PRESSURIZED WATER REACTOR OWNERS GROUP

PWROG PWR Owners Group

PWROG-17033-NP-A
Revision 1

WESTINGHOUSE NON-PROPRIETARY CLASS 3

Update for Subsequent License Renewal: WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems"

Materials Committee

PA-MSC-1498, Revision 0

November 2019

Westinghouse

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November 2019

Authors: Geoffrey Loy*
Reactor Vessel / Containment Vessel Design & Analysis

Reviewer: Anees Udyawar *
Reactor Vessel / Containment Vessel Design & Analysis

Approved: Lynn Patterson*, Manager
Reactor Vessel / Containment Vessel Design & Analysis

James Molkenthin*, Program Director
PWR Owners Group PMO

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

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

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

September 26, 2019

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 TOPICAL REPORT PWROG-17033, REVISION 1, "UPDATE FOR SUBSEQUENT LICENSE RENEWAL WCAP-13045, COMPLIANCE TO ASME CODE CASE N-481 OF THE PRIMARY LOOP PUMP CASINGS OF WESTINGHOUSE TYPE NUCLEAR STEAM SUPPLY SYSTEMS" (EPID: L-2018-TOP-0025)

Dear Mr. Nowinowski:

By letter dated June 14, 2018 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML18170A113), the Pressurized Water Reactor Owners Group (PWROG) submitted for U.S. Nuclear Regulatory Commission (NRC) review and approval Topical Report (TR) PWROG-17033-P and -NP, Revision 1, "Update for Subsequent License Renewal: WCAP-13045, Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems." By letter dated March 19, 2019, the PWROG submitted a supplement to PWROG-17033, "Generic Responses to the U.S. NRC Requests for Additional Information on PWROG Report, PWROG-17033, for Westinghouse Reactor Coolant Pump [(RCP)] Casings" (ADAMS Package Accession No. ML19091A098)

The NRC staff determines that PWROG-17033, Revision 1, has demonstrated structural integrity of the Westinghouse-designed RCP casings for the subsequent period of extended operation (80 years) based on the crack stability analysis and fatigue crack growth analysis. Therefore, the NRC staff concludes that PWROG-17033, Revision 1, is acceptable for generic use to address the time-limited aging analysis of the RCP casing integrity to satisfy the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 54.21(c)(1).

The NRC staff also concludes that the TR may be referenced in subsequent license renewal applications for the plants within scope of the report in accordance with 10 CFR 54.21(c)(1) Applicants who utilize the report will be required to adhere to the conditions that the NRC staff imposed in this safety evaluation (SE).

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By letter dated May 29, 2019 (ADAMS Accession No. ML19142A218), the NRC staff provided the draft SE to the PWROG for review and comment. By letter dated June 27, 2019 (ADAMS Accession No. ML19184A115), the PWROG provided comments to the draft SE.

In accordance with the guidance provided on the NRC website, we request that the PWROG publish approved versions of PWROG-17033-P and -NP, Revision 1 within 3 months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page. For -NP versions, the PWROG shall strike the proprietary information markings in this letter and make the appropriate redactions and adjustments to document security classifications to the enclosed SE. Also, the approved versions must contain historical review information, including NRC requests for additional information (RAIs) and the 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 Jason Drake at 301-415-8378.

Sincerely,

/RA/

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

Docket No. 99902037

Enclosure: Final SE

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SUBJECT: FINAL SAFETY EVALUATION BY THE OFICE OF NUCLEAR REACTOR REGULATION TOPICAL REPORT PWROG-17033, REVISION 1, "UPDATE FOR SUBSEQUENT LICENSE RENEWAL: WCAP-13045, COMPLIANCE TO ASME CODE CASE N-481 OF THE PRIMARY LOOP PUMP CASINGS OF WESTINGHOUSE TYPE NUCLEAR STEAM SUPPLY SYSTEMS" (EPID: L-2018-TOP-0025) DATE: SEPTEMBER 26, 2019

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ADAMS Accession Nos.
ML19266A669 -Package
ML19266A567 -Cover Letter
ML19266A668 -Enclosure ***concurrence via e-mail** **NRR-106**

OFFICE	NRR/DLP/PLPB/PM	NRR/DLP/PLPB/LA*	NRR/DMLR/MPHB/BC	NRR/DLP/PLPB/BC
NAME	JDrake	DHarrison	JTsao	DMorey
DATE	9/3/19	9/24/19	9/25/19	9/26/19

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT PWROG-17033, REVISION 1

"UPDATE FOR SUBSEQUENT LICENSE RENEWAL: WCAP-13045, COMPLIANCE TO ASME CODE CASE N-481 OF THE PRIMARY LOOP PUMP CASINGS OF WESTINGHOUSE TYPE NUCLEAR STEAM SUPPLY SYSTEMS"

PRESSURIZED WATER REACTOR OWNERS GROUP

DOCKET NO. 99902037

EPID: L-2018-TOP-0025

1.0 INTRODUCTION

By letter dated June 14, 2018 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML18170A113), the Pressurized Water Reactor Owners Group (PWROG) submitted for U.S. Nuclear Regulatory Commission (NRC) review and approval topical report (TR) PWROG-17033-P and -NP, Revision 1, "Update for Subsequent License Renewal: WCAP-13045, Compliance to ASME (American Society of Mechanical Engineers) Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems." By letter dated March 19, 2019, the PWROG submitted a supplement to PWROG-17033, "Generic Responses to the U.S. NRC Requests for Additional Information (RAIs) on PWROG Report, PWROG-17033, for Westinghouse Reactor Coolant Pump Casings" (ADAMS Package Accession No. ML19091A098).

The PWROG submitted PWROG-17033, Revision 1, to support individual applicants in the subsequent license renewal applications to satisfy the time-limited aging analysis (TLAA) requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 54.21(c)(1). Specifically, the PWROG report is to address TLAA for Westinghouse-designed RCP casings through the subsequent license renewal period of operation to 80 years.

The PWROG has published Revision 0 and Revision 1 of the PWROG-17033 report. This safety evaluation (SE) is only applicable to PWROG-17033, Revision 1, and is referred to in this SE as the PWROG-17033 report. The submittal package is available in ADAMS under Package Accession No. ML18170A089. The non-proprietary (-NP) and proprietary (-P) versions of the TR are available in ADAMS at Package Accession No. ML18170A114.

2.0 REGULATORY EVALUATION

The regulation in 10 CFR 54.21(c) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the NRC staff must make to issue a renewed license (10 CFR 54.29(a)) is that adequate actions have been identified and have been or will be taken with respect to managing the effects of aging during the subsequent period of extended operation on the functionality of structures and components that have been identified to require review under

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10 CFR 54.21. There should be reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. As described in the Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants (SRP-SLR), NUREG-2192, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report, NUREG-2191.

The regulation in 10 CFR 54.21(c)(1)(i) states that the analyses remain valid for the period of extended operation. The regulation in 10 CFR 54.21(c)(1)(ii) states that, for a specific TLAA that is dispositioned in accordance with this regulation, applicants must demonstrate that the analysis has been projected to the end of the period of extended operation.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the PWROG to submit and for the Commission to review and disposition the TR PWROG-17033, Revision 1.

3.0 SUMMARY OF PWROG-17033, REVISION 1

The ASME Code, Section XI, Table IWB-2500-1, requires periodic volumetric examinations of the welds of the RCP casing in nuclear power plants. Westinghouse-designed RCP casing is fabricated with heavy-wall cast stainless steel. The PWROG stated that a volumetric inspection of the full thickness of the welds in the RCP casing using the ultrasonic test method from the outside diameter surface is impractical due to the severe attenuation associated with the large grain structures. A volumetric inspection of the full thickness of the welds would require unconventional approaches (inside diameter and outside diameter ultrasonic testing or radiographic testing) that require access to the internal side of the pump casing.

Since these inspections result in radiation exposure to the personnel performing the inspections and require significant resources to perform the inspections, the ASME Code Committee approved Code Case N-481, "Alternative Examination Requirements for Austenitic Pump Casings," in March 1990 that allows an alternative to the volumetric inspection requirement. The NRC endorsed Code Case N-481 in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability ASME Code Section XI Division 1," in April 1992.

ASME Code Case N-481 allowed replacing the volumetric examination of RCP casing with a fracture mechanics-based integrity evaluation supplemented by specific visual inspections. In September 1991, Westinghouse published TR WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems" (ADAMS Accession No. 9111080158) which contains the structural integrity evaluation that was performed to demonstrate compliance with ASME Code Case N-481 for 40 years of operation.

Since the issuance of WCAP-13045, the ASME Code, Section XI, Table IWB-2500-1 has been updated over time to be consistent with the guidance in Code Case N-481 to require visual inspections of the RCP casing. In March 2004, Code Case N-481 was annulled by ASME, and the information in Code Case N-481 was incorporated into the 2008 Addenda of ASME Code Section XI. The 2000 Addenda of ASME Code, Section XI replaced the Category B-L-1 volumetric examinations of pump casing welds with visual examinations, while the ASME Code, Section XI, 2008 Addenda eliminated the volumetric examinations of pump casing welds completely. The only required examination is a visual examination of the pump casing (Examination Category B-L-2) when the pump is disassembled for maintenance or repair.

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The purpose of the PWROG-17033 report is to extend the fracture mechanics integrity evaluation in WCAP-13045 through subsequent license renewal of 80 years. The goal of the PWROG-17033 report is to demonstrate that the structural integrity analyses performed in WCAP-13045 adequately represented and qualified the Westinghouse-designed pump casing for all loadings and service conditions to the end of 80 years.

In the past, plants have used the generic fracture mechanics evaluation performed in WCAP-13045 to comply with the requirements of ASME Code Case N-481 for plant-specific license renewal applications to extend plant operations to 60 years. In the PWROG-17033 report, the fracture mechanics evaluation provides the justification for performing visual inspections, in lieu of volumetric inspections, for the RCP casing as incorporated in the ASME Code Section XI, and extend the applicability of WCAP-13045 to 80 years of operation for all plants with Westinghouse-designed RCP pump casing.

The PWROG stated that there are eight different models of RCPs in Westinghouse-type pressurized water reactors (PWR), Models 63, 70, 93, 93A, 93A-1, 93D, 100A, and 100D. Models 63, 70, 93, and 93D all have a tangent outlet nozzle. Models 93A and 93A-1 have outlet nozzles that are radially orientated. Models 100A and 100D are similar to the general shape of Model 93 but with a radially oriented outlet nozzle like Model 93A. Models 93, 93A, and 93A-1 are the most common, making up around 90 percent of the total domestic Westinghouse-type PWR plants.

WCAP-13045 chose an analytical model representative of each of the outlet nozzle configurations for a three-dimensional finite element stress analysis and fracture mechanics evaluation. The representative models chosen were the Model 93A (radial outlet nozzle) and Model 93 (tangential outlet nozzle). The pump casings are fabricated from SA-351 CF8 except for the pumps of three plants which were fabricated from SA-351 CF8M. The SA-351 CF8M and CF8 are known to be susceptible to thermal aging.

WCAP-13045 used the limiting pump casing CF8M fracture toughness and the tearing modulus in J-integral stability evaluations based on elastic-plastic fracture mechanics. The PWROG stated that the WCAP-13045 fracture toughness determinations are applicable to 80 years of service because the fracture toughness parameters are at full-aged saturated conditions; therefore, any additional aging past 40 years does not have an impact on the fracture toughness parameters. Section 3.2 of the PWROG-17033 report compared the minimum fracture toughness in WCAP-13045 with the latest industry toughness correlations in NRC published report, NUREG/CR-4513, Revision 2, O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," May 2016. The PWROG demonstrated that the fracture toughness values in WCAP-13045 are less than (more conservative and limiting) than the fracture toughness values in NUREG/CR-4513, Revision 2. The PWROG contended that, as compared with the current industry standards, the fracture toughness values in WCAP-13045 are limiting and applicable to 80 years of operation. The PWROG concluded that the elastic-plastic fracture mechanics analysis for the pump casings analyzed in WCAP-13045 is valid for 80 years because the minimum fracture toughness used in the crack stability analysis is applicable to 80 years of service life.

The PWROG also performed a qualitative evaluation for the fatigue crack growth (FCG) evaluation. The PWROG determined that the current FCG rate for stainless steel in a water environment based on the 2013 Edition of the ASME Code, Section XI, are comparable to the rates used in WCAP-13045, and there will be an insignificant impact of the extended operation on the crack growth analysis. The PWROG stated that the stresses used in the FCG analysis

are generic and envelop the various pump designs. Additionally, these stresses do not change for 80 years of operation, as they are not time-dependent. The PWROG further stated that the stress intensity factors used in the FCG analysis are consistent with the current industry standards for similar FCG evaluations.

According to the PWROG-17033 report, the transient definitions in the FCG analysis are not expected to change over the design life of the plant, and the transient cycles used in the WCAP-13045 are also assumed to bound the predicted 80-year transient cycles. The PWROG-17033 report determined that there is sufficient margin between the final crack growth and the flaw size used for stability. Therefore, even if the 40-year transient cycles are doubled for 80 years of operation, the final flaw size after FCG from the initial flaw size of 0.3 inches will be less than the stability flaw size of 1/4T flaw depth (T is the wall thickness of the pump casing) for the stability analysis in WCAP-13045.

The PWROG-17033 report concluded that the fracture mechanics integrity evaluation in WCAP-13045 for pump casings is applicable to 80 years of design life. The fracture mechanics evaluation for 80 years in WCAP-13045 justifies continuing to perform visual inspections, in lieu of volumetric inspections, for pump casings.

4.0 TECHNICAL EVALUATION

As stated above, Westinghouse performed structural integrity analyses of the RCP casing as shown in WCAP-13045 to satisfy the requirements of ASME Code Case N-481. The analyses in WCAP-13045 are applicable to the 40-year operating license. As part of the 60-year license renewal application, PWR applicants are required to submit crack stability and fatigue crack growth analyses of the RCP casing as a time-limited aging analysis item pursuant to 10 CFR 54.21(c)(1). The affected PWR applicants submitted plant-specific structural integrity analyses of the Westinghouse-designed RCP casing as part of 60-year license renewal applications. Those analyses were based on an extension of the fracture mechanics methodology used in WCAP-13045.

For the efficiency of subsequent license renewal applications, the PWROG submitted the TR PWROG-17033, Revision 1, to satisfy the time-limited aging analysis requirements of 10 CFR 54.21(c)(1).

For the current evaluation, the NRC staff verifies (1) whether the input parameters in the crack stability analysis and FCG analysis in the PWROG-17033 and WCAP-13045 reports are acceptable to demonstrate structural integrity of the Westinghouse-designed RCP casing at the end of 80 years, and (2) whether the input parameters are bounding for the plants that are covered by the PWROG-17033 and WCAP-13045 reports.

4.1 General Information

The Westinghouse-designed RCP casing is fabricated from cast austenitic stainless steel (CASS), which is resistant to corrosion and exhibits certain material fracture toughness values (i.e., J_{Ic}, J_{max}, and T_{mat}) in the pre-service product form. However, CASS is subject to thermal aging embrittlement at nominal operating temperatures of nuclear plants, meaning that material fracture toughness will be reduced with age and may not resist crack propagation. The PWROG designed the pump casing to the ASME Code at the time of fabrication and the stresses of the RCP casing met ASME Code conditions and allowable stresses. The PWROG explained that design, normal, operating, and faulted applied loads on the RCP nozzles are

conservative when compared to actual service conditions. The PWROG indicated that it is not aware of service-induced degradation of RCP casings from corrosion or other cracking mechanisms.

4.2 Crack Stability Analysis

4.2.1 *Acceptance Criteria*

As discussed above, Code Case N-481 allows replacing the volumetric examinations with a fracture mechanics-based integrity evaluation supplemented by specific visual inspections. Specifically, Code Case N-481(d) requires that licensees and applicants "...perform an evaluation to demonstrate the safety and serviceability of the pump casing. The evaluation shall include the following: (1) evaluating material properties, including fracture toughness values; (2) performing a stress analysis of the pump casing; (3) reviewing the operating history of the pump; (4) selecting locations for postulating flaws; (5) postulating one-quarter thickness reference flaw with a length six times its depth; (6) establishing the stability of the selected flaw under the governing stress conditions; and (7) considering thermal aging embrittlement and any other processes that may degrade the properties of the pump casing during service...."

The NRC staff notes that structural integrity of the RCP casing depends on how well the casing material resists crack initiation and propagation. Code Case N-481 allows a crack of one-quarter casing wall thickness in depth to exist. The question is whether the RCP casing material can resist the propagation of such crack. Structural integrity of the RCP casing is established and accepted when the crack is demonstrated to be stable and not propagate uncontrollably. To address the crack propagation concern, the industry (Westinghouse) performed a crack stability analysis as shown in WCAP-13045.

For the crack stability analysis, the NRC staff evaluated the input parameters, specifically postulated flaw size, applied loading, and material fracture toughness to ensure that the parameters are bounding and are acceptable to the end of 80-year operating license. The NRC staff evaluated thermal aging of the CASS pump casing (CF8M and CF8) to verify that the crack stability analysis used the fully-aged, saturated fracture toughness values at the end of 80 years as the stability acceptance criteria. The fracture toughness properties of the CASS materials are crack initiation toughness (J_{Ic}), the maximum fracture toughness, $J_{maximum}$ (or J_{max}), and the tearing modulus, $T_{material}$ (or T_{mat}), which are determined based on J-integral resistance curves.

The crack is demonstrated stable and acceptable when the applied loading in terms of applied J-integral $J_{applied}$ (or J_{app}) and applied tearing modulus $T_{applied}$ (or T_{app}) at the crack tip are less than the material fracture toughness values J_{Ic}, T_{mat}, and J_{max}. The crack stability acceptance criteria are as follows:

(1) $J_{app} < J_{Ic}$, or (2) if $J_{app} \geq J_{Ic}$, then $T_{app} < T_{mat}$ and $J_{app} \leq J_{max}$

4.2.2 *Postulated Flaw Size*

To demonstrate crack stability in the RCP casing, WCAP-13045 postulated flaws of 1/4T (1/4 thickness of RCP casing wall, T) deep with a six-to-one (6:1) aspect ratio (i.e., the flaw length is 6 times the flaw depth). This postulated flaw size is consistent with the required flaw size of Code Case N-481(d)(5). In addition, WCAP-13045 modeled the postulated 1/4T flaws at various high stressed regions in accordance with Code Case N-481(d)(6) and various locations in welds not affected by discontinuities such as nozzles.

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In RAI-1(a), the NRC staff asked the PWROG to confirm that the locations of the postulated flaws in WCAP-13045 represent and bound the high stress areas of the RCP casings that were analyzed. In its response, the PWROG stated that the postulated flaw locations in WCAP-13045 represent and bound the high stress areas of all applicable Westinghouse RCP casing designs in the U.S. PWR operating fleet. The PWROG also stated that the geometry of the Model 93A-1 and Model 100A pump casings in the high stress locations is adequately represented by the Model 93 and Model 93A casings. By explicitly modeling and analyzing the Model 93 and Model 93A pump casings using finite element techniques, the high stress areas of the casings for all plants were adequately represented in WCAP-13045.

In RAI-1(c), the NRC staff asked about the critical crack depth in the RCP casing that would exceed the crack stability acceptance criteria. In its response, the PWROG stated that WCAP-13045 identified that flaw location [ ] is the limiting flaw location for the Model 93 pump casing, and flaw location [ ] is the limiting flaw location for the Model 93A pump casing with regard to fracture toughness. The postulated flaw sizes were sufficiently large, with flaw depths on the order of 1/4T, which provided flaw depths greater than 1 inch. Furthermore, WCAP-13045 stated that SA-351 CF8M material was selected for analysis because it possesses limiting fracture toughness values resulting from higher levels of molybdenum in the chemistry compared to CF8. The PWROG explained that higher levels of molybdenum result in lower values for J_{Ic}, T_{mat}, and J_{max}. The PWROG stated that the plant-specific evaluations for the plants with CF8M showed that [

] The PWROG indicated that postulated flaws larger than the range mentioned above would have difficulty satisfying the crack-stability criteria, unless other parameters can be reassessed (i.e., refinement in loads, stresses, material properties, etc.). The PWROG stated that for the CF8M plant-specific analyses, the crack stability criteria were met per the guidance of Code Case N-481 for a postulated flaw depth of 1/4T. For CF8 pump casings, which have very low molybdenum, the critical flaw depth would be generically larger than that based on CF8M. The NRC staff finds that the postulated 1/4T flaw depth with an aspect ratio of 6 to 1 is fairly large. The NRC staff finds that there is still a margin between the postulated flaw depth and critical flaw depth.

In RAI-3(a), the NRC staff asked about the degradations in the RCP casings. In its response, the PWROG explained that to date, Westinghouse-designed RCP casings have not had service-related degradation from corrosion or other cracking mechanisms.

In RAI-6(c), the NRC staff asked the PWROG to discuss the likelihood of postulated flaw with the applied loading occurring in any Westinghouse RCP casing. In its response, the PWROG stated that the likelihood of postulated flaw [ ] occurring with the applied screening loads from WCAP-13045 is low in the actual conditions. Therefore, the flaw at location [ ] is assumed in a theoretical, worst case scenario used for analysis purposes to generically bound those PWR plants which use Westinghouse-designed RCP casings, and to satisfy the fracture mechanics evaluation requirements in Code Case N-481.

The NRC staff finds that the postulated flaw size in WCAP-13045 has satisfied the provisions of Code Case N-481(d)(4), (d)(5) and (d)(6) because the crack stability analysis postulated 1/4T flaw with a length six times its depth in the governing stress location and conditions.

4.2.3 *Applied Loading*

In RAI-1(b), the NRC staff asked the PWROG to discuss whether the J_{app} values in WCAP-13045 bound the J_{app} values from the RCPs in all the plants that were analyzed. In its response, the PWROG explained that as discussed in Section 6 of WCAP-13045, conservative loads (internal force and moment loading) for most of the plants were selected to form the basis for the crack stability analysis. The J_{app} values reported in Section 11 of WCAP-13045 were based on the screening loads in Loading Level A (faulted condition, safe shutdown earthquake) and Loading Level C (upset condition, loss of load transient). Those screening loads were developed as a set of representative bounding loads based on a sampling of plant-specific loads. The NRC staff notes that the definition of loading levels A, B, and C in WCAP-13045 is different from that of the ASME Code, Sections III and XI. As stated in WCAP-13045, plants were expected to reconcile their plant-specific loads (forces and moments) with the screening loads. If plant-specific loads were larger than the screening loads used in the crack stability analysis in WCAP-13045, then updated J_{app} values would be calculated using plant-specific loads to confirm the crack stability criteria. The PWROG stated that those reconciliations were performed in plant-specific reports, and it was determined that some plants had loads and J_{app} values that exceeded the screening values in WCAP-13045. The PWROG further stated that however, the J_{app} values associated with those higher plant-specific loads were still qualified without exception in plant-specific evaluations using either the limiting fracture toughness values in WCAP-13405 or the fracture toughness values calculated with the methodology in NUREG/CR-4513, Revision 1, "Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR systems." The PWROG stated that for the majority of the plant-specific evaluations, the screening stresses were sufficiently higher and more limiting than the plant-specific stresses. Hence, the J_{app} values in WCAP-13045 bound the J_{app} values for the majority of the plant-specific cases. The NRC staff has imposed conditions as shown further in the SE to handle those cases where plant-specific J_{app} is higher than the J_{app} used in WCAP-13045.

The PWROG noted that, over time, loads have not changed significantly because there have been only a few major nuclear steam supply system (NSSS) component replacements and refurbishments, and power uprates do not have a significant impact on stability evaluations. The PWROG stated that the screening loadings considered in WCAP-13045 are representative of plant-specific values that were considered in the existing crack stability analyses.

The PWROG further stated that the applied J-integral, J_{app}, and applied tearing modulus, T_{app}, are calculated based on the Electric Power Research Institute (EPRI) report, Kumar, V., German, M. D., and Shih, C. P., "An Engineering Approach for Elastic-Plastic Fracture Analysis," EPRI Report NP-1931, Project 1237-1, dated July 1981. This method is based on various combinations of loading parameters and material properties for various pump casing designs.

The PWROG indicated that applied J_{app} and T_{app} will not be affected by the extension of the operating period to 80 years. The NRC staff determines that the applied J_{app} and T_{app} are not affected by the operating time period unless operating conditions have been changed as a result of power uprate or the piping connected to the RCP is modified. The NRC staff has imposed conditions as shown further in the SE to require applicants to discuss whether the applied loadings have changed such that they exceed the screening loadings used in WCAP-13045 in their subsequent license renewal application.

In RAI-7, the NRC noted that Table A-1 of WCAP-13045 contains the names of 29 nuclear units that were considered in the fracture mechanics calculation. The NRC staff requested the PWROG discuss whether the applied loads and stresses of all the 29 nuclear units were considered and the bounding values from the 29 plants were used in the crack stability and FCG calculations in WCAP-13045. In its response, the PWROG explained that the applied loads for 29 plants were considered in Table 6-1 of WCAP-13045. The PWROG stated that to cover a wide range of plants, representative force and moment values were chosen and are listed in Table 6-2 of WCAP-13045. Furthermore, as mentioned in Sections 6 and 8 of WCAP-13045, the major contributor to the high stresses in the discontinuity regions of the pump casings is the internal pressure, which is set at a value of 2,635 psig or 2,485 psig for all plants based on the loading condition (i.e., Loading Levels A, B, or C). The PWROG explained that any differences in the force and moment loadings shown in Table 6-1 (note the load magnitudes are similar) will not have a significant impact on the final stability conclusions because the crack stability evaluation is related to total stress, not individual force and moment components. The NRC staff notes that the crack stability evaluation is based on stresses, not forces and moments.

In its response to RAI-2, the PWROG stated that the screening loads developed in WCAP-13045 were bounding for most of the RCP pumps. For plant-specific fracture mechanics analyses where the plant-specific loads were higher than the screening loads, the stability criteria were reanalyzed and shown to meet all fracture toughness margins. For the majority of plant-specific evaluations performed, it was determined that the screening stresses in WCAP-13045 were sufficiently higher than and bounded the plant-specific stresses. The PWROG stated that the applied loadings and stresses for all plants considered in WCAP-13045 have been qualified.

The PWROG indicated that plant-specific fracture mechanics evaluations on RCP casings have been performed for more than half of the U.S. operating PWRs with Westinghouse RCPs. Based on plant-specific evaluations, the majority of the plants were bounded by the screening loadings of WCAP-13045. For the few plants whose loads were not bounded by the screening loads, the casings of those plants were re-analyzed and shown to meet crack stability margins. The PWROG stated that for the three plants that have CF8M pump casings, all plant-specific loads were bounded by the screening loads of WCAP-13045.

The PWROG concluded that the screening loads considered in Table 6-2 of WCAP-13045, which are based on 29 plants, can be considered a good representation of the plants in the PWR fleet with Westinghouse RCP casings, as evident by the wide range of plant-specific evaluations that have met the crack stability criteria using the screening loads of WCAP-13045. The PWROG further concluded that other conservatisms are also present in the fracture mechanics evaluations, such as the use of ASME Code properties and the use of a large postulated flaw that has never been observed during 1400 reactor-years of operation.

In its response to RAI-6(c), the PWROG stated that the conservative screening loads are based on the maximum loss of load design transient pressure of 2,635 psig (400 psi above the normal operating pressure of 2,235 psig) and at a conservative temperature of 590 °F (40 °F above the normal operating temperature of 550 °F). As described in Section 6.1 of WCAP-13045, there are three systems in place to mitigate pressure increases of around 400 psi during a Loss of Load transient. First, there are power-operated relief valves (PORVs). These valves are activated at 2,335 psig as recommended in the technical specifications. The pressurizer safety valves are set for 2,485 psig, but allow the pressure to exceed 2,485 psig by 100 psi or more. The reactor trip high pressure setup is 2,385 psig. This set point is chosen so as to avoid

challenging the pressurizer safety valves. Thus, the pressure during a loss of load transient is actually limited to 2,485 psig in lieu of 2,635 psig, which was conservatively considered for certain cases in WCAP-13045.

The PWROG-17033 report indicated that the pump casing evaluations in WCAP-13045 are a generic integrity evaluation applicable to all Westinghouse-designed pump casings which demonstrates compliance to ASME Code Case N-481. WCAP-13045 provides enveloping or bounding criteria whereby a specific plant need only show that the pump casings of interest fall under the umbrella established in WCAP-13045. The loads used in the crack stability analysis in WCAP-13045 were selected to be conservative for a majority of the plants. PWROG-17033 explained that the plant-specific loads were not adequately covered by the umbrella loads considered in WCAP-13045; thus, the J_{app} values of WCAP-13045 do not completely bound certain plants. Therefore, a plant-specific flaw evaluation was performed. The J_{app} values have been recalculated using plant-specific loads and material properties. The PWROG stated that the plant-specific evaluations continue to meet the crack stability criteria.

The NRC staff finds that the screening loads used in WCAP-13045 are bounding in most cases and, therefore, are acceptable for most of Westinghouse plants. For plants whose plant-specific loadings exceeded the screening loads in WCAP-13045, plant-specific crack stability analyses have been performed and the analyses have shown those plants are within the crack stability acceptance criteria. The NRC staff has imposed conditions to address the scenario where the plant-specific applied loadings exceed the screening loads in WCAP-13045 as shown further in the SE.

4.2.4 *Fracture Toughness*

Table 5-1 of WCAP-13045 presented the end-of-life saturated fracture toughness (J_{Ic}, T_{mat}, and J_{max}) of the pump casings based on the limiting CASS (SA-351, CF8M) material. The PWROG stated that the fracture toughness properties shown in Table 5-1 of WCAP-13045 are already at the fully-aged saturated condition and, therefore, can represent the material conditions for the 80-year operation. The saturation room temperature impact energies of the CASS materials are determined from the chemical compositions. The PWROG-17033 report stated that the fracture toughness data in Table 5-1 of WCAP-13045 is based on the chemistry data from Appendix A of WCAP-13045. Table 5-1 presents the material fracture toughness values at end-of-service life for 3 high stress locations in model 93 RCP casings having CF8M material and 1 high stress location in all remaining RCP casings having CF8 material. The fracture toughness of CASS material is derived based on the chemistry such as Silicon, Chromium, Molybdenum, Nickel, Carbon, Manganese, and Nitrogen in percent weight and percent delta ferrite.

The PWROG used the fracture toughness values in WCAP-13045 as part of the crack stability analysis based on the J-integral approach. The PWROG explained that the fracture toughness material properties based on the CF8M (high molybdenum) material is more susceptible (limiting) to thermal aging than CF8 material. Therefore, the CF8M material is used in the generic fracture mechanics assessment in WCAP-13045. PWROG-17033 stated that the crack stability analysis results using CF8M fracture toughness will apply for the CF8 material as well.

The PWROG-17033 report indicated that its crack stability evaluation considers the latest fracture toughness correlations that have been developed for the CASS pump casings based on NUREG/CR-4513, Revision 2, "Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR systems," by Omesh Chopra, published in May 2016.

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The PWROG-17033 report explained that NUREG/CR-4513, Revision 2, provides a large database for CASS material and thermal aging, and builds on the work performed in 1994 for Revision 1 of NUREG/CR-4513. In 1994, the Argonne National Laboratory (ANL) completed an extensive research program in assessing the extent of thermal aging of CASS materials. The ANL research program measured mechanical properties of CASS materials after they had been heated in controlled ovens for long periods of time. ANL compiled a data base, both from data within ANL and from international sources, of about 85 compositions of CASS exposed to a temperature range of 290-400 °C (550-750 °F) for up to 58,000 hours (6.5 years). In 2015, the work done by ANL was augmented, and the fracture toughness database for CASS materials was aged to 100,000 hours at 290-350 °C (554-633 °F). The methodology for estimating fracture properties has been extended to cover CASS materials with a ferrite content of up to 40 percent. From the database in NUREG/CR-4513, Revision 2, ANL developed lower bound correlations for estimating the extent of thermal aging of CASS. ANL developed the fracture toughness estimation procedures by correlating data in the database conservatively. After developing the correlations, ANL validated the estimation procedures by comparing the estimated fracture toughness with the measured value for several CASS components harvested from actual plant service.

The PWROG used the procedure developed by ANL to derive the fully saturated fracture toughness values for CASS material of the RCP casing. The PWROG indicated that its confirmatory analysis showed that the minimum fracture toughness values based on the original methodology in WCAP-13045 is still limiting as compared to the fracture toughness properties based on NUREG/CR-4513, Revision 2.

The PWROG-17033 report stated that NUREG/CR-4513, Revision 2, provides the fracture toughness correlations for the fully aged condition of CASS materials and the correlations are applicable for the CF8M pump casings of the plants operating at and beyond 15 effective full power years (EFPYs). The PWROG-17033 report further states that currently, RCPs in the Westinghouse fleet are operating well beyond the 15 EFPYs service life; therefore, the use of the fracture toughness correlations described in NUREG/CR-4513, Revision 2, is applicable for the fully aged or saturated conditions.

In RAI-4, the NRC staff noted that Section 11.2 of WCAP-13045 indicates that a postulated flaw location [ ] is subject to the loss of load transient (upset condition) and is identified as the highest stressed location. WCAP-13045 also indicates that on a plant-specific basis, a yield strength level slightly greater than 20 ksi (thousand pounds per square inch) is sufficient to confirm the flaw stability at this flaw location. WCAP-13045 further indicates that such a yield strength level (greater than 20 ksi) will ensure that the stability criteria regarding the fracture toughness and tearing modulus are satisfied (i.e., $J_{app} < J_{max}$, and $T_{app} < T_{max}$). In contrast, the NRC staff noted that Table 2 of the PWROG-17033 report uses a yield strength level less than 20 ksi to calculate the fracture toughness properties of the CF8M material in the crack stability analysis. The NRC staff also noted that the PWROG-17033 report does not address whether the postulated flaw location [ ] in Section 11.2 of WCAP-13045 meets the crack stability criteria for 80 years of operation when the crack stability analysis uses the updated fracture toughness properties. The NRC staff questioned the stability of postulated flaw location [ ], considering the plant-specific yield strength of the material, actual loading conditions, and the fracture toughness values of J_{Ic}, J_{max} and T_{mat}.

In addition, the NRC staff notes that the PWROG-17033 report used NUREG/CR-4513, Revision 2, to derive material fracture toughness values. WCAP-13045 used its own method to derive a set of limiting fracture toughness values. The NRC staff asked the PWROG to derive

material fracture toughness values based on NUREG/CR-4513, Revision 1, as a comparison to the values derived from NUREG/CR-4513, Revision 2. The NRC staff asked the PWROG to compare the fracture toughness values derived from the three sources--WCAP-13045, NUREG/CR-4513, Revision 1, and NUREG/CR-4513, Revision 2. The NRC staff's intent is to ensure that the fracture toughness values used in the crack stability analysis in WCAP-13045 are truly aged and saturated (limiting and bounding) for 80 years.

In its response to RAI-4, the PWROG stated that of all the domestic operating plants considered in WCAP-13045, only three plants have pump casings made from SA-351 CF8M material. The fracture toughness values from the most limiting heat were selected from the three CF8M plants to represent the highest stressed flaw location [ ] as shown in Table 5-1 of WCAP-13045.

The PWROG used the chemistry of the limiting heat of CF8M material at the highest stressed flaw location to recalculate the fracture toughness values based on NUREG/CR-4513, Revision 1. Based on NUREG/CR-4513, Revision 1, the PWROG derived the fracture toughness values at operating conditions. The PWROG noted that the saturated fracture toughness values based on NUREG/CR-4513, Revision 1, are larger than the values determined in Table 5-1 of WCAP-13045 for high stress flaw location [ ].

Based on WCAP-13045, Table 11-2, for CF8M material, the highest stress flaw location [ ] for loading level C has applied loadings of [ ] when the pressure is limited to 2,485 psi due to reactor trip. The NRC staff notes that WCAP-13045, Tables 11-2 to 11-5, Table 11-7, Table 11-9 present J_{Ic}, T_{mat}, J_{max}, J_{app}, and T_{app} values associated with various postulated flaws for various RCP models. Comparing the calculated J_{app} and T_{app} values to the material fracture toughness values derived from NUREG/CR-4513, Revision 1, the NRC staff finds that the high stress flaw location satisfies the acceptance criteria for crack stability.

With regard to loading conditions, the plant-specific evaluation performed for the plant with the limiting CF8M material heat for high stress flaw location [ ] reports that the actual plant-specific loads are below the screening loads used in WCAP-13045. As a result, the screening loads bound the plant-specific loads, and the J_{app} and T_{app} values in WCAP-13045 remain bounding for the plant with the limiting CF8M casing at the high stress flaw location.

Thus, the generic crack stability analysis in WCAP-13045 confirms the stability of postulated flaw location [ ] taking into account conservative ASME Code yield strength (in lieu of plant-specific yield strength) of the material, screening loads (in lieu of less limiting actual loading conditions), and the fracture toughness values (J_{Ic}, J_{max}, and Tmat) based on NUREG/CR-4513, Revision 1.

In RAI-2(a), the NRC staff asked the PWROG to provide guidance to applicants/licensees to demonstrate that the input parameters used in WCAP-13045 analyses are bounding. In its response, the PWROG stated that the majority of domestic operating plants with Westinghouse pumps are made from CF8 material. Table 5-1 of WCAP-13045 presents the limiting fracture toughness values (J_{Ic}, T_{mat}, and J_{max}) in CF8 material. The PWROG stated that as can be seen in WCAP-13045, Table A-2, those values are the limiting fracture toughness properties for any of the plants whose pump casings are made from CF8 material, regardless of the plant-specific chemistry of the pump casing material. Only three U.S. PWR plants have pump casings that are made from CF8M material. Table 5-1 of WCAP-13045 presents the limiting CF8M fracture toughness values for the flange inner quarter, the nozzle outer quarter, and the nozzle inner

quarter locations. Each of those limiting sets of values in Table 5-1 is based on the pump casing chemistry of one of the CF8M casings, and, as can be seen from Table A-2, pages A-29, A-30, and A-41 of WCAP-13045, they bound the fracture toughness values of the other CF8M casings at those locations.

The PWROG explained that the three plants that have the CF8M casings were analyzed on a plant-specific basis and were shown to meet the crack stability margins. As a result, the fracture toughness values reported in WCAP-13045 bound the fracture toughness values of all the CF8M pump casings. Over time, the fracture toughness values for SA-351 CF8M and CF8 materials reach a saturated state based on chemistry (molybdenum, delta ferrite), casting method (static or centrifugal), temperature, and time as discussed in NUREG/CR-4513, Revision 2 (see Figure 12 of NUREG/CR-4513, Revision 2). The PWROG-17033 report used a lower bound saturated fracture toughness correlation curve from NUREG/CR-4513, Revision 2, to demonstrate that the bounding CF8M material heat from all Westinghouse RCPs, as shown in Table 5-1 of WCAP-13045, still has sufficient fracture toughness to meet the stability acceptance criteria and screening loadings of WCAP-13045.

The NRC staff finds that the minimum fracture toughness values based on the original methodology in WCAP-13045 are still limiting as compared to the fracture toughness properties derived from NUREG/CR-4513, Revision 2. However, the NRC staff questioned how the fracture toughness values derived from NUREG/CR-4513, Revision 2, compared to those derived from NUREG/CR-4513, Revision 1. The discussion below provides additional technical basis.

4.2.5 *Saturated Fracture Toughness Using NUREG/CR-4513, Revisions 1 and 2*

In RAI-8, the NRC staff noted that Section 3.2 of the PWROG-17033 report states that the material fracture toughness values were derived from NUREG/CR-4513, Revision 2. The NRC staff notes that Aging Management Program XI.M12, *Thermal Aging Embrittlement of Cast Austenitic Stainless Steel*, in Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) report, NUREG-2191, Volume 2, discusses fracture toughness values based on the prediction method in NUREG/CR-4513, Revision 1, not Revision 2. The NRC staff questioned whether the saturated fracture toughness value used in the crack stability analysis in WCAP-13045 would still be bounding as compared to the saturated fracture toughness values predicted in accordance with NUREG/CR-4513, Revision 1 or Revision 2.

In RAI-9, the NRC staff noted that the fracture toughness (J_{Ic}, J_{max}, and T_{mat}) values in Table 1 of the PWROG-17033 report are bounding and were used to demonstrate the crack stability of RCP casings. However, the J_{Ic}, J_{max}, and T_{mat} values used in Tables 11-2 and 11-3 of WCAP-13045 for high stress flaw location [ ] were higher (i.e., not conservative) than the J_{Ic}, J_{max}, and T_{mat} values listed in Table 1 of the PWROG-17033 report. The NRC staff finds that if the lower J_{Ic}, J_{max}, and T_{mat} values in Table 1 of the PWROG-17033 report were used, the postulated crack at the highest stressed location may not satisfy the crack stability acceptance criteria. The NRC staff also noted that it appears that the fracture toughness values in Table 5-1 of WCAP-13045 should be compared to the fracture toughness values predicted based on the method in NUREG/CR-4513, Revision 1, not Revision 2.

To respond to RAI-8 and RAI-9, the PWROG re-calculated the four sets of fracture toughness values reported in Table 5-1 of WCAP-13045 based on the methodology outlined in NUREG/CR-4513, Revisions 1 and 2. The PWROG reviewed and reconciled any major

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differences in the fracture toughness values calculated by using the methodology of WCAP 13045 and the two revisions of NUREG/CR-4513 as discussed below.

4.2.6 *NUREG/CR-4513, Revision 1*

The PWROG re-calculated all four sets of fracture toughness values (J_{Ic}, J_{max}, and T_{mat}) in Table 5-1 of WCAP-13045 using the prediction method in NUREG/CR-4513, Revision 1, and the results were shown in Table 1 of the PWROG's response to the March 19, 2019, letter. The PWROG re-evaluated all crack stability results given in Tables 11-2 through 11-5, Table 11-7, and Table 11-9 of WCAP-13045 using the fracture toughness values derived based on the methodology of NUREG/CR-4513, Revision 1. The PWROG noted that the fracture toughness values using NUREG/CR-4513, Revision 1, are higher than (not as conservative as) those from WCAP-13045, except for T_{mat} for two cases.

The PWROG stated that based on the re-evaluation of the fracture stability margins in Section 11 of WCAP-13045, the crack stability criteria were satisfied in all but one case. The one case that does not show acceptable stability results is flaw location [ ] (Level C loading). As shown in Table 11-7 of WCAP-13045, J_{app} is below J_{max} for this flaw case, but T_{app} is slightly above the T_{mat} calculated by using NUREG/CR-4513, Revision 1. The PWROG stated that as a more realistic scenario, it is reasonable to assume a reactor trip transient with a pressure limit of 2,485 psig in lieu of the highly conservative pressure of 2,635 psig. As shown by Note (b) in Table 11-7 of WCAP-13045, this condition was assumed in order to show acceptability of flaw location [ ] for Level C loading. As a result of using the lower pressure, the T_{app} will be reduced to a value below the bounding criteria limit of T_{mat} as calculated using NUREG/CR-4513, Revision 1, methodology. The PWROG stated that using the technical justification above, all flaw locations reported in Tables 11-2 through 11-5, Table 11-7, and Table 11-9 of WCAP-13045 satisfy the fracture toughness bounding criteria calculated using the methodology of NUREG/CR-4513, Revision 1 and are, therefore, shown to be stable.

4.2.7 *NUREG/CR-4513, Revision 2*

The PWROG re-calculated all 4 sets of fracture toughness values (J_{IC}, J_{max}, and T_{mat}) in Table 5-1 of WCAP-13045 using the prediction method in NUREG/CR-4513, Revision 2, and the results were shown in Table 2 of the response to RAI-9 in the March 19, 2019, letter. The PWROG stated that NUREG/CR-4513, Revision 2, produces more limiting (lower) material fracture toughness values than NUREG/CR-4513, Revision 1. However, comparing the fracture toughness values produced by the two methodologies, only a few NUREG/CR-4513, Revision 2, fracture toughness values are less than WCAP-13045.

The first three cases (i.e., nozzle inner quarter, nozzle outer quarter, and flange inner quarter) which are similar to Table 5-1 of WCAP-13045, consider SA-351, CF8M stability criteria. Based on the fact that there are CF8M pump casings from three different plants in the entire Westinghouse PWR operating fleet, the PWROG analyzed each of the above cases separately for crack stability based on Tables 11-2, 11-5, 11-7, and 11-9 of WCAP-13045. The PWROG also evaluated the CF8 material in Table 2 of the response to RAI-9 in the March 19, 2019, letter with the crack stability criteria of WCAP-13045 to show acceptable margins as discussed below.

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4.2.8 *Nozzle Inner Quarter – CF8M Material*

The PWROG stated that the fracture toughness values (J_{Ic}, J_{max}, and T_{mat}) per NUREG/CR-4513, Revision 2, are slightly less than those in WCAP-13045. However, the J_{app} and T_{app} values reported in Table 11-2 of WCAP-13045 are still sufficiently below the NUREG/CR-4513, Revision 2, values for J_{max} and T_{mat}. In addition, plant-specific analysis shows that the plant- specific loads are bounded by the screening loads reported in WCAP-13045. As a result, all stability margins are met, and crack stability has been demonstrated for this flaw location.

4.2.9 *Nozzle Outer Quarter–CF8M*

The PWROG stated that all fracture toughness values (J_{Ic}, J_{max}, and T_{mat}) per NUREG/CR-4513, Revision 2, are higher than those reported in WCAP-13045. Therefore, the values reported in WCAP-13045 remain limiting and bounding. Furthermore, plant-specific analysis shows that the plant-specific loads are bounded by the screening loads reported in WCAP-13045. As a result, all stability margins are met, and crack stability has been demonstrated for this flaw location.

4.2.10 *Flange Inner Quarter –CF8M*

The PWROG stated that the fracture toughness value of J_{max} per NUREG/CR-4513, Revision 2, is higher than that reported in WCAP-13045; however, the J_{Ic} and T_{mat} values per NUREG/CR-4513, Revision 2, are less than those in WCAP-13045. From Table 11-2 of WCAP-13045, the limiting flaw location [ ] has a J_{app} value of [ ] and a T_{app} value of [ ] based on the use of a reactor trip pressure of 2,485 psig as shown in Note (e) of Table 11-2. The J_{max} value of [ ] per NUREG/CR-4513, Revision 2, is greater than the J_{app} value of [ ]; however, the T_{mat} value of [ ] per NUREG/CR-4513, Revision 2, is less than the T_{app} value of [ ]. The PWROG stated that those limiting CF8M fracture toughness values are specific to one pump casing of one plant. The other two CF8M plants showed acceptable stability results at flaw location [ ], as they have different material heats with higher calculated fracture toughness values. The PWROG stated that the plant-specific analysis shows that the screening loads in WCAP-13045 are higher than the plant-specific loads for the plant with the limiting fracture toughness values as shown in Table 3 of the response to RAI-9 in the March 19, 2019, letter. Therefore, the stresses used in the T_{app} value are conservative in WCAP-13045 for flaw location [ ]. Furthermore, ASME Code properties have been used to determine the T_{app} in WCAP-13045, but actual plant-specific yield strength is typically higher than the minimum Code yield strength. As a result, the T_{app} values in WCAP-13045 are conservative relative to actual material yield strengths, and if the actual plant-specific T_{app} value was calculated using realistic plant-specific inputs (i.e., loads and yield strength), T_{app} would fall below the T_{mat}.

The PWROG explained that for this particular flaw location [ ], only one plant has a CF8M heat that is limiting for flaw stability criteria. However, that plant demonstrates acceptable flaw stability using the methodology of WCAP-13045 and NUREG/CR-4513, Revision 1. Furthermore, that heat of the CF8M material would likely demonstrate acceptable margins using NUREG/CR-4513, Revision 2, methodology if actual plant-specific loads and yield strength were used. The PWROG explained that based on the previous discussion and no known service degradation of this limiting CF8M casing, sufficient crack stability for this flaw location has been demonstrated.

With respect to CF8 material, the PWROG stated that the fracture toughness values per NUREG/CR-4513, Revision 2, are all higher than the WCAP-13045 values with the exception of T_{mat}, which is discussed below. The SA-351 CF8 fracture toughness properties are considered in WCAP-13045, Table 11-5 (Model 93), Table 11-7 (Model 93A) and Table 11-9 (Models 63 and 100A). In Table 11-5 for Model 93 pump casings, all stability criteria ($T_{app} < T_{mat}$ and $J_{app} < J_{max}$) are met using fracture toughness values per NUREG/CR-4513, Revision 2. In Table 11-9 for Model 100A pump casings, all stability criteria are also met with the use of fracture toughness values from NUREG/CR-4513, Revision 2. There are no Model 63 pumps in the U.S. PWR operating fleet; therefore, the stability results for Model 63 in Table 11-9 are not applicable.

In WCAP-13045, Table 11-7 (Model 93A), the stability criteria for flaw location [ ] (Level C) are met with a reactor trip pressure limit of 2,485 psig as shown in Note (b) in Table 11-7, using fracture toughness values per NUREG/CR-4513, Revision 2. In Table 11-7, flaw location [ ] (Level C) has a T_{app} value of [ ], which is higher than the T_{mat} value of [ ] based on NUREG/CR-4513, Revision 2. However, if the Reactor Trip pressure limit is used for flaw location [ ], then the T_{app} would drop consistently with the T_{app} drop seen for flaw location [ ] (i.e., T_{app} will be reduced by a value of approximately 21). As a result, all the stability results will be met for the Model 93A casing using the methodology of NUREG/CR-4513, Revision 2.

In RAI-6(b), the NRC staff asked the PWROG to discuss how a plant can demonstrate structural integrity of its RCP casings using PWROG-17033 if the crack stability criteria are exceeded. In its response, the PWROG stated that crack stability is demonstrated for the highest stressed flaw location [ ] and the limiting CF8M material casings using the ASME Code minimum properties, conservative screening loads, and updated fracture toughness values based on NUREG/CR-4513, Revision 1. The PWROG explained that if, hypothetically, the crack stability criteria are exceeded, then plant-specific yield strength along with the use of actual plant-specific loadings can be used to reduce conservatism in the evaluation. Use of a smaller flaw size, instead of 1/4T flaw, can also be considered, as the service experience for pump casings has shown no degradation or detected indications.

The PWROG concluded that the use of the methodology in NUREG/CR-4513, Revision 1 or Revision 2, will produce material fracture toughness values that are typically higher and less limiting than the material fracture toughness values derived in WCAP-13045. For cases where the fracture toughness values based on NUREG/CR-4513, Revision 2, are lower than those in WCAP-13045, the PWROG demonstrated that all flaw stability margins have been met based on plant-specific evaluations and realistic inputs for reactor trip pressure, plant-specific loads, and yield strength.

The PWROG further concluded that degradation or crack-like indications have not been found in Westinghouse designed RCP casings. Thus, based on the above discussions, the stresses, applied loadings, fracture toughness values, and transients considered in WCAP-13045 are bounding for an individual plant's pump casing or have been reconciled through plant-specific evaluations.

The NRC staff notes that if a crack stability analysis uses conservative parameters, the postulated crack may not satisfy the stability acceptance criteria. In this case, the crack analysis may not be realistic, and it may not be reasonable for such analysis to demonstrate structural integrity of a safety-related component, if the component has not experienced active degradation. It is unreasonable to expect WCAP-13045 to use the highest applied loadings and

a conservative postulated flaw to meet the lowest saturated fracture toughness values. However, it is NRC staff's expectation that a crack stability analysis does need to demonstrate that certain safety margin is maintained in the safety-related component.

The NRC staff further noted that the method to derive saturated fracture toughness values of CASS material may be changing in the future depending on the data in the future experiments. The saturated fracture toughness values are predicted based on the statistical analysis of the experimental thermal embrittlement data. As discussed above, ANL revised the derivation method of fracture toughness values between Revision 1 and Revision 2 of NUREG/CR-4513 based on additional specimen data. In the future, the derivation method may be changed again such that the saturated fracture toughness values may be lower than the values derived in WCAP-13045. However, based on the current data and derivation methods, the NRC staff finds that the fracture toughness values derived in WCAP-13045 are reasonable because in most cases they are lower than values derived from NUREG/CR-4315, Revisions 1 and 2.

4.2.11 *Crack Stability Analysis Summary*

The NRC staff determines that for the crack stability analysis:

(1) the PWROG-17033 and WCAP-13045 reports have used, in general, the limiting material fracture toughness values to demonstrate crack stability,

(2) the material fracture toughness values used in the crack stability analysis are shown to be at fully-aged, saturated conditions such that any additional aging will not lower the fracture toughness values and, therefore, will continue to remain applicable for 80 years of operation,

(3) the fracture toughness values used in WCAP-13045 are acceptable because they are, in most of cases, less (more conservative and limiting) than the fracture toughness values obtained by the method of NUREG/CR-4513, Revision 1 or Revision 2,

(4) for those cases where the fracture toughness values derived from NUREG/CR-4513, Revision 2, are lower than WCAP-13045, the PWROG provided reasonable explanation for the acceptance of the crack stability analysis in WCAP-13045,

(5) the applied loads and stresses used in the crack stability analysis are appropriate. For the cases where the applied loading (J_{app} or T_{app}) exceeded the material fracture toughness values (J_{IC}, J_{max} and T_{mat}), plant-specific analyses have shown that crack stability acceptance criteria are satisfied,

(6) Some plants may not satisfy the crack stability acceptance criteria or may have their applied loads exceeding the screening loading in WCAP-13045. These plants have performed a plant-specific crack stability analysis. The NRC staff has imposed conditions to address scenarios where plant-specific data exceed the inputs in the crack stability analysis of the WCAP-13045 or PWROG-17033 report. The NRC staff's imposed conditions are discussed further in the SE, and

(7) the crack stability analysis for RCP pump casings performed in the WCAP-13045 and PWROG-17033 reports is acceptable for the 80 years of operation.

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4.3 Fatigue Crack Growth Analysis

The PWROG-17033 report re-evaluated and confirmed that the fatigue crack growth (FCG) calculation in WCAP-13045 is still valid for the subsequent period of extended operation. Specifically, the PWROG-17033 report confirmed that the generic FCG analysis performed in WCAP-13045 remains applicable for 80 years of operation in terms of the stresses, stress intensity factor equations, transient definitions and cycles, and FCG rates.

The PWROG stated that two FCG analyses were originally performed in Section 12 of WCAP-13045 for the postulated crack in the highest stressed [ ] of the Model 93 and 93A pump casings. The PWROG also stated that the FCG analysis of the Model 93A pump was performed prior to the publication of the stainless steel FCG law in the ASME Code, Section XI. Therefore, the FCG law from the technical paper by Bamford, W. H., "Fatigue Crack Growth of Stainless Steel Piping in a Pressurized Water Reactor Environment," Trans ASME, Journal of Pressure Vessel Technology, dated February 1979, was initially considered in WCAP-13045. However, the FCG law used for the postulated flaw in Model 93 was based on Figure C-3210-1 of Article C-3000 of the 1989 Edition of the ASME Code, Section XI.

The PWROG compared the FCG rate used in WCAP-13045 with the FCG rate for stainless steel in air environment from the NRC-approved 2007 Edition with 2008 Addenda of ASME Code, Section XI, Appendix C. The PWROG applied a factor of two to the FCG rate to account for environmental effects of a postulated flaw in water as described in "Evaluation of Flaws in Austenitic Steel Piping," Trans ASME, Journal of Pressure Vessel Technology, Vol. 108, pp. 352-366, dated 1986.

The PWROG stated that there are no significant differences between the FCG rate in stainless steel in water in the NRC-approved 2007 Edition with 2008 Addenda of ASME Code, Section XI, Appendix C and the FCG rate used in WCAP-13045 for the postulated flaws in Model 93 RCPs. The PWROG further stated that the difference between the current stainless steel FCG rate in water and the FCG rate considered in WCAP-13045 for the postulated flaw in Model 93A RCP casing is also insignificant. The PWROG concluded that the existing FCG rates in Section 12 of WCAP-13045 are acceptable based on current industry standards for FCG for stainless steel material in a water environment.

The PWROG stated that other inputs required for an FCG analysis are the stress intensity factors (SIF), stresses, transient cycles, and transient definitions. The SIF correlations used for the FCG analysis in WCAP-13045 are consistent with the current correlations provided in the ASME Code, Section XI, Appendix A. The PWROG indicated that the transient stresses used in the FCG analysis are generic and encompass the various pump designs. These stresses have not changed for the subsequent period of extended operation. The PWROG further indicated that the numbers of predicted cycles for 80 years of extended operation are assumed to be bounded by the transient cycles considered in Table 12-2 of WCAP-13045. The PWROG stated that the transient definitions are also not expected to change over the 80-year operation. The PWROG-17033 report concludes that the generic transient descriptions and numbers of transient cycles in Table 12-2 of WCAP-13045 envelop the transient conditions for 80 years of operation.

The PWROG explained that plants considering the 80-year life extension count cycles typically comply with original design basis cycles for 40 years; therefore, there will be no significant increases in the number of cycles for 80 years. For a confirmatory evaluation, the PWROG

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doubled transient cycles designed for 40 years to apply to the 80-year FCG analysis to account for any large differences in the transient cycles. The PWROG-17033 report demonstrated that the final flaw size at the end of 80 years would still be less than the critical flaw size for crack instability at the high stress location.

In RAI-2(b), the NRC staff asked about the flaw size, flaw orientation, and direction of its growth. In its response, the PWROG stated that for the Model 93 pump casing, the FCG evaluations which produced the most limiting results were for postulated flaws at the [ ]. For the Model 93A pump casing, the limiting FCG results were for a postulated flaw oriented in the [ ].

The PWROG explained that the FCG analysis results for postulated flaw depths (flaw length is based on the aspect ratio, flaw length/flaw depth, of 6:1) into the pump casing wall thickness are provided in Tables 12-1 and 12-3 of WCAP-13045 for the Model 93A and Model 93 pump designs, respectively. A small postulated flaw depth evaluated in the WCAP-13045 and PWROG-17033 reports is based on an initial depth of 0.3 inches. Other initial flaw depths were also considered for sensitivity analyses, as shown in Tables 12-1 and 12-3 of WCAP-13045, to demonstrate that the growth due to FCG is small.

The PWROG stated that the particular flaw depth of 0.3 inches was the maximum acceptable flaw size in the acceptance standards in Table IWB-3518-2 (for pressure retaining welds in pump casings) in the editions up to the 2007 Edition of the ASME Code, Section XI. The flaw depth of 0.3 inches is still the maximum acceptable flaw size in the acceptance standards of Table IWB-3519.2-2 (for pump casings) in later editions of the ASME Code, Section XI. The PWROG used additional postulated flaw sizes that are equal to and exceed the maximum acceptable flaw size of 0.3 inches in sensitivity studies to demonstrate that the flaws do not grow significantly over time.

The NRC staff finds that the postulated flaw with a depth of 0.3 inches is acceptable to be used in the FCG analysis because it is the maximum acceptable flaw size in accordance with Table IWB-3519.2-2 of the ASME Code, Section XI.

The PWROG indicated that if all other inputs for an FCG analysis (SIFs, stress, transient cycles, and transient definitions) are bounded by or are similar to the inputs of WCAP-13045, then the fatigue crack growths for 80 years are expected to be similar to Table 12-1 and Table 12-3 of WCAP-13045 for 40 years. The PWROG reported that even if the transient cycles for 40 years are doubled for 80 years to account for any large differences in the transient cycles, the final flaw size would continue to be less than the minimum stability flaw size of 1/4T flaw depth provided in Table 11-6 of WCAP-13045, at the location of the highest stressed region. The PWROG-17033 report concluded that the FCG analysis provided in Section 12 of WCAP-13045 continues to remain valid for 80 years.

In RAI-2(a), the NRC staff asked whether the input parameters used in the FCG analysis are bounding. In its response, PWR stated that one of the inputs in a FCG analysis is transient stress ranges. As discussed in the plant-specific evaluations, the generic transients considered in the FCG evaluation of WCAP-13045 were reviewed against the severity and frequency of the plant-specific operating transients. The PWROG also stated that the typical design transients and cycles used in WCAP-13045 are generally applicable to the plant-specific transients and

cycles. The PWROG further stated that [

]. The PWROG noted that uprate conditions will not impact the loads and transients as previously determined by plant-specific evaluations.

As stated above, the PWROG doubled the number of transient cycles used in WCAP-13045 to account for the increase in plant operation to 80 years, thereby demonstrating that the FCG analyses performed in WCAP-13045 and in the plant-specific evaluations remain valid. Plants applying for 80-year operation typically have retained the same number of design cycles as their 60 year design cycles. The PWROG stated that as a result, the fatigue crack growth assessment in the PWROG-17033 report will be applicable for plants submitting their 80-year subsequent license renewal application

4.3.1 *Fatigue Crack Growth Analysis Summary*

The NRC staff determines that for the FCG analysis:

(1) the FCG rates used in WCAP-13045 are comparable to the FCG rate for stainless steel in water environment as shown in ASME Section XI,

(2) the stresses used are appropriate,

(3) the transient cycles, in general, used in the WCAP-13045 bound the predicted 80 year transient cycles,

(4) when the transient cycles were doubled, the final flaw size for the FCG analysis would still have sufficient margin with respect to the stability flaw size of 1/4T flaw depth.

(5) to confirm whether the transient cycles used in WCAP-13045 are truly bounding for the plant-specific transient cycles, the NRC staff has imposed a condition for the use of the PWROG-17033 report in the subsequent license renewal application regarding the plant-specific FCG analysis input. The NRC staff's-imposed condition is discussed further in the SE, and

(6) the FCG analysis for RCP pump casings performed in the WCAP-13045 and PWROG-17033 reports is acceptable for the 80 years of operation.

4.4 Inspection Requirements

The NRC staff notes that inspection of the RCP casing is specified in the ASME Code, Section XI, Table IWB-2500-1. The RCP casing is under Examination Category B-L-2, Item Number B12.20 as shown in the latest NRC-approved 2013 Edition of the ASME Code. Item Number B12.20 requires a VT-3 visual examination of internal surfaces of the RCP casing once every 10-year inservice interval.

In addition, Note 2 to Table 2500-1, Examination Category B-L-2, requires that VT-3 visual examinations of the internal surfaces of the RCP casings be performed during refueling outages when the pumps are disassembled. These examinations are performed on accessible portions of the internal surfaces of the pump casings.

4.5 Compensatory Measures

In RAI-3(b), the NRC staff asked the PWROG to discuss measures that are in place to alert the operators to take corrective actions should leakage occur at the RCP casing. In its response, the PWROG stated that based on typical service conditions, flaws are not expected to occur during the life of the plant. If small flaws are present, they will grow only a small amount throughout the service life. If a flaw should happen to penetrate the wall, it will leak at a very detectable rate and will not be unstable, thus allowing a shutdown of the plant per the requirements in plant Technical Specifications (TSs). The PWROG stated that plants rigorously follow RG 1.45, which provides guidance on monitoring and responding to reactor coolant system leakage.

The NRC staff noted that the aging management program XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," of GALL-SLR report, NURGE-2191, Volume 1, is required to manage the aging effect of thermal embrittlement for the RCP casings.

The PWROG stated that the RCP casings in Westinghouse-type PWR designs have an operating history that demonstrates the inherent flaw tolerance and structural stability of the pump casings. Based on operating experience, there have been no detectable service induced flaws nor discernible degradation of the CASS pump casings and welds in the PWR operating history.

The NRC staff finds that based on operating experience, the Westinghouse-designed RCP casings at nuclear plants have not experienced active degradation mechanisms. Should degradation occur in the future, each plant has various compensatory measures to monitor structural integrity of the RCP casing. In addition, the plant TSs require operators to take corrective actions within specific allowable time should leakage occur at the RCP.

4.6 Applicability of PWROG-17033, Revision 1

The PWROG stated that from 1991 until around 2008, in order to supplement the generic WCAP-13045, several plant-specific evaluations (more than half of the U.S. PWR operating fleet and several foreign plants) were performed for the casings of the Westinghouse-designed pumps, either for the original design life of the plant or for 60-year life extensions. Those plant-specific evaluations were based on the pump configurations analyzed in WCAP-13045 (i.e., each of the plant-specific pump models have design drawings that are consistent with the pumps analyzed in WCAP-13045) and demonstrated structural integrity of the casings at the critical locations.

In RAI-5(a), the NRC staff asked about the applicability of the PWROG-17033 report. In its response, the PWROG stated that the scope of the WCAP-13045 and PWROG-17033 reports is only applicable to Westinghouse-designed RCPs with Model 63, 70, 93, 93A, 93A-1, 93D, 100A, and 100D casings fabricated from SA-351 CF8 or CF8M material. All operating U.S. PWRs with Westinghouse NSSS have pumps designed by Westinghouse. Two of the operating U.S. PWRs with Babcock and Wilcox (B&W) NSSS also have Westinghouse designed pumps. However, none of the operating U.S. PWRs with Combustion Engineering NSSS have

Westinghouse-designed pumps. Therefore, the provisions of the WCAP-13045 and PWROG-17033 reports are applicable to all operating U.S. PWRs with Westinghouse NSSS and two operating U.S. PWRs with B&W NSSS.

In RAI-2(a), the NRC staff asked the PWROG to provide guidance to applicants/licensees on how to demonstrate that plant-specific data are bounded by the parameters used in the analyses in WCAP-13045. In its response, the PWROG provided guidance on the use of the PWROG-17033 report to satisfy the requirements of 10 CFR 54.21(c)(1). The PWROG's guidance is incorporated as part of NRC imposed conditions as discussed below.

4.7 Summary

Based on the review and conditions above, the NRC staff concludes that: (1) the crack stability and FCG analyses of the RCP casings in WCAP-13045 and PWROG-17033, Revision 1, have used, in most cases, bounding and limiting parameters, (2) PWROG-17033, Revision 1, has demonstrated pursuant to 10 CFR 54.21(c)(1)(ii) that the RCP casing structural integrity analyses have been projected to the end of 80 years, (3) the structural integrity analyses in WCAP-13045 and PWROG-17033, Revision 1, have satisfied the TLAA acceptance criteria in SRP-SLR Section 4.7 because the fully-aged, saturated fracture toughness for the RCP casings and the FCG have been projected to the end of 80 years, and (4) the crack stability analysis and FCG analysis in WCAP-13045 and PWROG-17033, Revision 1, have demonstrated that structural integrity of the RCP casings will be maintained to the end of 80 years.

5.0 <u>LIMITATIONS AND CONDITIONS</u>

If a licensee uses PWROG-17033, Revision 1, to satisfy the requirements of 10 CFR 54.21(c)(1), the following conditions must be satisfied:

1. The licensee must confirm that its RCPs are Westinghouse-designed models.

2. The licensee must confirm that the Westinghouse-designed RCP is either a Model 63, Model 70, Model 93, Model 93A, Model 93A-1, Model 93D, Model 100A, or Model 100D, and fabricated with SA-351 CF8 or CF8M material.

3. For the crack stability analysis, the licensee must confirm that the screening loadings (forces, moments, J_{app} and T_{app}) used in WCAP-13045 bound the plant-specific loadings. The licensee must confirm the limiting material fracture toughness values (J_{Ic}, T_{mat}, and J_{max}) used in WCAP-13045 and PWROG-17033, Revision 1, bound the plant-specific fracture toughness values. If the screen loadings and material fracture toughness values in the WCAP-13045 and PWROG-17033 reports bound plant-specific values, the licensee needs to discuss how the TRs are bounding in the subsequent license renewal application. If the screening loadings or material fracture toughness values in the WCAP-13045 and PWROG-17033 reports do not bound plant-specific values, the licensee needs to submit a plant-specific crack stability analysis to demonstrate structural integrity of the RCP casing as part of the subsequent license renewal application.

4. For the FCG analysis, the licensee must confirm that the transient cycles specified in the WCAP-13045 or PWROG-17033 report bound the plant-specific transient cycles for the 80 years of operation. The licensee must confirm that the loadings used in the FCG analysis in WCAP-13045 bound the plant-specific applied loadings, considering potential increase in applied loading caused by plant-specific system operational changes, power uprate or piping

modifications. If the FCG analysis inputs in WCAP-13045 bound the plant-specific conditions, the licensee must discuss how they are bounding in the subsequent license renewal application. If the FCG analysis inputs in WCAP-13045 do not bound the plant-specific conditions, the plant owner must provide a plant-specific analysis to demonstrate the FCG of the postulated flaw is within acceptable criteria as part of the subsequent license renewal application.

5.0 CONCLUSION

On the basis of its review, the NRC staff determines that PWROG-17033, Revision 1, has demonstrated structural integrity of the Westinghouse-designed RCP casings for the subsequent period of extended operation (80 years) based on the crack stability analysis and FCG analysis. Therefore, the NRC staff concludes that PWROG-17033, Revision 1, is acceptable for generic use to address the time-limited aging analysis of the RCP casing integrity to satisfy the requirements of 10 CFR 54.21(c)(1).

The NRC staff further concludes that an applicant that utilizes PWROG-17033, Revision 1, in its subsequent license renewal application needs to follow the conditions that the NRC staff imposed as specified in this SE.

For the regulatory efficiency, the NRC staff suggests that the PWROG incorporate, in their entirety, the PWROG's RAI responses dated March 19, 2019, and the NRC's SE into PWROG-17033, Revision 1, as the final version, which would be considered as the NRC-approved document for generic use. The NRC staff suggests that applicants use the NRC-approved version of PWROG-17033, Revision 1, as part of its technical basis to address the RCP casing examination issue in their subsequent license renewal application.

Attachment: Comment Resolution Table

Principal Contributor: John Tsao, NRR/DMLR/MPHB

Date: September 26, 2019

ACKNOWLEDGEMENTS

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

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PWR Owners Group
United States Member Participation* for PA-MSC-1498, Revision 0 [4]

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 Units 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, 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	
Luminant Power	Comanche Peak 1 & 2 (W)		X

PWR Owners Group
United States Member Participation* for PA-MSC-1498, Revision 0 [4]

Utility Member	Plant Site(s)	Participant Yes	Participant No
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	

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PWR Owners Group
International Member Participation* for PA-MSC-1498, Revision 0

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
Elektriciteits Produktiemaatschappij Zuid-Nederland	Borssele 1 (Siemens)		X
Eletronuclear-Eletrobras	Angra 1 (W)		X
Emirates Nuclear Energy Corporation	Barakah 1 & 2		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

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1 PURPOSE

The purpose of this Topical Report (TR) is to extend the fracture mechanics integrity evaluation in WCAP-13045 [1], "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems," through Subsequent License Renewal (SLR), 80 years of operation. In the past, plants have used the generic fracture mechanics evaluation performed in WCAP-13045 to comply with the requirements of ASME Code Case N-481 for plant-specific license renewal applications to extend plant operations to 60 years. In this TR, the fracture mechanics evaluation provides the justification for performing visual inspections, in lieu of volumetric inspections, for reactor coolant pump (RCP) casings as incorporated in the ASME Code Section XI, and extend the applicability of WCAP-13045 to 80 years of operation for all plants with Westinghouse pump casings.

ASME Section XI Table IWB-2500-1, Examination Categories [2] required performing periodic volumetric inspections of the welds of the primary loop pump casings in nuclear power plants. Since these inspections result in radiation exposure to the personnel performing the inspections and require significant resources to perform the inspections, the ASME Code Committee approved Code Case N-481 [3] in March 1990 (see Appendix A of this report) that allows an alternative to the volumetric inspection requirement. The NRC endorsed Code Case N-481 in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability ASME Code Section XI Division 1," in April 1992.

Westinghouse design primary loop pump casings are heavy-wall cast stainless steel. A volumetric inspection of the full thickness of the welds using the ultrasonic test method from the outside diameter surface is impractical due to the severe attenuation associated with the large grain structures. A volumetric inspection of the full thickness of the welds would require unconventional approaches (inside diameter and outside diameter ultrasonic testing or radiographic testing) that require access to the internal side of the pump casing.

ASME Code Case N-481 [3] allowed replacing the volumetric examination of primary loop pump casings with a fracture mechanics-based integrity evaluation supplemented by specific visual inspections. WCAP-13045 [1] contains the integrity evaluation that was performed to demonstrate compliance with ASME Code Case N-481 for 40 years of operation.

Since WCAP-13045 [1] was issued in September 1991, the ASME Code tables have been updated over time to be consistent with the guidance in Code Case N-481 to require visual inspections of the primary loop pump casings. In March 2004, Code Case N-481 was annulled by ASME, and the information in Code Case N-481 was implemented into the 2008 Addenda of ASME Code Section XI. Note that the ASME Section XI 2000 Addenda replaced the pump casing weld B-L-1 volumetric examinations with visual examinations, while the ASME Section XI 2008 Addenda eliminated the pump casing weld (B-L-1) examinations completely. The only required examination is a visual examination of the pump casing (B-L-2) when the pump is disassembled for maintenance or repair.

The technical basis for WCAP-13045 was based on experience with evaluations performed for an assumed 40 year life. Due to the SLR program to extend an operating license to 80 years, the

integrity evaluations in WCAP-13045 were reviewed and confirmed to be applicable for 80 years of service. The fracture mechanics integrity evaluation in this report, as well as the requirements in Code Case N-481 (now incorporated into the ASME Code Section XI) were reviewed to confirm that the visual inspections for pump casings continue to remain valid for an 80 year life.

WCAP-13045 [1] contains the fracture mechanics based integrity evaluation for cast austenitic stainless steel pump casings as required by Code Case N-481 [3]. The evaluations contained in WCAP-13045 are applicable to Westinghouse design RCPs. There are eight different models of pumps: Models 63, 70, 93, 93A, 93A-1, 93D, 100A, and 100D. Models 63, 70, 93 and 93D, which have a tangent outlet nozzle. Models 93A and 93A-1 have outlet nozzles that are radially orientated. Models 100A and 100D are similar to the general design of the Model 93, except for a radially oriented outlet nozzle that is consistent with the Model 93A. Models 93, 93A and 93A-1 are the most commonly used RCPs in Westinghouse Nuclear Steam Supply System (NSSS) plants.

WCAP-13045 contains a model that is representative of each of the outlet nozzle configurations that was used for a 3D finite element stress analysis and fracture evaluation (the inlet nozzles are reasonably axisymmetric with the pump casing proper). The representative models chosen were the Model 93A (radial outlet nozzle) and Model 93 (tangential outlet nozzle). The material of the pump casings is fabricated from SA-351 CF8, except for the pumps of three plants which were fabricated from SA-351 CF8M. The SA-351 CF8 and CF8M are known to be susceptible to thermal aging.

WCAP-13045 addressed thermal aging of the cast austenitic stainless steel pump casings (CF8 and CF8M) by using end of life (40 years) fracture toughness values for all Westinghouse design pump casings. The fracture toughness criteria were established using the lowest toughness for each pump component. This report justifies the continued use of the end of life fracture toughness values determined in WCAP-13045 for 80 years of service.

The fracture toughness is used in WCAP-13045 as part of the elastic plastic fracture mechanics (EPFM) analysis based on the J-integral approach; therefore, it is necessary to confirm the fracture toughness values for an 80 year evaluation, and demonstrate that the EPFM analysis continues to remain valid for 80 years. The J-integral evaluation also used bounding loads that covered a wide range of pump casing nozzle loads from the various different plants. This report only confirms the toughness properties for 80 years, and not the applied loads, since the applied loadings in the J-integral analysis considered in WCAP-13045 will not be impacted by license extension to 80 years of operation because they are not time dependent. Note that the fracture toughness properties based on the CF8M (high molybdenum content) material is more susceptible (limiting) to thermal aging than the CF8 material. Therefore, the CF8M material is used in the fracture mechanics evaluation in this TR; and the conclusions for the CF8M fracture toughness determined in this report also apply to the CF8 material.

Fatigue crack growth evaluations were also determined in WCAP-13045 for the high stress outlet nozzle crotch regions. Various crack sizes were considered in the evaluations, and based on the evaluations, it was demonstrated that the fatigue crack growth was small. The discussion in this

report also justifies that the fatigue crack growth evaluations contained in WCAP-13045 are still applicable for the 80 year design life.

The following two items were reviewed to confirm the applicability of WCAP-13045 [1] for 80 years of service:

1. Confirm that the fracture toughness (J_{Ic} – J at crack initiation and J_{max} – J at maximum crack extension) used in WCAP-13045 [1] and the associated tearing modulus (T_{mat}) for the stability analysis are applicable for 80 years of service. The postulated 1/4T flaw size used in the stability analysis is compared to the final flaw size due to fatigue crack growth.

2. Confirm that the generic fatigue crack growth (FCG) analysis performed in WCAP-13045 is applicable for 80 years of service, specifically; the stresses, stress intensity factor (SIF) equations, transient definitions and cycles, and the FCG rate.

The stability calculations in WCAP-13045 were reviewed based on the applicability of the fracture toughness for SLR in Section 3 of this report. The fatigue crack growth analysis in WCAP-13045 was reviewed based on for FCG rates, stresses, and transient definitions and cycles in Section 4. The final conclusions of this report are provided in Section 5, with all cited references provided in Section 6.

Appendix A of this report also provides the ASME Code Section XI Code Case N-481, for reference, which was approved and published in March 1990, and later annulled in March 2004, as the requirements of the code case were incorporated in the 2000 Addenda and 2008 Addenda of the ASME Code Section XI IWB-2500.

This document contains Westinghouse Electric Company LLC proprietary information and data which has been identified by brackets. Coding (a,c,e) associated with the brackets sets forth the basis on which the information is considered proprietary. These code letters are listed with their meanings in BMS-LGL-84 [12].

2 BACKGROUND

Plants have used the generic fracture mechanics evaluation performed in WCAP-13045 to comply with the requirements of ASME Code Case N-481 for the license renewal applications to extend plant operations to 60 years. Before Code Case N-481 was annulled and the requirements of the Code Case were incorporated into ASME Code Section XI, utilities were furthermore required to perform a plant-specific evaluation using the generic fracture mechanics analysis in WCAP-13045 to demonstrate compliance with N-481. Following the incorporation of Code Case N-481 into Section XI, no further fracture mechanics evaluations have been performed as plants were not required to perform volumetric or routine internal visual examinations of RCP casing welds, and the only ASME Code Section XI inspection requirement is to perform external surface examinations of the pump casing welds and visual examinations of the internal surfaces of the pump casing welds when the RCP is disassembled for other reasons (e.g. maintenance or refurbishments).

The thermal aging and fatigue crack growth methodology in WCAP-13045 was reviewed and is provided in this TR to demonstrate that the time-limiting aging analysis aspects of WCAP-13045 continue to remain bounding and acceptable for 80 years of plant life. The NRC has approved several plant license renewal applications for 60 years of operation that utilized the generic fracture mechanics analyses and conclusions in WCAP-13045 as discussed below:

1. The NRC Safety Evaluation Report (Section 4.4.4) for Salem Units 1 and 2 [13] discusses the use of the generic fracture mechanics analysis in WCAP-13045 to meet the requirements of ASME Code Case N-481 for a 60 year license renewal. The NRC staff concluded that the generic analysis in WCAP-13045 is applicable to the Salem design of the RCP casings. The generic analysis, WCAP-13045, bounds the plant-specific analysis, WCAP-16957-P [17], as approved by the staff in the SER [13]. The analysis was shown to remain valid for extended operation.

2. In the 60 year license renewal NRC Safety Evaluation Report for D.C. Cook Units 1 and 2, NUREG-1831, Section 4.7.2 [14], the NRC staff discusses the use of WCAP-13045 to satisfy the requirements of ASME Code Case N-481 based on a plant-specific analysis, WCAP-13128 [18]. In Section 4.7.2.4 of [14], the NRC concludes that the time-limited aging analysis (TLAA) regarding ASME Code Case N-481 that was provided is acceptable.

3. The 60 year license renewal Safety Evaluation Report (Section 4.3.2.10) for Diablo Canyon Units 1 and 2 [15] discusses that WCAP-13045 was used to demonstrate compliance with Code Case N-481 based on a plant-specific evaluation, WCAP-13895 [19]. In Section 4.3.2.10.4 of [15], the NRC concludes that it was shown that the aging of the RCP pump casings will be adequately managed for extended operation.

4. Based on the Sequoyah 60 license renewal Safety Evaluation Report [16], the NRC concluded that the plant's LRA did not need to include a TLAA related to Code Case N-481 because the licensing basis is per the ASME Code Section XI Edition, which no longer relies on N-481 for ISI (in-service inspection interval) requirements. Thus, it was not

necessary to perform any TLAA analysis for RCP pump casings and N-481, as it does not meet Criterion 4 or 6 in 10 CFR 54.3(a).

In summary, WCAP-13045 has been reviewed by NRC to support 60 year license renewal applications for several plants that used Code Case N-481 as a basis for their ISI examination programs. This report reviews the TLAA aspects of WCAP-13045 to demonstrate its continued applicability for 80 years of plant operation.

2.1 OPERATING EXPERIENCE AND APPLICATION OF LATEST FRACTURE TOUGHNESS DATABASE

The Westinghouse RCP casings have an operating history that demonstrates the inherent flaw tolerance and structural stability of the pump casings. No detectable service-induced flaws nor discernable degradation of the cast austenitic stainless steel (CASS) pump casings and welds in the Westinghouse pump operating history have been identified.

The fracture mechanics evaluation contained in this TR (later discussed in Section 3) considers the latest fracture toughness correlations that have been developed for the CASS pump casings. The end of life fracture toughness properties for the pump casing materials are determined based on NUREG/CR-4513, Revision 2, "Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR systems," by Omesh Chopra, published in May 2016 [11].

Revision 2 of NUREG/CR-4513 provides a large database for CASS material and thermal aging, and builds on the work performed in 1994 for Revision 1 of NUREG/CR-4513. In 1994, the Argonne National Laboratory (ANL) completed an extensive research program to evaluate the extent of thermal aging of cast stainless steel materials [11]. The ANL research program measured mechanical properties of cast stainless steel materials after they had been heated in controlled ovens for long periods of time. ANL compiled a database, both from data within ANL and from international sources, of approximately 85 compositions of cast stainless steel exposed to a temperature range of 290-400°C (550-750°F) for up to 58,000 hours (6.5 years). In 2015, the work done by ANL was augmented, and the fracture toughness database for CASS materials was aged to 100,000 hours at 290-350°C (554-633°F). The methodology for estimating fracture properties has been extended to cover CASS materials with a ferrite content of up to 40%. From this database in NUREG/CR-4513, Revision 2, ANL developed lower bound correlations for estimating the extent of thermal aging of cast stainless steel [11].

ANL developed the fracture toughness estimation procedures by correlating data in the database conservatively. After developing the correlations, ANL validated the estimation procedures by comparing the estimated fracture toughness with the measured value for several cast stainless steel plant components that were removed from service. The procedure developed by ANL in [11] was used for the end of life fracture toughness values contained in this TR. The ANL research program was sponsored and the procedure was accepted by the NRC.

3 STABILITY ANALYSIS AND FRACTURE TOUGHNESS

3.1 FRACTURE TOUGHNESS DETERMINED IN WCAP-13045

The fracture mechanics integrity evaluation is based on the Elastic Plastic Fracture Mechanics (EPFM) methodology as discussed in Sections 10 and 11 of WCAP-13045 [1]. The EPFM is determined for a postulated 1/4T (1/4 thickness) flaw size with a six-to-one (6:1) aspect ratio. This particular flaw size is consistent with the guidelines of Code Case N-481 Part (d)(5). The location of the postulated 1/4T flaws are either at highest stressed region, regions of significant stress concentrations, or locations in welds not affected by discontinuities such as nozzles. Additional discussion on flaw postulation is contained in Section 9 of WCAP-13045.

The criterion for establishing stability is based on the fracture toughness of the pump casings, as well as the tearing modulus, T, as discussed in Section 10.4 of WCAP-13045 and shown below:

A crack is stable if either

1) $J_{applied} < J_{Ic}$ or if
2) $J_{applied} > J_{Ic}$ then $T_{applied} < T_{material}$ and $J_{applied} \leq J_{maximum}$

The applied toughness ($J_{applied}$ or J_{app}) and applied tearing modulus ($T_{applied}$ or T_{app}) are calculated with the EPRI handbook methodology [10], based on various combinations of loading parameters and material properties for the various pump designs as discussed in WCAP-13045. The J_{app} and T_{app} are not impacted by a design life to 80 years as they are not time dependent; however, the fracture toughness material parameters such as the crack initiation toughness (J_{Ic}), which is based on a J-integral resistance curve, the maximum fracture toughness ($J_{maximum}$ or J_{max}), and the tearing modulus ($T_{material}$ or T_{mat}) need be reviewed to confirm that these parameters are not impacted by 80 years of operation.

The end of service (40 years) fracture toughness (J_{Ic}, T_{mat}, and J_{max}) of the pump casings are calculated in Section 5 of WCAP-13045 [1]. The lower bound toughness criteria selected from among all the pump casing (and welds) and models (Models 63, 70, 93, 93A, 93A-1, 93D, 100A, and 100D) are given in Table 5-1 of WCAP-13045. The minimum fracture toughness values based on the most limiting SA-351 CF8M component from Table 5-1 of WCAP-13045 which were considered in this TR are shown in Table 1 below. The fracture toughness properties for CF8M in WCAP-13045 bound the CF8 material, because the thermal aging is more limiting for the CF8M material than the CF8 material.

Table 1: End-of-Service Fracture Toughness from WCAP-13045 (Table 5-1) [1]

Material	J_{Ic} (in-lb/in^2)	T_{mat} (dimensionless)	J_{max} (in-lb/in^2)
Model 93 SA-351 CF8M	[ ]^{a,c,e}	[ ]^{a,c,e}	[ ]^{a,c,e}
<u>Chemistry</u>: [ ]^{a,c,e}			

Note: The values J_{Ic}, T_{mat}, and J_{max} shown for the CF8M material above are the same values in Table 5-1 and on page A-30 of WCAP-13045.

The fracture toughness data in Table 1 (from Table 5-1 of WCAP-13045) is based on the chemistry data from Appendix A (page A-30) of WCAP-13045. Based on the chemistry of the limiting heat of cast austenitic stainless steel (i.e., Silicon (Si), Chromium (Cr), Molybdenum (Mo), Nickel (Ni), Carbon (C), Manganese (Mn), and Nitrogen (N) in percent weight, and percent delta ferrite), the fracture toughness is calculated in accordance with WCAP-10931 [5], Slama [6], and WCAP-10456 [7].

Based on the results in Slama [6] and WCAP-10456 [7], the minimum (saturated) fracture toughness properties were obtained after [ ]^{a,c,e} Therefore, the fracture toughness properties shown in Table 1 (per Table 5-1 of WCAP-13045) are at the full-aged saturated condition material values, and are therefore applicable to 80 years of service life, because the resulting minimum (saturated) properties are reached by [ ]^{a,c,e}

Therefore, the EPFM stability analysis and conclusions in WCAP-13045 [1], based on the saturated fracture toughness values in Table 1, are applicable for 80 years of operation.

The minimum fracture toughness values based on Table 5-1 of WCAP-13045 (as shown in Table 1) were compared with the latest fracture toughness correlations for thermal aging of cast austenitic stainless steel CF8M material per NUREG/CR-4513, Revision 2 [11]; see Section 3.2 below. This comparison confirms that the fracture toughness values in WCAP-13045 [1] are bounding and applicable to the 80 year design life.

3.2 FRACTURE TOUGHNESS BASED ON NUREG/CR-4513, REVISION 2

In this section of the TR, a calculation is performed to determine the J_{Ic}, T_{mat} and J_{max} values based on latest industry guidelines and fracture toughness correlations for thermal aging of cast austenitic stainless steel from NUREG/CR-4513 Revision 2 [11] for the most limiting CF8M pump casing material, using the limiting heat specific chemistry values provided in Table 1 (originally from WCAP-13045). The fracture toughness values calculated in accordance with NUREG/CR-4513 will be compared to the values determined in Table 1, from WCAP-13045. This comparison of fracture toughness values will demonstrate if the toughness properties from WCAP-13045 remain bounding and acceptable for 80 years. Note that the impact of thermal aging on the CF8 fracture toughness properties is less than that for the CF8M material; therefore the evaluation herein only considered the CF8M material because it bounds the fracture toughness values for the CF8 material.

The following equations are contained in NUREG/CR-4513, Revision 2 [11] and are applicable to the CF8M material. The calculated fracture toughness values based on [11] are shown in Table 2.

Cr_{eq} = Cr + 1.21(Mo) + 0.48(Si) - 4.99 = (Chromium equivalent)

Ni_{eq} = (Ni) + 0.11(Mn) - 0.0086(Mn)^2 + 18.4(N) + 24.5(C) + 2.77 = (Nickel equivalent)

δ_c =100.3(Cr_{eq} / Ni_{eq})^2-170.72(Cr_{eq} / Ni_{eq})+74.22 = (Ferrite Content)

The elements are in percent weight and δ_c is ferrite in percent volume.

The saturation room temperature (RT) impact energies for the cast stainless steel materials are determined from the chemical composition.

For CF8M steel with < 10% Ni, the saturation value of RT impact energy Cv_{sat} (J/cm^2) is the lower value determined from:

$\log_{10}Cv_{sat} = 0.27 + 2.81 \exp(-0.022\phi)$

where the material parameter ϕ is expressed as:

$\phi = \delta_c (Ni + Si + Mn)^2(C + 0.4N)/5.0$

and using:

$\log_{10}Cv_{sat} = 7.28 - 0.011\delta_c - 0.185Cr - 0.369Mo - 0.451S - 0.007Ni - 4.71(C + 0.4N)$

For CF8M steel with $\geq$ 10% Ni, the saturation value of RT impact energy Cv_{sat} (J/cm^2) is the lower value determined from:

$\log_{10}Cv_{sat} = 0.84 + 2.54 \exp(-0.047\phi)$

where the material parameter ϕ is expressed as:

$\phi = \delta_c (Ni + Si + Mn)^2(C + 0.4N)/5.0$

and using:

$\log_{10}Cv_{sat} = 7.28 - 0.011\delta_c - 0.185Cr - 0.369Mo - 0.451Si - 0.007Ni - 4.71(C + 0.4N)$

The saturation J-R curve at RT, for static-cast CF8M steel is given by:

$J_d = 1.44\ (Cv_{sat})^{1.35}(\Delta a)^n$ for $Cv_{sat} < 35$ J/cm^2

$J_d = 16\ (Cv_{sat})^{0.67}(\Delta a)^n$ for $Cv_{sat} \geq 35$ J/cm^2

$n = 0.20 + 0.08 \log_{10} (Cv_{sat})$

where J_d is the "deformation J" in kJ/m^2 and Δa is the crack extension in mm.

The saturation J-R curve at 290-320°C (554-608°F), for static-cast CF8M steel is given by:

$J_d = 5.5\ (Cv_{sat})^{0.98}(\Delta a)^n$ for $Cv_{sat} < 46$ J/cm^2

$J_d = 49\ (Cv_{sat})^{0.41}(\Delta a)^n$ for $Cv_{sat} \geq 46$ J/cm^2

$n = 0.19 + 0.07 \log_{10} (Cv_{sat})$

where J_d is the "deformation J" in kJ/m^2 and Δa is the crack extension in mm.

[

]^{a,c,e} The tearing modulus, $T_{material}$ is calculated by T = dJ/da * E/σ_f^2, where dJ/da is the slope of the J-R curve, E is elastic modulus, and σ_f is the flow strength (average of the yield strength and ultimate strength). Applying the NUREG/CR-4513, Revision 2 [11] correlations, the fracture toughness properties are given in Table 2. NUREG/CR-4513, Revision 2 discusses that the fracture toughness correlations used for the full aged condition are applicable to plants that have been operating for greater than or equal to 15 Effective Full Power Years (EFPY) for the CF8M material. The Westinghouse NSSS plants have been operating for greater than 15 EFPY; therefore, the use of the fracture toughness correlations discussed above is applicable to the fully aged or saturated conditions.

Therefore, the fracture toughness values based on the original methodology in WCAP-13045 are limiting (see Table 1) as compared to the fracture toughness values based on NUREG/CR-4513 (see Table 2). By calculating the latest industry correlation for the fracture toughness values for aged cast austenitic stainless steel from NUREG/CR-4513, it can be concluded that the aged toughness values in Table 1 (from Table 5-1 of WCAP-13045) are bounding and limiting for 80 years of operation.

Table 2: NUREG/CR-4513 [11] Fully-Aged (Saturated) Fracture Toughness

Material	J_{Ic} (in-lb/in^2)	T_{mat} (dimensionless)	J_{max} (in-lb/in^2)
Model 93 SA-351 CF8M	[ ]^{a,c,e}	[ ]^{a,c,e}	[ ]^{a,c,e}
Chemistry (from Table 1): [ ]^{a,c,e} ASME Code properties are used for E (modulus of elasticity) = 25.6 x 10^6 psi and Yield Strength = 19,350 psi. Ultimate strength = 67,000 psi per Table 6-3 of WCAP-13045, which is a close approximation to the ASME Code values. The values are for the cold leg temperature, approximately 550°F.			

4 FATIGUE CRACK GROWTH ANALYSIS

Two fatigue crack growth analyses were originally performed in Section 12 of WCAP-13045 [1], one for a postulated crack in the highest stressed outlet nozzle knuckle in a Model 93A pump casing and the other in the highest stressed outlet nozzle knuckle in a Model 93 pump casing. The FCG analysis for the Model 93A pump was performed prior to the publication of the stainless steel FCG rate in the ASME Code Section XI; therefore, the FCG rate in [Equation 1] from [9] was initially considered in WCAP-13045. However, the FCG analysis for a Model 93A RCP was calculated based on [Equation 2], which was an updated version of [Equation 1] for stainless steel in a water environment. [Equation 1] in WCAP-13045 was provided for information purposes and is as follows:

$$\frac{da}{dN} = 5.4 \times 10^{-12} (K_{eff})^{4.48} \quad \text{(inches/cycle)} \qquad \text{[Equation 1]}$$

$$K_{eff} = K_{max}(1-R)^{0.5}$$

$$R = K_{min}/K_{max} \qquad K_{max} \text{ and } K_{min} \text{ is in units of ksi } \sqrt{in}$$

The FCG rate used for the Model 93 postulated flaw in [1] was based on Figure C-3210-1 of Article C-3000 of Section XI of the ASME Code (1989 Edition) and is shown in Equation 2 below:

$$\frac{da}{dN} = C\ F\ S\ E\ (\Delta K)^{3.30} \text{ (inches/cycle)} \qquad \text{[Equation 2]}$$

Where: $C = 2.42 \times 10^{-20}$, F = 1 for temperatures below 800°F, S = the R ratio correction (see the definition of 'S' in Equation 3 below), R = the ratio of the minimum stress intensity factor (SIF) to the maximum SIF, ΔK = the range of stress intensity factors (psi $\sqrt{in}$), and E = 2 based on stainless steel in water [8].

The current NRC approved 2013 Edition of the ASME Code Section XI Appendix C fatigue crack growth for stainless steel is shown below. A factor of 2 is applied to the da/dN rate to account for the environmental effects of a postulated flaw in water as discussed in [8]:

$$\frac{da}{dN} = 2{*}C_o\ S\ (\Delta K_I)^{3.3} \text{ (inches/cycle)} \qquad \text{[Equation 3]}$$

where $C_o = 10$ ^ $[-10.009 + 8.12 \times 10^{-4}\, T - 1.13 \times 10^{-6}\, T^2 + 1.02 \times 10^{-9}\, T^3]$

T = Temperature (°F) = 550°F (see Table 11-1 and Table 11-6 of [1])

S = 1.0 for $R \leq 0$, S = 1 + 1.8R when $0 < R < 0.79$, and S = -43.35 + 57.97R when $0.79 < R < 1$

ΔK_I is in units of ksi $\sqrt{in}$, $R = K_{min}/K_{max}$

There are no significant differences between the current stainless steel FCG rate in water, [Equation 3] and the FCG rate used in WCAP-13045 [1], and [Equation 2] for the Model 93 postulated flaws. The difference between the current stainless steel FCG rate in water [Equation

3] and the FCG rate considered in WCAP-13045, [Equation 1] for the Model 93A postulated flaw is also insignificant. Therefore, the existing FCG rates in Section 12 of WCAP-13045 are acceptable based on current industry standards for fatigue crack growth for stainless steel material in a water environment.

Other inputs required for an FCG analysis are the stress intensity factors, stresses, transient cycles and transient definitions. The stress intensity factor (SIF) correlations used for the FCG analysis in WCAP-13045 are consistent with the current correlations provided in 2013 Edition of ASME Code Section XI Appendix A. The transient stresses used in the FCG analysis are generic and encompass the various pump models. These stresses are not impacted for 80 years of operation, as plant operations for this extended period will not vary greatly from current operations. The number of predicted cycles for 80 years of service for each applicable transient is assumed to be bounded by the number of transient cycles considered in the 40 year life of the plant and shown in Table 12-2 of WCAP-13045. However, to ensure conservatism in the results presented in this TR, the FCG cycles for 40 years were doubled for 80 years to account for any large differences in the transient cycles. It was concluded that the final flaw size would still be less than the stability flaw size for this region as discussed below.

The calculated FCG for four flaw sizes in the outlet nozzle knuckle region of a Model 93A pump casing is given in Table 12-1 of WCAP-13045. The calculated FCG for three flaw sizes in the outlet nozzle knuckle region of a Model 93 pump casing is given in Table 12-3 of WCAP-13045. One of the flaw size cases for FCG was for an initial flaw depth of 0.3"; this particular flaw depth was the maximum acceptable flaw size in the Acceptance Standards in Table IWB-3518-2 (for pressure retaining welds in pump casings) up to the 2007 Edition of the ASME Code Section XI. The flaw depth of 0.3" is also the maximum acceptable flaw size in the Acceptance Standards Table IWB-3519.2-2 (for pump casings) in later editions of the ASME Code Section XI. Therefore, the flaw depth of 0.3" was an important flaw size case to consider in WCAP-13045 for the flaw tolerance evaluation, and any actual as-found flaws larger than this depth would need to be evaluated based on fracture mechanics. The other FCG postulated flaw size cases considered in WCAP-13045 were provided as sensitivity studies to demonstrate that the flaws do not grow significantly over time.

Based on the fatigue crack growth analyses in Tables 12-1 and 12-3 of WCAP-13045, the flaw depth of 0.3" grows to a maximum value of [ ]^{a,c,e} over 40 years of service. All other inputs for the FCG analysis (stress intensity factors, stress, transient cycles and transient definitions) are bounded by, or are similar to, the inputs of WCAP-13045, then the fatigue crack growths for 80 years are similar to or less than the crack growth in Table 12-1 and Table 12-3 of WCAP-13045 for 40 years. Additionally, if the number of transient cycles for 40 years are doubled for 80 years to account for any large differences in the transient cycles, then the final flaw size would continue to be less than the minimum stability flaw size of 1/4T flaw depth ([ ]^{a,c,e} in Table 11-6 of WCAP-13045), which is associated with the location of the highest stressed region. Therefore, the FCG analysis provided in Section 12 of WCAP-13045 remains valid for 80 years of operation.

5 CONCLUSIONS

The objective of this report is to review the RCP casing fracture mechanics integrity evaluations and the fatigue crack growth analysis contained in WCAP-13045 [1], and to confirm that they remain valid for 80 years of service

Section 3.1 discussed the most limiting pump casing CF8M fracture toughness and the tearing modulus used in the WCAP-13045 J-integral stability evaluations (EPFM). The WCAP-13045 fracture toughness determinations are applicable to 80 years of service because the fracture toughness parameters are at full-aged saturated conditions, therefore, any additional aging past 40 years does not have an impact on the fracture toughness parameters. Section 3.2 compared the minimum fracture toughness in WCAP-13045 with the latest industry toughness correlations in NUREG/CR-4513, Revision 2 [11]. Based on the conclusions in Section 3.2, it is demonstrated that the fracture toughness values in WCAP-13045 are less than (more conservative and limiting) than the fracture toughness values in NUREG/CR-4513. Therefore, as compared with the current industry standards, the fracture toughness values in WCAP-13045 are limiting and applicable to 80 years of operation. Thus, the EPFM analysis for the pump casings contained in WCAP-13045 is valid for 80 years because the minimum facture toughness used in the stability analysis is applicable to 80 years of service life.

A qualitative evaluation for the fatigue crack growth evaluation was performed in Section 4 of this report for the pump casings. It was determined that the current FCG rate for stainless steel in a water environment based on the 2013 Edition of the ASME Code Section XI, when compared to the rates used in WCAP-13045, are comparable and there will be an insignificant impact on the crack growth analysis. Furthermore, the stresses used in the FCG analysis are generic and envelop the various pump designs. Additionally, these stresses do not change for 80 years of operation, as they are not time dependent. The stress intensity factors used in the FCG analysis are consistent with the current industry standards for similar FCG evaluations. The transient definitions in the FCG analysis are also not expected to change over the design life, and the cycles used in the WCAP-13045 are assumed to bound the predicted 80 year transient cycles. Finally, there is sufficient margin between the final crack growth and the flaw size used for stability; therefore, even if the 40 year transient cycles are doubled for 80 years of operation, the final flaw size after FCG will be less than the stability flaw size, 1/4T flaw depth, for the stability analysis in WCAP-13045.

In conclusion, the fracture mechanics integrity evaluation in WCAP-13045 [1] for pump casings is applicable to 80 years of design life. The fracture mechanics evaluation for SLR in this TR justifies continuing to perform visual inspections, in lieu of volumetric inspections, for pump casings as incorporated in ASME Code Section XI.

6 REFERENCES

1) Westinghouse Document, WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply System," September 1991.
2) ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."
3) Code Case N-481, "Alternative Examination Requirements for Cast Austenitic Pump Casings," Section XI, Division 1, Approval Date: March 5, 1990.
4) PWROG Project Authorization, PA-MSC-1498, Revision 0, "Update for Subsequent License Renewal: WCAP-13045, 'Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems'."
5) Westinghouse Document, WCAP-10931, Revision 1, "Toughness Criteria for Thermally Aged Cast Stainless Steel," July 1986.
6) Slama, G., Petrequin, P., Masson, S. H. and Mager, T. R., "Effect of Aging on Mechanical Properties of Austenitic Stainless Steel Casting Welds," presented at SMIRT 7 Post Conference Seminar 6 – Assuring Structural Integrity of Steel Reactor Pressure Boundary Components, August 29/30, 1983, Monterey, CA.
7) Westinghouse Document, WCAP-10456, "The Effects of Thermal Aging on Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems," November 1983.
8) "Evaluation of Flaws in Austenitic Steel Piping," Trans. ASME, Journal of Pressure Vessel Technology, Vol. 108, pp. 352-366, 1986.
9) Bamford, W. H., "Fatigue Crack Growth of Stainless Steel Piping in a Pressurized Water Reactor Environment," Trans ASME, Journal of Pressure Vessel Technology, February 1979.
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12) BMS-LGL-84, Revision 0.00, "Protection of Proprietary Information Regarding Submittals to the USNRC including Safety Analysis Reports for Commercial Nuclear Power Plants," Effective Date: April 15, 2017.
13) U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station," Docket Numbers 50-272 and 50-311, March 2011. (NRC ADAMS Accession No. ML110900295)
14) U.S. Nuclear Regulatory Commission, NUREG-1831, "Safety Evaluation Report Related to the License Renewal of the Donald C. Cook Nuclear Plant, Units 1 and 2," Docket Nos. 50-315 and 50-316, July 2005. (NRC ADAMS Accession No. ML052230442)

15) U.S Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Diablo Canyon Nuclear Power Plant, Units 1 and 2," Docket Nos. 50-275 and 50-323, June 2011. (NRC ADAMS Accession No. ML11153A103)

16) U.S. Nuclear Regulatory Commission, NUREG-2181, "Safety Evaluation Report Related to the License Renewal of Sequoyah Nuclear Plant, Units 1 and 2," Docket Nos. 50-327 and 50-328, July 2015. (NRC ADAMS Accession No. ML15187A206)

17) Westinghouse Document, WCAP-16957-P, "A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casing of Salem Generating Station Units 1 and 2 for the License Renewal Program," March 2009.

18) Westinghouse Document, WCAP-13128, "A Demonstration of Compliance of the Primary Loop Pump Casings of the D. C. Cook Units 1 and 2 to ASME Code Case N-481," March 1992.

19) Westinghouse Document, WCAP-13895, "A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of the Diablo Canyon Nuclear Power Plants Units 1 and 2," October 1993.

APPENDIX A : ASME CODE SECTION XI CODE CASE N-481

Approval Date: March 5, 1990

See Numerical Index for expiration and any reaffirmation dates.

Case N-481
E **Alternative Examination Requirements for Cast Austenitic Pump Casings**
Section XI, Division 1

Inquiry: When conducting examination of cast austenitic pump casings in accordance with Section XI, Division 1, what examinations may be performed in lieu of the volumetric examinations specified in Table IWB-2500-1, Examination Category B-L-1, Item B12.10?

Reply: It is the opinion of the Committee that the following requirements shall be met in lieu of performing the volumetric examination specified in Table IWB-2500-1, Examination Category B-L-1, Item B12.10:

E *(a)* Perform a VT-2 visual examination of the exterior of all pumps during the hydrostatic pressure test required by Table IWB-2500-1, Category B-P.

E *(b)* Perform a VT-1 visual examination of the external surfaces of the weld of one pump casing.

(c) Perform a VT-3 visual examination of the internal surfaces whenever a pump is disassembled for maintenance.

(d) Perform an evaluation to demonstrate the safety and serviceability of the pump casing. The evaluation shall include the following:

(1) evaluating material properties, including fracture toughness values;

(2) performing a stress analysis of the pump casing;

(3) reviewing the operating history of the pump;

(4) selecting locations for postulating flaws;

(5) postulating one-quarter thickness reference flaw with a length six times its depth;

(6) establishing the stability of the selected flaw under the governing stress conditions;

(7) considering thermal aging embrittlement and any other processes that may degrade the properties of the pump casing during service.

(e) A report of this evaluation shall be submitted to the regulatory and enforcement authorities having jurisdiction at the plant site for review. E

APPENDIX B : CORRESPONDENCE WITH U.S. NRC REGARDING THE REVIEW OF PWROG-17033-NP, REVISION 1

40TH ANNIVERSARY
PWROG
PWR Owners Group

Program Management Office
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

PWROG-17033-P/NP, Revision 1
Project Number 99902037

March 27, 2019

OG-19-58

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 1-9 Associated with PWROG-17033-P (& NP), Revision 1, "Update for Subsequent License Renewal: WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems", PA-MSC-1498

References:

1. Letter OG-18-142, Transmittal of PWROG-17033-P (& NP), Revision 1, "Update for Subsequent License Renewal: WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems" (PA-MSC-1498), dated June 14, 2018

2. NRC Letter of Acceptance for Review of PWROG-17033-P (& NP), Revision 1, "Update for Subsequent License Renewal: WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems" dated July 9, 2018

3. Email from the NRC (Drake) to the PWROG (Holderbaum), Request for Additional Information, RAIs 1-9, RE: PWROG-17033-P (& NP), Revision 1, "Update for Subsequent License Renewal: WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems" dated October 30, 2018

4. Email from the PWROG (Holderbaum) to the NRC (Drake). DRAFT Response to NRC RAI for the Review of Generic Topical Report No. PWROG-17033-P and NP, Rev. 1, dated March 6, 2019

On June 14, 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-17033-P & NP, Revision 1 for referencing in regulatory actions (Reference 1). The report was accepted for review on

U.S. Nuclear Regulatory Commission March 27, 2019
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July 9, 2018 (Reference 2). The NRC Staff has determined that additional information is needed to complete the review per the email dated October 30, 2018 (Reference 3). Draft responses were provided via email to the NRC on March 6, 2019 (Reference 4).

Enclosures 1 and 2 to this letter provides formal responses to NRC RAIs 1-9 (Reference 3) associated with PWROG-17033-P & NP, Revision 1, "Update for Subsequent License Renewal: WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems".

Also enclosed is the Westinghouse Application for Withholding Proprietary Information from Public Disclosure, CAW-19-4881, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Enclosure 1 contains information proprietary to Westinghouse Electric Company LLC ("Westinghouse"), it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission ("Commission") and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the item listed above or the supporting Westinghouse Affidavit should reference CAW-19-4881 and should be addressed to Camille Zozula, Manager, Infrastructure & Facilities Licensing, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2 Suite 259, Cranberry Township, Pennsylvania 16066.

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

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March 27, 2019
Page 3 of 3

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-1498)
PWROG PMO
PWROG Steering and Management Committee
J Drake, US NRC
P Atkin, DOM
J Andrachek, Westinghouse
T. Zalewski, Westinghouse
L Patterson, Westinghouse
G. Demetri, Westinghouse
A. Udyawar, Westinghouse

Enclosures 3: LTR-SDA-18-127-P/NP, Revision 0, RAIs 1-9 Responses for PWROG-17033-P & NP, Revision 1 (PA-MSC-1498) and CAW-19-4881

Westinghouse Non-Proprietary Class 3

CAW-19-4881
Page 1 of 3

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:
COUNTY OF BUTLER:

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

(2) I am requesting the proprietary portions of LTR-SDA-18-127-P Rev.0 be withheld from public disclosure under 10 CFR 2.390.

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

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

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

(ii) Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

Westinghouse Non-Proprietary Class 3

CAW-19-4881
Page 2 of 3

AFFIDAVIT

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

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

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

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

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

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

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

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

AFFIDAVIT

lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

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

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

Executed on: 21 MARCH 2019

Camille T. Zozula, Manager
Infrastructure & Facilities Licensing

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

Utility Customer Instructions

Include the following information in the TRANSMITTAL LETTER to NRC. This is not part of the affidavit.

Enclosed is:

CAW-19-4881, which includes: the Affidavit, Proprietary Information Notice, Copyright Notice

The enclosure contains information proprietary to Westinghouse Electric Company LLC ("Westinghouse"), it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission ("Commission") and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-19-4881 and should be addressed to Camille T. Zozula, Manager, Infrastructure & Facilities Licensing, Westinghouse Electric Company, 1000 Westinghouse Drive, Suite 165, Cranberry Township, Pennsylvania 16066.

Westinghouse Non-Proprietary Class 3

Westinghouse

To: James P. Molkenthin
cc:
From: George J. Demetri
Ext: (412) 374-6257
Fax: (724) 940-8565

Date: March 19, 2019
Your: N/A
Our: LTR-SDA-18-127-NP, Rev. 0

Subject: **Generic Responses to the U.S. NRC Requests for Additional Information on PWROG Report, PWROG-17033, for Westinghouse Reactor Coolant Pump Casings**

This letter provides a non-proprietary version of the Westinghouse responses to the Requests for Additional Information (RAI) RAI-1 through RAI-9 from the U.S. Nuclear Regulatory Commission (NRC) for the Reactor Coolant Pump Casing evaluation performed in the Pressurized Water Reactor Owners Group (PWROG) report, PWROG-17033. Please transmit these responses to the PWROG. If there are any questions, please contact the undersigned.

Authors:
Electronically Approved*
George J. Demetri
Structural Design & Analysis III

Electronically Approved*
Geoffrey M. Loy
Structural Design & Analysis III

Electronically Approved*
Anees Udyawar
Structural Design & Analysis III

Reviewers:
Electronically Approved*
Alexandria M. Carolan
Structural Design & Analysis III

Approved By:
Electronically Approved*
Lynn A. Patterson, Manager
Structural Design & Analysis III

** Electronically approved records are authenticated in the Electronic Document Management System.*

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

Generic RAI Responses (Non-Proprietary)

March 19, 2019

Reactor Coolant Pump Integrity Analysis, GALL TLAA 4.7

Regulatory Basis:

10 CFR 54.21(c) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. NRC staff must be able to find that actions have been identified, and have been, or will be taken to manage the effects of aging during the subsequent period of extended operation (SPEO) on the functionality of structures and components such that there is reasonable assurance that the activities authorized by the subsequent renewed license will continue to be conducted in accordance with the CLB. As described in the Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants (SRP-SLR), an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report.

The regulation in 10 CFR 54.21(c)(1)(ii) states that, for a specific time-limited aging analysis (TLAA) that is dispositioned in accordance with this regulation, applicants must demonstrate that the analysis has been projected to the end of the SPEO. Many PWR applicants have identified the crack stability analysis of cast austenitic stainless steel (CASS) reactor coolant pump (RCP) casings as a TLAA item.

Background:

The ASME Code, Section XI, Table IWB-2500-1, requires periodic volumetric inspections of the welds of the RCP casings in nuclear power plants. These inspections result in radiation exposure to the personnel performing the inspections and require significant resources to perform the inspections. As a result, the ASME Code Committee approved Code Case N-481 in March 1990 that allows replacing the volumetric examinations with a fracture mechanics-based integrity evaluation supplemented by specific visual inspections. Previously, industry provided topical report WCAP-13045 which contains the crack stability evaluation and fatigue crack growth calculation of the RCP casings to demonstrate compliance with ASME Code Case N-481 for 40 years of operation. To demonstrate continued compliance during SPEO, the PWROG re-evaluated the analyses in WCAP-13045 for 80 years as documented in PWROG-17033, Revision 1.

The NRC staff notes that ASME Code Committees annulled Code Case N-481 in 2004 because the inspection requirements of the code case have been incorporated into the ASME Code, Section XI. The NRC staff finds that it is acceptable to reference Code Case N-481 herein because the specific analytical method for the crack stability analysis in the code case was used for the original 40-year license.

RAI-1

Issue:

WCAP-13045 postulated a flaw of ¼ T depth (T is the wall thickness of the RCP casing) having an aspect ratio of 6 to 1 in accordance with Code Case N-481. The locations of the flaws are identified in Section 9 of WCAP-13045. The NRC staff notes that the applied loadings to analyze the postulated flaw are generic in nature. Tables 11-2 to 11-9 of WCAP-13045 provided the generic J_{app} values (the applied J-integral) for various flaws under various loading conditions for various RCP models.

A crack is stable if either

(1) $J_{app} < J_{Ic}$ or
(2) If $J_{app} \geq J_{Ic}$, then $T_{app} < T_{material}$ and $J_{app} \leq J_{max}$

Request:

(a) Confirm that the locations of the postulated flaws in WCAP-13045 represent and bound the high stress areas of the RCP casings of all the nuclear plants that were analyzed.

(b) Discuss whether the J_{app} values in WCAP-13045 bound the J_{app} values from the RCPs in all the plants that were analyzed.

(c) Provide the critical crack depth that would exceed the crack stability criteria (i.e., what is the critical crack depth?).

Response:

(a) The postulated flaw locations in WCAP-13045 represent and bound the high stress areas of all applicable Westinghouse reactor coolant pump (RCP) casing designs in the U.S. PWR operating fleet. The Westinghouse RCP models in the current U.S. PWR operating fleet are 93, 93A, 93A-1, and 100A, with the majority being Model 93 or Model 93A. As stated in WCAP-13045, the geometry of the Model 93A-1 and Model 100A pump casings in the high stress locations is adequately represented by the Model 93 and Model 93A casings. By explicitly modeling and analyzing the Model 93 and Model 93A pump casings using finite element techniques, the high stress areas of the casings for all plants considered in WCAP-13045 were adequately represented. The locations of the postulated flaws were chosen based on those high stress areas of the representative casings, as determined by the finite element analyses.

WCAP-13045 was issued in September 1991 to generically evaluate the structural integrity of the Westinghouse-designed RCP casings for long term operation in accordance with Code Case N-481, which was approved in 1990 by the ASME Code Committee. In March 2004, Code Case N-481 was annulled by ASME and the visual inspection guidelines of the Code Case were incorporated into the 2000 Edition (for pump casings) and 2008 Addenda (for valve bodies) of ASME Section XI. From 1991 until around 2008, in order to supplement the generic WCAP-13045, several plant specific evaluations (more than half of the U.S. PWR operating fleet and several foreign plants) were performed for the casings of the Westinghouse-designed pumps, either for the original design life of

the plant or for 60 year life extensions. Those plant specific evaluations were based on the pump configurations analyzed in WCAP-13045 (i.e. each of the plant specific pump models have design drawings that are consistent with the pumps analyzed in WCAP-13045), and demonstrated the structural integrity of the casings at the critical locations.

(b) As discussed in Section 6 of WCAP-13045, conservative loads (internal force and moment loading) for a majority of plants were selected to form the basis for the fracture mechanics analysis. The J_{app} values reported in Section 11 of WCAP-13045 were based on the Loading Level A (faulted condition, safe shutdown earthquake) and Loading Level C (upset condition, loss of load transient) screening loads which were developed in that report. Those screening loads were developed as a set of representative bounding loads based on a sampling of plant-specific loads. As stated in WCAP-13045, plants were expected to reconcile their plant-specific loads (forces and moments) with the screening loads. If plant specific loads were larger than the screening loads, then updated J_{app} values would be calculated using plant specific loads to confirm the crack stability criteria. Those reconciliations were performed in plant specific topical reports, and it was determined that some plants had loads and J_{app} values that exceeded the screening values in WCAP-13045. However, the J_{app} values associated with those higher plant specific loads were still qualified without exception in plant specific evaluations using either the limiting fracture toughness values in WCAP-13405 or the fracture toughness values calculated with the methodology in NUREG/CR-4513, Revision 1. For the majority of the plant specific evaluations, it was determined that the screening stresses were sufficiently higher and more limiting than the plant specific stresses; hence the J_{app} values in WCAP-13045 bound the J_{app} values for the majority of the plant specific cases.

It should be noted that, over time, loads have not changed significantly since there have been only a few major NSSS (Nuclear Steam Supply System) component replacements and refurbishments, and uprates do not have a significant impact on stability evaluations. Therefore, the representative screening loadings considered in WCAP-13045 are representative of plant specific values that were considered in the existing N-481 analyses.

(c) As reported in WCAP-13045, flaw location [ ]^{a,c,e} was determined to be the most limiting flaw location for the Model 93 pump casing, and flaw location [ ]^{a,c,e} was determined to be the most limiting flaw location for the Model 93A pump casing with regard to fracture toughness. The postulated flaw sizes were sufficiently large, with flaw depths on the order of 1/4T (or 25% of the wall thickness), which provided flaw depths greater than 1 inch. Furthermore, WCAP-13045 puts more emphasis on the analysis of SA-351 CF8M material due to its more limiting fracture toughness values resulting from higher levels of molybdenum in the chemistry compared to CF8. Higher levels of molybdenum result in lower values for J_{Ic}, T_{mat}, J_{max}. By reviewing the plant specific evaluations for the plants that have CF8M, [

]^{a,c,e} Postulated flaws larger than the range mentioned above would have difficulty satisfying the crack stability criteria, unless other parameters can be reassessed (i.e. refinement in loads, stresses, material properties, etc.). However, for the CF8M plant specific analyses, the stability criteria were met per the guidance of Code Case

N-481 for a postulated flaw depth of 1/4T. For CF8 pump casings, which have very low molybdenum, the critical flaw depth would be generically larger than that based on CF8M.

RAI-2

Issue:

The second paragraph on Page 4-2 of PWROG-17033 states that (1) the transient stresses used in the fatigue crack growth are generic and encompass the various RCP models; (2) these stresses have not impacted for 80 years of operation; and (3) the number of predicted cycles for 80 years of service is assumed to be bounded by the transient cycles considered in Table 12-2 of WCAP-13045. It is not clear to the NRC staff that the generic stresses used in WCAP-13045 bound the transient stresses in the RCP casings of all the plants that will use this topical report. Also, it is not clear that the number of transient cycles used in the fatigue growth calculations in WCAP-13045 bound the transient cycles that are predicted to the end of 80 years for the plants that will use this topical report. The NRC staff understands that licensees who uses the topical report will use the results of fatigue crack growth and crack stability calculations in WCAP-13045 for their plants and will not perform plant-specific analyses for their RCP casings. In this scenario, the topic report should provide guidance for a licensee to demonstrate that the input parameters such as applied loading, stresses, fracture toughness values and transient cycles associated with the RCP casings at its plant are bounded by the corresponding input parameters used in the analyses in WCAP-13045.

In addition, the fourth paragraph on Page 4-2 of PWROG-17033 states that a flaw depth of 0.3 inches is the maximum acceptable flaw size for the RCP casing.

Request

(a) For licensees who plans to use PWROG-17033 as part of subsequent license renewal application, provide guidance for the licensees to demonstrate that the input parameters such as applied loading, stresses, fracture toughness values and transient cycles associated with the RCP casings at their plant are bounded by the corresponding input parameters used in the fatigue crack growth and crack stability analyses in WCAP-13045.

(b) Discuss the length of the postulate flaw, orientation of the flaw, and the direction of its growth (e.g., crack grows axially or circumferentially into the wall thickness) in the fatigue crack growth and crack stability calculations in WCAP-13045.

Response:

(a) In general, the goal of PWROG-17033 was to demonstrate that the crack stability analyses performed in WCAP-13045 adequately represented and qualified the Westinghouse-designed pump casings for all loadings and service conditions. This conclusion was reached based on the following factors:

1. After the inception of Code Case N-481 in 1990, several plant specific fracture mechanics evaluations were performed and showed adequate crack stability margins. The casings of more than half of the Westinghouse pumps currently operating in the U.S. PWR fleet were evaluated. In the year 2000, the Code Case was incorporated into the body of the ASME Code. However, the flaw tolerance evaluation requirements were removed from the Code because multiple flaw tolerance calculations had been completed and reviewed by the NRC.

Those calculations demonstrated that the pump casings were both flaw tolerant and resistant to degradation caused by corrosion mechanisms. At that time, the NRC representative on the appropriate Code Committee insisted on removing the requirements for flaw tolerance stability calculations because he considered the continued review of such calculations an unwarranted expense.

2. The majority of domestic operating plants with Westinghouse pumps are made from CF8 material. As reported in Table 5-1 of WCAP-13045, the CF8 material results in the limiting fracture toughness values of [ ]^{a,c,e}. As can be seen in WCAP-13045, Table A-2, those values are limiting for any of the plants whose pump casings are made from CF8 material, regardless of the plant-specific chemistry of the pump casing material. Only three pump casings are made from CF8M material. Table 5-1 of WCAP-13045 reports the most limiting CF8M fracture toughness values for the flange inner quarter, the nozzle outer quarter, and the nozzle inner quarter locations. Each of those limiting sets of values in Table 5-1 is based on the pump casing chemistry of one of the three CF8M casings, and, as can be seen from Table A-2, pages A-29, A-30, and A-41 of WCAP-13045, they bound the fracture toughness values of the other two casings at those locations. The three plants that have the CF8M materials were analyzed on a plant specific basis, and were shown to meet the flaw stability margins. As a result, the fracture toughness values reported in WCAP-13045, as calculated by the methodology referenced therein, bound the fracture toughness values that would be calculated for a specific plant's pump casing using the methodology of WCAP-13045.

   Over time, the fracture toughness values for SA-351 CF8M and CF8 materials reach a saturated state based on chemistry (molybdenum, delta ferrite), casting method (static, centrifugal), temperature, and time as discussed in NUREG/CR-4513, Revision 2 (see Figure 12 of NUREG/CR-4513, Revision 2). The work in PWROG-17033 used a lower bound saturated fracture toughness correlation curve from NUREG/CR-4513, Revision 2 to demonstrate that the bounding CF8M material heat of Table 5-1 of WCAP-13045 from all Westinghouse RCPs still has sufficient fracture toughness to meet the stability criteria and screening loadings of WCAP-13045. The other limiting CF8M and CF8 materials in Table 5-1 of WCAP-13045 were also reviewed in the RAI-9 response based on NUREG/CR-4513, Revision 1 and NUREG/CR-4513, Revision 2, and shown to have acceptable flaw stability. Therefore, any further aging of CF8M material or CF8 material beyond 40 years at PWR operating temperatures will not decrease the fracture toughness values because those materials have already reached a lower bound saturated fracture toughness state.

3. The screening loads developed in WCAP-13045 were bounding for the majority of the RCP pumps. For plant specific fracture mechanics analysis cases where the plant specific loads were higher than the screening loads, the stability criteria were reanalyzed and shown to meet all fracture toughness margins. For the majority of plant specific evaluations performed, it was determined that the screening stresses in WCAP-13045 were sufficiently higher than and bounded the plant specific stresses. Therefore, the applied loadings and stresses for all plants considered in WCAP-13045 have been qualified.

4. The Westinghouse-designed RCP models in the current operating U.S. PWR fleet are Model 93, Model 93A, Model 93A-1, and Model 100A. The majority of the operating Westinghouse RCPs are Model 93 or 93A. As a result, the Model 93 and Model 93A pump casings were explicitly modelled and analyzed using finite element techniques in WCAP-13045 to determine the most critical stress locations in the casings (and for postulation of flaws). RCP Models 93A-1 and 100A are adequately represented by the Model 93 and Model 93A RCPs and therefore the finite element models of the Model 93 and Model 93A pump casings were sufficient to cover the Model 93A-1 and Model 100A casings. It should be noted that all the pump casings were fabricated from SA-351 CF8, with the exception of 3 plants whose casing material is CF8M. The casings made from CF8M have been analyzed on a plant specific basis and shown to meet the requirements of Code Case N-481 for their design life. Therefore, the appropriate critical stress locations were considered and analyzed in WCAP-13045 to encompass all of the Westinghouse-designed pumps currently operating in the U.S. PWR fleet.

5. One of the inputs in a fatigue crack growth (FCG) analysis is transient stress ranges. As discussed in the plant-specific evaluations, the generic transients considered in the FCG evaluation of WCAP-13045 were reviewed against the severity and frequency of the plant specific operating transients. Based on that review, it was concluded that the typical design transients and cycles used in WCAP-13045 are generally applicable to the plant specific transients and cycles. [

]^{a,c,e}. It should also be noted that uprate conditions will not impact the loads and transients as previously determined by plant specific evaluations.

   As an additional measure of conservatism, the FCG analysis performed in PWROG-17033 doubled the number of transient cycles used in WCAP-13045 to account for the increase in plant life from 60 to 80 years of operation, thereby demonstrating that the FCG analyses performed in WCAP-13045 and in the plant specific evaluations remain valid. Plants applying for 80 year life typically have retained the same number of design cycles as their 60 year design cycles. As a result, the fatigue crack growth assessment in PWROG-17033 will be applicable for plants submitting their 80 year subsequent license renewal application.

6. Based on a review of Westinghouse records, the service life of the RCP casings has been of no major concern as there has been no degradation or detection of crack-like indications.

Thus, based on the above discussions, the stresses, applied loadings, fracture toughness values, and transients considered in WCAP-13045 are bounding for an individual plant's pump casing or have been reconciled through plant specific evaluations.

It is recommended that the following guidance be considered for licensees who plan to use PWROG-17033 and WCAP-13045 as part of their subsequent license renewal application:

1. Confirm that the licensees' pumps are Westinghouse-designed models.

   The Combustion Engineering and some of the Babcock & Wilcox NSSS plants do not have Westinghouse-designed pumps and therefore do not fall within the scope of WCAP-13045 and PWROG-17033.

2. Confirm that the licensees' Westinghouse-designed pump is either a Model 63, Model 70, Model 93, Model 93A, Model 93A-1, Model 93D, Model 100A, or Model 100D, and fabricated with SA-351 CF8 or CF8M material.

   The above Westinghouse pump models and materials were considered in the fracture mechanics analysis of WCAP-13045, as these designs have been encompassed by the finite element models and loading conditions.

3. For plants with Westinghouse-designed pumps whose pump material heats are not explicitly listed in Appendix A of WCAP-13045, perform a plant-specific analysis or reconciliation to ensure the fracture toughness values based on actual material chemistry are bounded by the values in WCAP-13045 or PWROG-17033.

   The fracture toughness values derived in WCAP-13045 were based on available material certification records for Westinghouse-designed pump casings, both domestic and foreign. The limiting fracture toughness values in Table 5-1 of WCAP-13045 were based on the heats reported in Appendix A of WCAP-13045.

(b) For the Model 93 pump casing, the fatigue crack growth evaluations which produced the most limiting results were for postulated flaws at the [

]^{a,c,e}. For the Model 93A pump casing, the limiting fatigue crack growth results were for a postulated flaw oriented in the [

]^{a,c,e}.

The FCG analysis results for postulated flaw depths (flaw length is based on the aspect ratio, flaw length/flaw depth, of 6:1) into the pump casing wall thickness are provided in Tables 12-1 and 12-3 of WCAP-13045 for the Model 93A and Model 93 pump designs, respectively. A small, postulated flaw depth evaluated in WCAP-13045 and PWROG-17033 is based on an initial depth of 0.3". Other initial flaw depths were also considered for sensitivity analyses, as shown in Tables 12-1 and 12-3 of WCAP-13045, to demonstrate that the growth due to FCG is small. Therefore, the postulated flaw depths used in the FCG analysis are equal to and well in excess of the maximum acceptable flaw size (flaw depth of 0.3") in the Acceptance Standards in Table IWB-3518-2 (for pressure retaining welds in pump casings) up to the 2007 Edition of the ASME Section XI code. The flaw depth of 0.3" is

still the maximum acceptable flaw size in the Acceptance Standards Table IWB-3519.2-2 (for pump casings) in later editions of the ASME Section XI Code. Therefore, the flaw depth of 0.3" and the other larger postulated flaw size cases in WCAP-13045 were provided as sensitivity studies to demonstrate that the flaws do not grow significantly over time.

Further details of the flaw locations and postulations in the stability analysis are provided in Section 9 of WCAP-13045 for the various Westinghouse pump casings.

RAI-3

Issue

The topical report specifies that RCP casings be periodically visually examined in accordance with Code Case N-481. The visual examinations of RCP casings are performed once every 10-year interval. The NRC staff questioned whether there have been degradations observed in RCP casings in PWRs. In addition, the NRC staff questioned whether there are defense-in-depth measures that can alert the operators if any pump casing is degraded during the 10-year interval between two visual examinations.

Request

(a) Discuss any degradation in RCP casings that have occurred in any of PWRs.

(b) Discuss defense-in-depth measures that are in place to alert the operators to take corrective actions should leakage occur at the RCP casing during the SPEO.

Response:

(a) Westinghouse has no knowledge of service degradation of pump casings from corrosion or other cracking mechanisms.

The Westinghouse-designed pump casings are fabricated from cast stainless steel. Such materials are highly resistant to corrosion issues, and exhibit high fracture toughness values (i.e., J_{Ic} J_{max}, and T_{mat}) in the pre-service product form. However, such materials are subject to thermal aging embrittlement at nominal operating temperatures of nuclear plants. As a result, the materials for the pump casings were evaluated for thermal aging embrittlement consistent with the methodology in WCAP-13045, PWROG-17033, and plant specific analyses, and all were found to meet the acceptance criteria.

The pump casings were designed to the ASME Code at the time of fabrication. Design, normal, operating, and faulted applied loads on the nozzles are very conservative when compared to actual service conditions. The stresses are such that Code conditions are met.

(b) Based on typical service conditions, flaws are not expected to occur during the life of the plant. If small flaws are present, they will grow only a small amount throughout the service life. If a flaw should happen to penetrate the wall, it will leak at a very detectable rate and will not be unstable, thus allowing a shutdown of the plant per the requirements in plant Technical Specifications. Plants rigorously follow Regulatory Guide 1.45, which provides guidance on monitoring and responding to reactor coolant system leakage.

RAI-4

Issue

Section 11.2 of WCAP-13045 addresses the crack stability analysis results for postulated flaws in the Model 93 RCP casings made with CF8M cast austenitic stainless steel. Specifically, the stability analysis in the WCAP-13045 report indicates that postulated flaw location 5-93 is subject to the Loss of Load transient (upset condition) and is identified as the highest stressed location.

WCAP-13045 also indicates that on a plant-specific basis, a yield strength level slightly greater than 20 ksi is sufficient to confirm the flaw stability at flaw location 5-93. The WCAP report further indicates that such a yield strength level (greater than 20 ksi) will ensure that the stability criteria regarding the fracture toughness and tearing modulus are met (i.e., applied J-integral < J_{max} of the material, and applied tearing modulus $T < T_{max}$ of the material).

In contrast, Table 2 of PWROG-17033, Revision 1 uses a yield strength level less than 20 ksi to calculate the fracture toughness properties of the CF8M material in the crack stability analysis. In addition, PWROG-17033, Revision 1, does not address whether postulated flaw location 5-93 in Section 11.2 of WCAP-13045 meets the crack stability criteria for 80 years of operation when the crack stability analysis uses the updated fracture toughness properties, such as those in NUREG/CR-4513, Revision 1.

Request

Describe how the generic crack stability analysis in WCAP-13045 confirms the stability of postulated flaw location 5-93 (CF8M material at the highest stressed location) taking into account the plant-specific yield strength of the material, actual loading conditions, and the fracture toughness values (J_{IC}, J_{max} and T_{mat}).

Response:

Of all the domestic operating plants considered in WCAP-13045, only 3 plants have pump casings made from SA-351 CF8M material. The most limiting heat was picked from the three CF8M plants to represent the highest stressed flaw location [ ]^{a,c,e} as shown in Table 5-1 of WCAP-13045. Therefore, Table 5-1 of WCAP-13045 reports the limiting fracture toughness values of flaw location [ ]^{a,c,e}

The chemistry of the limiting heat used for the highest stress flaw location [ ]^{a,c,e} for the CF8M material was used to recalculate the fracture toughness values based on NUREG/CR-4513 Revision 1. Based on NUREG/CR-4513, Revision 1, the fracture toughness values are [ ]^{a,c,e} at operating conditions. As a result, the saturated fracture toughness values based on NUREG/CR-4513 Revision 1 are larger than the values determined in Table 5-1 of WCAP-13045 for flaw location [ ]^{a,c,e}.

Based on WCAP-13045, Table 11-2 (CF8M material with ASME Code properties), the highest stress flaw location [ ]^{a,c,e} for loading level C has applied loadings of [ ]^{a,c,e} when the pressure is limited to 2,485 psi due to a Reactor Trip. Comparing these calculated J_{app} and T_{app} values to the NUREG/CR-4513, Revision 1 fracture toughness values demonstrates that the high stress location of flaw [ ]^{a,c,e} meets the crack stability margins.

With regard to loading conditions, the plant specific evaluation performed for the plant with the limiting CF8M material heat for location [ ]^{a,c,e} reports that the actual plant specific loads are below the screening loads used in WCAP-13045. As a result, the screening loads bound the plant-specific loads, and the J_{app} and T_{app} values in WCAP-13045 remain bounding for the plant with the limiting CF8M casing at the high stress flaw location [ ]^{a,c,e}.

Thus, the generic crack stability analysis in WCAP-13045 confirms the stability of postulated flaw location [ ]^{a,c,e} (CF8M material at the highest stressed location) taking into account conservative ASME Code yield strength (in lieu of plant specific yield strength) of the material, conservative screening loads (in lieu of less limiting actual loading conditions), and the fracture toughness values (J_{Ic}, J_{max} and T_{mat}) based on NUREG/CR-4513, Revision 1.

RAI-5

Issue

Page 3-4 of PWROG-17033 stated that "...the fracture toughness correlations used for the full aged condition is applicable for plants operating at and beyond 15 EFPY (Effective Full Power Years) for the CF8M materials... The Westinghouse NSSS plants have been operating for greater than 15 EFPY..."

Request

(a) Clarify whether the use of fully-aged fracture toughness for plants beyond 15 EFPY are also applicable to the RCPs in Babcock and Wilcox plants and Combustion Engineering plants.

(b) Confirm that PWROG-17033 is applicable only to RCPs that are manufactured by Westinghouse or its designated-vendor that follows Westinghouse's specifications. Discuss whether any domestic nuclear power plant uses a RCP that is not manufactured by Westinghouse.

Response:

(a) The scope of WCAP-13045 and PWROG-17033 is only applicable to Westinghouse-designed RCPs with Model 63, 70, 93, 93A, 93A-1, 93D, 100A, and 100D casings fabricated from SA-351 CF8 or CF8M material.

All operating U.S. PWRs with Westinghouse NSSS have pumps designed by Westinghouse. Furthermore, two of the operating U.S. PWRs with Babcock/Wilcox NSSS also have Westinghouse-designed pumps. However, none of the operating U.S. PWRs with Combustion Engineering NSSS have Westinghouse-designed pumps.

Therefore, the provisions of WCAP-13045 and PWROG-17033 are applicable to all operating U.S. PWRs with Westinghouse NSSS and two operating U.S. PWRs with Babcock &Wilcox NSSS.

(b) See response to RAI-5 (a) above.

<u>RAI-6</u>

<u>Issue</u>

Section 11.2 of WCAP-13045 (bottom of page 11-1 and top of page 11-2) discusses the crack stability analyses for Model 93 RCP casings. The report discusses a postulated flaw (Number 5-93) that exceeded the stability criteria under certain assumptions on yield stress and operating conditions (i.e., crack stability cannot be demonstrated for Flaw Number 5-93).

<u>Request</u>

(a) Certain assumptions on yield stress and operating conditions for Flaw Number 5-93 may result in crack instability as discussed in Section 11.2 of WCAP-13045. Discuss whether these assumptions and resultant crack instability are applicable to any domestic PWR plants that may use PWROG-17033.

(b) Discuss how a plant can demonstrate structural integrity of its RCP casings using PWROG-17033 if the crack stability criteria are exceeded.

(c) Discuss the likelihood of postulated Flaw Number 5-93 with the applied loading occurring in any Westinghouse RCP casing. Is Flaw Number 5-93 a theoretical, worst case scenario flaw, or it is a possible flaw that would likely to occur in an operating RCP in the field?

<u>Response:</u>

(a) Section 11.2 of WCAP-13045 discusses Model 93 pumps fabricated from SA-351 CF8 and CF8M materials; however, CF8M material is more limiting than CF8 material and is the crux of the discussion herein, especially for the highest stressed flaw location [$]^{a,c,e}$. As previously identified in the RAI-4 response, there are only three operating U.S. PWR plants that have CF8M casings. Furthermore, as discussed in the RAI-4 response, the plant which has the limiting heat that was used to determine the fracture toughness values for flaw location [$]^{a,c,e}$ in Table 5-1 of WCAP-13045 was re-evaluated based on updated fracture toughness values from NUREG/CR-4513, Rev 1. As concluded in the RAI-4 response, crack stability was demonstrated for postulated flaw location [$]^{a,c,e}$, taking into account conservative ASME Code yield strength (in lieu of plant specific yield strength) of the material, conservative screening loads (in lieu of less limiting actual loading conditions), and fracture toughness values (J_{Ic}, J_{max} and T_{mat}) based on NUREG/CR-4513, Revision 1. The other two domestic PWR pump casings with CF8M have plant specific fracture toughness values that are considerably higher for flaw location [$]^{a,c,e}$ than the limiting values in Table 5-1 of WCAP-13045. For those two casings, plant specific crack stability evaluations using the J_{app} and T_{app} values in Table 11-2 of WCAP-13045 have demonstrated that flaw location [$]^{a,c,e}$ is stable without exception (i.e., the ASME Code minimum yield strength is assumed and the screening loads are based on WCAP-13045).

Therefore, all domestic operating PWR plants with pump casings made from CF8M material have met the stability criteria at flaw location [ ]^{a,c,e} based on plant specific evaluations which used ASME Code properties for yield strength, conservative screening loads to generate the J_{app} and T_{app} values in Section 11.2 of WCAP-13045, and fracture toughness values based on WCAP-13045 and NUREG/CR-4513, Revision 1.

(b) Based on the response to RAI-6(a), crack stability is demonstrated for the highest stressed flaw location [ ]^{a,c,e} and the limiting CF8M material casings using the ASME Code minimum properties, conservative screening loads, and updated fracture toughness values based on NUREG/CR-4513, Revision 1. Therefore, there is no concern for domestic operating PWR plants with CF8M pump casings.

If, hypothetically, the crack stability criteria are exceeded, then plant specific yield strength along with the use of actual plant specific loadings can be used to reduce conservatism in the evaluation. Use of a smaller flaw size can also be considered, as the service experience for pump casings has shown no degradation or detected indications.

(c) The likelihood of postulated flaw location [ ]^{a,c,e} with the applied screening loads from WCAP-13045 is low. To date, Westinghouse has no knowledge of service degradation of pump casings from corrosion or other cracking mechanisms. Therefore, flaw location [ ]^{a,c,e} is a theoretical, worst case scenario used for analysis purposes to generically bound those PWR plants which use Westinghouse-designed pumps, and to resolve the fracture mechanics evaluation condition in Code Case N-481.

Furthermore, the conservative screening loads are based on the maximum Loss of Load design transient pressure of 2,635 psig (400 psi above the normal operating pressure of 2,235 psig) and at a conservative temperature of 590°F (40°F above the normal operating temperature of 550°F). As described in Section 6.1 of WCAP-13045, there are three systems in place to mitigate pressure increases of around 400 psi during a Loss of Load transient. First, there are power operated relief valves (PORVs). These valves are activated at 2,335 psig as recommended in the technical specification. The pressurizer safety valves are set for 2,485 psig, but allow the pressure to exceed 2,485 psig by 100 psi or more. The reactor trip high pressure setup is 2,385 psig. This set point is chosen so as to avoid challenging the pressurizer safety valves. Thus, the pressure during a Loss of Load transient is actually limited to 2,485 psig in lieu of 2,635 psig, which was conservatively considered for certain cases in WCAP-13045.

RAI-7

Issue

Table A-1 of WCAP-13045 contains the names of 28 nuclear units. It is not clear whether all the applied loadings from the 28 nuclear units were considered in the crack stability and fatigue crack growth calculations such that the most limiting loadings were used in these calculations to demonstrate the structural integrity of the RCP casing.

Request

Discuss whether the applied loads and stresses of all the 28 nuclear units were considered and the bounding values from the 28 plants were used in the crack stability and fatigue crack growth calculations in WCAP-13045. If not all 28 nuclear units were considered, discuss how many nuclear units whose loads and stresses were considered and used in these calculations and whether the input values used are bounding for the 28 nuclear units.

Response:

The applied loads for 29 plants were considered in Table 6-1 of WCAP-13045. In general, to cover a wide range of plants, representative bounding force and moment values were chosen and are listed in Table 6-2 of WCAP-13045. Furthermore, as mentioned in Sections 6 and 8 of WCAP-13045, the major contributor to the high stresses in the discontinuity regions of the pump casings is the internal pressure, which is set at a value of 2,635 psig or 2,485 psig for all plants based on the loading condition (i.e. Loading Levels A, B or C). Any differences in the force and moment loadings shown in Table 6-1 (note the load magnitudes are similar) will not have a significant impact on the final stability conclusions since the fracture mechanics evaluation is related to total stress, not individual force and moment components.

As mentioned in the response to RAI-1, plant specific fracture mechanics evaluations on RCP casings have been performed for more than half of the U.S. operating PWRs (and also several foreign plants) with Westinghouse RCPs. Based on plant specific evaluations, the majority of the plants were bounded by the screening loadings of WCAP-13045. For the few plants whose loads were not bounded by the screening loads, the casings of those plants were re-analyzed and shown to meet fracture stability margins with no concern. For the 3 plants that have CF8M pump casings, all plant specific loads were bounded by the screening loads of WCAP-13045.

Thus, the screening loads considered in Table 6-2 of WCAP-13045, which are based on 29 plants, can be considered a good representation of the plants in the PWR fleet with Westinghouse RCP casings, as evident by the wide range of plant specific evaluations that have met the fracture stability criteria using the screening loads of WCAP-13045. As noted earlier, other conservatisms are also present in the fracture mechanics evaluations, such as the use of ASME Code properties and the use of a large postulated flaw that has never been observed during 1400 reactor-years of operation.

RAI-8

Issue

Section 3.2 of PWROG-17033 discusses fracture toughness calculations based on NUREG/CR-4513, Revision 2. The NRC staff notes that Aging Management Program XI.M12, *Thermal Aging Embrittlement of Cast Austenitic Stainless Steel*, in Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) report, NUREG-2191, Volume 2, discusses fracture toughness values based on the prediction method in NUREG/CR-4513, Revision 1. The NRC staff notes that the GALL-SLR report does not reference NUREG/CR-4513, Revision 2.

Request

Discuss whether the saturated fracture toughness value used in the crack stability analysis of RCP casings in WCAP-13045 would still be limiting and bounding as compared to the saturated fracture toughness values predicted in accordance with NUREG/CR-4513, Revision 1 or Revision 2.

Response:

See Response to RAI-9

RAI-9

Issue

PWROG-17033, Revision 1 indicates that the J_{IC}, J_{max} and T_{mat} values in Table 1 of PWROG-17033 are bounding and were used to demonstrate the crack stability of pump casings. However, the J_{IC}, J_{max} and T_{mat} values used in Tables 11-2 and 11-3 of WCAP-13045 to demonstrate the crack stability of flaw location 5-93 were higher than the J_{IC}, J_{max} and T_{mat} values listed in Table 1 of PWROG-17033. If the lower J_{IC}, J_{max} and T_{mat} values in Table 1 of PWROG-17033 were used to analyze flaw location 5-93, crack stability may not be demonstrated for Flaw Number 5-93. It appears that separate fracture toughness value criteria are needed to qualify various flaws to demonstrate crack stability, not a single set of fracture toughness value as specified in Table 1 of PWROG-17033-P. Table 5-1 of WCAP-13045 does provide the 4 sets of fracture toughness values as end-of-service life criteria. Therefore, it appears that the fracture toughness values in Table 5-1 of WCAP-13045 should be compared to the fracture toughness values predicted based on the method in NUREG/CR-4513, Revision 1.

Request

Discuss whether the 4 sets of fracture toughness values in Table 5-1 of WCAP-13045 satisfy the fracture toughness values as predicted using the method in NUREG/CR-4513, Revision 1. If not, please provide technical justification.

Response:

The 4 sets of fracture toughness values reported in Table 5-1 of WCAP-13045 were recalculated based on the methodology outlined in NUREG/CR-4513, Revisions 1 and 2. Any major differences in the fracture toughness values calculated by using the methodology of WCAP-13045 and the two revisions of NUREG/CR-4513 were reviewed and reconciled in the paragraphs that follow.

NUREG/CR-4513, Revision 1

All 4 sets of fracture toughness values in Table 5-1 of WCAP-13045 were recalculated using NUREG/CR-4513, Revision 1, and the values are shown in Table 1 below.

Table 1
Bounding End-of-Service Fracture Toughness Criteria (NUREG/CR-4513, Revision 1)

	Material	Part / Category	WCAP-13045, Table 5-1			NUREG/CR-4513, Revision 1			
			J_{Ic}	T_{mat}	J_{max}	J_{Ic}	T_{mat}	J_{max}	
			(in-lb/in²)	(-)	(in-lb/in²)	(in-lb/in²)	(-)	(in-lb/in²)	
[									]a,c,e
[									]a,c,e
[									]a,c,e
[									]a,c,e

All stability results given in Tables 11-2 through 11-5, Table 11-7, and Table 11-9 of WCAP-13045 were re-evaluated using the fracture toughness values calculated with the methodology of NUREG/CR-4513, Revision 1. As shown by Table 1, the values using NUREG/CR-4513, Revision 1 are higher than those from WCAP-13045, with the exception of T_{mat} for two cases.

Based on the re-evaluation of the fracture stability margins in Section 11 of WCAP-13045, the stability criteria were met in all but one case. The one case that does not show acceptable stability results is [ ]^{a,c,e} (Level C loading). As shown in Table 11-7 of WCAP-13045, J_{app} is below J_{max} for this flaw case, but [ ]^{a,c,e}), is slightly above the T_{mat} value of [ ]^{a,c,e} calculated by using NUREG/CR-4513, Revision 1. However, as a more realistic scenario, it is reasonable to assume a Reactor Trip transient with a pressure limit of 2,485 psig in lieu of the highly conservative pressure of 2,635 psig. As shown by Note (b) in Table 11-7 of WCAP-13045, this condition was assumed in order to show acceptability of flaw location [ ]^{a,c,e} for Level C loading. As a result of this lower pressure, the T_{app} will be substantially reduced from [ ]^{a,c,e} to a value significantly below the bounding criteria limit of [ ]^{a,c,e}, as calculated using NUREG/CR-4513, Revision 1 methodology. Therefore, using the technical justification above, all flaw locations reported in Tables 11-2 through 11-5, Table 11-7, and Table 11-9 of WCAP-13045 meet the fracture toughness bounding criteria calculated using the methodology of NUREG/CR-4513, Revision 1 and are therefore shown to be stable.

NUREG/CR-4513, Revision 2

Next, all 4 sets of fracture toughness values in Table 5-1 of WCAP-13045 were recalculated using NUREG/CR-4513, Revision 2, and the values are shown in Table 2 below.

Table 2
Bounding End-of-Service Fracture Toughness Criteria (NUREG/CR-4513, Revision 2)

	Material	Part / Category	WCAP-13045, Table 5-1			NUREG/CR-4513, Revision 2			
			J_{Ic} (in-lb/in^2)	T_{mat} (-)	J_{max} (in-lb/in^2)	J_{Ic} (in-lb/in^2)	T_{mat} (-)	J_{max} (in-lb/in^2)	
[									]^{a,c,e}
[									]^{a,c,e}
[									]^{a,c,e}
[									]^{a,c,e}

Based on a review of Tables 1 and 2, it is observed that NUREG/CR-4513, Revision 2 produces more limiting fracture toughness values than Revision 1. However, comparing the fracture toughness values produced by the two methodologies, only a few NUREG/CR-4513, Revision 2 fracture toughness values are slightly less than WCAP-13045. A review of the fracture toughness values reported in Tables 1 and 2 is performed below to demonstrate acceptable flaw stability criteria.

The first 3 cases (i.e. nozzle inner quarter, nozzle outer quarter, and flange inner quarter) in Table 2, which is similar to Table 5-1 of WCAP-13045, consider SA-351 CF8M stability criteria. Based on the fact that there are only 3 CF8M pump casings from three different plants in the entire Westinghouse PWR operating fleet, each of the above cases can be analyzed separately for flaw stability based on

Tables 11-2, 11-5, 11-7, and 11-9. The CF8 material in Table 2 will also be evaluated with the stability criteria of WCAP-13045 to show acceptable margins.

Nozzle Inner Quarter (1st row of Table 2) –CF8M

Based on Table 2, the fracture toughness values per NUREG/CR-4513, Revision 2 are slightly less than those in WCAP-13045. However, the J_{app} and T_{app} values reported in Table 11-2 of WCAP-13045 are still sufficiently below the NUREG/CR-4513, Revision 2 values for J_{max} and T_{mat}. In addition, plant specific analysis shows that the plant specific loads are bounded by the screening loads reported in WCAP-13045. As a result, all stability margins are met, and flaw stability has been demonstrated for this flaw location.

Nozzle Outer Quarter (2nd row of Table 2) - CF8M

Based on Table 2, all fracture toughness values per NUREG/CR-4513, Revision 2 are higher than those reported in WCAP-13045; therefore, the values reported in WCAP-13045 remain limiting and bounding. Furthermore, plant specific analysis shows that the plant specific loads are bounded by the screening loads reported in WCAP-13045. As a result, all stability margins are met, and flaw stability has been demonstrated for this flaw location.

Flange Inner Quarter (3rd row of Table 2) –CF8M

Based on Table 2, the fracture toughness value of J_{max} per NUREG/CR-4513, Revision 2 is higher than that reported in WCAP-13045; however, the J_{Ic} and T_{mat} values per NUREG/CR-4513 Revision 2 are less than those in WCAP-13045. From Table 11-2 of WCAP-13045, the limiting flaw location [ ]^{a,c,e} has a J_{app} value of [ ]^{a,c,e} and a T_{app} value of [ ]^{a,c,e} (based on the use of a Reactor Trip pressure of 2,485 psig – see Note (c) of Table 11-2). The J_{max} value of [ ]^{a,c,e} per NUREG/CR-4513, Revision 2 is greater than the J_{app} value of [ ]^{a,c,e}; however, the T_{mat} value of [ ]^{a,c,e} per NUREG/CR-4513, Revision 2 is less than the T_{app} value of [ ]^{a,c,e}. It is important to note that those limiting CF8M fracture toughness values are specific to one pump casing of one plant. The other two CF8M plants showed acceptable stability results at flaw location [ ]^{a,c,e}, as they have different material heats with higher calculated fracture toughness values. A review of the plant specific analysis shows that the screening loads in WCAP-13045 are higher than the plant specific loads for the plant with the limiting fracture toughness values (see Table 3); therefore, the stresses used in the T_{app} value are conservative in WCAP-13045 for flaw location [ ]^{a,c,e}. Furthermore, ASME Code properties have been used to determine the T_{app} in WCAP-13045, but actual plant specific yield strength is typically higher than the minimum Code yield strength. As a result, the T_{app} values in WCAP-13045 are conservative relative to actual material yield strengths, and if the actual plant specific T_{app} value was calculated using realistic plant specific inputs (i.e. loads and yield strength), T_{app} would fall below the T_{mat}.

Therefore, for this particular flaw location, only one plant has a CF8M heat that is limiting for flaw stability criteria; however, that plant demonstrates acceptable flaw stability using the methodology of WCAP-13045 and NUREG/CR-4513, Revision 1. Furthermore, that heat would likely demonstrate acceptable margins using NUREG/CR-4513, Revision 2 methodology if actual plant specific loads and yield strength were used. Based on the previous discussion and no known service degradation of this limiting CF8M casing, sufficient crack stability for this flaw location has been demonstrated.

Table 3
Comparison of Level C Screening Loads with Plant Specific Loads for RCP Casing Nozzles

Load	Inlet Nozzle Force (kips)	Inlet Nozzle Moment (in-kip)	Outlet Nozzle Force (kips)	Outlet Nozzle Moment (in-kip)
[				]^{a,c,e}
[				]^{a,c,e}

SA-351 CF8 (4th row of Table 2)

Based on Table 2, the fracture toughness values per NUREG/CR-4513, Revision 2 are all higher than the WCAP-13045 values with the exception of T_{mat}, which is discussed below.

The SA-351 CF8 fracture toughness properties are considered in WCAP-13045, Table 11-5 (Model 93), Table 11-7 (Model 93A) and Table 11-9 (Models 63, 100A). In Table 11-5 for Model 93 pump casings, all stability criteria ($T_{app} < T_{mat}$ and $J_{app} < J_{max}$) are met using fracture toughness values per NUREG/CR-4513, Revision 2 In Table 11-9 for Model 100A pump casings, all stability criteria ($T_{app} < T_{mat}$ and $J_{app} < J_{max}$) are also met with the use of fracture toughness values from NUREG/CR-4513, Revision 2. There are no Model 63 pumps in the U.S. PWR operating fleet; therefore, the stability results for Model 63 in Table 11-9 are not applicable.

Lastly, in WCAP-13045, Table 11-7 (Model 93A), the stability criteria for flaw location [ ]^{a,c,e} (Level C) are met with a Reactor Trip pressure limit of 2,485 psig (Note (b) in Table 11-7), using fracture toughness values per NUREG/CR-4513, Revision 2. In Table 11-7, flaw location [ ]^{a,c,e} (Level C) has a T_{app} value of [ ]^{a,c,e}, which is higher than the T_{mat} value of [ ]^{a,c,e} based on NUREG/CR-4513, Revision 2. However, if the Reactor Trip pressure limit is used for flaw location [ ]^{a,c,e}, then the T_{app} would drop consistently with the T_{app} drop seen for flaw location [ ]^{a,c,e} (i.e., T_{app} will be reduced by a value of approximately 21). As a result, all the stability results will be met for the Model 93A casing using the methodology of NUREG/CR-4513, Revision 2.

Conclusion

In conclusion, the use of the methodology in NUREG/CR-4513, Revision 1 or Revision 2 will produce fracture toughness values that are typically higher and less limiting than the fracture toughness values in WCAP-13045. For instances where the fracture toughness values based on NUREG/CR-4513, Revision 2 are lower than those in WCAP-13045, a case-by-case investigation was conducted as discussed above. Based on a review of plant specific evaluations and consideration of more realistic inputs (i.e. Reactor Trip pressure, plant specific loads and yield strength), it was demonstrated that all flaw stability margins have been met.

Program Management Office
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

PWROG-17033-P and NP, Revision 1
Project Number 99902037

June 14, 2018

OG-18-142

Document Control Desk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Subject: PWR Owners Group
Transmittal of PWROG-17033-P (&NP), Revision 1, "Update for Subsequent License Renewal: WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems" PA-MSC-1498

The purpose of this letter is to transmit Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR), PWROG-17033-P (& NP), Revision 1, "Update for Subsequent License Renewal: WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems" in accordance with the Nuclear Regulatory Commission (NRC) TR program for review and acceptance for referencing in regulatory actions (Enclosures 1 and 2).

Topical Report Summary

ASME Code Case N-481 allowed replacing the volumetric examination of primary loop pump casings with a fracture mechanics-based integrity evaluation supplemented by specific visual inspections. WCAP-13045 contains the integrity evaluation that was performed to demonstrate compliance with ASME Code Case N-481 for 40 years of operation.

1. The NRC has approved several plant license renewal applications for 60 years of operation that utilized the generic fracture mechanics analyses and conclusions in WCAP-13045 as discussed below:

2. The NRC Safety Evaluation Report for Salem Units 1 and 2 discusses the use of the generic fracture mechanics analysis in WCAP-13045 to meet the requirements of ASME Code Case N-481 for a 60 year license renewal in Section 4.4.4 of the SER. The NRC staff concluded that the generic analysis in WCAP-13045 is applicable to the Salem design of the RCP casings. The generic analysis, WCAP-13045, bounds the plant-specific analysis, WCAP-16957-P, as approved by the staff in the SER. The analysis was shown to remain valid for extended operation.

3. In the 60 year license renewal NRC Safety Evaluation Report for D.C. Cook Units 1 and 2, NUREG-1831, Section 4.7.2, the NRC staff discusses the use of WCAP-13045 to satisfy the requirements of ASME Code Case N-481 based on a plant-specific analysis. WCAP-13128. In Section 4.7.2.4 of the SER, the NRC concludes that the time-limited aging analysis (TLAA) regarding ASME Code Case N-481 that was provided is acceptable.

4. The 60 year license renewal SER (Section 4.3.2.10) for Diablo Canyon Units 1 and 2 discusses that WCAP-13045 was used to demonstrate compliance with Code Case N-481 based on a plant-specific evaluation, WCAP-13895. In Section 4.3.2.10.4 of the SER, the NRC concludes that it was shown that the aging of the RCP pump casings will be adequately managed for extended operation.

5. Based on the Sequoyah 60 license renewal SER, the NRC concluded that the plant's LRA did not need to include a TLAA related to Code Case N-481 because the licensing basis is per the ASME Code Section XI Edition, which no longer relies on N-481 for ISI (in-service inspection interval) requirements. Thus, it was not necessary to perform any TLAA analysis for RCP pump casings and N-481, as it does not meet Criterion 4 or 6 in 10 CFR 54.3(a).

In summary, WCAP-13045 has been reviewed by NRC to support 60 year license renewal applications for several plants that used Code Case N-481 as a basis for their ISI examination programs. The purpose of this TR is to extend the fracture mechanics integrity evaluation in WCAP-13045 through Subsequent License Renewal (SLR), 80 years of operation 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-13045-A is applicable to all Westinghouse Design primary loop pump casings. This same applicability is carried over for the TR presented herein. This TR is applicable to all Westinghouse design primary loop pump casings for 80 years of operation.

Intended Application

Licensees will reference PWROG-17033-P (&NP) 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 Westinghouse design primary loop pump casings through the subsequent license renewal period of operation (80 years).

Industry Implementation

PWROG-17033-P (&NP) can be implemented by all applicable U. S. PWRs as listed in the Limits of Applicability section above.

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-17033-P (&NP) 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 Westinghouse design primary loop pump casings 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.

<u>NRC Review Schedule</u>

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

This letter transmits two copies of PWROG-17033-P, Revision 1 (Enclosure 1) and one copy of PWROG-17033-NP, Revision 1 (Enclosure 2). PWROG-17033-P, Revision 1 (Enclosure 1) contains information proprietary to Westinghouse which is supported by an affidavit signed by Westinghouse, owner of the information. The affidavit, CAW-18-4762, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's regulations. The affidavit is included as Enclosure 3.

Accordingly, it is respectfully requested that this information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive, Suite 386
Cranberry Township, Pennsylvania, 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, Chief Operating Officer and Chairman
PWR Owners Group

KS:WAN:am

Electronically Approved Records Are Authenticated in the Electronic Document Management System

*** This record was final approved on 7/6/2018 9 53 38 AM. (This statement was added by the PRIME system upon its validation)

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Enclosures: 1. PWROG-17033-P Revision 1 "Update for Subsequent License Renewal: WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems" (Proprietary, 2 copies)

2. PWROG-17033-NP Revision 1 "Update for Subsequent License Renewal: WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems" (Non-Proprietary, 1 copy)

3. Application of Withholding, CAW-18-4762

cc: PWROG Management Committee
PWROG Materials Committee
PWROG Steering Committee
PWROG Licensing Committee
PWROG PMO
B. Benney, US NRC
P. Atkin, DOM
J. Andrachek, Westinghouse
J. Gresham, Westinghouse
G. Hall, Westinghouse
E. Shen, Westinghouse
S. Rigby, Westinghouse
M. Higby, Westinghouse
T. Zalewski, Westinghouse
P. Brusamonti, Westinghouse
J. Whelan, Westinghouse