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

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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

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

Sincerely yours,

Ken Schrader, 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

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.

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 <u>TRANSMITTAL LETTER to NRC</u>. 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

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

Date March 19, 2019
You 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*
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** 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 $J_{app} > J_{Ic}$, 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.

RAI-6

Issue

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).

Request

(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?

Response:

(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} (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

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 (e) 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

	Inlet Nozzle		Outlet Nozzle		
Load	Force (kips)	Moment (in-kip)	Force (kips)	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.

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