Program Management Office
1000 Westinghouse Drive, Suite 172
Cranberry Township, PA 16066

PWROG-17031-NP, Revision 1
Project Number 99902037

December 5, 2019

OG-19-258

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: PWR Owners Group
Transmittal of PWROG Comments on the Draft SE 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

5. Letter OG-19-184, 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, dated August 29, 2019

D048
NRR

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 2). The NRC Staff identified additional information needed to complete the review per the email dated October 30, 2018 and February 21, 2019 (References 3 and 4). Formal responses were provided to the NRC on August 29, 2019 (Reference 5). On November 1, 2019 the NRC provided the draft SE for review and comment.

Attachment 1 to Enclosure 1 to this letter provides PWROG comments on the draft SE 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." Attachment 2 to Enclosure 1 is a mark-up of the draft SE.

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Executive Director
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

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
PWR Owners Group

JKS:am

Enclosure: Comments on draft SE and mark-up of the draft SE (non-proprietary), LTR-SDA-19-098, Revision 0

cc: PWROG Analysis Committee (Participants of PA-MSC-1497)
PWROG PMO
PWROG Steering and Management Committee
L. Fields, US NRC
J. Andrachek, Westinghouse
T. Zalewski, Westinghouse
L. Patterson, Westinghouse
G. Hall, Westinghouse
A. Udyawar, Westinghouse
M. Semmler, Westinghouse
B. Mays, Westinghouse

Westinghouse

To: Jim Molkenthin
cc: Chad M. Holderbaum

Date: December 3, 2019

From: Benjamin E. Mays
Ext: 412-342-1793

Our ref: LTR-SDA-19-098, Revision 0

Subject: PWROG Comments on the Draft Safety Evaluation for PWROG-17031-NP, Revision 1

By letter dated May 31, 2018 (Agencywide Documents and Access Management System [ADAMS] Accession No. ML18164A034), as supplemented by letter dated August 29, 2019 (ADAMS Accession No. ML19253B327), the Pressurized Water Reactor (PWR) Owners Group (PWROG) submitted to the U.S. Nuclear Regulatory Commission (NRC) topical report (TR) 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'," (ADAMS Accession No. ML18164A035) for review and approval. The U.S. NRC staff transmitted its draft safety evaluation (DSE) via a letter dated October 29, 2019 (ADAMS Accession No. ML19300A001).

Attachment 1 contains the PWROG comments on the DSE and identifies the specific changes to the DSE to address the comments. Attachment 2 contains a markup of the DSE to reflect the changes identified in Attachment 1.

Please transmit the comments to the NRC so that the comments can be incorporated in the Final Safety Evaluation (FSE).

Electronically Approved*
Author: Benjamin E. Mays
License Renewal, Radiation Analysis, and Nuclear Operations

Electronically Approved*
Reviewer: Gordon Z. Hall
Structural Design & Analysis

Electronically Approved*
Reviewer: James D. Andrachek
Licensing Engineering

Electronically Approved*
Manager: Michael G. Semmler
License Renewal, Radiation Analysis, And Nuclear Operations

***Electronically approved records are authenticated in the electronic document management system.**

Attachment 1

Table 1 – Comments on the Draft Safety Evaluation (DSE) for PWROG-17031-NP, Revision 1

Comment #	DSE Page No.	Comment Type	PWROG Comment	NRC Response
1	1	Clarification	Please add: "Nuclear Steam Supply System" after "Westinghouse Electric Company (Westinghouse)"	
2	8	Editorial	Please remove capital letter in the word "Initial".	
3	8	Editorial	Please add: "in" before "WCAP-15338-A" in the final paragraph on this page.	
4	8	Clarification	Please replace "it" with "this method" in the sentence "Therefore, it is acceptable."	
5	11	Clarification. This clarification addresses the use of 200 ksi√in as an upper-shelf value.	Please add: "limited to a value no greater than" before "two-hundred thousand pounds per square inch times the square root of an inch" on the first line of this page.	
6	11	Editorial	Please add: "of" between "use" and "200 ksi√in".	
7	11	Clarification	Please replace "Revision 0" with "Revision 1" after "PWROG-17031-NP".	
8	11	Clarification	Please define "PTS" as "Pressurized Thermal Shock (PTS)".	
9	12	Clarification	Please add: "for reactor coolant system primary loop piping" after "confirm the implementation of LBB".	
10	12	Editorial	Please replace "bring equal" with "being equal".	
11	13	Clarification	Please replace "the a 60 to 80" with "the 60 to 80-year".	
12	13	Editorial	Please replace "Ki" with "K_I".	
13	15	Clarification.	Please add: "The NRC staff notes that it is unlikely that actual RCS transients and cycles for SPEO will exceed the number of design cycles conservatively considered in the transient table in Reference 2. However," to the beginning of Action Item 1.	
14	15	Clarification. Not every transient in the table applies to every reactor vessel in the PWR fleet.	Please add: "for the actual applicable transients" before "for the SPEO" in Action Item 1.	
15	15	Clarification	Please replace "the 270°F limit" with "the PTS screening criterion of 270°F".	
16	16	Clarification	Please add: "alloy" between "low" and "steel".	
17	16	Clarification	Please replace "continuous length" with "continuous flaw shape".	
18	16	Editorial	Please replace "low allow steel" with "low alloy steel".	

Attachment 2

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

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT 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'"

EPID: L-2018-TOP-0022

Nuclear Steam Supply System

1.0 INTRODUCTION AND BACKGROUND

By letter dated May 31, 2018, the Pressurized Water Reactor Owners Group (PWROG) submitted Topical Report (TR) PWROG-17031-NP, Revision (Rev.) 1, "Update for Subsequent License Renewal: WCAP-15338-A, 'A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants'" (Ref. 1), dated May 2018 for U.S. Nuclear Regulatory Commission (NRC) review and approval. Additional information related to PWROG-17031-NP, Rev. 1 (also referred to herein as the TR) was submitted by letter dated August 29, 2019 (Ref. 2) in response to a request for additional information (RAI) from the NRC staff.

This TR proposes to extend the applicability of the reactor pressure vessel (RPV) underclad crack analysis methodology in WCAP-15338-A (Ref. 3) from 60 to 80 years of operation to support applications of subsequent license renewal (SLR) for all U.S. Westinghouse Electric Company (Westinghouse) plants. WCAP-15338-A, dated October 2002, provides the NRC-approved generic methodology for analysis of the impact of underclad cracks on RPV structural integrity for 60-year operating periods. The underclad crack analysis in WCAP-15338-A is based on the methods and acceptance criteria of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, IWB-3610 for analytical evaluation of RPV flaws using linear elastic fracture mechanics (LEFM). In its September 2002 safety evaluation (SE) accompanying WCAP-15338-A, the NRC staff concluded that the report adequately demonstrated that RPVs with underclad cracks in Westinghouse 2-Loop, 3-Loop, and 4-Loop plants are acceptable for 60-year operating terms. The NRC staff's conclusion was based on its review the 60-year crack growth analysis considering the bounding RPV flaw characteristics from industry studies, and its determination that the bounding projected crack size satisfies the ASME Code, Section XI, IWB-3610 acceptance criteria for 60-year terms. The NRC staff's SE also concluded that upon completion of the license renewal applicant action items specified therein, the WCAP-15338-A report is acceptable for referencing as a basis for time-limited aging analysis (TLAA) of RPV underclad cracks in initial license renewal applications for Westinghouse plants.

The original evaluation of the impact of cracks beneath austenitic stainless-steel cladding (underclad cracks) on RPV structural integrity is documented in Topical Report WCAP-7733, dated April 1971 (Ref. 4), wherein Westinghouse presented a fracture mechanics analysis to justify the operation of Westinghouse plants for 32 effective full power years (EFPY) with the underclad cracks in the RPVs. The U.S. Atomic Energy Commission (AEC) staff accepted the TR in 1972 as a technical basis for demonstrating the acceptability of underclad cracks in Westinghouse plant RPVs for the original 40-year license term.

2.0 BACKGROUND AND REGULATORY EVALUATION

PWROG-17031-NP, Revision 1, was submitted for NRC review and approval in accordance with the Commission's TR review program to provide the regulatory and technical basis for analysis of RPV underclad cracks for referencing in plant licensing applications. The TR is to be implemented as the basis for a TLAA of RPV underclad cracks in applications for SLR under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

The regulation at 10 CFR Part 54, Section 54.3 (10 CFR 54.3), "Definitions," defines TLAAs as those licensee calculations and analyses that:

(1) Involve systems, structures, and components (SSCs) within the scope of license renewal, as delineated in 10 CFR 54.4(a);

(2) Consider the effects of aging;

(3) Involve time-limited assumptions defined by the current operating term;

(4) Were determined to be relevant by the licensee in making a safety determination;

(5) Involve conclusions or provide the basis for conclusions related to the capability of the SSC to perform its intended functions, as delineated in 10 CFR 54.4(b); and

(6) Are contained or incorporated by reference in the current licensing basis (CLB), as defined in 10 CFR 54.3.

Pursuant to 10 CFR 54.21(c), each application for license renewal (LR), including applications for SLR, shall include an evaluation of TLAAs. Pursuant to 10 CFR 54.21(c)(1), 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.

The TR proposes to extend the applicability of the RPV underclad crack analysis methodology in WCAP-15338-A from 60 to 80 years of operation. The 80-year operating term generally bounds the subsequent period of extended operation (SPEO) for SLR applications. As addressed in the NRC staff's technical evaluation below, this extension is based on generic projection of certain time-limited inputs into the analysis, such that SLR applicants invoking this methodology, as approved by the NRC staff, would have a generic basis for determining that their RPV underclad crack analyses have been projected to the end of the SPEO.

Therefore, based on the above regulatory requirements, and subject to the following technical evaluation, the NRC staff finds that the TR would constitute a regulatory and technical basis for analysis of RPV underclad cracks for Westinghouse plants in accordance with 10 CFR 54.21(c)(1)(ii).

3.0 SUMMARY OF PWROG-17031-NP, REVISION 1

The PWROG-17031-NP, Revision 1, report provides an analytical evaluation to generically demonstrate that RPV forgings with underclad cracks in Westinghouse plants are projected to satisfy the LEFM acceptance criteria of the ASME Code, Section XI, Paragraph IWB-3610 for 80-year operating periods. A summary of the TR is provided below.

Section 2 of the TR addresses the mechanisms of RPV underclad cracking. This discussion is consistent with the information on underclad cracking mechanisms provided in WCAP-15338-A. RPV underclad cracking was initially detected in 1970 and has been extensively investigated by industry over a 30-year period. Underclad cracking is a fabrication defect that has occurred in the base metal heat affected zone (HAZ) of low alloy steel (LAS) RPV forgings directly beneath the austenitic stainless-steel cladding; the stainless-steel cladding was deposited by weld overlay on the LAS base metal to protect the RPV interior from general corrosion. Underclad cracking has been identified only in ASME Code, SA508, Class 2 and Class 3 RPV forgings. Underclad cracks initiated at or near the clad/base metal weld fusion line and penetrated the RPV forging base metal.

The "reheat cracking" mechanism has occurred in SA508, Class 2 forgings because of post weld reheating after cladding was applied using certain high-heat-input welding processes. Reheat cracks were detected and evaluated primarily by destructive evaluation of both laboratory samples and clad nozzle forging cutouts. The cracks are often numerous and are confined to a region that is about 0.165-inch-deep and about 0.5-inch-long in the forging base metal directly beneath the cladding. The "cold cracking" mechanism has occurred in SA508, Class 3 forgings after deposition of the second and third layers of cladding, where no pre-heating or post-heating was applied to subsequent cladding layers. Cold cracking was attributed to weld residual stresses near the yield strength in the weld/base metal interface after cladding deposition, combined with crack-sensitive HAZ microstructure and high levels of diffusible hydrogen in the austenitic stainless steel; the hydrogen diffused into the HAZ and caused cold hydrogen-induced cracking as the HAZ cooled. Destructive analyses revealed 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.

Section 3 of the TR reviews and updates industry operating experience (OpE) associated with underclad cracking as well as industry OpE associated with RPV interior surface flaws that involve degraded or missing cladding. The TR cites the historical OpE discussion in WCAP-15338-A for underclad cracking mechanisms described above and provides additional updates for OpE with cladding defects in PWRs since 1999. The historical OpE information presented in WCAP-15338-A documents extensive investigations by the industry during the 1970's through the 1990's, including fabrication surveys, preservice inspections to baseline the condition of RPVs with underclad cracks, and subsequent inservice inspection (ISI) per the ASME Code, Section XI. ISI conducted on RPVs during this time revealed no measurable growth in the known crack indications. All reported underclad crack indications met the ISI acceptance standards of the ASME Code, Section XI, IWB-3500 (i.e., allowable flaw criteria). Thus, all RPVs with documented underclad cracks were acceptable for continued operation without analytical evaluation of RPV structural integrity using LEFM analysis techniques per

IWB-3610. This OpE review identifies that the maximum flaw depth of 0.295 inch and the maximum flaw length of 2.0 inches were established based on destructive analyses to examine cold cracking in SA508, Class 3 forgings.

WCAP-15338-A also reviewed OpE for RPVs with cladding defects that involve exposed regions of LAS base metal. In some cases, the removal of cladding in some nozzle forgings was carried out many years ago as a corrective action to determine the extent of underclad crack penetration, resulting in a relatively flat LAS surface exposed to RCS water; these types of interior cladding flaws do not challenge the structural integrity of the RPV. Historical OpE has shown that RPV interior cladding flaws involving exposed LAS base metal have exhibited no significant penetration into the base metal or other detrimental effects.

In addition to the historical OpE information presented in WCAP-15338-A, the Section 3 of the TR provides an update to the OpE information regarding RPV cladding defects that involve exposed RPV base metal. This update documents additional cladding defects that were discovered at Callaway, Diablo Canyon, and a Chinese plant in the 2000s and early 2010s, and it addresses the status of the earlier cladding defects documented in WCAP-15338-A. This OpE information indicates that these types of cladding defects have not shown a significant change during plant operation, and they continue satisfy applicable acceptance standards.

Section 4 of the TR briefly summarizes earlier experimental studies of the effects of cladding on RPV fracture behavior and fracture mechanics analysis. The TR cites Section 4 of WCAP-15338-A, which documents fracture tests and cladding residual stress measurements. These experiments included three-point bending fracture tests conducted on clad LAS bend bar test specimens with machined-in surface flaws thru the cladding into the base metal, and measurements of cladding residual stress profiles in and near the weld fusion zone of clad pressure vessel steel using the hole-drilling residual stress measurement technique. The three-point bending fracture tests and cladding residual stress experiments were designed to measure fracture behavior and changes in residual stress profile over a range of temperatures and were conducted on specially-designed test specimens from RPV nozzle forging cutouts. As discussed in the WCAP, it was determined that the unfavorable effects of cladding residual stress on fracture behavior are more significant at lower temperatures, in and below the lower ductile-to-brittle transition region for low alloy pressure vessel steel. At temperatures greater than the transition temperature of the low alloy steel base metal, WCAP-15338-A determined that the cladding would not have a significant impact on fracture behavior.

Section 5 of the TR provides the generic structural integrity evaluation of RPVs with underclad cracks for 80 years of plant operation. Sections 5.1 and 5.2 of the TR briefly address factors that could potentially result in the inservice exposure of RPV LAS base metal to the reactor cooling water environment. These sections identify that damage to RPV cladding can potentially occur due to mechanical impact loads, and any exposed base metal could undergo some degree of general corrosion. The TR identified that the amount of corrosion of exposed LAS base metal has been shown to be insignificant based on OpE with such flaws and laboratory studies of LAS corrosion in PWR water environments. Consistent with WCAP-15338-A, the TR determines that inservice material aging mechanisms such as SCC and fatigue are expected to remain non-credible as mechanisms for the formation of new flaws for 80 years of plant operation. This determination is based on the substantial absence of oxidizing conditions and aggressive anion species in the reactor coolant and very low cumulative usage factors (CUF).

Consistent with the WCAP-15338-A, Section 5.3 of the TR establishes that RPV underclad cracks have been found to be limited to a maximum depth of 0.295 inch based on the OpE discussed in Section 3 and that all reported underclad crack indications are within the flaw acceptance standards of the ASME Code, Section XI, IWB-3500. However, the TR notes that the NRC staff had requested in its 2001 RAI that the underclad crack evaluation in the WCAP-15338 submittal be supplemented to include an analytical evaluation of RPV structural integrity using the LEFM methods and acceptance criteria of IWB-3610. The staff's RAI and the industry's RAI response were included in Section 8 of WCAP-15338-A under the appendix, "ASME Code Section XI Flaw Evaluation." Sections 5.4, 5.5, and 5.6 of TR provide the updated analytical evaluation to generically demonstrate that U.S. Westinghouse RPVs with underclad cracks are projected to satisfy the LEFM acceptance criteria of IWB-3610 for 80-years, consistent with the 60-year methods in WCAP-15338-A.

Section 5.4 of the TR provides generic 80-year fatigue crack growth (FCG) calculations for underclad cracks in RPV forgings. The FCG calculations are based on ASME Code, Section XI, Appendix A FCG rate curves for low alloy ferritic steel in a water environment and application of a generic set of 40-year design transient cycles times a factor of 2.0 to account for 80-years of operation. 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. These allowable flaw sizes were determined in accordance with acceptance criteria and methods for analytical evaluation RPV flaws in the ASME Code, Section XI, IWB-3610 and Appendix A. The allowable flaw sizes were determined based on the same governing transient characteristics for normal, upset, and test conditions (Service Levels A and B), and emergency and faulted conditions (Service Levels C and D), as well as the continued use of certain assumptions for RPV beltline fracture toughness for 80-year applications.

The NRC staff's review of the industry OpE for RPV flaws, analytical flaw evaluation methods, time-dependent inputs, and assumptions that were used for the 80-year flaw evaluations is documented in Section 4.0 of this SE.

4.0 NRC STAFF TECHNICAL EVALUATION

4.1 Background – NRC Position on RPV Underclad Cracking

Regulatory Guide (RG) 1.43, Revision 1, March 2011 (Ref. 5), provides the NRC staff's regulatory position on the control of welding processes for cladding of SA508, Class 2 RPV forgings. The recommendations of RG 1.43 are limited to cladding fabrication process and weld procedure qualification to avoid conditions that result in crack formation due to the reheat cracking mechanism in SA508, Class 2 forging materials. It should be noted that *the RG does not take a position on methods for preexisting flaw evaluation for operating plant RPVs*; however it does briefly address industry experience with inspections and evaluations of underclad cracks for operating plants and it cites the AEC and NRC-approved generic studies and fracture mechanics evaluations of underclad cracks provided in the earlier topical reports, WCAP-7733 and WCAP-15338-A for 40-year and 60-year operating periods, respectively. The RG identifies that underclad cracks are difficult to detect using conventional non-destructive examination NDE techniques, and adequate detection often requires destructively removing the cladding to the weld fusion line and examining the exposed base metal with metallographic techniques, or with liquid penetrant or magnetic particle testing methods. Therefore, as established in generic evaluations cited below, the existence of underclad cracks in SA508, Class 2 and Class 3 RPV forging materials cannot be conclusively ruled out even if a given plant has not detected them during fabrication and pre-service exams, or during subsequent ISI activities.

Topical Report WCAP-7733 provided the original evaluation of the effects of underclad cracks on RPV structural integrity. This analysis was accepted by AEC staff in 1972 as the technical basis for determining that all Westinghouse plant RPVs with underclad cracks would remain acceptable for 32 EFPYs, which corresponded to the terms of the original 40-year operating licenses. The generic evaluation in WCAP-7733 was limited to the analysis RPV underclad cracks caused by the reheat cracking mechanism. Subsequently, underclad cold cracking was discovered in 1979 for SA508, Class 3 nozzle forgings that were clad using multilayer welding processes, where no heat treatment was applied to subsequent layers. The RPV nozzle bores in six U.S. plants considered to be susceptible to cold cracking were examined using a special ultrasonic testing (UT) technique developed to detect underclad cracks. These UT exams confirmed the existence of flaws in the nozzle bores that were indicative of cold cracking. All these flaws met the allowable flaw acceptance standards of the ASME Code, Section, IWB-3500. Four of these flaws were destructively evaluated to determine that the cold cracks in SA508, Class 3 nozzles are limited to a maximum flaw depth of 0.295 inch and the maximum flaw length of 2.0 inches.

Since 1972, fracture mechanics analysis techniques have improved significantly. To reflect this improvement, Westinghouse developed WCAP-15338, which was submitted for NRC review in 2001. As documented in its SE accompanying the NRC-approved version, WCAP-15338-A (October 2002), the NRC staff found the report acceptable for referencing as the generic basis for underclad crack analysis in initial LR applications covering 60-year operating terms. The report was approved by the NRC for generic application to RPV underclad crack TLAAs for all U.S. Westinghouse plants. This report employed modern LEFM and FCG analysis techniques that are now considered to be the standard for conservative analytical evaluation of RPV flaws that are detected during plant ISI, as required by the ASME Code, Section XI and 10 CFR 50.55a. As the basis for its generic analytical evaluation of underclad cracks, WCAP-15338-A included a comprehensive review of industry OpE for RPV flaws that were detected both in and underneath RPV cladding. To ensure the analysis encompasses industry experience with these types of RPV flaws, the WCAP report analyzed a series of crack sizes and shapes, crack orientations, and crack locations. The most bounding initial crack size used in the analysis had a depth 0.30 inch, which slightly exceeds the maximum 0.295-inch flaw depth that was observed for all detected underclad cracks based on destructive evaluation of cold cracking in SA508, Class 3 nozzle forgings.

4.2 NRC Staff Review of Industry OpE – Defects in and Underneath RPV Cladding

For its evaluation of the 80-year TR, the NRC staff reviewed the historical OpE information presented in WCAP-15338-A (as cited in the TR) and additional OpE for RPV cladding defects since 1999. The NRC staff's review of industry OpE with detected RPV flaws located in and below cladding generally confirms that these types of flaws have not yet been shown to be a structural integrity problem for RPVs in operating U.S. plants. This is based on the fact that the occurrence and behavior of RPV underclad cracks and interior surface flaws has been extensively investigated and monitored over many years; and licensees for all U.S. plants continue to perform comprehensive inspection and evaluation of their RPVs in accordance with ASME Code Section XI and 10 CFR 50.55a requirements. All detected RPV flaws, whether they are in the cladding, thru the cladding, or underneath the cladding, continue to satisfy the ASME Code, Section XI acceptance standards; and there have been no detected underclad cracks or interior surface flaws with dimensions that exceed the maximum flaw depth of

0.295 inch and maximum flaw length of 2.0 inches based on the destruction evaluation of cold cracking in SA508, Class 3 nozzle forgings described above.

The NRC staff confirmed that the initial crack depth of 0.30 inch is still considered to be bounding for industry OpE with detected underclad cracks based on the maximum crack depth of 0.295 inch, where the full depth of the flaw is considered to be completely through the thickness of the low alloy steel. As established through the NRC staff's review of WCAP-15338-A, the flaw is analyzed as an interior surface flaw that is exposed to the RCS water environment[1]. Based on the factors discussed below, the staff determined that this remains a conservative and appropriate assumption for SLR applications.

It should be noted that RPV interior surface flaws that involve degraded or missing cladding do not originate from the underclad cracking mechanism. The root cause of such RPV cladding defects varies. The NRC staff's review of more recent OpE in this area has shown that they were likely caused by disparate factors such as excessive mechanical grinding during RPV fabrication, surface damage due to mechanical impact—or most notably in certain instances, deliberate inservice removal of cladding and base metal in the interior of RPV nozzle forgings to assess the extent of actual underclad cracks. None of these factors are related to material aging mechanisms. These types of flaws are often characterized as flat, shallow regions of exposed LAS base metal. While the exposed LAS regions are theoretically susceptible to general erosion and corrosion due to interaction with PWR reactor cooling water, the rate of corrosion is extremely low due to the low oxidizing potential in the PWR water environment. Further, general erosion and corrosion of exposed LAS does result in the formation of new cracks in the exposed region. The NRC staff's review of this OpE confirms that these types of flaws have not shown a significant amount of penetration into the RPV base metal, and they have continued to satisfy ASME Code, Section XI acceptance standards during plant service.

The NRC staff also confirmed that there are no credible material aging mechanisms for the formation of new interior surface flaws for intact cladding in PWR operating environments. The low oxidizing potential and absence of chlorides in PWR water environments precludes new flaw formation in cladding by SCC, and RPV design requirements for CUF preclude flaw formation by metal fatigue. Therefore, with respect to material aging, actual underclad cracks are generally expected to remain embedded beneath the cladding and are unlikely to become directly exposed to the RCS water environment.

Notwithstanding the above factors, the NRC staff had determined through its review of WCAP-15338-A that it cannot be ensured that existing cracks will not penetrate through the clad. Accordingly, the WCAP-15338-A analysis assumes that the full depth of the initial underclad crack is thru the RPV low alloy steel base metal, and that the crack is a surface flaw exposed to a water environment. This is necessary to ensure that flaw growth and LEFM analyses per the methods in IWB-3610 and Appendix A of the ASME Code, Section XI are bounding for these cases. Based on its review of the OpE for RPV interior flaws that are located in the cladding, thru the cladding, or underneath the cladding, staff finds that the TR's continued use of WCAP-15338-A assumptions regarding the initial crack sizes and characteristics are acceptable for 80-year applications.

---

[1] The NRC staff's RAI for the WCAP-15338 submittal requested that the evaluation be supplemented to address LEFM and FCG analysis of underclad cracks by conservatively assuming they are interior surface flaws. This surface crack analysis was provided in an "Appendix A" located in Section 8 of WCAP-15338-A and forms the basis for the bounding analytical flaw evaluation per IWB-3610.

4.3 RPV Flaw Analytical Evaluation, ASME Section XI, IWB-3610

Both the WCAP-15338-A report and the TR use consistent methods for the analytical evaluation of RPV interior surface flaws for 60-year and 80-year applications. These methods are based on the acceptance standards for analytical evaluation of RPV flaws in the ASME Code, Section XI, IWB-3610, including the LEFM procedures and FCG calculation methods in Appendix A of Code, as specified by IWB-3610. These methods are generally required for analytical evaluation of RPV flaws that are detected during plant ISI under 10 CFR 50.55a if the flaws exceed IWB-3500 acceptance limits. As established in WCAP-15338-A, the use of these methods is appropriate for referencing in TLAAs of RPV underclad cracking because even if a plant has not detected any underclad cracks, the potential for their existence in susceptible SA508, Class 2 and Class 3 RPV forgings cannot be ruled out. Therefore, implementation of the ASME Code, Section XI, IWB-3610 and Appendix A, considering bounding initial flaw parameters from OpE addressed above, remains the appropriate method for 80-year applications, provided that the analysis generically accounts for 60 to 80-year extension of time-dependent inputs. The NRC staff's evaluation of the 60 to 80-year extension of the generic analytical evaluation per IWB-3610 and Appendix A is addressed below.

80-Year Flaw Growth Projections

The 60-year WCAP-15338-A and 80-year PWROG-17031-NP, Revision 1 reports perform a generic FCG evaluation for a series of postulated RPV cracks, which consider various flaw depths and aspect ratios, axial and circumferential crack orientations, as well as crack locations in both the RPV beltline shell and the inlet nozzle regions. The initial crack sizes and characteristics used in the generic FCG evaluation are considered to be applicable to the preservice condition. The initial flaw parameters used for the 80-year TR are the same as those used in WCAP-15338-A. The ~~Initial~~ initial crack depths through the RPV low alloy steel range from 0.05 inch to 0.30 inch, which is the bounding initial crack depth based on the evaluation of industry OpE from destructive evaluation of underclad cold cracks discussed above. The initial crack lengths are established based on consideration of three flaw aspect ratios (length-to-depth ratios) of 2, 6, and 100. The aspect ratio of 100 is referred to as the "continuous" flaw shape, and it always provides the most bounding FCG result for any given initial flaw depth. Therefore, the most bounding initial crack size considered for the FCG analysis has a depth of 0.30 inch and an effectively continuous length of 30 inches, which exceeds the length of any flaw ever detected.

The 80-year FCG analysis in the TR projects that the bounding axial crack in the RPV beltline shell region, with initial depth of 0.30 inch and continuous crack length, will grow to about 0.43 inch in depth after 80-years of operation, based on FCG rate curves in the ASME Code, Section XI, Appendix A for low allow steel exposed to RCS water environments. This is the most bounding crack growth result for all crack shapes and crack orientations considering both RPV beltline shell and inlet nozzle locations. The use of FCG rate curves for a water environment is conservative for FCG rate calculations since realistically, actual underclad cracks are most likely not directly exposed to RCS water. As discussed above, the NRC staff's review of this FCG rate method for 60-year applications in WCAP-15338-A established that the water environment assumption is necessary if an underclad crack were to become a surface flaw. The staff determined that this FCG rate method is consistent with the FCG rate method in WCAP-15338-A, and it remains bounding for generic application to 80-year operating terms. Therefore, ~~it~~ this method is acceptable.

Using the FCG rate curves for water environments, the TR determined cumulative FCG by analyzing a generic set of transient cycles projected out to 80 years. The TR applied the full set of design transients for normal, upset, and test conditions over an 80-year period by multiplying the 40-year transient cycles by a factor of two to generically account for an 80-year term. For the initial flaw parameters discussed above, the TR projected the 80-year flaw sizes using the following iterative process: First, the range of fluctuation in the applied stress intensity factor (ΔK_I) for each transient load cycle is calculated; next, the incremental crack growth corresponding to ΔK_I is determined using the FCG rate data; third, the crack growth increment is added to the flaw size. The process is repeated for subsequent transient cycles until all cycles have been accounted. The staff verified that this process is consistent with the methods specified in the ASME Code, Section XI, Appendix A, Paragraph A-5200 to establish end-of-period flaw size for evaluation in accordance with IWB-3610 acceptance criteria.

In supplemental correspondence (Ref. 2), provided in response to the staff's RAI (RAI-1), the PWROG submitted its updated transient table identifying the reactor coolant system transients and the number of transient cycles for normal, upset, and test conditions; these transient cycles were applied for the 80-year cumulative FCG calculation in the TR. Reference 2 states that the 80-year transient cycles are twice the 40-year transient cycles specified in the Westinghouse Systems Standard Design Criteria, and they are meant to be generically representative for Westinghouse plants. For the 60-year cumulative FCG analysis, WCAP-15338-A had multiplied the 40-year standard set of transient cycles for normal, upset, and test conditions by a factor of 1.5 to account for 60-years of operation; therefore this 80-year transient cycle projection is consistent with the method used to project 60-year transient cycles for the cumulative FCG projection in WCAP-15338-A.

The NRC staff confirmed that the multiplication factors (1.5 for 60 years and 2.0 for 80 years) used for determining generic transient cycle projections are conservative because the 40-year standard transient set represents the number of cycles that were generically analyzed for meeting design requirements. It should be noted that plant-specific CUF TLAAs for primary RCS components incorporate a comparison of the accumulated number of transient cycles and projected transient cycles for PEOs and SPEOs to the number of transient cycles that were analyzed for meeting original 40-year design requirements for CUF; cycle count management is often used to ensure that corrective action is taken if the number of cycles accumulated during PEOs and SPEOs exceeds (or comes close to exceeding) the number of cycles that were analyzed for meeting design criteria for the original 40-year license. The staff noted that it is unlikely that actual RCS transient cycles would exceed 1.5 times the number of design cycles over 60 years and 2.0 times the number of design cycles over 80-years. However, individual SLR applicants should make this determination as part of a plant-specific TLAA in accordance with 10 CFR 54.21(c)(1)(ii). Therefore, consistent with the staff's basis for approving this generic projection for 60-year terms, as documented in its September 25, 2002, SE accompanying WCAP-15338-A, individual SLR applicants referencing this TR as the basis for a TLAA evaluation under 10 CFR 54.21(c)(1)(ii), should verify that their plant is bounded by the transient cycle inputs into the PWROG-17031-NP, Revision 1 FCG analysis for 80 years. Specifically, in their plant-specific TLAAs for RPV underclad cracks, SLR applicants are to indicate whether the generic transient types and projected number of transient cycles listed in Reference 2 for the 80-year FCG projection bounds the projected number of transient cycles for the SPEO. **This is TLAA Action Item 1**

ASME Section XI IWB-3610 Allowable Flaw Sizes for 80-Years Operating Periods

PWROG-17031-NP, Revision 1 documents the results of LEFM analyses for a representative Westinghouse 3-Loop plant to determine the allowable flaw sizes based on the ASME Code, Section XI IWB-3610 acceptance criteria. In accordance with IWB-3610, the Code-allowable flaw sizes are calculated based on the evaluation of transient loadings for normal, upset, and test conditions (Service Levels A and B), and emergency and faulted conditions (Service Levels C and D). The Code-allowable flaw sizes reported in the 80-year TR are the same as those reported in WCAP-15338-A for 60-year applications for all transient analyses.

Consistent with WCAP-15338-A, the 80-year TR reports that the most limiting allowable flaw depth for Service Levels A and B is 0.67 inch through a 7.75-inch-thick RPV beltline shell forging based on LEFM analysis of the continuous axial flaw for the Excessive Feedwater Flow Transient. Consistent with WCAP-15338-A, the most limiting allowable flaw depth for Service Levels C and D is 1.25 inch (7.75-inch-thick RPV beltline shell forging) based on LEFM analysis of a continuous axial flaw for the Large Steamline Break Transient. For the bounding continuous axial crack, the TR determined that the 80-year projected crack depth (0.43 inch) based on the FCG analysis is less than the limiting allowable flaw depths for Service Levels A and B (0.67 inch) and Service Levels C and D (1.25 inch). On this basis, the TR concludes that RPV underclad cracks for all U.S. Westinghouse plants are acceptable for 80-year operating periods.

Since the Code-allowable flaw sizes have not changed between the 60-year and 80-year versions of this methodology, the staff reviewed the time-dependent inputs and assumptions for determining allowable flaw sizes based on the methods in IWB-3610 and Appendix A of the Code, to address whether they would remain the same for 60-year and 80-year operating periods.

The TR indicates that the governing transient characteristics for determining Code-allowable flaw sizes for 80-year applications are the same as those in WCAP-15338-A for 60-year applications. As established in WCAP-15338-A, applied stress intensity factor (K_I) calculations for Service Levels A, B, C, and D were performed based on analysis of RPV transient loadings for a representative Westinghouse 3-Loop plant. The NRC staff's review of WCAP-15338-A determined that the analysis of the governing 3-Loop transients for determining applied K_I values and Code-allowable flaw sizes is acceptable for generic application to all Westinghouse Plants, including the 2-Loop and 4-Loop designs.

With respect to transient loadings on RPV cracks, the staff noted that there are no time-dependent aging affects. The severity of loadings on RPV cracks in the beltline shell region are primarily determined based on transient characteristics for RPV pressure and temperature versus time (e.g., pressurized rapid cooldown events lead to greater flaw loadings for the more severe transients). As such, the staff found that the WCAP-15338-A transient analyses for determining the applied K_I values for RPV beltline shell flaws will continue to remain valid for 80-years.

With respect to RPV beltline material fracture toughness (K_{IC}), the staff noted that the TR indicates that the Code-allowable flaw sizes for all transients were determined based on the following assumptions:

1. The RPV beltline material is in the upper shelf temperature regime for all transients evaluated in the TR for Service Levels A, B, C, and D;

limited to a value no greater than

2. The K_{IC} value used for all transient analyses is two-hundred thousand pounds per square inch times the square root of an inch (200 ksi$\sqrt{in}$); and

3. Any increase in the adjusted reference temperature (RT_{NDT}) caused by RPV beltline material embrittlement for SPEOs (60- to 80-year extended license terms) would be insignificant, relative to the impact on K_{IC} and the determination of allowable flaw size.

The staff noted that a K_{IC} value of 200 ksi$\sqrt{in}$ is, by convention, considered to be a conservative upper limit on K_{IC} for LEFM analysis of flaws in ferritic RPV materials per IWB-3610 and Code Appendix A; this K_{IC} value is valid only if certain criteria for material temperature and RT_{NDT} are satisfied. Specifically, for LEFM analysis under various transient conditions, the ASME Code, Section XI, Appendix A, Paragraph A-4200 (Code Paragraph A-4200) specifies that the K_{IC} value should be determined based on the lower bound K_{IC} curve. The equation specified in Code Paragraph A-4200 shows that the K_{IC} value increases as an exponential function of the metal temperature at the analyzed flaw depth minus the RT_{NDT} at the analyzed flaw depth. Therefore, K_{IC} should be determined based on the crack tip metal temperature for the analyzed transient conditions; and for RPV beltline materials, the adjusted RT_{NDT} at the crack tip should be used to account for the effects of neutron embrittlement, as specified in Code Paragraph A-4400. Based on the equation for K_{IC} specified in Code Paragraph A-4200, the RPV metal temperature must exceed the adjusted RT_{NDT} value for the limiting RPV beltline material by at least 104.25 °F for the analyzed flaw depths in order for the K_{IC} value to be
of greater than or equal to 200 ksi$\sqrt{in}$ for the analyzed transient conditions.

Considering the above ASME Code criteria for K_{IC}, and the projected state of RPV beltline neutron embrittlement for the SPEO, the staff requested that the PWROG justify the continued use 200 ksi$\sqrt{in}$ for RPV beltline materials to determine allowable flaw sizes for the transients evaluated in PWROG-17031-NP, ~~Revision 0~~. Revision 1

In supplemental correspondence (Ref. 2) provided in response to the staff's RAI (RAI-2), the PWROG reported the following aspects concerning its K_{IC} calculation for the transients evaluated in the TR. The first part of the RAI-2 response addresses all Service Level A, B, C, and D transients except for the Large Steamline Break transient:

- Both PWROG-17031-NP and WCAP-15338-A calculate K_{IC} in accordance with Code Paragraph A-4200.
- 200 ksi$\sqrt{in}$ was conservatively used as a maximum value (or "upper shelf") for K_{IC}, even if the calculated K_{IC} is higher per the equation in Code Paragraph A-4200.
- All limiting transients for Service Levels A and B have high fluid temperatures, and K_{IC} calculated per Code Paragraph A-4200 exceeds 200 ksi$\sqrt{in}$ even if the 10 CFR 50.61 ~~PTS~~ screening criterion of 270 °F is used for RT_{NDT}. Therefore, K_{IC} was limited to 200 ksi$\sqrt{in}$ to maintain conservatism and be in line with industry practices.
- For transients of emergency and faulted conditions (Service Level C and D transients), if the metal temperature minus RT_{NDT} is greater than 104.25 °F, 200 ksi$\sqrt{in}$ is used; otherwise, the K_{IC} equation per Code Paragraph A-4200 is used.
- For the TR evaluation of two of the Level C and D transients, Steam Generator Tube Rupture and Small LOCA, the calculated K_{IC} also exceeds 200 ksi$\sqrt{in}$ when using the 270°F PTS screening criterion for RT_{NDT}.
- With respect to the Large LOCA, typical Westinghouse plants have performed Leak Before Break (LBB) analysis, and the implementation of LBB eliminates Large LOCA

Pressurized Thermal Shock (PTS)

for reactor coolant system primary loop piping

from the design basis. The RAI-2 response specifies that individual plants should confirm the implementation of LBB when referencing this report.

Given the relatively high fluid temperatures for all Service Level A and B transients and the Level C Steam Generator Tube Rupture and Small LOCA transients, the staff confirmed that 200 ksi$\sqrt{in}$ would remain bounding based on using the 10 CFR 50.61 PTS screening limit of 270 °F for RT_{NDT}. Therefore, the staff finds that the K_{IC} value of 200 ksi$\sqrt{in}$ and thus the IWB-3610 allowable flaw depths would remain the same for 60-year and 80-year applications given these transient characteristics, and assuming the RPV beltline forgings meet the 10 CFR 50.61 screening limit for 80-year periods. RAI-2 is adequately addressed by the PWROG for these relatively high temperature transients. In order to apply this part of the RAI-2 response to plant-specific underclad crack TLAAs for 80-year applications, individual plants using the TR for 80-year underclad crack analyses should confirm that their limiting RPV beltline forgings that are of the SA508, Class 2 or Class 3 specification meet the PTS screening limit of 270 °F in 10 CFR 50.61. **This is TLAA Action Item 2.**

For the Large LOCA transient analysis, the staff noted that the fluid temperature would not be high enough to justify continued use of 200 ksi$\sqrt{in}$ as a bounding K_{IC} value; however, the staff confirmed that plant-specific implementation of the LBB analysis would eliminate the Large LOCA from consideration. Therefore, to apply this part of the RAI-2 response to 80-year underclad crack TLAAs, individual plants using the TR should address implementation of LBB for reactor coolant system primary loop piping as part of their 80-year SLR applications. **This is TLAA Action Item 3.**

Evaluation of the Large Steamline Break Transient

The second part of the RAI-2 response addresses the K_{IC} analysis for the Large Steamline Break (LSB) transient for determining the limiting allowable flaw size of 1.25 inch for Service Levels C and D. The LSB has low temperature characteristics, and it cannot be eliminated from consideration with the application of LBB. The PWROG RAI-2 response references the generic Westinghouse LSB transient data provided to the NRC staff in response to a similar RAI on this issue for the Turkey Point SLR application. The LSB transient starts at approximately the cold leg operating temperature and decreases to the boiling point of water at atmospheric conditions. The response identifies that transient temperatures are not exclusively in the upper-shelf regime, and K_{IC} calculated per Code Paragraph A-4200 is used to determine the critical flaw size. The response states that critical flaw size calculations for the Level C and D transients "*are based upon a typical Westinghouse Pressurized Water Reactor (PWR) for 60 years*," and the RT_{NDT} used for the calculation "*is not expected to change significantly from 60 to 80 years as the rate of material embrittlement decreases at higher fluence levels*." The response cited the generic neutron fluence factor (FF) curve in Figure 1 of RG 1.99, Rev. 2, which generally shows that for a given set of RPV beltline material properties (initial RT_{NDT}, chemistry factor (CF), margin term inputs), the rate of increase in the adjusted RT_{NDT} as a function of neutron fluence decreases at higher fluence values. The staff noted that Figure 1 of RG also shows that the actual increase in adjusted RT_{NDT} for a given set of properties is still significant over a 60 to 80-year extended operating period—even if it is less drastic than the increase over a 40 to 60-operting with all other inputs ~~bring~~ equal—and therefore this increase still needs to be evaluated to address the 60 to 80-year period. The staff noted that this response did not cite any 60 to 80-year adjusted RT_{NDT} calculation as the basis for its claim that the RT_{NDT} used for the LSB allowable flaw size calculation "*is not expected to change significantly from 60 to 80 years*." Instead the response indicates that the TR relies on the continued implementation of 60-year RT_{NDT} for determining the allowable flaw size of 1.25 inch for the LSB transient.

being

The RAI-2 response also indicates that a small increase in adjusted RT_{NDT} as a result of any additional neutron embrittlement can be accommodated given that the maximum projected flaw depth due to fatigue crack growth for 80 years is 0.4267 inches. This projected flaw depth is determined to be acceptable based on the most limiting of the allowable flaw sizes for all transients evaluated in the TR, which is the 0.67-inch depth continuous flaw for the Excessive Feedwater Flow transient (Service Level B). This transient will maintain a RT_{NDT} value of 200 ksi√in based on using the PTS screening limit for the RT_{NDT}. The response adds that, as a further conservatism, the analysis assumes that RPV underclad cracks are surface flaws, which results in a conservative value for K_I and a higher crack growth rate due to the assumption that the flaw is exposed to the RCS water environment.

The RAI-2 response emphasizes that the limiting allowable flaw size for Service Levels A and B is 0.67 inch for the Excessive Feedwater Flow transient, whereas the limiting allowable flaw size for Service Levels C and D is 1.25 inch for the LSB transient. Based on the difference between these allowable flaw sizes, the response identifies that the 60 to 80-year reduction in K_{IC} due to a fluence increase and the corresponding reduction in allowable flaw size for Levels C and D *would have to be more than 46 percent in order for the Level C and D allowable flaw size of 1.25 inch to become smaller than the Level A and B allowable flaw size of 0.67 inch*. This response states that this reduction is unlikely given the change in fluence and radiation damage from 60 years to 80 years. Therefore, the RAI-2 response concludes that the limiting Level A and B allowable flaw size of 0.67 inch would remain bounding for all transients evaluated in the TR; and the maximum projected flaw depth of 0.4267 inches for 80-years would be acceptable based on this bounding allowable flaw for all transients evaluated in the TR.

The NRC staff reviewed the PWROG's assertion that the ~~a~~ 60 to 80 -year increase in neutron fluence would not be significant enough to result in a decrease in the allowable flaw size for the LSB transient from 1.25 inch to 0.67 inch (i.e., a 46.5 percent decrease). To substantiate the PWORG's claim, the staff performed independent calculations of 60 to 80-year increase in RT_{NDT}, decease in K_{IC}, and decrease in allowable flaw size by using bounding material properties for SA508, Class 2 RPV beltline shell forgings in Westinghouse plants from its database and conservative estimates for the 60 to 80-year neutron fluence increase.

For Westinghouse SA508, Class 2 RPV beltline shell forgings, the staff's calculations considered how bounding material property inputs (i.e., CF, initial RT_{NDT}, and margin term inputs) and neutron fluence affect the 60 to 80-year increase in RT_{NDT}, and the corresponding decrease in K_{IC} and the allowable flaw size. The staff performed several sets of calculations to explore how these input parameters affect these changes. For its allowable flaw size calculations, the staff directly applied the stress intensity factor (K_I) equations for surface flaws in Paragraph A-3320 of the ASME Code, Section XI, Appendix A to assess the decrease in allowable flaw sizes based on the 60 to 80-year decrease in K_{IC}. For Service Level C and D transients, the allowable flaw depth is always one-half of the critical flaw depth, where ~~Ki~~ K_I is equal to K_{IC} at the critical flaw depth, per IWB-3611 (Code Acceptance Criteria Based on Flaw Size).

For its K_{IC} calculation, the staff evaluated a range of LSB transient temperatures to ensure that 60 and 80-year K_{IC} values up to 200 ksi√in (the upper bound value that is used for all LEFM analyses in the TR) are considered, based on its consideration of the 60-year and 80-year adjusted RT_{NDT} values. If the transient temperature and the projected 80-year RT_{NDT} value results in the 80-year K_{IC} being greater than or equal to 200 ksi√in (given that metal temperature exceeds transient temperature during rapid cooldown), then the percentage decrease in allowable flaw depth is zero since the 60-year K_{IC} would also be bounded by 200 ksi$\sqrt{in}$. For these cases, the 60-year allowable flaw size would remain the same for 80 years.

For cases where transient temperature minus 80-year adjusted RT_{NDT} is less than 104.25 °F (200 ksi$\sqrt{in}$ is not bounding) at 80 years, the staff's independent calculations showed the following:

- Considering a conservative 50 percent increase in neutron fluence for 60 to 80 years, the most limiting adjusted RT_{NDT} value for SA508, Class 2 RPV beltline shell forgings increases from 253 °F to 265 °F. Based on the relevant range of LSB transient temperatures (where 200 ksi$\sqrt{in}$ is not bounding), the highest percentage decrease in K_{IC} is 18 percent, and the highest percentage decrease in allowable flaw depth is about 30 percent. Therefore, for this case, the assertions made in the RAI-2 response are valid.

- Considering a conservative 50 percent increase in neutron fluence, the most non-limiting 60-year adjusted RT_{NDT} value for SA508, Class 2 shell forgings that also exhibits the most significant RT_{NDT} *increase* (generally about 12 °F maximum for all shell forgings), shows an RT_{NDT} increase from 133 °F to 145 °F. Based on the relevant range of LSB transient temperatures, the highest percentage decrease in K_{IC} is 17 percent, and the highest percentage decrease in allowable flaw depth is about 35 percent. Therefore, for this case, the assertions made in the RAI-2 response are valid.

- The staff also examined the relative sensitivity of the decrease in critical flaw depth to decreases in K_{IC}. Based on its calculation of K_I as a function flaw depth (using the surface flaw K_I equations in Paragraph A-3320 of Code Appendix A), the staff determined that in order for the critical flaw depth (and therefore the allowable flaw depth) for the LSB transient to decrease by 46.5 percent, the 60 to 80-year decrease in K_{IC} would need to be greater than about 20 percent. For the SA508, Class 2 RPV beltline shell forgings in the staff's database, none of the K_{IC} values decreases by more than about 18 percent for the relevant range of LSB transient temperatures (where 200 ksi$\sqrt{in}$ is not bounding); and in most cases, the decrease in K_{IC} is significantly less than 18 percent. Therefore, the staff determined that the decrease in allowable flaw depth for the database forgings, while significant in many cases, would be less than 46.5 percent.

Based on its own independent calculations, as documented above, the staff determined that the allowable flaw depth for the LSB transient is expected to decrease by a significant amount (depending on the inputs) when analyzed for a 60 to 80-year term. However, the staff also determined that there is reasonable assurance that it would *not be expected to decrease from 1.25 inch (the established value for 60-years per WCAP-15338-A) to less than the limiting allowable depth for the Service Level A and B transients, which is 0.67 inch*. Therefore, the staff finds that the limiting allowable flaw size, with depth of 0.67 inch and continuous length for Service Levels A and B, is expected to remain the most limiting allowable flaw size for all transients that were analyzed in the TR. On this basis, the staff finds that there is reasonable assurance that the largest 80-year projected crack depth (0.43 inch) for the continuous axial flaw from the FCG analysis will continue to be bounded by the limiting allowable flaw sizes for all governing transients in the TR for Service Levels A, B, C, and D, based on the IWB-3610 acceptance criteria, and subject to TLAA action items discussed above. On this basis, and subject to the action items above, the staff finds that RPV underclad cracks in SA508 forgings for all U.S. Westinghouse plants are projected to be acceptable for service for 80-year operating periods.

In addition to the consideration of projected decrease in allowable flaw depth for the LSB transient, the RAI-2 response also discussed other regulatory requirements and associated methods for protection of RPV integrity. The response addressed PTS requirements in 10 CFR 50.61 and alternate PTS requirements 10 CFR 50.61a, which are based on probabilistic fracture mechanics (PFM) analysis techniques. The response also addressed 10 CFR Part 50, Appendix G requirements for equivalent margins analysis (EMA) of RPVs that have projected upper shelf impact energies less than 50 ft-lbs. The EMA methods are based on elastic-plastic fracture mechanics analysis techniques.

The NRC staff's independent calculations, as documented above, were able to confirm the acceptability of the projected flaw size for 80 years of operation based on the most limiting allowable flaw size for all transients evaluated in the TR. Considering that these calculations utilized the deterministic LEFM methods of the ASME Code, Section XI, IWB-3610 and Appendix A, the staff found that there is no need to fully consider the merits of the alternate approaches to this issue, as proposed in the RAI-2 response. As such, the NRC staff makes no findings concerning the validity or applicability of these approaches to the subject RPV flaw analyses.

5.0 CONDITIONS, LIMITATIONS, AND/OR ACTION ITEMS

There is no NRC staff-imposed condition or limitation for the generic RPV underclad crack evaluation in the TR. However, SLR [illegible] of RPV underclad cracks need to ver[illegible] 10 CFR 54.21(c)(1)(ii):

The NRC staff notes that it is unlikely that actual RCS transients and cycles for the SPEO will exceed the number of design cycles conservatively considered i[illegible] the transient table in Reference 2. However,

- TLAA Action Item 1: In their plant-specific TLAAs for RPV underclad cracks, SLR applicants are to confirm that the generic transient types and number of transient cycles used for the 80-year FCG calculation, as listed in the RCS transient table in Reference 2, bounds the projected number of transient cycles for the SPEO.

  for the actual applicable transients

- TLAA Action Item 2: To ensure the continued validity of 200 ksi√in toughness for RPV beltline forgings, based on an adjusted RT_{NDT} less than or equal to 270 °F for the high fluid temperature transients addressed in Reference 2, SLR applicants are to confirm that their limiting SA508, Class 2 or Class 3 RPV beltline forgings meet the 270 °F ~~limit~~ in 10 CFR 50.61.

- TLAA Action Item 3: To ensure that the Large LOCA may be eliminated from consideration in the TR flaw evaluation based on plant-specific implementation of the LBB analysis (Ref. 2), SLR applicants are to confirm their implementation of the LBB analyses for primary loop piping as part of their 80-year SLR applications.

The above TLAA actions must be addressed for all underclad crack TLAAs in SLR applications to fulfill the TR requirements, as supplemented by the RAI response in Reference 2. It should be noted that TR requirements and associated SLR action items are not considered to be SE conditions and limitations imposed by the NRC staff.

PTS screening criterion of

6.0 CONCLUSIONS

alloy flaw shape

As set forth above, the NRC staff has reviewed the PWROG-17031-NP, Revision 1 report and has determined the following:

- The initial crack depth of 0.30 inch through the low steel and continuous ~~length~~ is still considered to be bounding for industry OpE with detected underclad cracks.

- Industry OpE continues to show that RPV underclad cracks and interior cladding defects do not pose a structural integrity concern for the RPV, and the known flaws continue to meet ASME Code, Section XI acceptance standards.

- The 80-year FCG analysis in the TR conservatively projects that the bounding axial crack in the RPV beltline shell region, with initial depth of 0.30 inch and continuous crack length, will grow to about 0.43 inch in depth after 80-years of operation. Subject to TLAA action items discussed above, the staff finds that this a conservative projection based on the number of transient cycles assumed for the FCG calculation and the TR's use of FCG rate curves in the ASME Code, Section XI, Appendix A for low ~~allow~~ steel exposed to RCS water environments.
  alloy

- There is reasonable assurance that the largest 80-year projected crack depth (0.43 inch) for the continuous axial flaw from the FCG analysis will continue to be bounded by the limiting allowable flaw sizes for all governing transients in the TR for Service Levels A, B, C, and D, subject to TLAA action items discussed above.

On this basis, and subject to completion of the action items above, the staff concludes that the TR is acceptable for referencing in plant-specific TLAAs of RPV underclad cracks for SLR applications pursuant to 10 CFR 54.21(c)(1)(ii).

7.0 REFERENCES

1. Letter from Ken Schrader, Pressurized Water Reactor Owners Group, to USNRC Document Control Desk, May 31, 2018, 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 (ADAMS Package Accession No. ML18164A025).

2. Letter from Ken Schrader, Pressurized Water Reactor Owners Group, to USNRC Document Control Desk, August 29, 2019, 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 (ADAMS Accession No. ML19253B327).

3. Topical Report WCAP-15338-A, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," October 2002, Includes NRC Approval SE (ADAMS Accession No. ML083530289).

4. Topical Report WCAP-7733, "Reactor Vessels: Weld Cladding-Base Metal Interaction," by T.R. Mager, E. Landerman, and C.J. Kubit, Westinghouse Nuclear Energy Systems Report, April 1971.

5. Regulatory Guide 1.43, Revision 1, "Control of Stainless Steel Weld Cladding of Low Alloy Steel Components," March 2011 (ADAMS Accession No. ML101670458).

Principle Contributor: Christopher R. Sydnor, NRR/DMLR/MVIB

Date: October 29, 2019