transient

Issue:

Section 5.6 of the TR cites transient analyses for Level C and D service conditions as the basis for the allowable axial flaw sizes in Table 5-5 of the TR. The staff noted that analysis results for the "Large Steamline Break" transient in Section A-5, Table A-5.1 of WCAP-15338-A show a more limiting critical flaw depth for the continuous circumferential flaw (2.21 inches) compared to the critical flaw depth for the continuous axial flaw (2.50 in.). Considering its reliance on the assumed K_{Ic} fracture toughness of 200 ksi$\sqrt{in}$ for the upper shelf temperature regime, this analysis result is inconsistent with expected RPV beltline shell stress due to internal pressure. In theory, for the RPV shell region, it would be expected that the hoop stress from RCS pressure acting on axial flaws is about twice the axial stress acting on circumferential flaws.

Request:

Considering the RPV shell axial stress versus RPV shell hoop stress due to RCS pressure and a fixed K_{Ic} value of 200 ksi$\sqrt{in}$, please explain how the IWB-3610 analysis of the Large Streamline Break transient can result in a more limiting critical flaw depth (2.21 in.) for the continuous circumferential flaw compared to the 2.50 in. critical flaw depth for the continuous axial flaw. If this is a typographical error, please correct it in WCAP-15338-A and in the TR.

Westinghouse Response

Westinghouse agrees with NRC that the pressure hoop stress for axial flaws is higher than the pressure axial stress for circumferential flaws. The large steam line break transient results in a continuous circumferential flaw size of 2.64 inches.

This is a typographical error that is in both WCAP-15338-A [1] and PWROG-17031-NP Rev. 0. An errata letter, OG-18-267 [3] was issued for WCAP-15338-A documenting the typographical correction. Report PWROG-17031-NP Revision 0 has been revised to a Revision 1. The table containing this typographical error has been removed from Revision 1 of PWROG-17031-NP, and no further action is required for this report.

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. Westinghouse Topical Report, PWROG-17031-NP, Rev. 1, "Update for Subsequent License Renewal: WCAP-15338-A, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants"," May 2018.
3. PWROG Letter, OG-18-267, "Submittal of Errata Page for WCAP-15338-A, 'A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants' (PA-MSC-1497)," October 31, 2018.
4. Areva Report, BAW-2178, Revision 0 Supplement 1NP-A, Rev. 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads," December 2018.
5. Westinghouse Letter, LTR-SDA-18-131-P, Rev. 0, "Westinghouse Response to NRC Request for Additional Information on Turkey Point Subsequent License Renewal (RAI 4.3.4-1a)," January 17, 2019.

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