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2012-058 BWR Vessel & Internals Project (BWRVIP)

February 29, 2012

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

Attention: Andrew Hon

Subject: Project No. 704 – Draft NRC Safety Evaluation of BWRVIP-138, Revision 1

References: Letter from John R. Jolicoeur (NRC) to Mr. David Czufin (BWRVIP Chairman), "Electric Power Research Institute Draft Safety Evaluation for Technical Report 1016574 BWRVIP-138, Revision 1: BWR [Boiling Water Reactor] Vessel and Internals Project Updated Jet Pump Beam Inspection and Flaw Evaluation Guidelines (TAC NO. ME2191)," dated January 30, 2012.

The NRC letter referenced above transmitted a draft NRC Safety Evaluation (SE) of the BWRVIP document entitled "BWRVIP-138, Revision 1: BWR Vessel and Internals Project, Updated Jet Pump Beam Inspection and Flaw Evaluation Guidelines" to the BWRVIP and requested the BWRVIP to identify any proprietary information in the draft SE.

The BWRVIP has reviewed the draft SE and has determined that there is EPRI proprietary information in the draft SE. Enclosed are two copies of a proprietary version of the draft SE with EPRI proprietary information identified with yellow shading and the letters TS in the margin indicating the information is considered "Trade Secrets" in accordance with 10CFR2.390. A letter requesting that the proprietary enclosure be withheld from public disclosure and an affidavit describing the basis for withholding this information are provided as Attachment 1.

Two copies of a non-proprietary version of the draft SE are also enclosed. This non-proprietary version is identical to the enclosed proprietary version except that the proprietary information has been deleted.

The BWRVIP has only one comment regarding factual errors or clarity concerns in the draft SE. Section 4.1.1 of the draft SE states: "As discussed in Section 3.1 of this SE, Table 2.1 of the submittal should be revised to reflect..." However, Section 3.1 of the draft SE does not discuss Table 2.1 of the submittal. Section 4.1.1 or Section 3.1 of the draft SE should be revised to correct the misstatement in the current Section 4.1.1.

Extra copy sent to PM

G004
NRR

If you have any questions on this subject please call Randy Schmidt (PSEG Nuclear, BWRVIP Assessment Committee Technical Chairman) at 856.339.3740.

Sincerely,

Dave Czufin
Exelon
Chairman, BWR Vessel and Internals Project

ELECTRIC POWER
RESEARCH INSTITUTE

Attachment 1

February 24, 2012

Document Control Desk
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Request for Withholding of the following Proprietary Document:

Draft Safety Evaluation by the Office of Nuclear Reactor Regulation for Technical Report TR-1016574, "BWRVIP-138, Revision 1: Boiling Water Reactor Vessel and Internals Project, Updated Jet Pump Beam Inspection and Flaw Evaluation Guidelines" (TAC NO. ME2191)

To Whom It May Concern:

This is a request under 10 C.F.R. §2.390(a)(4) that the U.S. Nuclear Regulatory Commission ("NRC") withhold from public disclosure the information identified in the enclosed Affidavit consisting of the proprietary information owned by Electric Power Research Institute, Inc. ("EPRI") identified above (the "Report"). Proprietary and non-proprietary versions of the Correspondence and the Affidavit in support of this request are enclosed.

EPRI desires to disclose the Report in confidence to assist the NRC. The Report is not to be divulged to anyone outside of the NRC or to any of its contractors, nor shall any copies be made of the Report provided herein. EPRI welcomes any discussions and/or questions relating to the information enclosed.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (704) 704-595-2630. Questions on the content of the Report should be directed to ***Randy Stark*** of EPRI at (650) 855-2122.

Sincerely,

Steven M. Swilley
Senior Business Operations Manager, Nuclear

Attachment(s)

cc: Sheldon Stuchell, NRC (sheldon.stuchell@nrc.gov)

ELECTRIC POWER RESEARCH INSTITUTE

AFFIDAVIT

RE: Request for Withholding of the Following Proprietary Document:

Draft Safety Evaluation by the Office of Nuclear Reactor Regulation for Technical Report TR-1016574, "BWRVIP-138, Revision 1: Boiling Water Reactor Vessel and Internals Project, Updated Jet Pump Beam Inspection and Flaw Evaluation Guidelines" (TAC NO. ME2191)

I, Steven M. Swilley, being duly sworn, depose and state as follows:

I am the Senior Business Operations Manager for the Nuclear Sector at Electric Power Research Institute, Inc. whose principal office is located at 3420 Hillview Avenue, Palo Alto, CA 94304 ("EPRI") and I have been specifically delegated responsibility for the above-listed Report that is sought under this Affidavit to be withheld (the "Report"). I am authorized to apply to the U.S. Nuclear Regulatory Commission ("NRC") for the withholding of the Report on behalf of EPRI.

EPRI requests that the Report be withheld from the public on the following bases:

Withholding Based Upon Privileged And Confidential Trade Secrets Or Commercial Or Financial Information:

a. The Report is owned by EPRI and has been held in confidence by EPRI. All entities accepting copies of the Report do so subject to written agreements imposing an obligation upon the recipient to maintain the confidentiality of the Report. The Report is disclosed only to parties who agree, in writing, to preserve the confidentiality thereof.

b. EPRI considers the Report and the proprietary information contained therein (the "Proprietary Information") to constitute trade secrets of EPRI. As such, EPRI holds the Report in confidence and disclosure thereof is strictly limited to individuals and entities who have agreed, in writing, to maintain the confidentiality of the Report. EPRI made a substantial economic investment to develop the Report, and, by prohibiting public disclosure, EPRI derives an economic benefit in the form of licensing royalties and other additional fees from the confidential nature of the Report. If the Report and the Proprietary Information were publicly available to consultants and/or other businesses providing services in the electric and/or nuclear power industry, they would be able to use the Report for their own commercial benefit and profit and without expending the substantial economic resources required of EPRI to develop the Report.

c. EPRI's classification of the Report and the Proprietary Information as trade secrets is justified by the Uniform Trade Secrets Act which California adopted in 1984 and a version of which has been adopted by over forty states. The California Uniform Trade Secrets Act, California Civil Code §§3426 – 3426.11, defines a "trade secret" as follows:

"'Trade secret' means information, including a formula, pattern, compilation, program device, method, technique, or process, that:

(1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and

(2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy."

d. The Report and the Proprietary Information contained therein are not generally known or available to the public. EPRI developed the Report only after making a determination that the Proprietary Information was not available from public sources. EPRI made a substantial investment of both money and employee hours in the development of the Report. EPRI was required to devote these resources and effort to derive the Proprietary Information and the Report. As a result of such effort and cost, both in terms of dollars spent and dedicated employee time, the Report is highly valuable to EPRI.

e. A public disclosure of the Proprietary Information would be highly likely to cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Proprietary Information both domestically and internationally. The Proprietary Information and Report can only be acquired and/or duplicated by others using an equivalent investment of time and effort.

I have read the foregoing and the matters stated herein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 1300 W WT Harris Blvd being the premises and place of business of Electric Power Research Institute, Inc.

Date: 2/24/2012

Steven M. Swilley

(State of North Carolina)
(County of Mecklenburg)

Subscribed and sworn to (or affirmed) before me on this 24th day of February, 2012 by Steven M. Swilley, proved to me on the basis of satisfactory evidence to be the person(s) who appeared before me.

Signature Deborah H. Rouse (Seal)

My Commission Expires 2nd day of April, 2016

Draft Safety Evaluation by the Office of Nuclear Reactor Regulation for Technical Report TR-1016574, "BWRVIP-138, Revision 1 Boiling Water Reactor Vessel and Internals Project, Updated Jet Pump Beam Inspection and Flaw Evaluation Guidelines" (TAC NO. ME2191)

(Non-Proprietary Version)

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR TECHNICAL REPORT TR-1016574, "BWRVIP-138, REVISION 1 BOILING WATER

REACTOR VESSEL AND INTERNALS PROJECT,

'UPDATED JET PUMP BEAM INSPECTION AND FLAW EVALUATION GUIDELINES'"

(TAC NO. ME2191)

1.0 INTRODUCTION

1.1 Background

By letter dated March 9, 2009, the Boiling Water Reactor (BWR) Vessel and Internals Project (BWRVIP) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review and approval of Electric Power Research Institute (EPRI) Technical Report (TR) 1016574, "BWRVIP-138, Revision 1 'Updated Jet Pump Beam Inspection and Flaw Evaluation Guidelines'" dated December 2008. This submittal was supplemented by letter dated May 17, 2011 (ADAMS Accession No. ML11139A150), in response to the staff's request for additional information (RAI) dated June 2, 2010 (ADAMS Accession No. ML101230464).

The goal for BWRVIP-138, Revision 1 (referred to as the submittal) was to develop an updated inspection strategy for Jet Pump Beams (JPBs) to assure continued integrity of jet pump safety functions and to maintain the design basis. The recommendations included in the submittal supersede previously documented BWRVIP guidance for inspection of JPBs contained in preceding versions of BWRVIP-41, "BWR Vessel and Internals Project, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines", guidance for inspection of the remaining jet pump components can be found in preceding versions of BWRVIP-41 and remain unaffected. The BWRVIP-41 report did not provide sufficient detail on the key design parameters, heat treatments and stress state, or the new General Electric beam design. In addition, the BWRVIP-41 report did not cover the effect of water chemistry, and a new region of the beam has been shown to fail due to intergranular stress corrosion cracking (IGSCC) in the earlier design.

In this submittal, BWRVIP compiled and evaluated information on JPB design and configurations, field experience with IGSCC in the three different regions of the beam, and inspection capabilities. BWRVIP performed stress and fracture mechanics analyses to establish the flaw tolerance of the designs currently installed in the BWR fleet, which in turn was used to demonstrate the effectiveness of nondestructive evaluation (NDE) techniques and for establishing appropriate inspection intervals. The two types of beams that are currently installed in the BWR fleet were considered for both normal water chemistry (NWC) and hydrogen water chemistry (HWC) environments.

1.2 Purpose

The staff reviewed the submittal and the supplemental information to determine whether the guidance in the document provides acceptable levels of quality for flaw inspection and evaluation (I&E) of the JPBs. The review considered the past service history, the potential degradation mechanism, consequences of failure, and the ability of the proposed inspections to detect degradation in a timely manner.

1.3 Organization of this report

Although the BWRVIP-138, Revision 1 TR is proprietary, this SE was written not to repeat proprietary information contained in the report. The staff does not discuss, in any detail, the provisions of the guidelines nor the parts of the guidelines that it finds acceptable. A brief summary of the contents of the submittal is given in Section 1.4 of this safety evaluation (SE,) with the evaluation presented in Sections 2.0 - 4.0. The conclusions are summarized in Section 5.0.

1.4 Summary of BWRVIP-138, Revision 1 TR

The submittal discusses the following topics:

Section 1: Introduction and Background – provides a brief discussion of the history of JPB failures, states the objectives and scope of the submittal, and lists the installation status for the JPBs as of June 1, 2004, in the BWR fleet. This report supersedes guidance contained in previously documented versions of BWRVIP-41 related to JPBs.

Section 2: Beam Susceptibility – explains the design/manufacturing history of JPBs, makes the point that all of the beams that have experienced IGSCC problems have been in the equalized and aged (EQA) condition, and states that all of the EQA beams have been replaced as of June 1, 2004. All of the beams currently used in the plants were manufactured with the high temperature anneal and aged (HTA) heat treatment instead of the EQA treatment. This section also describes the analysis for a conservative estimation of IGSCC initiation life that are used for justification for the baseline inspection intervals.

The recommended baseline inspection intervals are shown in Table 1 and are good for both NWC and HWC conditions. The recommendations are based on the prediction of crack initiation in a Group 2 beam, which is a function of the maximum stress in the beam, supported by data obtained under NWC conditions. No credit was taken for any potential increase in the time required to initiate a crack in the HWC environment. While effective HWC is expected to increase the initiation time, at this point, the effect of HWC on crack initiation is difficult to quantify. The statistical analysis and inspection recommendations were originally included in

BWRVIP-41, Revision 1 (released in September of 2005) and have been used by the nuclear power industry since that time.

Table 1. Recommendations for Baseline Inspection Intervals.

	Group 2	Group 3
BB-1 & BB-2		
BB-3		

Section 3: Field Experience – describes the IGSCC failures of JPBs, which were all beams with the EQA heat treatment. The earlier HTA-treated beams (referred to as the Group 2 beams) are dimensionally identical to EQA-treated beams that had experienced failures (referred to as Group 1 beams in the report). The excellent field experience with the Group 2 beams reflects the improved resistance to IGSCC of the HTA treatment, some of which have been in service since 1979. The later HTA-treated, Group 3 beams were made thicker in the center and the ends to reduce the applied stresses, thereby increasing the safety margins against IGSCC.

Section 4: Inspection Regions – defines the IGSCC-susceptible regions of the JPBs and those inspection techniques that will be used to monitor the beams for cracking. For the thinnest section around the bolt hole (BB-1 location), inspection is performed with ultrasonic testing (UT) at from an axis perpendicular to the beam axis, on both sides of the beam's center bolt hole. This is consistent with the observed field cracking pattern and the peak applied stress field. The UT inspection region for the BB-2 location is the transition region at each end of the beam, covering the entire radius region and extending to the hold down locations. For the BB-3 location, enhanced visual inspection (EVT-1), which can detect the fine cracking associated with IGSCC initiation, is recommended to cover the tapered region between BB-1 and BB-2 regions with the exception of the center of the top surface.

Section 5: Alloy X-750 Crack Growth Rate – describes the IGSCC crack growth rate (CGR) for the fracture mechanics evaluation of the JPBs. The data are limited with the measurements made under high oxygen conditions in laboratory tests of compact tension specimens with a relatively high applied stress intensity factor, K (minimum applied K was). The report makes use of the extensive understanding of the IGSCC behavior of austenitic stainless steels and Alloy 600 to benchmark the limited Alloy X-750 data and to demonstrate the significant benefit imparted to austenitic stainless steels by the effective HWC environment that is present in many of the operating BWRs. The data for other austenitic alloys suggest a factor of reduction in CGR for HWC conditions compare to NWC conditions. For this report, a factor of reduction in CGR for the Alloy X-750 beams was assumed for all of the CGR calculations in HWC compared to NWC conditions.

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Section 6: Flaw Evaluation Methodology – summarizes the stress analysis and fracture mechanics evaluation of JPBs with different postulated flaw locations, nominally associated with the three inspection locations. The following items are key points to note:

- Analyses of the stresses from the preload percentile of the maximum value, assumed to act for the life of the beam) were performed for each of the two designs with different crack locations.

- A compliance analysis on the Group 2 beam design shows that for most crack locations, there is no significant relaxation in load until the cracks grow significantly. Therefore, no load reduction was considered for the CGR analyses discussed in the next section.

- The allowable flaw size is calculated based on the hydraulic load with a safety factor of The calculation assumes that the final failure is due to plastic collapse in shear of the un-cracked ligament. Field experience from failed beams support the use of plastic collapse in shear as the failure method for the JPB.

- The results of CGR calculations, using linear elastic fracture mechanics (LEFM), for each design/crack location start from a small initial assumed flaw size such that the residual life of the beam, given any detectable flaw in the beam, could be determined from the results of the calculation.

Section 7: Flaw Acceptance and Re-inspection Criteria – describes the crack growth predictions and proposed inspection intervals for JPBs. For the Group 2 and 3 beams, the dates for the 1st (baseline) recommended inservice inspection (ISI) were chosen based on results described in Section 2 of the submittal.

The dates of subsequent ISI intervals depend on the new stress analysis and the CGR calculations from Section 6; the summary of the re-inspection intervals is shown below in Tables 2 and 3.

Table 2. Recommendations for Group 2 JPB Re-inspection Intervals.

	NWC	HWC
BB-1 & BB-2		
BB-3		

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Table 3. Recommendations for Group 3 JPB Re-inspection Intervals.

	NWC	HWC
All locations		

For the more aggressive NWC environment, the calculations use the best estimate of the upper-bound on the CGR data in Section 5. The specific conservatisms included in the selection of inspection intervals are:

- choosing an "initial" flaw size significantly larger than the detection capabilities of current NDE methods,
- for BB-1 and BB-2 locations, the re-inspection intervals are the same, based on the shortest predicted life from the CGR calculations at either of the two locations, and
- when the predicted life from the CGR calculations is greater than , the re-inspection interval is limited to .

For the less aggressive HWC environment, the submittal proposes longer re-inspection intervals than for the NWC environment, but less than would be supported by the CGR calculation results to retain additional conservatism in the approach.

Appendix A: Details of the Group 2 JPB analysis are presented, which included the compliance analysis as well as the stress and crack growth analyzes. The CGR calculations were performed on different combinations of possible crack plane and crack geometry (corner crack or center crack) in order to identify the limiting case for each JPB region (BB-1, BB-2 and BB-3). The limiting crack locations represent the critical inspection points of the JPB. A six year re-inspection frequency for the NWC environment was selected as a starting point.

For the BB-1 location and in the NWC environment, the limiting crack configuration is identified as a corner crack; smaller initial flaws would not cause failure within the re-inspection interval. The limiting crack size for the given interval is considered to be detectable with the procedures described in Section 4; therefore, a re-inspection interval should provide a reasonable opportunity to detect an existing IGSCC crack in a Group 2 HTA JPB.

In the HWC environment, the evaluation would predict failure from IGSCC after of service; however, a limit of was selected for the BB-1 and BB-2 locations to retain additional conservatism in the approach.

The analysis demonstrates that the stress in the BB-3 region is much lower in magnitude than for the BB-1 or BB-2 regions and thus supports much longer re-inspection intervals than

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proposed for the BB-1 and BB-2 locations. Given these factors, the submittal recommends a maximum inspection interval of for the NWC environment and for the HWC environment, which are considered conservative and technically justified.

Appendix B: Details of the Group 3 JPB analysis are presented, which included the compliance analysis as well as the stress and crack growth analyzes. The CGR calculations were performed on different combinations of possible crack plane and crack geometry (corner crack or center crack) in order to identify the limiting case for each JPB region (BB-1, BB-2 and BB-3). The limiting crack locations represent the critical inspection points of the JPB. An re-inspection frequency for the NWC environment was selected as a starting point.

For the BB-2 location and in the NWC environment, the limiting crack configuration is identified as a center crack; smaller initial flaws would not cause failure within the re-inspection interval. The limiting crack size for the given interval is considered to be detectable with the procedures described in Section 4; therefore, an re-inspection interval should provide a reasonable opportunity to detect an existing IGSCC crack in a Group 3 HTA JPB.

In the HWC environment, the evaluation would predict failure from IGSCC after of service; however, a limit of was selected for the BB-1 and BB-2 locations to retain additional conservatism in the approach.

For the BB-3 location, the analysis demonstrates that the flaw tolerance is only slightly higher than that for the BB-1 and BB-2 locations; the section size at the BB-3 location is smaller than the size at the same location in the Group 2 JPB. Given these factors, the submittal recommends the same re-inspection intervals (for NWC and for HWC) for all three locations, which is conservative and can be technically justified. TS

2.0 REGULATORY EVALUATION

The BWRVIP guidance regarding JPB inspections is a voluntary program pursued by industry in order to address past failures in BWR units. These failures could cause significant damage to surrounding internal components, but do not represent a reactor safety issue. If a failure occurs, existing jet pump operability surveillance procedures required by plant Technical Specifications will detect a beam failure and shut down the reactor. The purpose of this inspection guideline is to avoid mid-cycle failures and possible damage to reactor internals. The creation of the BWRVIP was, at least in part, motivated by a desire to demonstrate that no increased specificity in NRC regulation for BWR internals aging management would be necessary.

3.0 TECHNICAL EVALUATION

The staffs' review is focused on the technical basis for the recommended initial inspection intervals as well as the re-inspection intervals. During its review of the submittal, the staff issued one RAI that addressed technical issues. The details of the staff's RAI and the corresponding responses are available in ADAMS. However, the staff did not include all the RAI questions and the responses in this SE; it included only those salient RAI questions and BWRVIP responses that address specific points of emphasis.

3.1 Baseline Inspection Interval

Section 2.4 of the submittal references the 1981 proprietary report that covers the testing that is the technical basis for setting the baseline inspection intervals for Groups 2 and 3 JPBs. The staff recognizes that the applied stress level is the main factor that will control the time to initiate an IGSCC crack and requested in RAI 2 that the BWRVIP summarize the maximum stress for each of the inspection locations in each beam design. The RAI response was included in the BWRVIP's May 17, 2011 letter. The salient information regarding IGSCC crack initiation from the RAI response and the original submittal are included below in Table 4. The applied loads for each design are the same and the highest stress is found in the BB-1 location for each beam design; the reported maximum stress in the BB-2 and BB-3 regions occurs at the boundary between the two regions, so therefore, the maximum stress is the same for both regions.

The staff also asked in RAI questions 4 and 5 for additional information on the crack initiation testing and the statistical evaluation used to justify the recommended initial baseline inspection interval. In its May 17, 2011 response, the BWRVIP summarized the crack initiation testing and stated that the American Society of Mechanical Engineers specifications and General Electric-Hitachi internal material design specifications have been used to procure the materials for the JPBs. Tight control on manufacturing and procurement are vital to minimize the effect of material property variability in the actual hardware on the conclusions drawn from the statistical evaluation. Additionally, a re-assessment of the ratio of applied stress to yield stress (referred to as the stress ratio) versus time to initiate IGSCC was performed which accounted for material property variability. The staff accepts these responses to the RAI questions and the issues related to the stress analysis, crack initiation testing, and statistical evaluation are considered resolved.

Table 4. Summary of Stress Analysis for Baseline Inspection Intervals.

	Group 2	Group 3
Max stress @ BB-1, ksi		
Max stress @ BB-2/BB-3, ksi		
Max applied stress / Yield stress++		
Statistical analysis for highest stress region		
Mean IGSCC initiation life # of years		
Lower bound (Mean – 3σ) # of years		

+ Does not include thermal relaxation of the preload.
++ Yield stress at 550°F is a minimum of .

3.1.1 Group 2 Beams

The recommended baseline inspection interval is based on the stress ratio of the higher stressed BB-1 region. The statistical evaluation showed that the mean time to failure for the maximum applied stress ratio is significantly greater than the recommended baseline interval, but when the scatter in results is considered (mean value minus 3σ), the time for initiation is slightly less than the baseline interval ( ). The baseline intervals for the BB-2 and BB-3 regions are less than the minimum (mean value minus 3σ) time to failure, given the maximum stress in these regions, and provide additional conservatism in the re-inspection frequencies.

For the BB-1 region, several factors need to be considered when comparing the discrepancy between the result from the analysis and the maximum recommended interval length ( ). The investigators noted that the crack initiation testing was performed in an environment with high oxygen ( ) and high conductivity water chemistry. The staff agrees that the dissolved oxygen (DO) environment is significantly more aggressive than typical NWC where the DO content could vary from (Reference 1). The staff also notes that the IGSCC would have to grow a significant distance through the cross section after initiation before failure could occur. In the limited number of field failures of the EQA-treated JPBs, the beams demonstrated significant flaw tolerance before failure. Finally, the stress ratio

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at the BB-1 location is less than the recommended [Reference 2] found in the current version of BWRVIP-84 (Reference 2). Finally, the staff notes that Group 2 JPBs have been used in 31 BWRs in the United States, some for as long as 30 years, without any confirmed cases of IGSCC.

Taking all of these factors into consideration, the staff agrees that the maximum recommended baseline inspection intervals for the Group 2 beams (for the BB-1 and BB-2 locations and for the BB-3 locations) are acceptable.

3.1.2 <u>Group 3 Beams</u>

In Table 1, the stress analysis showed that the latest beam design does reduce the overall stresses in the beam; the highest stressed region is still around the center bolt hole (BB-1), but now the maximum stress in the rest of the beam (found at the boundary between regions BB-2 and BB-3) is less. The recommended baseline inspection interval in Table 1 for all three regions ( ) is considerably less than the results from the statistical evaluation ( ). The staff notes that Group 3 JPBs have been used in 6 BWRs in the United States without any confirmed instances of IGSCC; the longest service time is currently 10 years.

Based on the stress analysis and the May 17, 2011, response to RAIs, the staff agrees that the maximum recommended baseline inspection interval for the Group 3 beams for all locations) is acceptable.

3.2 Re-inspection Intervals

The recommended re-inspection intervals were based on the new crack growth calculations described in Section 6 of the submittal with the conservative assumptions summarized below:

1) there is a small, pre-existing flaw present,

2) this small flaw will grow in a stable fashion due to IGSCC,

3) the stable IGSCC crack growth ends when the remaining ligament fails due to plastic collapse of the remaining ligament.

The additional assumption that is made in the submittal is that the dependence of the CGR, da / dN, on the stress intensity factor K can be represented by a power law of the form:

$da / dN =$ Equation 1

where A for Alloy X-750 is a constant that depends on the water chemistry and n is . For the NWC environment, the value of A is assumed by the BWRVIP to be so that the

[1] BWRVIP-84 is in the process of being revised so that the maximum recommended stress ratio will be for threaded components and for non-threaded components like the JPBs. TS

curve will be an upper bound to the limited, existing test data. For the HWC environment, there are no test results for Alloy X-750, so the value of A is assumed to be a factor of less than that for NWC ; data for other austenitic alloys suggest a factor of reduction in CGR in the HWC environment compared to that for NWC.

In the submittal, the BWRVIP has compared the limited measured CGR for X-750 to the measured CGR for Alloy 600 and austenitic stainless steels (small range of applied K) as a function of yield strength (YS); the measured data is also compared to theoretical modeling of SCC to suggest that the effect of YS is consistent with the theory. In RAI 7, the staff has questioned why the YS dependence is important and if the theory supports the applied K dependence of CGR used in the submittal. In its May 17, 2011, response, the BWRVIP reviewed in more detail the comparisons made in the submittal and concludes that because the available data over the limited range of applied K values are similar, then the proposed CGR curves for Alloy X-750 are acceptable at this time; however, the BWRVIP has a project underway to measure CGRs on a representative heat of Alloy X-750 and plans to communicate those results to the NRC when they become available.

To assess the adequacy of the proposed CGR curves for Alloy X-750, the staff has reviewed the available data, and the relationship of the proposed CGR curves to other NRC-approved approaches to make conservative estimates of CGRs that are relevant to the re-inspection intervals in the submittal. In this case, the minimum assumed detected flaw size is deep and the critical flaw sizes for plastic collapse is about deep, which translates into an applied K that ranges from about

The staff has summarized in Figure 1 the CGRs for NWC from two other sources to compare with the assumed crack growth trends with those found in this submittal. The trend proposed in the submittal for Alloy X-750 has a lower absolute value than that which is used as an upper bound for sensitized pipe welds made from austenitic stainless steels (Reference 4); the trend for Alloy X-750 is equal to (at applied K) or higher than (at applied K) that used for Alloy 600 and 182 welds in BWR internals (Reference 5). Given the BWRVIP response to RAI 7 and the information in Figure 1, the staff concludes that the proposed CGR curve for Alloy X-750 in the NWC environment may not represent an absolute upper-bound to the CGR, but does provide for a reasonable extrapolation to lower applied K values and, at this point in time, is acceptable to use in the LEFM calculations. If the results of the supplemental CGR testing on a representative heat of Alloy X-750 that the BWRVIP is currently conducting do not confirm the assumed X-750 CGRs, then the BWRVIP should re-evaluate the proposed re-inspection intervals using the results of the supplemental testing. This resolves the issues in TS
RAI 7.

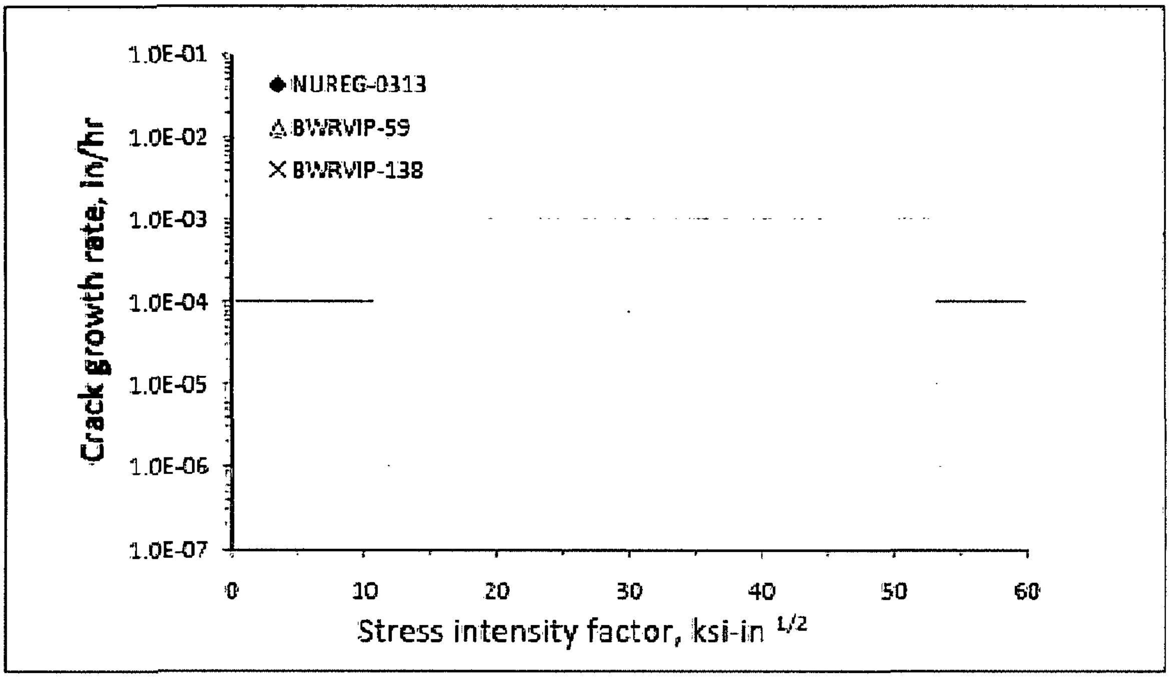

Figure 1. CGR in NWC environment vs. applied K trends from different sources.

The proposed reduction of CGR for the HWC environment (factor of reduction in the value of A in Equation 1) should be recognized as being dependent upon plant-specific factors. To take credit for an "effective" HWC environment, the staff considers the factor of reduction in the CGR acceptable when the plant operators follow the guidance included in BWRVIP-62 (Reference 6) and meet all of the conditions listed in the SE (Reference 7).

3.2.1 <u>Group 2 JPBs</u>

The proposed intervals are based on the crack growth analysis detailed in Appendix A of the submittal. The staff reviewed Appendix A and noted that the LEFM crack growth analysis, which detemines how fast the crack will grow, and the limit-load analysis of plastic collapse, which controls how far the crack will grow, are based on different loads. In RAI 10, the staff asked for clarification on why the crack growth should be based solely on the preload while the plastic collapse should be based on the hydraulic load. In its May 17, 2011, response, the BWRVIP described the assumptions that were used. For crack growth, the maximum load that the beam would see in service is assumed to be the higher of the two loads (Table 1 in Appendix A of the submittal shows the bolt preload and the hydraulic load), which is the bolt preload. For plastic collapse to occur, the analysis assumes that the beam would experience significant distortion, which would indicate that the preload is relieved and only the hydraulic load would remain. The staff finds that the BWRVIP has adequately clarified the assumptions controlling crack growth and the issue is resolved. TS

The staff reviewed the specific recommendations for Group 2 JPBs, which are summarized in Table 2 of this SE. The recommended re-inspection interval for the BB-1 region is based on the limiting case of crack growth from a corner crack in the BB-1 region. The recommended interval for the BB-2 region is slightly less than the results from the calculations; for the BB-3 region, the recommended interval is considerably less than the calculated service life see Table 7 in Appendix A of the submittal) because the stress is significantly lower at this location and represents part of the additional conservatism in the recommendations.

For the HWC environment, the results of the crack growth calculations at the BB-1 location would predict a remaining service life of (a factor of greater). The BWRVIP has recommended re-inspection in for the BB-1 and BB-2 locations, not taking full credit for the reduced CGR in the recommended re-inspection intervals to provide additional conservatism to the approach. Again, for the BB-3 location, the BWRVIP recommends re-inspection after , much less than predicted from the CGR results in the HWC environment.

The staff performed similar, independent calculations of crack growth for the limiting case and verified the acceptable flaw tolerance performance for the recommended interval under NWC conditions with the CGR curve used in the submittal; however, the calculations are sensitive to the exact CGR curve used. If the more conservative CGR curve for NWC from NUREG-0313 were used, the staff's calculations would predict failure in approximately 3.6 years rather than reduction in service life. Given the limited data available for Alloy X-750 in the HTA heat treated condition, the staff finds the recommended re-inspection intervals for Group 2 beams acceptable; however, additional test results in the range of applied K are advisable to verify the acceptability of the proposed CGR curve for Alloy X-750 in the HTA heat treated condition.

3.2.2 Group 3 JPBs

The proposed intervals are based on the crack growth analysis detailed in Appendix B of the submittal. The staff reviewed Appendix B and noted that the details are similar to those noted in Appendix A for the Group 2 JPBs.

The staff reviewed the specific recommendations for Group 3 JPBs, which are summarized in Table 3. The recommended re-inspection interval for all three regions is based on the limiting case of crack growth from a center crack in the BB-2 region. The recommended intervals for the BB-1 and BB-3 regions are slightly less than the results from the CGR calculations, which adds to the overall conservatism in the recommendations.

The results of the crack growth calculations for the HWC environment predict a factor of greater in the remaining service life. Again, the BWRVIP has chosen to not take full credit for the reduced CGR in the recommendations to provide additional conservatism to the approach. TS

The staff finds the recommended re-inspection intervals for Group 3 beams acceptable; however, as stated above in the case of the Group 2 beams, additional test results in the range of applied K are advisable to verify the acceptability of the proposed CGR curve for Alloy X-750 in the HTA heat treated condition. TS

3.3 Adequacy of the Overall Approach to I&E Guidelines

In the course of the staff's review of the submittal, two general questions were raised in the RAI regarding the scope and focus of the report. RAI 1 related to the effect of the surface condition on initiation of IGSCC. The initial Group 2 beams were manufactured from closed-die forgings, resulting in a combination of machined and as-forged surfaces on the installed beams. Most Group 2 beams manufactured after 1994 were supplied as open-die forgings and as a result were machined on all surfaces, removing any as-forged surfaces; Group 3 beams have always been manufactured from open-die forgings with all surfaces machined. In its May 17, 2011, response, the BWRVIP provided a thorough discussion of the effects of surface condition and surface films on the general corrosion and the initiation of IGSCC in Alloy X-750. The BWR industry, through the BWRVIP, has addressed these issues related to surface condition by publishing BWRVIP-84, which includes Alloy X-750 specifically and provides detailed manufacturing guidance. The staff has reviewed the response to RAI 1 and finds it acceptable because the proposed inspection schedule and inspection methodologies, along with the controls of the manufacturing process will provide adequate assurance that the surface condition of any Group 2 or Group 3 JPB will not adversely affect the performance of the beam.

The second general question, RAI 11, from the NRC staff was related to the neutron radiation conditions in the vicinity of the JPB and whether irradiation-assisted stress corrosion cracking (IASCC) was a potential aging mechanism that should be considered. In its May 17, 2011, response, the BWRVIP provided a detailed account of testing that they have done to characterize the behavior of Alloy X-750 as a function of the neutron radiation conditions. They summarized the work by stating that IASCC is not considered a potential aging mechanism for the Groups 2 and 3 beams. The staff has reviewed the information provided in the May 17, 2011, response and similar crack growth related issues in Ni-based alloys used in BWR internal components (BWRVIP-59). The staff agrees that the neutron radiation levels for these Alloy X-750 beams are low compared to other internal components and the impact of fluence on the JPBs would be limited; therefore, IASCC is not a significant issue for X-750 JPBs in the HTA heat treated condition.

3.4 Summary of Staff's Evaluation

Based on the review of the available information in the submittal, the May 17, 2011, response to the staff's RAIs, the extensive history of inspections on Group 2 and 3 JPBs, and the staff's independent calculations, the staff finds that the recommended inspection intervals are acceptable.

4.0 CONDITIONS/LIMITATIONS AND LICENSEE PLANT-SPECIFIC ACTION ITEMS

Based on its review, the staff identified two issues and concerns in Section 3.0 of this SE that were not adequately addressed regarding the implementation of the submittal. The conditions and limitations address deficiencies in the submittal are identified in Section 4.1 of this SE. One plant-specific action item that addresses the implementation of the submittal is identified in Section 4.2 of this SE.

4.1 Limitations and Conditions on the Use of BWRVIP-138, Revision 1

4.1.1 Condition 1.

As discussed in Section 3.1 of this SE, Table 2.1 of the submittal should be revised to reflect the May 17, 2011, response to RAI 2 in the approved (-A) version of this TR .

4.1.2 Condition 2.

As discussed in Section 3.2 of this SE, the staff requests that the BWRVIP reassess the proposed CGR curve for the applied K range of . If the results of the supplemental TS CGR testing on a representative heat of Alloy X-750 that the BWRVIP is currently conducting are not bound by the assumed X-750 CGRs, then the BWRVIP should re-evaluate the proposed re-inspection intervals using the results of the supplemental testing.

4.2 Licensee Plant-Specific Action Items for the Use of BWRVIP-138, Revision 1

As discussed in Section 3.2 of this SE, the staff requests that to take credit for an "effective" HWC environment and the longer re-inspection intervals described in the submittal, the plant operators must follow the guidance included in BWRVIP-62 (Reference 6) and meet all of the conditions listed in the SE.

5.0 CONCLUSIONS

The staff has reviewed the BWRVIP-138, Revision 1, TR and the supplemental information that was transmitted in the BWRVIP letter on May 17, 2011. The staff found that the TR, as clarified to incorporate the BWRVIP responses to RAI questions 2 and 8-11, provides an acceptable technical justification for the proposed I&E of the JPBs manufactured from Alloy X-750 in the HTA heat treated condition.

As described previously in Section 3.2 of this SE, should the results of the supplemental testing not confirm the assumed X-750 CGRs, then the BWRVIP would need to re-evaluate the proposed re-inspection intervals using the results of the supplemental testing.

6.0 REFERENCES

1. H.-P. Seifert and S. Ritter, in Proceedings of the 12th International Conference on Environmental Degradation of Materials in Nuclear Power System – Water Reactors – Edited by T.R. Allen, P.J. King, and L. Nelson, TMS, 2005.

2. BWR Vessels and Internals Project, "Guidelines for Selection and Use of Materials For Repairs to BWR Internal Components, (BWRVIP-84)" October 2000.

3. W.S. Hazelton, W.H. Koo, NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," 1988.

4. BWR Vessels and Internals Project, "Evaluation of Crack Growth in BWR Nickel Base Austenitic Alloys in RPV Internals (BWRVIP-59-A)," May 2007.

5. BWR Vessels and Internals Project, "Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection (BWRVIP-62)," December 1998.

6. NRC Safety Evaluation by the Office of Nuclear Reactor Regulation, Boiling Water Reactor (BWR) Vessel and Internals Project (BWRVIP) Report TR-108705 (BWRVIP-62): "Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection," July 21, 2009.

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