PRESSURIZED WATER REACTOR OWNERS GROUP

PWROG-17031-NP-A
Revision 1

WESTINGHOUSE NON-PROPRIETARY CLASS 3

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

Materials Committee

PA-MSC-1497

May 2020

PWROG-17031-NP-A
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

Benjamin E. Mays*
License Renewal, Radiation Analysis, and Nuclear Operations

May 2020

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U.S. NRC SAFETY EVALUATION

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

March 30, 2020

Mr. W. Anthony Nowinowski
Executive Director
PWR Owners Group,
Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

SUBJECT: FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR THE PRESSURIZED WATER REACTOR OWNERS GROUP 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)

Dear Mr. Nowinowski:

By letter dated May 31, 2018 (Agencywide Documents Access and Management System Accession No. (ADAMS Accession No. ML18164A025), 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 [pressurized water reactor] Plants,'" dated May 31, 2018 (ADAMS Accession No. ML18164A035) for the U.S. Nuclear Regulatory Commission (NRC) review and approval. Additional information related to PWROG-17031-NP, Rev. 1, was submitted by letter dated August 29, 2019 (ADAMS Accession No. ML19253B327), in response to a request for additional information (RAI) from the NRC staff. By letter dated December 5, 2019 (ADAMS Accession No. ML19347A422), the PWROG submitted comments to the draft safety evaluation (SE) and requested that the NRC prepare the final SE for PWROG-17031-NP against Rev. 1.

The NRC staff has completed its review of PWROG-17031-NP, Rev. 1, and found that the subject report, as modified by the conditions and limitations summarized in Section 4.0 of the enclosed final SE provides the technical and regulatory basis to extend the applicability of the reactor pressure vessel (RPV) underclad crack analysis methodology in WCAP-15338-A from 60 to 80 years of operation to support applications for subsequent license renewal (SLR) for all U.S. Westinghouse plants.

As stated in PWROG's submittal letter dated May 31, 2018, licensees will reference the TR PWROG-17031-NP, Rev. 1, in SLR applications to demonstrate compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) 54.21(c)(1), for the appropriate findings regarding the evaluation of time-limited aging analysis for the RPV components through the SLR period of operation (80 years). Applicants who utilize the TR will be required to adhere to the conditions that the NRC staff impose in the SE and shall be subject to NRC staff review and approval on a case-by-case basis.

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By letter dated October 29, 2019 (ADAMS Accession No. ML19300A002), the NRC staff provided the draft SE to the PWROG for review and comment. By letter dated December 5, 2019 (ADAMS Accession No. ML1922A259), the PWROG provided comments on the draft SE. The NRC staff's disposition table for the draft SE comments is provided in the final SE.

In accordance with the guidance provided on the NRC website, the NRC staff requests that the PWROG publish approved versions of PWROG-17031-NP, Rev. 1, within 3 months of receipt of this letter. The approved version shall incorporate this letter and the enclosed final SE after the title page. Also, the approved versions must contain historical review information, including NRC requests for additional information (RAIs) and the corresponding RAI responses. The approved versions shall include an "-A" (designating approved) following the TR identification symbol. As an alternative to including the request for RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and if the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of these TRs, PWROG will be expected to revise the TRs appropriately or justify their continued applicability for subsequent referencing. Licensees referencing these TRs would be expected to justify their continued applicability or evaluate their plant using the revised TRs.

If you have any questions, please contact Leslie Fields at 301-415-1186.

Sincerely,

/RA/

Dennis C. Morey, Chief
Licensing Processes Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 99902037

Enclosure:
Final SE – Nonproprietary

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SUBJECT: FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR THE PRESSURIZED WATER REACTOR OWNERS GROUP 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) DATED MARCH 30, 2020

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***via email**

OFFICE	NRR/DORL/LLPB/PM	NRR/DLP/PLPB/LA	NRR/DMLR/NVIB/BC	NRR/DORL/LLPB/BC
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DATE	03/30/2020	03/30/2020	03/30/2020	03/30/2020

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UNITED STATES
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WASHINGTON, D.C. 20555-0001

FINAL SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PRESSURIZED WATER OWNERS GROUP 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

1.0 INTRODUCTION

By letter dated May 31, 2018 (Ref. 1), as supplemented by letter dated August 29, 2019 (Ref. 2), the Pressurized Water Reactor Owners Group (PWROG) transmitted 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,'" dated May 2018 (Ref. 3, non-proprietary version), for U.S. Nuclear Regulatory Commission (NRC) review and approval

This TR proposes to extend the applicability of the reactor pressure vessel (RPV) underclad crack analysis methodology in WCAP-15338-A (Ref. 4) from 60 to 80 years of operation to support applications of subsequent license renewal (SLR) for all U.S. Westinghouse Electric Company (Westinghouse) Nuclear Steam Supply System 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). Based on the safety evaluation (SE) dated September 2002 (Ref. 5) of TR WCAP-15338-A, 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. Furthermore, the NRC staff's review verified that the 60-year crack growth analysis presented in the report 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. Additionally, the NRC staff's September 2002 SE 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

Enclosure

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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. 6), 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 NRC's TR review process 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 OVERVIEW 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 STAFF EVALUATION

4.1 Background – NRC Position on RPV Underclad Cracking

Regulatory Guide (RG) 1.43, Revision 1, March 2011 (Ref. 7), 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 pre-existing 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 the 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 not 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

[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.

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continued use of WCAP-15338-A assumptions regarding the initial crack sizes and characteristics are acceptable for 80-year applications.

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 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, 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;

2. The K_{IC} value used for all transient analyses is limited to a value no greater than two-hundred thousand pounds per square inch times the square root of an inch (200 ksi√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√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 greater than or equal to 200 ksi√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 of 200 ksi√in for RPV beltline materials to determine allowable flaw sizes for the transients evaluated in PWROG-17031-NP, 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√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√in even if the 10 CFR 50.61 Pressurized Thermal Shock (PTS) screening criterion of 270 °F is used for RT_{NDT}. Therefore, K_{IC} was limited to 200 ksi√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√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 Loss of Coolant Accident (LOCA), the calculated K_{IC} also exceeds 200 ksi√in when using the 270°F PTS screening criterion for RT_{NDT}.

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- With respect to the Large LOCA, typical Westinghouse plants have performed Leak Before Break (LBB) analysis, and the implementation of LBB eliminates Large LOCA from the design basis. The RAI-2 response specifies that individual plants should confirm the implementation of LBB for reactor coolant system primary loop piping 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√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√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√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 1.99 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-yearoperating period with all other inputs being 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.

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 result 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 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 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√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√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√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√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 CONDITIONSAND LIMITATIONS

There is no NRC staff-imposed condition or limitation for the generic RPV underclad crack evaluation in the TR. However, SLR applicants that implement this TR as the basis for TLAAs of RPV underclad cracks need to verify the following as part their TLAA evaluations pursuant to 10 CFR 54.21(c)(1)(ii):

- TLAA Action Item 1: The NRC notes that it is unlikely that actual RCS transients and cycles for the SPEO will exceed the number of design cycles conservatively considered in the transient table in Reference 2. However, 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 actual applicable transients for the SPEO.

- TLAA Action Item 2: To ensure the continued validity of 200 ksi$\sqrt{\text{in}}$ 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 PTS screening criterion of 270 °F 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.

6.0 CONCLUSIONS

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 alloy steel and continuous flaw shape 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 alloy steel exposed to RCS water environments.
- 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).

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3. 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'" dated May 2018 WCAP-15338-A, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," October 2002 (ADAMS Accession No. ML18164A035).

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

5. Letter from NRC to Roger A. Newton of Westinghouse Owners Group Safety Evaluation of WCAP-15338, "A Review of Cracking Associated With Weld Deposited Cladding In Operating Pressurized Water Reactor (PWR) Plants," September 2002 (ADAMS Package Accession No. ML022690375).

6. 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.

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

Attachment: Comment Resolution Table

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

Date: March 30, 2020

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TOPICAL REPORT PWROG-17031-NP, REVISION 1 COMMENT RESOLUTION TABLE			
Comment Number	Page Location	PWROG Comment	NRC Response
1	1 of 17	Please add: "Nuclear Steam Supply System" after "Westinghouse Electric Company (Westinghouse)"	The staff finds the change acceptable because it provides clarification, and it does not affect the staff's findings or conclusions.
2	8 of 17	Please remove capital letter in the word "Initial".	The staff finds the change acceptable because it corrects a minor typographical error.
3	8 of 17	Please add: "in" before "WCAP-15338-A" in the final paragraph on this page.	The staff finds the change acceptable because it corrects a minor typographical error.
4	8 of 17	Please replace "it" with "this method" in the sentence "Therefore, it is acceptable."	The staff finds the change acceptable because it provides clarification, and it does not affect the staff's findings or conclusions.
5	11 of 17	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.	The staff finds the change acceptable because it provides clarification, and it does not affect the staff's findings or conclusions.
6	11 of 17	Please add: "of" between "use" and "200 ksi√in"	The staff finds the change acceptable because it corrects a minor typographical error.
7	11 of 17	Please replace "Revision 0" with "Revision 1" after "PWROG-17031-NP".	The staff finds the change acceptable because it corrects a typographical error.
8	11 of 17	Please define "PTS" as "Pressurized Thermal Shock (PTS)".	The staff finds the change acceptable because it provides clarification, and it does not affect the staff's findings or conclusions.
9	12 of 17	Please add: "for reactor coolant system primary loop piping" after "confirm the implementation of LBB".	The staff finds the change acceptable because it provides clarification, and it does not affect the staff's findings or conclusions.
10	12 of 17	Please replace "bring equal" with "being equal".	The staff finds the change acceptable because it corrects a minor typographical error.
11	13 of 17	Please replace "the a 60 to 80" with "the 60 to 80-year"	The staff finds the change acceptable because it corrects a minor typographical error.

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TOPICAL REPORT PWROG-17031-NP, REVISION 1 COMMENT RESOLUTION TABLE			
Comment Number	Page Location	PWROG Comment	NRC Response
12	13 17	Please replace "Ki" of with "KI".	The staff finds the change acceptable because it corrects a minor typographical error.
13	15 of 17	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.	The staff finds the change acceptable because it provides clarification, and it does not affect the staff's findings or conclusions.
14	15 of 17	Please add: "for the actual applicable transients" before "for the SPEO" in Action Item 1.	The staff finds the change acceptable because it provides clarification, and it does not affect the staff's findings or conclusions
15	15 of 17	Please replace "the 270°F limit" with "the PTS screening criterion of 270°F".	The staff finds the change acceptable because it provides clarification, and it does not affect the staff's findings or conclusions
16	16 of 17	Please add: "alloy" between "low" and "steel".	The staff finds the change acceptable because it corrects a typographical error.
17	16 of 17	Please replace "continuous length" with "continuous flaw shape".	The staff finds the change acceptable because it provides clarification, and it does not affect the staff's findings or conclusions
18	16 of 17	Please replace "low allow steel" with "low alloy steel".	The staff finds the change acceptable because it corrects a minor typographical error.

ACKNOWLEDGEMENTS

This report was developed and funded by the PWR Owners Group under the leadership of the participating utility representatives of the Materials Committee.

WESTINGHOUSE ELECTRIC COMPANY LLC

PWR Owners Group
United States Member Participation* for PA-MSC-1497

Utility Member	Plant Site(s)	Participant	
		Yes	No
Ameren Missouri	Callaway (W)		X
American Electric Power	D.C. Cook 1 & 2 (W)	X	
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)		X
Dominion Connecticut	Millstone 2 (CE)		X
	Millstone 3 (W)	X	
Dominion VA	North Anna 1 & 2 (W)	X	
	Surry 1 & 2 (W)	X	
Duke Energy Carolinas	Catawba 1 & 2 (W)	X	
	McGuire 1 & 2 (W)	X	
	Oconee 1 (B&W)	X	
	Oconee 2, & 3 (B&W)		X
Duke Energy Progress	Robinson 2 (W)	X	
	Shearon Harris (W)	X	
Entergy Palisades	Palisades (CE)		X
Entergy Nuclear Northeast	Indian Point 2 & 3 (W)		X
Entergy Operations South	Arkansas 1 (B&W)		X
	Arkansas 2 (CE)		X
	Waterford 3 (CE)		X
Exelon Generation Co. LLC	Braidwood 1 & 2 (W)	X	
	Byron 1 & 2 (W)	X	
	TMI 1 (B&W)		X
	Calvert Cliffs 1 & 2 (CE)		X
	Ginna (W)		X
FirstEnergy Nuclear Operating Co.	Beaver Valley 1 & 2 (W)		X
	Davis-Besse (B&W)		X
Florida Power & Light \ NextEra	St. Lucie 1 & 2 (CE)		X
	Turkey Point 3 & 4 (W)	X	
	Seabrook (W)		X
	Pt. Beach 1 & 2 (W)	X	

PWR Owners Group
United States Member Participation* for PA-MSC-1497

Utility Member	Plant Site(s)	Participant	
		Yes	No
Luminant Power	Comanche Peak 1 & 2 (W)		X
Omaha Public Power District	Fort Calhoun (CE)		X
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)		X
PSEG – Nuclear	Salem 1 & 2 (W)		X
South Carolina Electric & Gas	V.C. Summer (W)	X	
So. Texas Project Nuclear Operating Co.	South Texas Project 1 & 2 (W)		X
Southern Nuclear Operating Co.	Farley 1 & 2 (W)		X
	Vogtle 1 & 2 (W)		X
Tennessee Valley Authority	Sequoyah 1 & 2 (W)		X
	Watts Bar 1 & 2 (W)		X
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)		X
Xcel Energy	Prairie Island 1 & 2 (W)	X	

* **Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.**

PWR Owners Group
International Member Participation* for PA-MSC-1497

Utility Member	Plant Site(s)	Participant	
		Yes	No
Asociación Nuclear Ascó-Vandellòs	Asco 1 & 2 (W)		X
	Vandellos 2 (W)		X
Axpo AG	Beznau 1 & 2 (W)		X
Centrales Nucleares Almaraz-Trillo	Almaraz 1 & 2 (W)		X
EDF Energy	Sizewell B (W)		X
Electrabel	Doel 1, 2 & 4 (W)		X
	Tihange 1 & 3 (W)		X
Electricite de France	58 Units		X
Eletronuclear-Eletrobras	Angra 1 (W)		X
Emirates Nuclear Energy Corporation	Barakah 1 & 2		X
EPZ	Borssele		X
Eskom	Koeberg 1 & 2 (W)		X
Hokkaido	Tomari 1, 2 & 3 (MHI)		X
Japan Atomic Power Company	Tsuruga 2 (MHI)		X
Kansai Electric Co., LTD	Mihama 3 (W)		X
	Ohi 1, 2, 3 & 4 (W & MHI)		X
	Takahama 1, 2, 3 & 4 (W & MHI)		X
Korea Hydro & Nuclear Power Corp.	Kori 1, 2, 3 & 4 (W)		X
	Hanbit 1 & 2 (W)		X
	Hanbit 3, 4, 5 & 6 (CE)		X
	Hanul 3, 4 , 5 & 6 (CE)		X
Kyushu	Genkai 2, 3 & 4 (MHI)		X
	Sendai 1 & 2 (MHI)		X
Nuklearna Electrarna KRSKO	Krsko (W)		X
Ringhals AB	Ringhals 2, 3 & 4 (W)		X
Shikoku	Ikata 1, 2 & 3 (MHI)		X
Taiwan Power Co.	Maanshan 1 & 2 (W)		X

* **Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.**

TABLE OF CONTENTS

1 Background and Introduction

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

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

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

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

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

2 Mechanisms of Cracking Associated with Weld Deposited Cladding

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

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

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

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

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

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

3.1 PWR Service Experience Since 1999

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

1. Callaway Reactor Vessel Bottom Head Region

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

2. Diablo Canyon Unit 1 Reactor Vessel Inlet Nozzle

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

3. Qinshan Reactor Vessel Bottom Head Region

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

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

4 Effects of Cladding on Fracture Analysis

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

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

5 Vessel Integrity Assessment

This section discusses the reactor vessel integrity evaluation and assessment.

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

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

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

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

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

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

5.2 Fatigue Usage

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

5.3 Acceptance Criteria

5.3.1 ASME Section XI – IWB-3500

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

5.3.2 ASME Section XI – IWB-3600

There are two alternative sets of flaw acceptance criteria for ferritic components, for continued service without repair in paragraph IWB-3600 of ASME Code Section XI. Either of the criteria below can be used as discussed in Appendix A of WCAP-15338-A [1].

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

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

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

5.3.2.1 Criteria Based on Flaw Size

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

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

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

where,

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

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

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

5.3.2.2 Criteria Based on Applied Stress Intensity Factors

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

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

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

where,

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

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

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

5.4 Fatigue crack growth

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

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

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

A series of flaw types were postulated to address the various possible shapes for the underclad cracks. Specifically, the postulated flaw depths ranged from 0.05 inch (1.3mm) to 0.30 inch (7.6mm), which is beyond the 0.295 inch (7.5mm) maximum depth of an underclad cold crack. The shape of the flaws analyzed (flaw depth/flaw length) ranged from 0.01 through 0.5. The results are shown in Table 5-1 through Table 5-3. The maximum flaw size of 0.4267 inch at the end of 80 years is less than the minimum allowable flaw size of 0.67 inch, presented in Section 5.5.

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

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

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

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

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

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

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

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

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

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

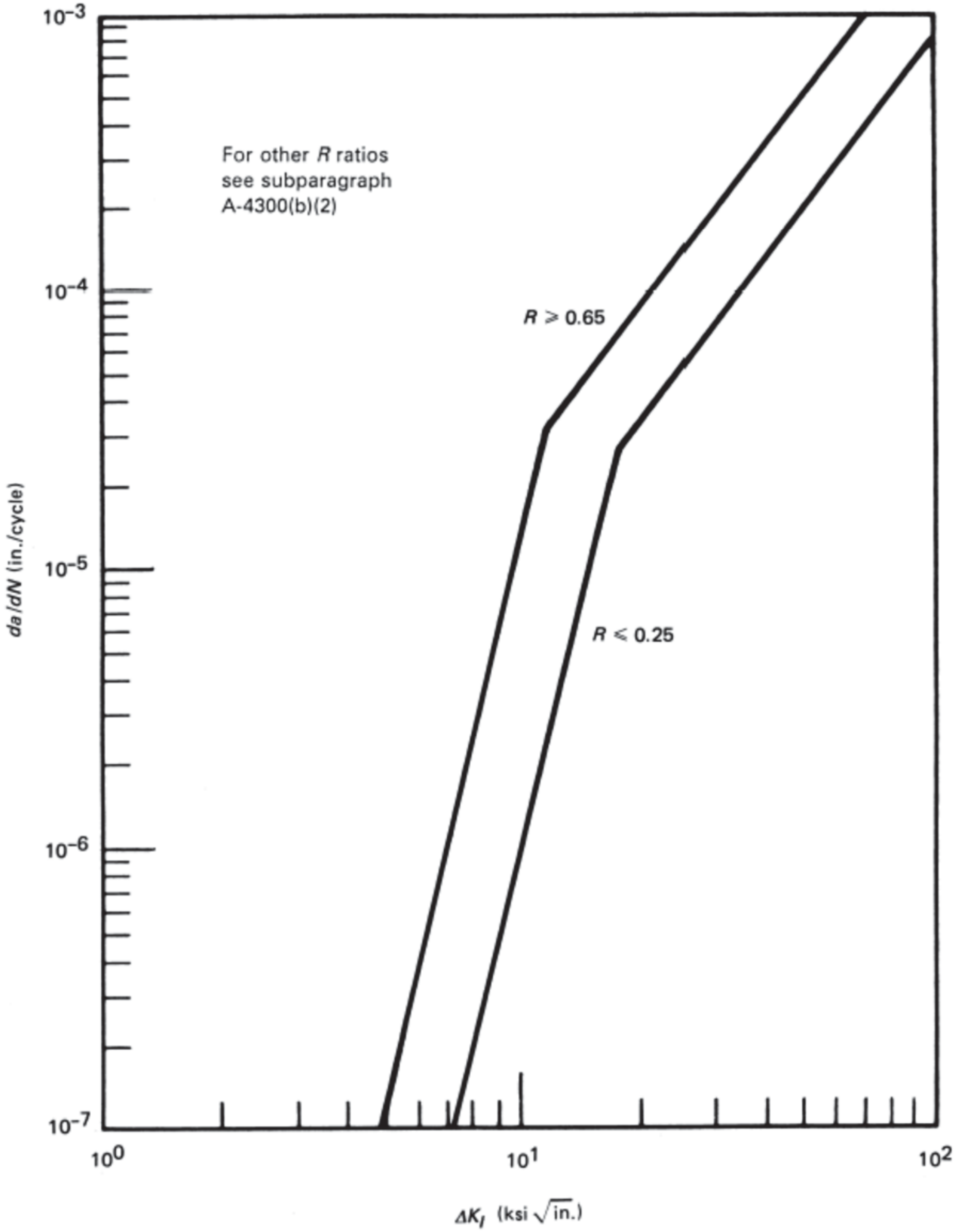

Figure 5-1: Reference Fatigue Crack Growth Curves for Carbon and Low Alloy Ferritic Steels Exposed to Water Environment [2, Fig. A-4300-2]

5.5 Allowable Flaw Size – Normal, Upset & Test Conditions

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

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

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

where K_{Ic} and K_{Ia} are in ksi√in.

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

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

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

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

Table 5-4: Allowable Flaw Size Summary for Beltline Region – Normal, Upset & Test Conditions

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

Note: A wall thickness of 7.75 inches was used.

5.6 Allowable Flaw Size – Emergency & Faulted Conditions

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

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

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

Note: A wall thickness of 7.75 inches was used.

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

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

6 Summary and Conclusions

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

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

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

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

Finally, underclad cracks identified during pre-service and inservice inspections have been evaluated in accordance with the acceptance criteria in the ASME Code, Section XI. The observed underclad cracks are very shallow, confined in depth to less than 0.295 inch and have lengths up to 2.0 inches. The FCG assessment for these small cracks concluded that there would be very little growth for 80 years of operation, even if they were exposed to the RCS water and with a crack tip pressure of 2,500 psi. For the worst case scenario, a 0.30-inch deep continuous axial flaw in the beltline region would grow to 0.43 inch after 80 years. The minimum allowable axial flaw size for normal, upset, and test conditions is 0.67 inch and for emergency and faulted conditions is 1.25 inches. Since the maximum flaw depth of 0.4267 inch after 80 years of FCG is less than the minimum allowable flaw size of 0.67 inch, underclad cracks of any shape are acceptable for 80 years, regardless of the size or orientation of the flaws. Therefore, it can be concluded that underclad cracks are acceptable relative to the structural integrity of the reactor vessel for 80 years.

7 References

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

APPENDIX A : CORRESPONDENCE WITH THE U.S. NRC REGARDING THE REVIEW OF PWROG-17031-NP, REVISION 1

40TH ANNIVERSARY
PWROG
PWR Owners Group

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

PWROG-17031-NP, Revision 1
Project Number 99902037

August 29, 2019

OG-19-184

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

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

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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:

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

cc: PWROG Analysis Committee (Participants of PA-MSC-1497)
PWROG PMO
PWROG Steering and Management Committee
J. Drake, US NRC
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G. Hall, Westinghouse
B. Mays, Westinghouse
S. Rigby, Westinghouse

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

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Westinghouse Non-Proprietary Class 3

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.

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

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.

*** This record was final approved on 8/26/2019 1:54:16 PM. (This statement was added by the PRIME system upon its validation)

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.

Westinghouse Response

Allowable Flaw Size Calculation

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

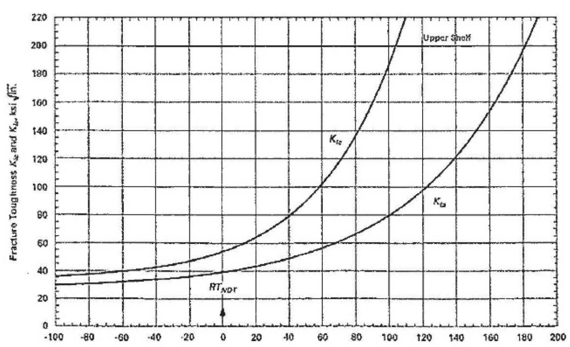

Figure 1. K_{Ic} Curve with 200 ksi√in Upper Shelf

*** This record was final approved on 8/26/2019 1:54:16 PM. (This statement was added by the PRIME system upon its validation)

(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$\sqrt{in}$ even if the 10CFR50.61 PTS screening criterion of 270°F is used. 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 (Level C and D transients), if $T - RT_{NDT} > 104.25$ °F, 200 ksi$\sqrt{in}$ 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$\sqrt{in}$ 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√in will be used to a temperature of approximately 375°F. When using a lower RT_{NDT}, 200 ksi√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 x 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.

<u>Westinghouse Response</u>

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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PWROG-17031-NP, Revision 1
Project Number 99902037

May 31, 2018

OG-18-118

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

Subject: PWR Owners Group
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

The purpose of this letter is to transmit Pressurized Water Reactor Owners Group (PWROG) 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" in accordance with the Nuclear Regulatory Commission (NRC) TR program for review and acceptance for referencing in regulatory actions (Enclosure 1).

Topical Report Summary

In WCAP-15338-A, Westinghouse provided the technical basis demonstrating that there is no structural integrity concern for the underclad cracking of the Westinghouse reactor vessels. The NRC issued a Safety Evaluation Report (SER) in September 25, 2002, accepting the technical arguments in WCAP-15338 and allow any Westinghouse plants to reference the report in a license renewal application to satisfy the requirements of 10 CFR 54.21(c)(1) for demonstrating the appropriate findings regarding the evaluation of time-limited aging analysis (TLAA) for the reactor pressure vessel (RPV) components for the period of extended operation (60 years).

The purpose of this TR is to update the fatigue crack growth analysis in WCAP-15338-A through 80 years of operation, and confirm that the evaluation remains applicable for subsequent license renewal (SLR) periods of operation through this time period.

Limits of Applicability

WCAP-15338-A is applicable to all Westinghouse Nuclear Steam Supply System (NSSS) plants. This same applicability is carried over for the TR presented herein. This TR is applicable to all Westinghouse NSSS plants for 80 years of operation.

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Intended Application

Licensees will reference PWROG-17031-NP, Revision 1 in subsequent license renewal applications to satisfy the requirements of 10 CFR 54.21(c)(1) for demonstrating the appropriate findings regarding the evaluation of time-limited aging analysis (TLAA) for the reactor pressure vessel (RPV) components through the subsequent license renewal period of operation (80 years).

Industry Implementation

PWROG-17031-NP, Revision 1 can be implemented by all applicable U. S. PWRs as listed in the Limits of Applicability section above.

Specialized Resource Availability

This TR is being submitted to the NRC for review and approval so that the NRC approved version can be utilized by licensees. Licensees will reference PWROG-17031-NP, Revision 1 in a license renewal application to satisfy the requirements of 10 CFR 54.21(c)(1) for demonstrating the appropriate findings regarding the evaluation of time-limited aging analysis (TLAA) for the reactor pressure vessel (RPV) components through the subsequent license renewal period of operation (80 years). NRC approval of the generic TR will reduce the impact on both licensee and NRC resources by eliminating the need for the preparation of and NRC review of plant specific justifications for the structural integrity of the Westinghouse RPVs due to underclad cracking.

NRC Review Schedule

The PWROG requests that the NRC complete their review of the TR by June, 2019.

This letter transmits one copy of PWROG-17031-NP, Revision 1 (Enclosure 1).

Correspondence related to this transmittal should be addressed to:

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1000 Westinghouse Drive, Suite 386
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U.S. Nuclear Regulatory Commission May 31, 2018
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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 Schrader, Chief Operating Officer and Chairman
PWR Owners Group
KS:WAN:am

Enclosure 1: One copy of PWROG-17031-NP, Revision 1

cc: PWROG Management Committee
PWROG Materials Committee
PWROG Steering Committee
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