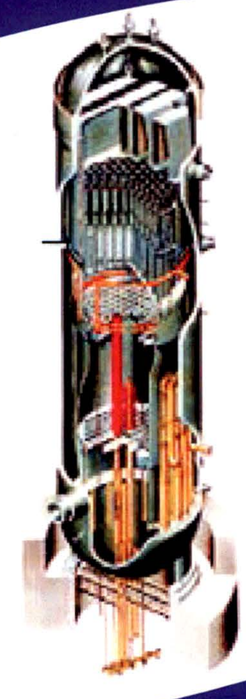
ELECTRIC POWER RESEARCH INSTITUTE

2017 TECHNICAL REPORT

BWRVIP-234NP-A: BWR Vessel and Internals Project

Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steels for BWR Internals



PREPARED UNDER THE NUCLEAR QA PROGRAM

LICENSED MATERIAL

BWRVIP-234NP-A: BWR Vessel and Internals Project

Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steels for BWR Internals

3002010550NP

Final Report, November 2017

EPRI Project Manager
R. Carter

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NRC SAFETY EVALUATION

In accordance with an NRC request, the NRC Safety Evaluation immediately follows this page. Other pertinent NRC and BWRVIP correspondence are included in appendices.

BWRVIP 2016-068A

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

June 22, 2016

Mr. Tim Hanley
Senior Vice President West Operations, Exelon
Chairman, BWR Vessel and Internals Project
3420 Hillview Avenue
Palo Alto, CA 94304-1395

SUBJECT: FINAL SAFETY EVALUATION OF THE BWRVIP-234: THERMAL AGING AND NEUTRON EMBRITTLEMENT EVALUATION OF CAST AUSTENITIC STAINLESS STEEL FOR BWR INTERNALS (TAC NO. ME5060)

Dear Mr. Hanley:

By letter dated September 10, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession Package No. ML102570749), the Boiling Water Reactor (BWR) Vessel Internals Project (BWRVIP) submitted the topical report (TR) "BWR Vessel and Internals Project: Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steels for BWR Internals (BWRVIP-234)." The original submittal was supplemented by three letters from the BWRVIP, dated September 18, 2012, May 23, 2014, and March 9, 2015 (ADAMS Accession Nos. ML12265A078, ML14174A841, and ML15334A267), in response to requests from the U.S. Nuclear Regulatory Commission (NRC) for additional information.

By letter dated March 8, 2016 (ADAMS Package Accession No. ML13044A154), an NRC draft safety evaluation (SE) was provided for your review and comment. By letter dated April 4, 2016 (ADAMS Accession No. ML16126A196), Electric Power Research Institute (EPRI) provided comments on the NRC draft SE. The comments provided by EPRI were related to clarifications and accuracy. No proprietary information was identified in the draft SE

The NRC staff has found that TR BWRVIP-234 is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that EPRI publish an approved version of TR BWRVIP-234 within six months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information (RAIs) and your responses. The approved versions shall include an "-A" (designating approved) following the TR identification symbol.

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TRs provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

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

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

Sincerely,

Kevin Hsueh

Kevin Hsueh, Chief
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 704

Enclosure:
Final SE

SAFETY EVALUATION OF THE BWRVIP-234 REPORT

"BWR VESSEL AND INTERNALS PROJECT: THERMAL AGING AND NEUTRON EMBRITTLEMENT EVALUATION OF CAST AUSTENITIC STAINLESS STEEL FOR BWR INTERNALS (BWRVIP-234)"

(TAC NO. ME5060)

1.0 INTRODUCTION

1.1 Background

By letter dated September 10, 2010, the Boiling Water Reactor (BWR) Vessel and Internals Program (BWRVIP) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review and approval the Electric Power Research Institute (EPRI) proprietary version of report TR-1016574 (BWRVIP-234, Ref. 1). This report was supplemented by letters dated September 18, 2012, May 23, 2014, and March 9, 2015 (Refs. 2, 3, and 4) in response to the NRC staff's request for additional information (RAI) questions. Additional background information was provided by a letter dated November 19, 2015 (Ref. 5).

The purpose of the BWRVIP-234 topical report (TR) is to evaluate the material composition, fluence, stresses and the field experience of Cast Austenitic Stainless Steel (CASS) components in BWR applications, and recommend augmented inspections if needed. This TR will also address the technical basis for the evaluation of loss of fracture toughness due to the potential synergistic effects of thermal embrittlement (TE) and neutron irradiation embrittlement[a] (IE) on BWR reactor vessel internal (RVI) components manufactured from CASS materials, as recommended in NUREG-1801, Rev. 1, Section XI.M13 (Generic Aging Lessons Learned (GALL) Report)[b]. The guidelines in the GALL Report are essentially the same as described initially in a letter from the NRC staff to the nuclear industry commonly referred to as the Grimes letter (Ref. 6).

In this TR, the investigators compiled and evaluated information on embrittlement issues with CASS materials for the RVI components in the BWR environment. The investigators applied a systematic screening criteria, based on the significant factors affecting embrittlement, to the components that are typically manufactured from CASS materials in all operating BWRs. The recommendations for any augmented inspections are based on a given component's susceptibility to loss of fracture toughness due to the combined effects of TE and IE.

1.2 Purpose

The NRC staff reviewed the TR and the supplemental information that was submitted to the NRC staff to determine whether the guidance in the document provides acceptable levels of quality for evaluation of the internal BWR component manufactured from CASS materials.

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[a] Sometimes referred to simply as neutron embrittlement.

[b] BWRVIP-234 was submitted before GALL, Revision 2 (December 2011) was approved to replace GALL Revision 1. Revision 2 includes a similar description except that the screening criteria for augmented inspections are found in GALL, Rev. 2, Section XI.M9, BWR Vessel Internals.

The review considered the microstructure, irradiated fracture toughness, the neutron fluence, the applied stress on each component, and the ability of current BWRVIP inspections to detect degradation.

1.3 Organization of this report

Because the TR is proprietary, this safety evaluation (SE) was written not to repeat proprietary information contained in the report. The NRC 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 TR is given in Section 2 of this SE, with the evaluation of the submittal along with the significant RAI responses presented in Section 3. The conclusions are summarized in Section 5.

2.0 SUMMARY OF THE BWRVIP-234 REPORT

The TR discusses the following topics:

Section 1: Introduction and Background – provides a brief discussion of the objectives and scope for the report. The GALL Report, Revision 1, Section XI.M13 states that an American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, VT-3 examination is required to be performed of RVI components. In addition, the GALL Report specifies that for the license renewal period these inspections shall be augmented by an aging management program to address the synergistic effects of thermal aging and neutron embrittlement in components manufactured from CASS materials. This aging management program consists of (a) identifying susceptible components; and (b) either performing additional inspections of these components, or performing a component-specific evaluation to confirm that the stresses (tensile loading) in the components are sufficiently low such that augmented inspections are not warranted.

This TR is for information only and the implementation requirements of Nuclear Energy Institute 03-08, "Guideline for the Management of Materials Issues," are not applicable.

Section 2: Literature Review – examines the factors that influence the TE mechanism, and how the potential synergism between TE and IE has been addressed in the past.

Section 3: Materials and Environment for BWR Internals – describes the specific factors (chemical composition and neutron fluence) as they relate to the BWR fleet. The key findings in this review were as follows:

- The majority of the components are manufactured according to either ASME SA-351 or ASTM A351 grade CF8 material.
- No molybdenum (Mo) or niobium (Nb) was specified in the grades of castings used for internal BWR components in the USA,
- The minimum ferrite volume fraction was originally specified to be 8 percent, as calculated from the Schaeffler diagram; no maximum ferrite was specified.
- based on review of about 80 available certified material test records (CMTRs), the calculated ferrite content (using Hull's equations for chromium (Cr) and nickel (Ni) equivalent factors) of the CASS components varied between 3 and 19 percent with an average value of 10 percent, and

- the estimated neutron fluence values, based on 60 years of operation, for each of the 10 components potentially fabricated from CASS materials were calculated for a BWR/6, 218-inch plant at extended power uprate (EPU) conditions, which was bounding for 251-inch BWR/5 and BWR/4 plants.

Section 4: Component Screening – described several factors, based on previously approved NRC documents, that could screen out BWR CASS components from augmented inspection requirements. The factors are listed below:

1. Ferrite content,
2. Neutron fluence,
3. Fracture toughness,
4. Stress, and
5. Current BWRVIP inspections.

The BWRVIP noted that operating experience (OE) for components fabricated from CASS materials shows that intergranular stress corrosion cracking (IGSCC) in BWRs is not a problem. Even in type 308 stainless steel welds (which are duplex materials similar to CASS), crack initiation has been extremely rare except in a few cases where the ferrite content was less than 5 percent. CASS components in BWRs are required to contain a minimum of 8 percent ferrite. The report concludes that the ferrite levels by themselves are not adequate to exempt RVI components fabricated with CASS materials from augmented inspections.

For fracture toughness, the Grimes letter considers a J-integral value of 255 kJ/m^2 at a crack extension of 2.5 mm as an adequate basis for a screening criteria to ensure function for a safety-significant component like Class 1 piping. The BWRVIP suggests that the same criteria could be applied as a basis for screening criteria to BWR RVI components fabricated from CASS materials that are potentially susceptible to degradation from TE and IE.

The TR states that little experimental data exists for fracture toughness data of actual CASS components that are known to be susceptible to TE and IE. The BWRVIP has presented two methods to estimate the fracture toughness. The first method is based on the Z-factor correction as defined in the ASME Code, Section XI, Nonmandatory Appendix C (Ref. 7) for austenitic stainless steel pipe welds. The second approach is to use the recommended lower bound properties in BWRVIP-100-A (Ref. 8), based on testing of base metal and core shroud welds, to predict the J-integral vs. crack extension curve, evaluated at 2.5 mm extension. Both the Z-factor correction and the BWRVIP-100-A methods would predict a fracture toughness in excess of the fracture toughness screening level, indicating that no evaluation or inspection is needed to manage the loss of fracture toughness.

The GALL Report allows for the use of a component-specific evaluation to determine the stress on the component during ASME Code Level A, B, C, and D conditions. If the component is loaded in compression or less than 5 kilo pounds per square inch (ksi) in tension, the GALL Report considers that adequate to prevent brittle fracture.

The requirement to address the embrittlement of CASS materials subject to both TE and IE is based on the interpretation that augmented inspections are required unless the screening criteria described above are met. However, the BWRVIP program already requires inspection of many of the CASS components.

Section 5: Stress Evaluation – summarizes the results of stress analysis for the orificed fuel support (OFS), the core spray sparger nozzle (CSSN) elbows, the jet pump (JP) assembly, the JP restrainer bracket, and the low-pressure coolant injection (LPCI) coupling. The analyses presented demonstrate that the tensile stresses in the CSSN elbows are less than 5 ksi and therefore, there is little chance of brittle failure, and no augmented inspections are needed. The other components could have tensile stress greater than 5 ksi,

For the control rod guide tube (CRGT) bases, no analysis was done because the maximum neutron fluence was orders of magnitude below the fluence screening criteria so that no synergistic interaction between TE and IE is anticipated.

Section 6: Combined Assessment of CASS Components– describes an assessment of the need for additional analysis or augmented inspection based on the factors presented in Section 5 of this report. In all cases where there are existing BWRVIP inspections required, the report credits those existing inspections as sufficient to make any additional inspections redundant. The results are summarized in Table 6.1 of the TR.

Section 7: Conclusions - given that all of the CASS components covered by this report are either CF-3 or CF-8 grade castings with an estimated delta ferrite between 3 percent to 19 percent, the recommendations are based on three critical screening criteria: stress, projected neutron fluence for 60 EFPY, and the estimated J-integral value at 2.5 mm of crack extension. Based on the screening criteria and the current BWRVIP inspections, no further augmented inspections of the CASS components are recommended.

Appendix A: The appendix includes the available certified material test records (CMTRs) and the mean, calculated delta ferrite content for each heat that was evaluated.

3.0 EVALUATION

The NRC staff's review focused on the technical basis for the recommendation that no augmented inspections are needed to manage the aging of CASS for the loss of fracture toughness due to the combined effects of TE and IE for the CASS components found in BWR RVI components. During its review of the TR, the NRC staff issued two RAIs that addressed technical issues. The details of the NRC staff's RAIs and the corresponding responses are available in Agencywide Documents Access and Management System (ADAMS). However, the NRC staff did not include all the RAI questions and the BWRVIP's responses in this SE; it included only those salient RAI questions and BWRVIP responses that address specific points of emphasis concerning the potential for augmented inspections.

3.1 NRC Staff Evaluation of Ferrite Content Uncertainty

The NRC staff notes that the ferrite content is an important consideration for CF-3/CF-8 materials. A material with a low ferrite content could be susceptible to IGSCC (similar to a wrought austenitic stainless steel), while a material with a high ferrite content could be susceptible to loss of fracture toughness (similar to an irradiated ferritic steel). The susceptibility to IGSCC is a separate concern that is not covered by the scope of the TR and is not considered further in this SE. This section of the SE will consider how to incorporate the uncertainty associated with the prediction of ferrite level for RVI components fabricated from CF-3/CF-8 materials to retain adequate fracture toughness when subject to both TE and IE.

To estimate the delta ferrite content, the Ni and Cr equivalent equations from Hull are calculated from the heat-specific CMTR. The NRC staff was concerned that the discussion of ferrite in the TR does not accurately reflect the uncertainty associated with the prediction. Specifically, the last sentence in Section 3.4 of the TR states that "there is a 99.8 percent confidence that the ferrite level will be below the 20 percent ferrite limit." Therefore in RAI 7, the NRC staff asked the BWRVIP to discuss how the estimated values compare with measured values for CASS components to demonstrate the level of confidence one can place on the calculations, and to provide additional discussion as to how the uncertainty in the calculations affects the screening process for the combined effects of TE + IE.

In the September 18, 2012, response to RAI 7, the BWRVIP stated that the measured delta ferrite content was not included on the CMTRs so that a comparison of measured to predicted delta ferrite for the RVI components in BWRs is not possible. From the CMTRs, there is no Mo content reported because Mo is not an intended alloying addition to the CF-3 and CF-8 grades used for BWR RVI components; thus, there is some uncertainty regarding the calculated ferrite. A residual level of Mo present in the heat of material would affect the delta ferrite content estimated with Hull's equations. Therefore, the BWRVIP examined how the estimated ferrite content could change given a residual Mo content of 0.5 percent by weight, which is the upper bound value for residual Mo in the CF-3 and CF-8 materials as given in SA-351. The results of the analysis demonstrated that the addition of 0.5 percent Mo would result in only a small increase in percent ferrite above that reported in the TR; therefore, the uncertainty in Mo was judged to not affect the ferrite screening process for loss of fracture toughness due to TE in CF-3 and CF-8 materials.

The NRC staff has reviewed the RAI 7 response and noted that in the May 19, 2000, Grimes letter, the calculated ferrite content from Hull's equations represents the mean value with a significant uncertainty (± 6 percent) when compared to the measured values for CASS materials. For example, heat 68 from Chen, et al., (Ref. 9), has a measured delta ferrite content of 23 percent but a predicted value of 15 percent based on Hull's equations. In RAI 7a, the NRC staff requested the BWRVIP to justify why 6 percent should not be added to the calculated ferrite values based on chemistry to represent a conservative upper-bound to the ferrite content and to provide additional discussion as to how the uncertainty in the prediction of ferrite affects the screening process for TE + IE.

In its May 23, 2014, response to RAI 7a, industry, BWRVIP and the Materials Reliability Program (MRP) provided the following conclusions relative to the prediction of ferrite and screening assessment:

1. It is not reasonable to add 6 percent to the delta ferrite values predicted by Hull's equations, since the available data clearly show that Hull's equations do not systematically underpredict measured values.

2. The standard error in the predictions from Hull's equations is of the order of 3 percent delta ferrite, that level of standard error provides a reasonable estimate of the uncertainty in the delta ferrite prediction.

3. The measured values can vary by 1 or 2 percent delta ferrite, depending on the method used.

4. Since the standard error in prediction of delta ferrite with Hull's equations is roughly of the same order as the accuracy of non-destructive measurements of delta ferrite, and

since some degree of uncertainty was included during the process of establishing the delta ferrite screening limits in the Grimes letter for TE alone, there is no need to alter the screening methodology in order to explicitly include any additional measure of uncertainty related to delta ferrite content in the screening for the combined effects of TE + IE.

The NRC staff has reviewed the response to RAI 7a and agrees with the industry that some degree of uncertainty in ferrite content was incorporated into the delta ferrite screening limits for TE alone, but it was not explicitly stated. In this case, considering the combined effects of TE plus IE, a similar level of uncertainty in the predicted ferrite content should be incorporated into the methodology used to predict fracture toughness as a function of neutron exposure, which in turn, will be compared to the acceptance criteria to determine if further evaluation is needed. As detailed in Section 3.3 of this SE, the NRC staff performed an independent confirmatory assessment of the fracture toughness of the RVI CASS components covered by the TR, using its own alternate screening criteria for IE+TE. The NRC staff's IE+TE screening criteria accounts for the uncertainty in ferrite prediction. The concern of RAI 7 and 7a are therefore resolved.

3.2 Staff Evaluation of Component Neutron Fluence Determination

The NRC staff noted that the material screening depends upon the neutron fluence, which was estimated for each of the ten CASS components in Table 3-3 of the TR. The fluence is an integral part of the fracture toughness evaluation so discussion of these assumed maximum values is incorporated into Section 3.3 of this SE.

3.3 Staff Evaluation of Fracture Toughness Basis for Material Screening

In the Grimes letter, the screening was based on measured properties from NUREG/CR-4513, Rev. 1 for samples aged to the saturation condition to account to TE alone. With the measured database, a lower-bound toughness (J integral value at 2.5 mm crack extension, referred to as J @ 2.5) for a CF-8 material with a calculated ferrite content of 15 to 25 percent was predicted to be 343 kJ/m^2. The ferrite screening criteria was chosen to ensure an end-of-life toughness level of 255 kJ/m^2, which was based on a flaw tolerance analysis of a highly stressed pressure boundary component (Ref. 10) determined to be conservative for pressure boundary components. Since the predicted lower bound saturated toughness of CF-8 material with up to 25 percent ferrite is 343 kJ/m^2, but a toughness level of 255 kJ/m^2 was determined to be sufficient, this would provide inherent margin for material with ferrite up to 25 percent. However, the ferrite screening criteria was set at 20 percent, which provides margin for uncertainty in the predicted ferrite content of 5 percent (which is similar to the stated uncertainty of Hull's equations of +/- 6 percent ferrite). Therefore, margin for uncertainty in the ferrite prediction is built into the Grimes letter ferrite screening criteria for low-molybdenum CASS material.

In the case of RVI components manufactured from CASS materials that receive significant neutron fluence, the screening must consider the combined effect of TE and IE, but the NRC staff and BWRVIP agree that there is little test data available to evaluate the combined effects. The BWRVIP has proposed two different screening methodologies to estimate the lower-bound toughness (the Z-factor correction and BWRVIP-100-A), which are reviewed by the NRC staff in the following subsections.

3.3.1 Fracture Toughness Estimates

The NRC staff considered the details of the two methodologies and had concerns regarding the applicability of the approaches. The NRC staff concerns, expressed in in RAI 9, were related to the basis used to estimate the fracture toughness, which could then be compared to the screening criteria for loss of toughness in CASS in the TR. The issues are summarized below along with the staff position on each.

3.3.1.1 *Z-Factor Correction*

Regarding the use of the Z-factor, it is approved in the ASME Code Section XI, Appendix C practice for flux welds in piping systems to account for an observed reduction in toughness of flux welds compared to non-flux welds. In the limit load evaluation described in Appendix C to the ASME Code, Section XI, the applied load is multiplied by the Z-factor to account for the lower toughness of flux welds. In the TR, the Z-factor approach is applied to CASS by dividing the measured values of J@2.5 for wrought stainless steel at the bounding fluence by Z squared to estimate the equivalent toughness of CF-8 with the same irradiation in the TR methodology (since the applied J is proportional to load squared). The wrought stainless steel J values were taken from the BWRVIP-100-A database. The NRC staff did not find the Z-factor approach to be appropriate to account for loss of fracture toughness due to the combined effects of TE + IE of CASS materials; it was meant to account for a reduction in toughness associated with fluxed welds.

3.3.1.2 *Use of BWRVIP-100-A Fracture Toughness Model*

The other method used in the TR to validate the toughness basis for the screening criteria is the BWRVIP-100-A lower bound curve. To determine whether the use of the BWRVIP-100-A model was appropriate for CASS, the NRC staff compared the prediction methodology for fracture toughness from BWRVIP-100-A to that of NUREG/CR-4513, Rev. 1 (Ref. 11) and NUREG/CR-6960 (Ref. 12). The comparison for J @ 2.5 is shown in Figure 1 along with the range of predicted J values for unirradiated CF-8 with no thermal embrittlement (TE) and maximum TE (no irradiation) for CF-8 with > 15 percent delta ferrite (Section 3.1.1 of Ref. 11) along with the Ref. 12 predicted toughness trend as a function of fluence. The lower-bound from Ref. 12 is lower, and therefore more conservative than the BWRVIP-100-A predicted toughness. The predicted toughness from Ref. 12 at 0 dpa (no irradiation) is also consistent with the minimum predicted toughness for CF-8 material due to TE alone, represented by the bottom of the vertical line at dpa = 0 on Figure 1. If extrapolated beyond where it was intended to be used, BWRVIP-100-A would predict a much higher toughness at 0 dpa. Based on this comparison, the NRC staff determined that the BWRVIP-100-A methodology maybe be an effective way to account for IE in welded, wrought structures like the core shroud, but the predicted toughness does not consider the effect of TE and therefore would not be a suitable methodology to use in the TR to account for the combined effects of TE + IE on CF-3 and CF-8 materials, especially those with greater than 15 percent but less than 20 percent ferrite (these would pass the screening for TE alone in the Grimes letter).

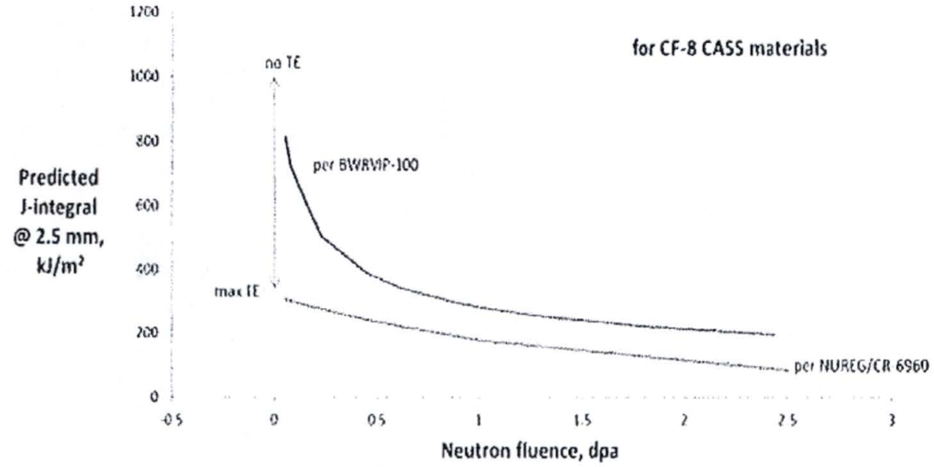

Figure 1. Plot of predicted toughness (J-integral value at 2.5 mm crack extension) from BWRVIP-100-A and NUREG/CR-6960 as a function of neutron fluence. The predicted lower-bound toughness of CF-8 due to TE alone is shown at the bottom of the vertical line at dpa = 0 for reference.

3.3.1.3 Summary

Because of the concerns expressed above with the methods used to validate the toughness screening criteria in the TR and the response to RAI 9, the NRC staff requested in RAI 9a that the BWRVIP review all available data on fracture toughness of irradiated CASS materials, and the associated uncertainties, and to either revise its methodology to predict the lower-bound toughness of CF-3 and CF-8 materials as a function of neutron fluence, or provide further justification that the methodology in the TR is sufficiently conservative.

3.3.2 *Response to RAI 9a*

In its May 23, 2014, response to RAI 9a, the industry provided a discussion of the available data in along with additional justification for why the methodology in the TR is sufficiently conservative; the response also highlighted the problems associated with the sub-size samples needed for evaluation of irradiation effects. The last section included supplemental information related to an industry proposal for screening of RVI components fabricated from CASS. The proposed process for assessment of potential TE and IE of CASS was developed based on a mechanistic model for the effect of which phase is controlling the embrittlement behavior of the CASS material. The proposal uses 1 dpa as a fluence screening limit for IE and is applied to those materials that do not screen in for TE; the format is patterned after that used in the Grimes letter.

The justification for using the TR methodology for CF-3 and CF-8 materials is discussed in the appendix to the RAI 9a response. The response compares the measured toughness for heat affected zone (HAZ) specimens irradiated to 2.15 dpa to the measured toughness of base metal irradiated to 1 dpa. Because the toughness for the base metal specimens at 1 dpa is almost a factor of 3 above the toughness of the HAZ specimens at 2.15 dpa, the industry asserts this is

an adequate demonstration of the adequacy of the methodology in the TR, which is based on dividing actual fracture toughness values from wrought materials in the BWRVIP-100-A database by the Z-factor squared (~ a factor of 2). The NRC staff has reviewed the information in the appendix and notes that the wrought materials and the HAZ samples do not contain any delta ferrite so it is not clear how this methodology could account to the aging effects of TE plus IE on CF-3/CF-8 materials with < 20 percent ferrite.

3.3.3 *Industry's Proposed Joint Revised Screening Criteria*

The May 23, 2014, letter contained supplemental information providing the basis for a proposed revised screening approach for TE and IE of CASS developed by a joint CASS working group comprised of both BWRVIP and MRP representatives. The revised criteria were intended to be applicable to both BWR and pressurized water reactor (PWR) internals. The revised criteria are different than the criteria used in the TR screening process, most notably in that a fluence threshold of 1 dpa is used for susceptibility to IE rather than 0.45 dpa used in the TR.

Regarding the proposed screening criteria for CASS to be used for RVIs, the industry examined the databases that were used in BWRVIP-100 and NUREG/CR-7027 (Ref. 13) in detail. For the range of fluences investigated, the data indicate very little difference between weld metal, base metal, and HAZ. The industry pointed out that this similarity would indicate that the criteria for IE screening for CASS materials should not be markedly different than that for wrought stainless steels. This is in agreement with the industry approach for screening, which proposed setting the fluence screening value for CASS materials to be 2/3 of that for the wrought stainless steels.

The industry approach to screening is supported by test results from Kim, et al. (Ref. 14) and the preliminary data of Chen, et al. (Ref. 15). The discussion highlights the relatively high fracture toughness of CF-8 and CF-3 and the fact that there are no data that demonstrates any exacerbation effect due to the combined thermal and irradiation effects. The industry noted that while the preliminary data of Chen et al. may display some greater loss of toughness on thermal plus irradiation exposures compared to irradiation or thermal exposure alone, these data do not necessarily reflect a significant combined effect. The industry also noted that the compositions of the materials tested by Chen et al., which did display good toughness, would have screened in for TE under the proposed industry screening hierarchy.

3.3.4 *Summary of Initial NRC Staff Review of Industry Response to RAI 9a*

Based on its review of BWRVIP's response, and the proposed joint industry screening criteria, the NRC staff identified several unresolved issues related to the basis for the TR CASS evaluation methodology, and therefore the NRC staff was unable to resolve RAI 9a. These issues are summarized in the following subsections.

3.3.4.1 <u>Use of Room Temperature Data</u>

All of the CF-3 and CF-8 data from Kim, et al. are from tests at room temperature, which is not conservative for the development of a lower bound estimate of fracture toughness at higher temperatures. This is graphically illustrated in Figure 7 of the May 23, 2014, response to RAI 10 where the test results a room temperature are all higher than that measured at higher temperature. It is the NRC staff's opinion that the room temperature results are qualitative evidence of ductile behavior, but not a quantitative assessment of fracture toughness at operating temperature as required for comparison to the acceptance criteria in the Grimes letter.

3.3.4.2 *Significance of Low-Fluence Aged and Irradiated Data*

The testing from Chen, et al. was done at typical operating temperature, but the irradiation conditions were only a fraction of the proposed fluence screening level. The data from Chen et al. suggests that there is a small combined effect of IE + TE at 0.08 dpa, but this does not provide any information about the properties at fluences near 1 dpa, which could support the industry's proposed fluence screening for CF-3 and CF-8 at 1 dpa. Given the points discussed above, the NRC staff still questioned whether the proposed methodology in the TR is sufficiently conservative for determining the lower-bound toughness of CF-3 and CF-8 materials as a function of neutron fluence.

3.3.5 *Development of NRC Proposed Screening Criteria*

With respect to the overall screening process and mechanistic model proposed by industry, the NRC staff developed a draft screening criteria for aged and irradiated CASS (Ref. 16). The NRC staff's criteria are based on a continuous reduction in fracture toughness of the austenite phase in CASS as a function of fluence, without a transition from ferrite controlled to austenite controlled fracture, consistent with the lack of any transition in the fracture appearance of test specimens. The NRC staff agreed with industry that the screening for IE + TE should retain as much consistency as possible with the Grimes letter screening for TE, but the limit on neutron fluence for screening should be reconsidered. Information from NUREG/CR-7027, an update on the data originally included in NUREG/CR-6960, was suggested by the NRC staff as a lower bound curve for the fracture toughness at 2.5 mm crack extension as a function of fluence. The NRC staff also suggested that the differences between the industry and the NRC staff in screening approaches could be resolved by component-specific toughness criteria, something less than 255 kJ/m^2 at 2.5 mm, which takes into account the lower crack driving forces in a RVI component.

3.3.6 *July 15, 2014, Public Meeting*

The NRC staff and industry held a public meeting on July 15, 2014, to discuss the two different proposed screening criteria for CASS. Industry used data from testing at room temperature along with the preliminary data from Chen, et al. for CASS thermally aged and irradiated to 0.08 dpa that showed minimal reduction in toughness, to support its technical basis for the conservatism of its screening methodology. The NRC staff disagreed with the hierarchy or sequence of screening first for TE, then for IE. The NRC staff also did not find that industry had provided a sufficient technical basis for increasing the fluence threshold for IE screening to 1 dpa.

Given the NRC staff's concerns with the industry's May 23, 2014, response to RAI 9a, and the discussions at the July 15, 2014, public meeting, the BWRVIP agreed to expand the response to RAI 9a to demonstrate the adequacy of the proposed screening for RVI components manufactured from CF-3/CF-8 materials, as documented in the meeting summary (Ref. 17).

3.3.7 *BWRVIP 2015-025*

The BWRVIP followed up with the NRC on the issues raised in the July 15, 2014, meeting on CASS with its March 9, 2015, letter. The main points of the supplement are summarized below:

- The industry recognizes that the NRC is intending to revise its original IE position in the Grimes letter from 0.00015 dpa to a position in the region of 0.5 to 1.5 dpa. The industry agrees that a change is appropriate and has proposed a value of 1 dpa.
- The NRC staff's proposal for screening will penalize CF-3 and CF-8 materials because of the lower properties associated with the Mo-containing CF-8M materials that are included in the NUREG/CR-7027 database.
- There is no data that demonstrates a synergistic effect for CF-3 and CF-8. As such the industry has proposed criteria that do not combine TE and IE. Each mechanism is considered distinct and separate.
- The introduction of a new criteria set for materials with ferrite content between 15 and 20 percent (having a lower proposed screening value for IE of 0.45 dpa) is unnecessary and technically unfounded. The introduction of this category of ferrite content is significantly burdensome to licensees since it will require more complex and potentially more error-prone assessments of components fabricated from CASS materials by virtue of having more categories. The industry maintains that the categories of materials and associated ferrite levels contained in the Grimes letter are appropriate.
- The criteria proposed by the industry (20 percent ferrite and 1 dpa) have been shown to project significant margin on toughness reduction and therefore safety when compared to measured embrittlement behavior for CF-3 and CF-8 materials.

The NRC staff has reviewed the March 9, 2015, letter and finds that the lack of high-temperature data above 0.08 dpa for CF-3 and CF-8 materials makes a robust technical basis for any position tenuous. Much of the supporting text in the March 9, 2015, letter is dealing with TE alone, yet the plot in Figure 1 of Attachment B to the March 9, 2015, letter does not include TE alone and depends on tests in air at room temperature to develop an estimate of J @ 2.5 as a function of neutron fluence for CF-3 and CF-8 materials under operating conditions; the proposed lower-bound curve starts at about 550 kJ/m^2 for 0.02 dpa. By comparison, the lower bound toughness for TE alone (0 dpa) in CF-8 with > 15 percent ferrite is < 400 kJ/m^2 which is lower than the BWRVIP's proposed value at 0.02 dpa. Because the trend curve in Figure 1 of Attachment B does not include the lower-bound estimate of toughness due to TE alone, the industry proposed technical basis effectively removes the significant margin that was approved in the Grimes letter, as discussed in the beginning of Section 3.3 of this SE.

The NRC staff agrees with the industry that there does not appear to be any "synergistic" effects, and that terminology should be eliminated. The mechanisms are clearly distinct and separate in the fact that TE mainly affects the ferrite and IE affects both ferrite and austenite. However, the NRC staff feels that any screening criteria should incorporate the bounding toughness for TE alone, and then include the effects of IE by reducing the estimated, lower-bound toughness for TE alone by a factor proportional to the neutron exposure.

3.3.8 *Staff's Revised Screening Criteria*

Based on the discussions with industry, the NRC staff intends to revise its screening criteria for RVIs fabricated from CASS materials based on a new methodology to predict J @ 2.5 as a function of neutron fluence, which takes into account chemical composition, ferrite content, and casting method; the format is similar to what was used in the Grimes letter for TE alone. There are two key differences in the revised screening criteria:

1. the application of a lower acceptance criteria, J @ 2.5 value of 200 kJ/m^2, based on the recognition that RVI components do not need the same level of toughness as pressure boundary components, and
2. the trend curve starts at a dpa = 0 that bounds the estimated J @ 2.5 from NUREG/CR-4513, Rev. 1.

The revised toughness basis supports screening for loss of fracture toughness for CF-3/CF-8 materials with < 20 percent ferrite and a fluence limit of 1 dpa. Therefore, the NRC staff's revised screening criteria are similar to those proposed by the industry in BWRVIP 2015-025, but by bounding the lower-bound estimate of toughness at a dpa = 0, the method maintains the consideration of ferrite uncertainty built into the approved screening from the Grimes letter.

The NRC staff's revised acceptance criteria is supported by the results of some generic flaw tolerance evaluations of representative RVI components found in Appendix D to the BWRVIP 2015-025 letter. It is likely these criteria will be published in the future in an NRC guidance document such as a license renewal interim NRC staff guidance document.

A summary of the technical bases for the NRC staff's screening criteria, is contained in Appendix A.

3.3.9 *Staff Independent Confirmatory Analysis*

The NRC staff applied its screening criteria described in Appendix A using the generic CASS component information in the TR (material grade, ferrite content, and neutron fluence) and determined that all the generic CASS components can be screened out with respect to significant loss of fracture toughness due to TE and IE. No augmented inspections would be required due to toughness considerations for up to 60 years of service (40 year original license plus 20 year renewal).

3.3.10 *Summary – Fracture Toughness Basis for Material Screening*

Based on its independent analysis using its revised screening criteria, the NRC staff determined that the aging management recommendations for the six generic CASS components (i.e., the six CASS components identified in Table 6-1) are acceptable, e.g., no additional inspections are necessary to manage aging due to loss of fracture toughness. However, the NRC staff does not accept the screening methodology as described in the TR, for generic use for screening for loss of fracture toughness in CASS RVI components. The NRC staff has developed an alternative screening process, described in Appendix A, which could be used to replace the screening described in Chapters 4 and 6 of the TR. Furthermore, with the NRC staff's revised methodology to predict the toughness outlined in Appendix A, the uncertainty in ferrite content incorporated into the delta ferrite screening limits for TE alone is also incorporated into the screening limits for TE plus IE. With this alternate screening process, the NRC staff's concern expressed in RAI 9a is resolved.

3.4 Staff Evaluation of Stress Effects and Material Screening

The NRC staff has reviewed the component-specific stress analyses for the RVI components manufactured from CASS materials in Section 5 of the TR. For CSSN elbows, the analyses demonstrated that the tensile stresses are less than 5 ksi and the NRC staff considers there is little chance of brittle failure when the tensile stresses are below 5 ksi. For the other

components that could be fabricated from CASS material, the stresses are above 5 ksi and therefore, could not be screened out on the basis of stress and would be subject to further evaluation for aging management. The NRC staff finds the screening for stress is satisfactory.

3.5 Staff Evaluation of Material Screening and Existing Inspections

The NDE methods of the existing examinations, discussed in Section 6, "Assessment of CASS Components," were justified because they were capable of detecting degradation other than loss of fracture toughness due to the combined effects of thermal and neutron embrittlement. Given that, based on the NRC staff's initial assessment, the lower-bound toughness for a CASS component subject to both thermal and neutron embrittlement could be less than the toughness screening criteria, the NRC staff asked the BWRVIP in RAI 11 to justify the adequacy of the existing BWRVIP inspections to detect subcritical flaws as discussed in the GALL Report, XI.M9 paragraph 4 -- *Detection of Aging Effects.*

By letter dated May 23, 2014, the industry concluded that while some BWR internal components might exhibit fracture toughness that approaches the lower bound value, this fact does not affect the effectiveness of the existing examinations with respect to detection of cracking before it reaches a critical flaw size that could affect the function of any safety-significant system. The inspections utilized by the BWRVIP for detection of flaws have been reviewed and approved by the NRC.

The BWRVIP also noted that the wrought material is more likely to crack than CASS and inspecting the wrought materials serves as a leading indicator. Furthermore, inspection of wrought to CASS weld joints would also reveal any flaw in the CASS portion of the weld joint. The applicable and NRC-approved BWRVIP guidelines recommend more stringent inspections, such as EVT-1 examinations or ultrasonic methods of volumetric inspection, for certain selected components and locations.

The BWRVIP further noted that even when comparing applied crack-driving forces to lower bound fracture toughness, there are no BWR internals that are discussed in Section 6 of the TR that will fail due to an undetectable flaw and thus challenge the integrity of the component, and that cracks would have to be visible before they are large enough to challenge integrity. Consequently, the aging effects due to TE and IE are considered by the BWRVIP to be adequately managed.

To summarize the BWRVIP's position expressed in the response to RAI 11, the NRC has accepted the aging management approach for BWR internals and no new experience has been observed to date that challenges that position. Based on the above discussions, the BWRVIP believes that the NRC accepts the inspection methodologies for BWR components as capable of performing their intended function of detecting flaws in the reactor vessel internals and detecting cracks of concern associated with non-pressure boundary RPV internals.

Based on the NRC staff's review of the industry response to RAI 11 and several of the existing BWRVIP reports (Refs. 18-20), the NRC staff notes that consideration of the uncertainty in the ferrite content, as discussed in Section 3.1 of this SE, was not included in any of the existing reports; the omission of any consideration of the uncertainty in the ferrite content in the existing BWRVIP reports is not within the scope of the NRC staff's review of the TR, but should be addressed in future updates to the existing BWRVIP reports. What is relevant here is the fact that, with this SE, the NRC staff has reached a satisfactory resolution for RAI 9a in Section 3.3 of this SE for BWRVIP-234; the components will have adequate fracture toughness so that the

structural integrity related to the loss of fracture toughness of the RVIs is not a concern for the license renewal period. Therefore, the issue that raised the NRC staff's concern expressed in RAI 11 is resolved by the fact that there is no need for any augmented inspections related to the loss of fracture toughness due to the combined effects of TE and IE.

3.6 Applicability of the Topical Report

The NRC staff's evaluation was performed using the sample of CMTRs, and conservative fluence analyses based on a BWR/6, 218-inch plant at EPU conditions that translates to a maximum neutron fluence for 60 years of operation. Therefore, application of the TR to the current licensing basis for any BWR to manage loss of fracture toughness of the six generic CASS components covered by the TR is dependent on the following condition. The licensee's plant-specific fluence assessment of the six generic CASS components must demonstrate that the projected neutron fluence is bounded by the maximum fluence stated in Table 3-3 of the TR.

When publishing the approved version of BWRVIP-234, Section 3.5 shall be revised to require a plant-specific fluence assessment of the six generic CASS components. **This is a Topical Report Condition.**

4.0 LIMITATIONS AND CONDITIONS

Any deviations from the aging management programs determined to be necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the components or other information presented in the report, i.e. accumulated neutron fluence, will have to be identified by the licensee/renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

5.0 SUMMARY AND CONCLUSIONS

The NRC staff has reviewed the BWRVIP-234 report and the supplemental information that was transmitted in the BWRVIP letters on September 18, 2012, May 23, 2014, March 9, 2015, and November 19, 2015. Based on its independent analysis using its revised screening criteria, the NRC staff determined that the aging management recommendations for the six generic CASS components listed in Table 6-1 of the TR are acceptable, e.g., no additional inspections are necessary to manage aging due to loss of fracture toughness. The NRC staff has found that this report, as modified by the **topical report condition** discussed in Section 3.6 of this SE, provides an acceptable basis for plant-specific AMPs for managing the aging effect of loss of fracture toughness in CASS in BWR RVI components.

As modified by this SE and approved by the NRC, any applicant may reference BWRVIP-234 in a License Renewal Application or other licensing action to satisfy the requirements of GALL, Rev. 2, Section XI.M9 for demonstrating that the effect of aging on the RVI components manufactured from CASS materials, within the scope of BWRVIP-234, will be adequately managed for loss of fracture toughness due to TE and IE and any possible combined effects of the two.

6.0 REFERENCES

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2. Letter, BWRVIP 2012-148, from Dennis Madison (BWRVIP Chairman) to Joseph Holonich (NRC), "Request for Additional Information on BWRVIP-234: Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals," September 18, 2012 (ADAMS Accession No. ML12265A078).

3. Letter, BWRVIP 2014-086, from Andrew McGehee, (BWRVIP Program Manager) and Dennis Madison (BWRVIP Chairman) to Joseph Holonich (NRC), "BWRVIP Response to NRC Request for Additional Information on BWRVIP-234," May 23, 2014 (ADAMS Accession No. ML14174A841).

4. Letter, BWRVIP 2015-025, from Andrew McGehee, (BWRVIP Program Manager) and Tim Hanley (BWRVIP Chairman) to Joseph Holonich (NRC), "Summary of Industry Position on Screening Criteria for Thermal and Irradiation Embrittlement for PWR and BWR Reactor Internals Fabricated of Cast Austenitic Stainless Steel," March 9, 2015 (ADAMS Accession No. ML15155B487).

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20. EPRI Report TR-108728, BWR Vessel and Internals Project, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41, Rev. 0)," Electric Power Research Institute, Palo Alto, California, October 1997.

Attachment: Appendix A

Principal Contributor: Patrick Purtscher, Division of Engineering

Date: June 22, 2016

Appendix A Screening of CF- 3/CF-8 for Loss of Fracture Toughness Due to TE + IE

The screening is based on the estimated, lower bound value of the J integral at 2.5 mm crack extension (J @ 2.5). Figure A1 plots the predicted values for J @ 2.5 from BWRVIP-100 as a function of neutron fluence along with a new, NRC staff-proposed curve for CF-3 and CF-8 materials where the J @ 2.5 for 0 dpa is set equal to the lower-bound J @ 2.5 value for CF-8 with 15 to 25 percent ferrite from NUREG/CR-4513, Rev. 1. CF-8 material with a calculated ferrite content ≤ 20 percent would not be considered susceptible to TE alone using the Grimes letter. The remainder of the proposed curve is obtained by shifting the BWRVIP-100 curve for J @ 2.5 to the left until it intersects the vertical axis. In this way, the proposed curve takes full account for the saturated lower bound toughness for TE used in the development of the Grimes letter and incorporates an additional margin to account for the further reduction in fracture toughness due to IE. Given that there is no evidence of a synergistic effect, this combined effect is acceptable to the NRC staff. With this method, the lower-bound estimate of J @ 2.5 for static-cast CF-8 with ≤ 20 percent ferrite equals the acceptance level at about 0.7 dpa.

Furthermore, the NRC staff notes that the discussion in BWRVIP-234 is applicable to an acceptance criteria where the J @ 2.5 is 255 kJ/m^2 (appropriate for pressure-boundary components like the reactor coolant system) yet the discussion in BWRVIP 2015-025, Attachment C, "Reactor Vessel CASS flaw tolerance," demonstrates that the loading for typical RVI components results in an applied J-integral far below the J @ 2.5 of 255 kJ/m^2 level. Furthermore, the NRC staff notes that EPRI report TR-112718[c] calculates applied J values for three different generic models that were designed to simulate RVI components. Both Attachment C and EPRI TR-112718 come to the similar conclusion that the crack driving forces in RVI components are much lower than the current acceptance criteria of J @ 2.5 is 255 kJ/m^2. These results suggest that there is a significant margin between the maximum applied J for RVIs (in Attachment C that would be 83 kJ/m^2) and either acceptance criteria (255 or 200 kJ/m^2). With an acceptance criteria of 200 kJ/m^2, the lower-bound estimate of J @ 2.5 for static-cast CF-8 with between ≤ 20 percent ferrite would exceed the acceptance level up to about 1.7 dpa.

Based on the proposed methodology to estimate fracture toughness of CASS that considers both TE and IE as well as the revised acceptance criteria, the NRC staff has developed a revised screening for low-Mo CASS materials, shown below as Table A1, which is similar to that found in Table 1 from Attachment D of BWRVIP 2015-025. There are two differences between Table A1 here and Table 1 of Attachment D. First, the acceptance criteria in Table A1 is 200 kJ/m^2; in Table 1 of Attachment D, the acceptance criteria is 255 kJ/m^2. The second difference can be found in the fluence limits. Table A1 is valid between 0.00015 and 1 dpa where Table 1 of the industry proposal goes from 0 to 1 dpa. In either case, CASS materials are potentially susceptible to significant loss of fracture toughness above 1 dpa of neutron exposure.

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[c] EPRI TR-112718, "Evaluation of Neutron Irradiation Embrittlement for PWR Stainless Steel Internal Component Supports," R. Nickell, M. Rinckel, and W. Pavinich, Electric Power Research Institute, Palo Alto, California, September, 1999.

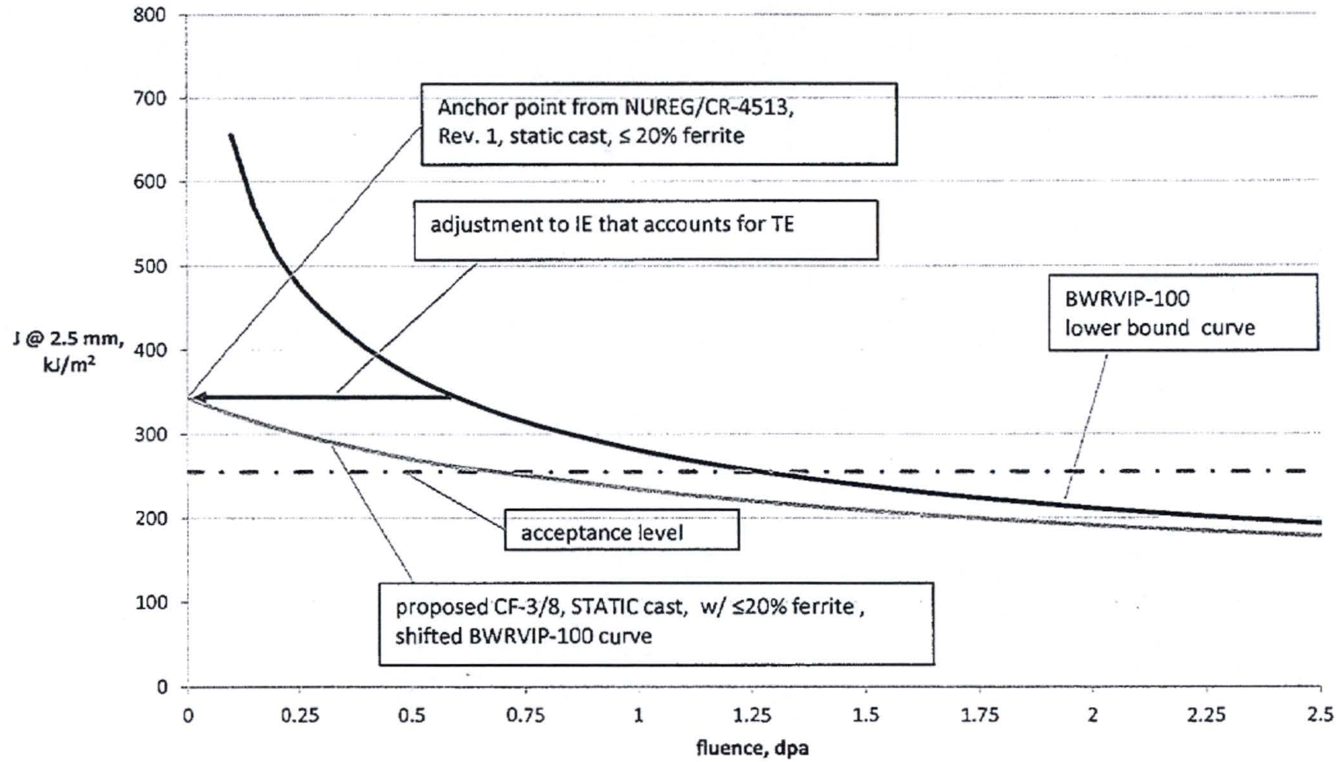

Figure A1. Plot of predicted lower-bound J @ 2.5 mm (BWRVIP-100 and NRC staff proposed) vs. neutron fluence. Arrow reflects shift of BWRVIP-100 curve to the left, represents assumed contribution to lower-bound J @ 2.5 mm due to IE.

Table A1. Screening for CF-3 AND CF-8 RVI Components with neutron exposure between 0.00015 and 1 dpa. ◊

Casting Method	Further Evaluation	Delta ferrite %⁺
static	Yes	> 20%
	No	≤ 20%
centrifugal	Yes	>25%
	No	≤ 25%⁺⁺

◊ All CASS materials need further evaluation above 1 dpa neutron fluence.

⁺ Estimate delta ferrite content from chemical composition with Hull's equivalent factors (discussed in Ref. 1). If chemical composition is unknown, further evaluation is required.

⁺⁺ Upper limit for validity of ferrite screening of CASS from NUREG/CR-4513, Rev. 1.

BWRVIP Comment Summary Table

Comment No.	Draft SE Location	Comment Type	Comment	NRC's Response
1	Pg. 12, Section 3.3.10, lines 34 and 35	Clarification	Recommend inserting the parenthetical "(i.e., the six CASS components identified in Table 6-1) to clarify that when the SE refers to "six generic CASS components it means those referenced in Table 6-1 of the TR. Alternatively, wherever "six generic CASS components" is stated in the SE, replace it with "the six CASS components identified in Table 6-1.	Agreed
2	Pg. 14, Section 3.6, line 13	Clarification	Recommend replacing "RVI" with "CASS" to be consistent with the other references to the six CASS components of Table 6-1.	Agreed
3	Pg. 14, Section 3.6, line 18	Clarification & Accuracy	Recommend replacing "for all plant-specific CASS RVI components" with "of the six generic CASS components" to be consistent with the first paragraph of Section 3.6.	Agreed
4	Pg. 17, first page of Appendix A, line 14	Editorial	Recommend deleting the word "between" as the sentence does not read correctly.	Agreed
5	Second page of Appendix A, Figure A1	Accuracy	The "< 20 % ferrite" values in Figure A1 need to be changed to "≤ 20% ferrite" to be consistent with Table A1.	Agreed
6	Headers throughout	Clarification	As the SE does not contain any EPRI proprietary information, "Proprietary Information" can be removed from the headers.	Agreed

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This report was prepared by

Electric Power Research Institute
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This publication is a corporate document that should be cited in the literature in the following manner:

BWRVIP-234NP-A: BWR Vessel and Internals Project, Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steels for BWR Internals. EPRI, Palo Alto, CA: 2017. 3002010550NP.

ABSTRACT

NUREG-1801 Rev. 1, Section XI.M13, (hereafter referred to as the GALL report) states that a Section XI VT-3 examination is required to be performed of reactor internal components. In addition, the GALL report specifies that for the license renewal period, these inspections shall be augmented by an aging management program to address the synergistic effects of thermal aging and neutron embrittlement in cast austenitic stainless steels (CASS). This aging management program consists of (a) identifying susceptible components; and (b) either performing additional inspections of these components, or performing a component-specific evaluation to confirm that the stresses (tensile loading) in the components are sufficiently low such that augmented inspections are not warranted. The purpose of this report is to evaluate the material composition, fluence, stresses and the field experience of CASS components in BWR applications and recommend augmented inspections if needed.

The report presents a review of the current literature on fracture toughness/embrittlement in cast austenitic stainless steel, including the roles of ferrite level, temperature effects, and irradiation on property degradation. The review considers the clear distinction between PWR and BWR operating conditions as well as the casting composition (ferrite content) of CASS components. In this context, the report then reviews the different CASS components used in all types of operating BWRs. These include the Orificed Fuel Support, Control Rod Guide Tube Base, Core Spray Sparger Nozzle Elbows, Jet Pump Assembly, Jet Pump Restrainer Bracket and Low Pressure Core Injection (LPCI) Coupling. The specifications used in their procurement and their compositions based on certified material test reports (CMTRs) of cast components are summarized based on internal GEH records. This CMTR information is used to determine ferrite content calculated using Hull's equivalent factors and substantiate the use of non-Mo containing alloys. Expected levels of fluence, including operation through the license renewal period (based on GEH assessments) have also been determined from specific plant evaluations.

The report then presents an evaluation of the different CASS components to determine if thermal aging and/or irradiation embrittlement could occur. A screening process was developed that included assessment of ferrite level, fluence, toughness, stress level and results of existing BWRVIP inspections. Given that many components will reach fluence levels where irradiation may have a potential effect on fracture toughness, the stress levels in each component and the predicted fracture toughness properties are included in the assessment to develop the recommended inspection guidance. The results are compared against NRC approved criteria to determine whether component-specific evaluation or augmented inspections are needed. In addition to the NRC criteria, the current BWRVIP inspections were also evaluated to determine whether they satisfy the intent of NUREG-1801, Rev. 1.

The evaluation shows that all the BWR CASS components have ferrite levels below the level for which aging embrittlement is a concern. Furthermore, for the Control Rod Guide Tube Base and Core Spray Sparger Nozzle Elbows, the end of life fluence is less than the threshold value for toughness loss. The end of life fluence levels for the orificed fuel support, the jet pump assembly castings and the LPCI couplings exceed the threshold, but the toughness data for irradiated austenitic stainless steel show that that these components will have sufficient fracture toughness at end of the license renewal period so that augmented inspection is not required. In summary, based on this assessment, it is concluded that augmented inspections are not required for the BWR CASS internals as long as a plant specific analysis confirms that predicted fluence at the end of the extended period of operation is bounded by that assumed in this report for the CASS components evaluated.

Keywords
Cast Austenitic Stainless Steel
BWR Internals
Thermal Aging
Neutron Embrittlement
Fracture Toughness

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EXECUTIVE SUMMARY

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PRIMARY AUDIENCE: BWR Vessel and Internals Project (BWRVIP) Program Owners

SECONDARY AUDIENCE: Engineers responsible for defining inspection requirements for Cast-Austenitic Stainless Steel components in BWR reactor internals.

KEY RESEARCH QUESTION

NUREG-1801 Rev. 1, Section XI.M13, states that an ASME Code Section XI VT-3 examination is required to be performed on reactor internal components. In addition, the report specifies that for the license renewal period, these inspections shall be augmented by an aging management program to address the synergistic effects of thermal aging and neutron embrittlement in cast austenitic stainless steels. This aging management program consists of (a) identifying susceptible components; and (b) either performing additional inspections of these components, or performing a component-specific evaluation to confirm that the stresses in the components are sufficiently low such that augmented inspections are not warranted.

The purpose of this report is to evaluate the potential synergistic effects of thermal aging and neutron embrittlement of BWR internal components fabricated of cast austenitic stainless steels (CASS) and to determine if augmented inspections are necessary to detect degradation.

RESEARCH OVERVIEW

The project team identified the BWR internal components fabricated of CASS, reviewed the material test reports for chemical compositions, and determined the relevant end of license fluence values for each component. They reviewed current literature on fracture toughness/embrittlement in cast austenitic stainless steel, including the roles of ferrite level, temperature effects, and irradiation on property degradation. Evaluation of the various CASS components was performed to determine if thermal aging and/or irradiation embrittlement could occur. A screening process was developed that included assessment of ferrite level, fluence, toughness, stress level and results of existing BWRVIP inspections. Given that many components will reach fluence levels where irradiation may have a potential effect on fracture toughness, the stress levels in each component and the predicted fracture toughness properties are included in the assessment. The results were compared against NRC approved criteria to determine whether component-specific evaluation or augmented inspections are needed. In addition to the NRC criteria, the current BWRVIP inspections were also evaluated to determine whether they satisfy the intent of NUREG-1801, Rev. 1. This NRC approved version of the report incorporates the NRC Safety Evaluation (SE) of the report as well as all pertinent EPRI/NRC correspondence during the NRC review of the report. In addition, changes to the report requested by the NRC as part of the SE have been incorporated in this version.

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EXECUTIVE SUMMARY

KEY FINDINGS

- The evaluation shows that all the BWR CASS components have ferrite levels below the level for which aging embrittlement is a concern.
- The Control Rod Guide Tube Base and Core Spray Sparger Nozzle Elbows, the end of life fluence is less than the threshold value for toughness loss.
- The end-of-life fluence levels for the orificed fuel support, the jet pump assembly castings and the LPCI couplings exceed the threshold, but the toughness data for irradiated austenitic stainless steel show that that these components will have sufficient fracture toughness at end of the license renewal period so that augmented inspection is not required.
- It is concluded that augmented inspections are not required for the BWR CASS internals, as long as, a plant specific analysis confirms that the predicted fluence at the end of the extended period of operation is bounded by that assumed in this report for the CASS components evaluated.

WHY THIS MATTERS

Embrittlement of CASS has been studied extensively, identifying many of the key parameters that can affect the fracture toughness with long term exposure to the high temperature light water reactor environment. Long-term plant life and license renewal require an understanding of thermal aging and neutron embrittlement susceptibility for BWR materials. Such information—also needed in aging management of plant components—reduces the risk of component failure by increasing awareness of material behavior over time and possible material failure mechanisms.

HOW TO APPLY RESULTS

This report provides the technical basis for BWRVIP to define appropriate inspection requirements for BWR CASS internals.

LEARNING AND ENGAGEMENT OPPORTUNITIES

- Boiling Water Reactor Vessel Internals Project (BWRVIP)

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PROGRAM: Boiling Water Reactor Vessel Internals Project (BWRVIP) 41.01.03

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BWRVIP-234NP-A: BWR Vessel and Internals Project, Thermal Aging and Irradiation Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals. EPRI, Palo Alto, CA: 2017. 3002010550NP.

RECORD OF REVISIONS

Revision Number	Revisions
BWRVIP-234	Original issue (EPRI report 1019060) published December 2009
BWRVIP-234-A	NRC approved version of BWRVIP-234 (EPRI report 3002010550) published November 2017. The report as originally published (1019060) was revised to incorporate changes proposed by the BWRVIP in responses to NRC Requests for Additional Information, recommendations in the NRC Safety Evaluation (SE), and other necessary revisions identified since the last issuance of the report. In accordance with a NRC request, the SE is included here in the report formatter and the report number includes an "-A" indicating the version of the report accepted by the NRC staff. Non-essential format changes were made to comply with the current EPRI publication guidelines. Appendix B added: NRC RAIs on BWRVIP-234 dated October 18, 2011. Appendix C added: BWRVIP Response to NRC RAIs on BWRVIP-234 dated November 20, 2012. Appendix D added: NRC RAIs on BWRVIP-234 dated May 7, 2013. Appendix E added: BWRVIP Response to NRC RAIs on BWRVIP-234 dated May 23, 2014. Appendix F added: BWRVIP submittal of Summary of Industry Position on Screening Criteria for Thermal and Irradiation Embrittlement for PWR and BWR Reactor Internals Fabricated of Cast Austenitic Stainless Steel. Details of the revision can be found in Appendix G.

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INTRODUCTION

1.1 Background

NUREG-1801 Rev. 1, Section XI.M13, (hereafter referred to as the GALL report) states that a Section XI VT-3 examination is required to be performed of reactor internal components. In addition, the GALL report specifies that for the license renewal period, these inspections shall be augmented by an aging management program to address the synergistic effects of thermal aging and neutron embrittlement in cast austenitic stainless steels (CASS). This aging management program consists of (a) identifying susceptible components; and (b) either performing additional inspections of these components, or performing a component-specific evaluation to confirm that the stresses (tensile loading) in the components are sufficiently low such that augmented inspections are not warranted.

The potential for aging embrittlement (and its detrimental effect on fracture toughness) is dependent on several material/environmental factors: ferrite level, operating temperature, chemical composition, casting methods, and fluence level. For example, molybdenum is known to have a synergistic effect on the formation of embrittling phases, such that the molybdenum-bearing grades are known to have a greater susceptibility to thermal aging. In order to evaluate the potential for embrittlement, all of these factors must be considered.

A second key contributor to evaluating the effects of embrittlement is the stress state of the individual components. There may be some components, for example, that do not experience high loads during service, and, as such, embrittlement would not be a significant concern for component performance.

1.2 Objectives and Scope

The purpose of this report is to evaluate the material composition, fluence, stresses and the field experience of CASS components in BWR applications and recommend augmented inspections if needed.

Ten (10) BWR internal components have been identified as fabricated of CASS, and within the scope of this evaluation (the BWRVIP document that contains the component description is included for reference):

- Orificed Fuel Support (OFS) (BWRVIP-47-A)
- Control Rod Guide Tube Base (BWRVIP-47-A)
- Core Spray Sparger Nozzle Elbows (BWRVIP-18, Rev.1)
- Jet Pump Transition Piece (BWRVIP-41, Rev. 1)
- Jet Pump Restrainer Bracket (BWRVIP-41, Rev. 1)

- Jet Pump Inlet Mixer Assembly (BWRVIP-41, Rev. 1)
- Jet Pump Elbow (BWRVIP-41, Rev. 1)
- Jet Pump Nozzle (BWRVIP-41, Rev. 1)
- Jet Pump Diffuser Collar/Guides (BWRVIP-41, Rev. 1)
- LPCI Coupling (BWRVIP-42-A)

This evaluation is composed of the following steps:

1. Literature review of CASS embrittlement
2. Material composition effects
3. Material data review from actual CASS components
4. Fluence effects
5. Development of generic screening criteria
6. Stress evaluation of typical component

Using the results of this evaluation, augmented inspection recommendations (if necessary) for CASS components have been developed.

1.3 Implementation Requirements

This report is provided for information only. Therefore, the implementation requirements of Nuclear Energy Institute (NEI) 03-08, Guideline for the Management of Materials Issues, are not applicable.

2
LITERATURE REVIEW

Embrittlement of cast austenitic stainless steels (CASS) has been studied extensively, identifying many of the key parameters that can affect the fracture toughness of CASS material with long term exposure to the high temperature light water reactor environment. This section reviews both thermal aging and neutron embrittlement and summarizes these key factors, which will be used in subsequent sections to evaluate the potential susceptibility of BWR CASS components to loss of fracture toughness.

2.1 Thermal Aging Embrittlement Phenomena

Thermal aging embrittlement (also termed as thermal embrittlement) of CASS components occurs as a result of long time aging at light water reactor operating temperatures [1]. Thermal embrittlement increases the hardness and tensile strength, and decreases ductility, impact strength, and fracture toughness. Austenitic welds and castings have a duplex microstructure consisting of austenite and δ-ferrite phases. Although the ferrite content is beneficial in preventing hot cracking and stress corrosion cracking, it is also the source of thermal embrittlement for austenitic welds and castings.

Studies investigating the cause of the thermal embrittlement of CASS have demonstrated that it can occur during the reactor design lifetime or life extension due to the precipitation of a Cr-rich α' phase as a result of spinodal decomposition, nucleation and growth of the γ' phase; precipitation of a G phase (a nickel and titanium rich silicide), formation of $M_{23}C_6$ carbides in the δ-ferrite, and/or additional precipitation and/or growth of existing carbides at the ferrite/austenite phase boundaries.

The effect of thermal embrittlement of austenitic stainless steel welds and castings is manifested in cleavage fracture in the ferrite phase or separation of the ferrite/austenite phase boundary. Materials with high ferrite content and/or large phase boundary carbides are more prone to thermal embrittlement. The effect of thermal aging is to decrease the lower and upper shelf Charpy energy of CASS. Significant reduction in fracture toughness is likely when the ferrite volume fraction exceeds 10% [2].

Even though there is potential for fracture toughness reduction, it is appropriate to consider a screening fracture toughness level to further differentiate materials subject to embrittlement, such that augmented inspections are not necessary. A screening crack growth resistance (J-R) value of 255 kJ/m^2 (1,450 in-lb/in^2) at a crack depth of 2.5 mm (0.1 in) is discussed in [4] and states these values can be used to differentiate between CASS materials that are non-susceptible and those that are subject to thermal aging embrittlement.

2.1.1 Temperature Effects

Chopra [3] discusses an Arrhenius model to determine the effect of aging temperature on the time to reach a given degree of embrittlement. A best estimate of the laboratory data at high temperatures suggests the following the equation:

$$t = 10^{P} \exp\left[\frac{Q}{R}\left\{\frac{1}{T} - \frac{1}{673}\right\}\right]$$

where Q is the activation energy, R the gas constant, T the absolute temperature, and P an aging parameter that represents the degree of aging reached after 10^P hours at 400°C (752°F). The activation energy for the process of embrittlement is a function of the chemical composition of the cast material. The relationship shows that the time to reach a given level of embrittlement is lower at higher temperatures, i.e. aging embrittlement is more significant at higher temperatures.

Sample data from Reference [4] (Figure 2-1) show clearly the effect of aging temperature on the fracture toughness of CF-8M material. While the saturation energy (i.e. Charpy energy at long times) does not depend on the aging temperature, the time to reach a given degree of embrittlement is much lower at higher aging temperatures. Based on this, the higher the aging temperature, the faster the aging effect reaches saturation. Since PWRs operate at a higher temperature than BWRs, thermal aging embrittlement effects for CASS components in a BWR are expected to occur later in life than for CASS components in a PWR.

2.1.2 Chemical Composition

Of the CASS materials commonly used in light water reactor applications (CF-3, CF-8, CF-3M and CF-8M), CF-8M is the most susceptible to aging embrittlement. The results indicate that thermal aging decreases the impact energy and shifts the ductile-to-brittle transition curves to higher temperatures. However, different heats exhibit different degrees of embrittlement. In general, the low-carbon CF-3 grades of cast materials are the most resistant, and the molybdenum-containing CF-8M grades are least resistant to embrittlement. For all grades of cast materials, the extent of embrittlement increases with an increase in ferrite content.

The quantity and distribution of the ferrite within the structure is an important aspect of thermal embrittlement of CASS materials. When the ferrite phase is continuous, e.g., large ferrite content, or the ferrite/austenite phase boundary provides an easy path for cracking, the material is much more susceptible to brittle failure than when the material contains lower ferrite levels. However, even with the decrease in toughness, the fracture margin is still significant due mainly to the high toughness of the austenite matrix.

Various studies have shown that higher ferrite contents (also called delta (δ) ferrite) correlate with increased susceptibility to embrittlement, and ferrite-forming elements such as Cr, Mo, Si, Cb and V increase the susceptibility to thermal embrittlement.

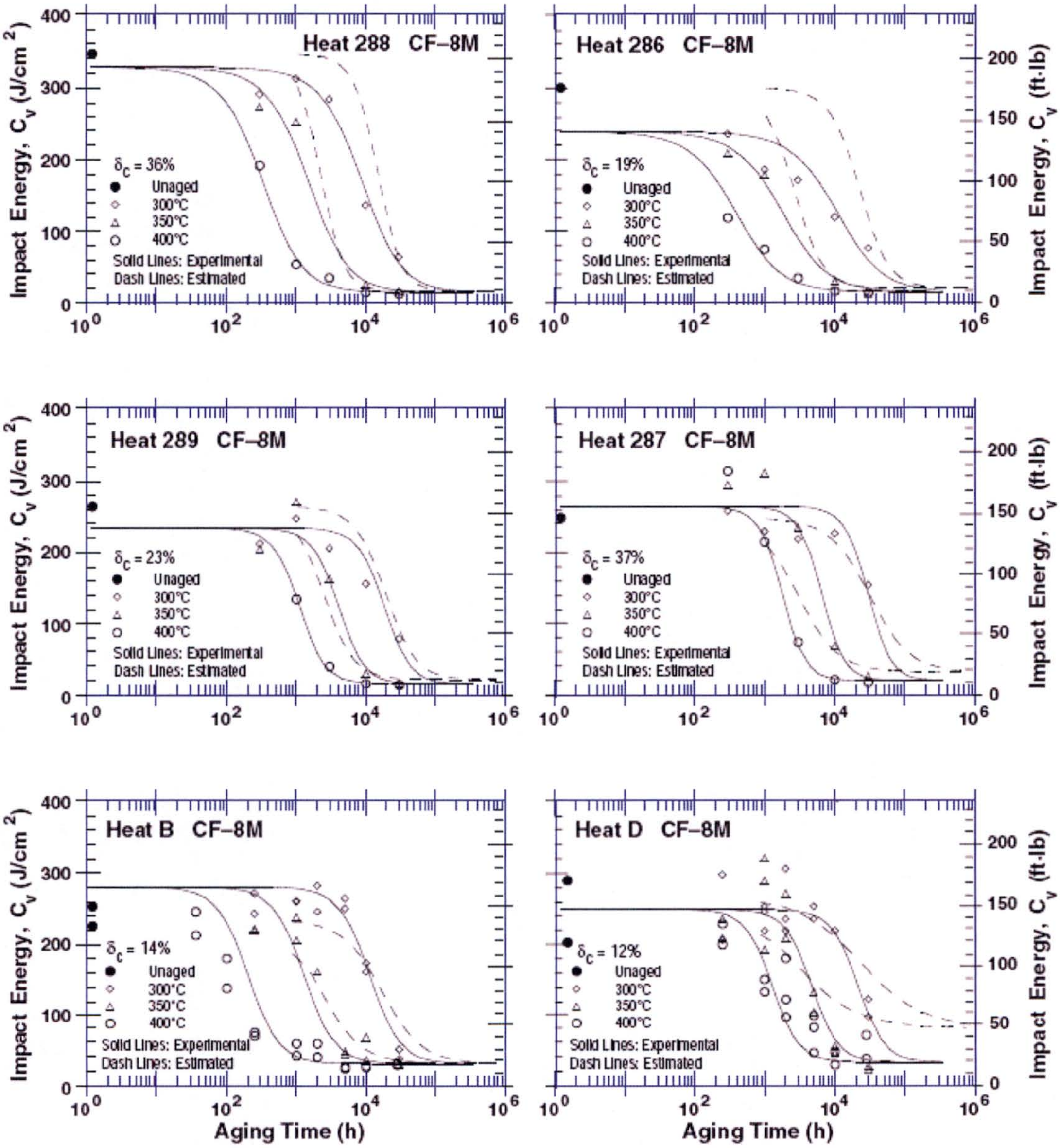

Figure 2-1
Effect of aging temperature on the fracture toughness of sample CF-8M heats

Because of the presence of ferrite, CASS materials show the typical brittle to ductile transition behavior that is often seen in ferritic steels. This is especially true in high ferrite CASS materials. Thermal aging causes the ductile-to-brittle transition temperature to increase and both room temperature and operating temperature toughness decrease with increasing embrittlement of the ferrite phase. Figures 2-2 and 2-3 from Reference [3] show charpy energy as a function of test temperature for different aging conditions for CF-3, CF-8, and CF-8M, respectively. A review of test data for different heats of material in Reference [3] leads to the following observations.

- For centrifugally-cast CF-8 material, 18% δ ferrite is acceptable even with the higher carbon (only 0.036 wt.% for this heat).

- Even at 28% δ ferrite, centrifugally cast low-molybdenum CASS (CF-8) has adequate toughness.
- Statically-cast, low-molybdenum material (CF-8) with relatively high δ ferrite content (≤ 20%) could be screened out from further evaluation.
- The fracture toughness of statically-cast CF-3 and CF-8 material is adequate and show very little aging effects for ferrite levels up to 15% for operating times close to the license renewal term.

The above study supports the traditional screening guideline for low temperature embrittlement of CASS, which is 20% ferrite for low molybdenum (CF-3 and CF-8) materials used in the BWR.

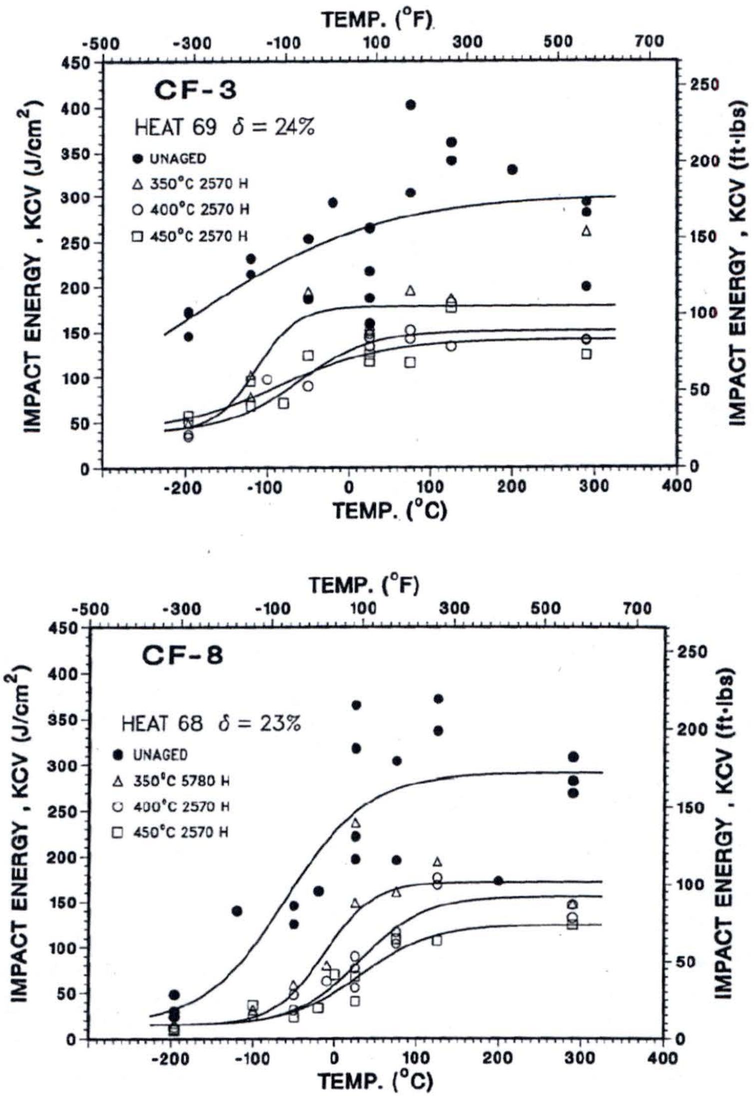

Figure 2-2
Charpy energy vs. test temperature for CF-3 and CF-8 under different aging conditions

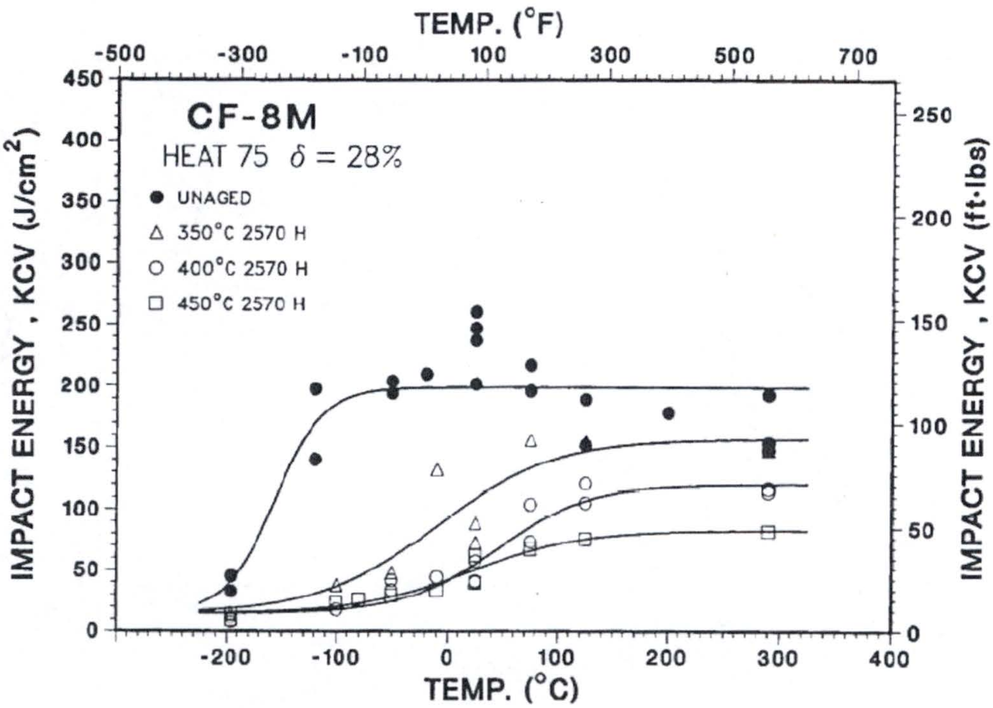

Figure 2-3
Charpy energy vs. test temperature for CF-8M under different aging conditions

The extent of the low temperature embrittlement is generally quantified by measuring the room temperature Charpy energy absorbed after aging at temperatures representative or slightly higher than those expected in service. Figure 2-4 from Reference [1] shows the effect of δ ferrite on material J-R curves for CF-8M. Table 2-1 shows the same data in terms of J_{IC} and J-R curve fits ($J = C\ \Delta a^n$). It is seen that there is a significant reduction in J-R properties at higher ferrite numbers. Figure 2-5 from Reference [4] shows the upper shelf Charpy energy as a function of ferrite number.

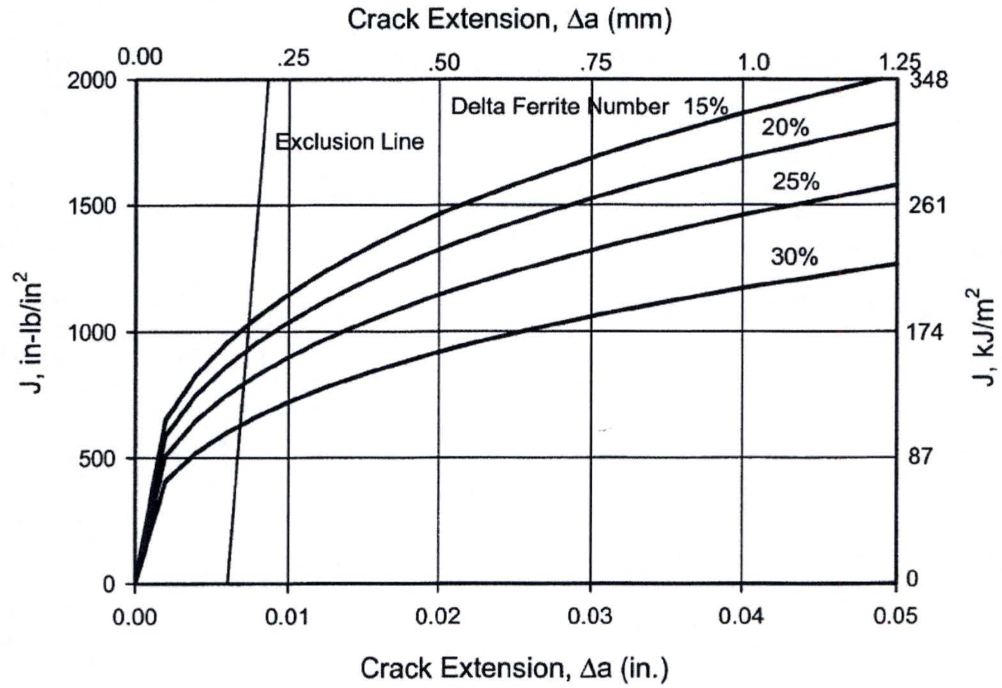

Figure 2-4
J-R curves for CF-8M CASS materials with various δ ferrite numbers

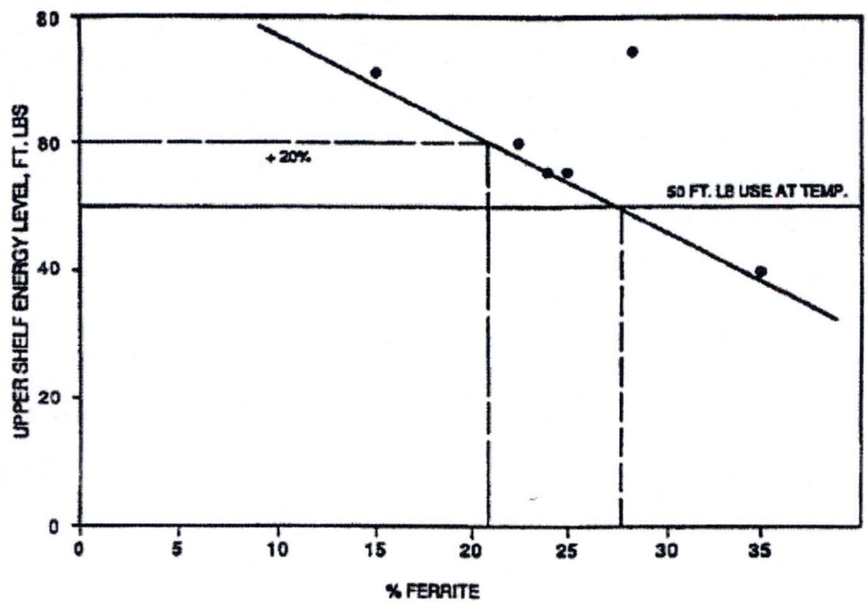

Figure 2-5
Upper shelf energy as a function of δ ferrite number

Table 2-1
J_{Ic} and J-R curve parameters ($J=C\Delta a^n$) as a function of δ ferrite number

DFN	J_{1C}		C		n
	in-lb/in²	kJ/m²	in-lb/in²	kJ/m²	
15	1000	175	5750	325	0.35
20	915	160	5200	294	0.35
25	800	140	4500	254	0.35
30	650	114	3610	204	0.35

For light water reactor applications, traditional guidelines have been that the low temperature embrittlement becomes a concern only when the volume fraction of the ferrite exceeds approximately 15 to 20%. The basis for this argument is that ferrite tends to form in pools in the austenite matrix and when ferrite levels are less than approximately 15%, there is significantly less likelihood that the embrittled ferrite phase can form a continuous path of embrittled material that can extend through the thickness of the cast component.

2.2 Neutron Fluence

As indicated in the GALL report, concurrent thermal embrittlement and exposure to neutron irradiation may result in a synergistic effect wherein the service-degraded fracture toughness can be less than that predicted for either of these processes independently. In the proposed resolution regarding the issue of thermal embrittlement of cast SS components [5], the NRC staff recommends that, to account for the synergistic loss of fracture toughness, "a program should be implemented consisting of either a supplemental examination of the affected components as part of the applicant's 10-year inservice inspection program during the license renewal term, or a component-specific evaluation to determine the susceptibility to loss of fracture toughness." The component-specific evaluation is based on the neutron fluence. The current guidance suggests that if the fluence is greater than 1×10^{17} n/cm^2 (E>1 MeV) (or ~0.15 mdpa) for a component, a supplemental examination is required. Alternatively, a mechanical loading assessment may be conducted to determine if the supplemental inspection program may be eliminated for the component. Considering that even for ferritic steels, the NRC position (as stated in Appendix G of 10CFR50) does not require fracture toughness assessment for fluences less than 1×10^{17} n/cm^2 (E>1 MeV), the threshold for the synergistic effects in CASS is overly conservative.

The EPRI assessment [6] by Nickell suggests that synergistic effects between thermal aging embrittlement and neutron irradiation embrittlement are not measurable for accumulated neutron irradiation at a fluence level of 1×10^{21} n/cm^2 (E>1 MeV) and below. Therefore, unless CASS internal components are subjected to higher neutron fluence levels, the potential synergistic effect of thermal aging embrittlement and neutron irradiation embrittlement need not be evaluated. Considering that radiation effects on mechanical properties of wrought stainless steel are significant at a fluence level of 1×10^{21} n/cm^2 (E>1 MeV), the suggested threshold for synergistic effects might be too high.

Similarly, MRP-175 [7] has identified a fluence level of 1 x 10^{21} n/cm^2 (E>1 MeV) as a screening criteria for the neutron embrittlement of wrought austenitic stainless steels, with a lower value of 6.7 x 10^{20} n/cm^2 for CASS components. These values, however, are based upon an acceptable fracture toughness value for PWR applications, and therefore may not representative of the synergistic effects for BWR applications.

Based on this discussion, it appears that neither the NRC threshold in [5], the EPRI threshold in Reference [6], or the MRP recommendation [7] are realistic or reasonable for BWR application. Clearly, the threshold for synergistic effects lies somewhere between the limiting numbers. In the absence of test data that validate the synergy threshold, a reasonable value of 3 x 10^{20} n/cm^2 is used as the upper bound fluence for application of limit load as defined in [8] which is approved by NRC. Currently, for shroud welds [9] (which are duplex materials similar to CASS) no fracture evaluation is needed for fluences less than 3 x 10^{20} n/cm^2 (E>1MeV). Since these welds are also subject to long term aging at reactor temperature and there are negligible fluence effects on mechanical properties below this value, it is reasonable to postulate that there will be no synergistic aging effects below 3 x 10^{20} n/cm^2 (E>1MeV). Chopra and Shack [10] state "Based on these very limited data and the general mechanism of embrittlement for cast SSs, the minimum fracture toughness of cast SSs can be taken as (a) the minimum predicted toughness for thermal aging for fluences less than 2 x 10^{20} n/cm^2 (E>1MeV) (or ~0.3 dpa) and (b) the lesser of the minimum predicted toughness for thermal aging or the lower bound curves for irradiated SS. The threshold fluence, taken as 0.3 dpa, is a slightly conservative value in light of the limited data and corresponding uncertainty." The suggested threshold of 3 x 10^{20} n/cm^2 (E>1MeV) is close enough to the 2 x 10^{20} n/cm^2 (E>1MeV) suggested by Chopra and Shack. The choice of the 3 x 10^{20} n/cm^2 threshold value is based on consistency with other BWRVIP documents. As discussed later in this report (Table 3-3) there are no CASS components where the 60-year fluence is within the range of 2-3 x 10^{20} n/cm^2 (E>1MeV). Thus, the evaluation remains unchanged regardless of whether the threshold is 2 x 10^{20} n/cm^2 suggested by Chopra and Shack [10] or the 3 x 10^{20} n/cm^2 value suggested in this report.

For fluences that exceed the 3 x 10^{20} n/cm^2 (E>1MeV) threshold, it is reasonable to use the CASS data in Reference [10] for highly irradiated welds. Since many of the irradiated weld test specimens were taken out of operating reactor components, the aging effect is indirectly included. Flaw evaluations based on Reference [11] would provide reasonable estimates of fracture margins and allowable flaw sizes.

3
MATERIALS AND ENVIRONMENT

There are two main forms of embrittlement for cast austenitic stainless steels (CASS): thermal and neutron. Thermal embrittlement is controlled by two key parameters: (1) material composition and (2) time exposed to elevated temperature. The role of material composition on thermal embrittlement is also affected by the secondary parameter of casting method. Neutron embrittlement of CASS is controlled by the chemical composition and fluence that the material will see during its lifetime. To evaluate the chemical composition, the material specifications and certified material test reports (CMTRs) for the cast components were reviewed. Given that the material requirements were similar across the components, and the materials were supplied to several plants, the CMTRs are representative of the complete scope of CASS internals components in all BWRs. In addition, the fluence at the locations for the CASS components was obtained.

3.1 Material Parameters

For the purpose of evaluating the effects of thermal embrittlement, material composition and casting method are the key parameters. Operating temperature also plays a role, but since all BWR CASS components are at a similar temperature, it is not used for the purposes of screening components that could be potentially susceptible to thermal embrittlement. For the cast components in the BWR, a review was conducted of the original material requirements (product specifications), as well as a representative sampling of CMTRs of the components identified as being potentially susceptible. The cast components for individual plants are identified in Table 3-1. As can be seen by this table, the material specified for the cast components was Grade CF8 in almost all cases.

3.2 Product Specifications

To further characterize the material specified for the individual components, a review of Jet Pump component drawings for three plants (BWR/3, 4 and 5) was conducted. All of the drawings specified GE specification B50YP43, which in turn specifies ASME SA-351 or ASTM A-351 grade CF-8 material. Early revisions of this GE specification contained a chemical requirement to control ferrite using a minimum chromium/nickel ratio of 1.9. By the early 1970s, ferrite was specified as a minimum 8% as calculated using the Schaeffler diagram [12]. No maximum ferrite was specified for these materials. It should also be noted that review of the revisions of this specification confirms that molybdenum-containing castings were never specified for these components.

Table 3-1
Material requirements

Component	Material	Applicable Plant (s)
Orificed Fuel Support	CF-8	All
Control Rod Guide Tube Base	CF-3, CF-8	All
Core Spray Sparger Nozzle Elbows	CF-8	All
Jet Pump Transition Piece	CF-8	All*
Jet Pump Restrainer Bracket	Type 304	BWR/3, Vermont Yankee
	CF-8	BWR/4 (except VY), BWR/5, BWR/6
Jet Pump Inlet Mixer Assembly	CF-8	All
Jet Pump Diffuser Collar/Guides	CF-8	All
Jet Pump Inlet Nozzle	CF-8	All
Jet Pump Inlet Elbow	CF-8	All
LPCI Coupling	CF-3, CF-8	BWR/4 (Hope Creek, Limerick) BWR/5 BWR/6

*As noted in BWRVIP-41, Rev. 1, some of the assemblies may be welded Type 304 material, but are assumed to be castings for this evaluation.

3.3 Casting Method

Casting method (centrifugal vs. static) has been identified as a screen for the effects of thermal aging. Centrifugally cast components are allowed a higher percent ferrite before thermal aging is of concern for the high Mo components, and centrifugally cast components that have low Mo content (such as CF-8) are not at risk for thermal aging following exposures of less than 320°C (608°F) for 525,000 hours [4]. Review of CMTRs for the CASS components supplied by GE could not confirm that any of the components were centrifugally cast. It should be noted, however, that this does not affect the final evaluation results, as casting method was not used as a screening criteria.

3.4 Ferrite Content

Since the specification limits do not adequately describe the ferrite content of the CASS materials present in the BWR, a review of CMTRs was performed. Approximately 80 heats of material chemistry were tabulated, as shown in Appendix A. All of the CMTR material was confirmed to be CF-8, consistent with the original product specifications.

Typically, the δ ferrite value for these materials was not reported. Therefore, in order to evaluate the ferrite levels, the δ ferrite, δ_c, was calculated in accordance with Hull's equivalent factors, and if nitrogen was not available, it was assumed to be 0.04% [13]:

$Cr_{eq} = Cr + 1.21\,(Mo) + 0.48\,(Si) - 4.99$

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

$\delta_c = 100.3(Cr_{eq}/Ni_{eq})^2 - 170.72(Creq/Nieq) + 74.22$

The ferrite calculations are summarized in Table 3-2, and the complete list of calculated ferrite values for the individual heats is tabulated in Appendix A. As can be seen, the ferrite levels are below the 20% threshold normally associated with thermal aging concerns [5]. In addition, a statistical evaluation of the data showed that there is a 99.8% confidence that the ferrite level will be below the 20% ferrite limit.

Table 3-2
Summary of CMTR data

Parameter	Value
Average % Ferrite	10.12
Standard deviation	3.37
Ferrite Range	3.21 to 18.8
Percent Ferrite Calculated from Average Chemistry	9.65
Percent of Castings Below 20% Limit for Ferrite	99.8

3.5 Fluence Effects

In order to evaluate the synergistic effects of neutron fluence, fluence values for CASS BWR components were obtained. Calculations have been performed for a BWR/6, 218 inch plant at extended power uprate (EPU) conditions in accordance with RG 1.190 methods [4]. Calculations for a BWR/4, 251 inch plant and BWR/5 251 inch plant for EPU conditions were also reviewed, and the BWR/6, 218 inch plant fluence values were bounding. For each of the ten CASS components of concern, the estimated peak fluence for each location is listed in Table 3-3; these values are for 60 years of operation. All of the cast components, except for the control rod guide tube base and core spray sparger elbows, are exposed to a neutron fluence greater than the 3×10^{20} n/cm^2 (E>1 MeV) discussed in Section 2.5.

Note: In order to apply the inspection requirements for the CASS components shown in Table 6-1, (i.e., no additional inspections, beyond those currently required in BWRVIP I&E Guidelines, are needed to manage the aging effect of loss of fracture toughness of BWR reactor vessel internal components constructed of CASS), the utility must demonstrate by plant-specific fluence assessment, that the projected neutron fluence for their plant is bounded by the maximum fluence stated in Table 3-3.

Table 3-3
Fluence values for CASS components

Component	
Orificed Fuel Support	
Control Rod Guide Tube Base	
Core Spray Sparger Nozzle Elbows	
Jet Pump Transition Piece	
Jet Pump Restrainer Bracket	
Jet Pump Inlet Mixer Assembly	
Jet Pump Diffuser Collar/Guides	
Jet Pump Inlet Nozzle	
Jet Pump Inlet Elbow	
LPCI Coupling	

4
COMPONENT SCREENING

This section describes several ways to screen out BWR CASS components from augmented inspection requirements. The screening options are based on NRC positions as stated in [5] and [15]. The screening criteria are based on several considerations: ferrite content, fluence, available fracture toughness, applied stress and current inspection practice. These are briefly discussed in the following paragraphs.

4.1 Screening Based on Ferrite Content

As part of the evaluation of passive, long-lived reactor structures for license renewal, the NRC staff has proposed screening criteria to determine the susceptibility of cast SS components to thermal aging embrittlement [3]. The thermal embrittlement criteria are outlined in Table 4-1. Essentially it states that for low Molybdenum statically cast CASS (<0.5 wt.% Mo) the threshold ferrite (below which thermal embrittlement is unlikely) is 20%. This means that CF-3, CF-3A and CF-8 with percent ferrite less than 20% are unlikely to experience significant loss of toughness due to thermal embrittlement. For statically cast CASS with Mo>2% wt.% (CF-3M and CF-8M) the ferrite threshold is 14%. For components found or assumed to be potentially susceptible, an aging management program is required for the license renewal period.

Table 4-1
Screening criteria based on ferrite number

CASS Thermal Aging Susceptibility Screening Criteria			
Mo Content (Wt.%)	**Casting Method**	**Ferrite Level or Content**	**Susceptibility Determination**
High (2.0 to 3.0)	Static	≤ 14%	Not susceptible
		> 14%	Potentially Susceptible
	Centrifugal	≤ 20%	Not susceptible
		> 20%	Potentially Susceptible
Low (0.5 max.)	Static	≤ 20%	Not susceptible
		> 20%	Potentially Susceptible
	Centrifugal	ALL	Not susceptible

Cast stainless steel components have not experienced IGSCC in BWRs. Even in Type 308 stainless steel welds (which are duplex materials similar to CASS), crack initiation has been extremely rare except in a few cases where the ferrite content was less than 5%. CASS components in BWRs require a minimum ferrite level of 8%. Hughes, Clarke and Delwiche [16]

performed an extensive study of the IGSCC resistance of CASS components in high temperature BWR environment. The tests involved CF-3, CF-3A, CF-8, CF-3M and CF-8M specimens in the as welded and furnace sensitized condition. Other tests considered the effects of nitriding (to simulate CRD CASS components) and stellite weld application (to simulate the inlet mixer belly band and the restrainer bracket). Several types of tests with aggressive loading were performed: i) constant extension rate tests (CERT), ii) pipe tests, iii) variable load tests and iv) constant load tests. Figure 4-1 from Reference [16] shows the influence of carbon content and percent ferrite on IGSCC susceptibility in CASS. The results of their tests are summarized here:

- Regardless of the percent ferrite there was no IGSCC in welded CASS with <0.05 wt.% carbon. This means that CF-3, CF-3M and CF-3A components are resistant to IGSCC regardless of the ferrite content.
- No SCC was observed beyond 12% ferrite, regardless of the carbon content. The curve representing the combination of carbon content and percent ferrite for IGSCC is somewhat lower for furnace sensitization compared to welded CASS, i.e. for a given percent ferrite, the maximum carbon content (above which IGSCC can occur) is lower for furnace sensitized CASS than for welded CASS.

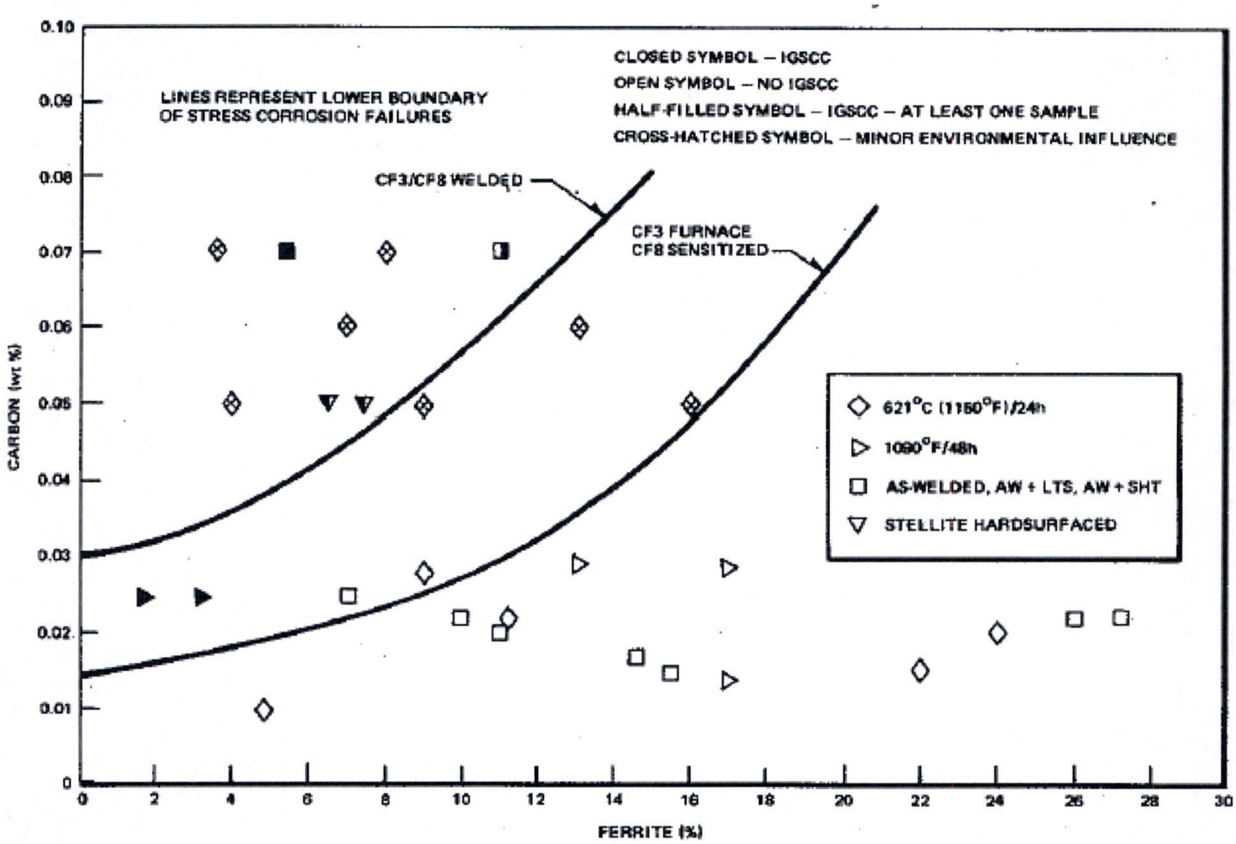

Figure 4-1
Role of carbon and ferrite content on IGSCC potential in CASS components [16]

Based on this assessment, the likelihood of crack initiation in CASS components is low. In particular, the case can be made that environmentally assisted cracking of low carbon CASS (CF-3, CF-3A and CF-3M) is extremely unlikely.

Fatigue due to system cycling is not a significant issue since all the CASS components in question are reactor internals and not subjected to either pressure or thermal cycling of any significance. The only source of fatigue usage is from potential vibrations in CASS jet pump components. The jet pump (JP) assembly (including the inlet mixer and restrainer bracket) is routinely inspected as part of the BWRVIP-41, Rev. 1 [17] inspections. No cracking has been found in CASS components in the JP assembly. Thus, while fatigue cracking cannot be ruled out, field experience in the past 30 years suggests that crack initiation due to fatigue in any of the CASS components is unlikely and the existing inspections per BWRVIP-41, Rev. 1 are sufficient.

Since the threshold for crack initiation by IGSCC is <12% ferrite based on the study in Reference [16] and the threshold for thermal aging embrittlement is 14% or higher, one can argue that concurrent crack initiation and loss of toughness due to thermal aging is unlikely and additional evaluation is not needed. However, many CASS reactor internals components are subject to neutron irradiation and therefore, the ferrite levels by themselves are not adequate to exempt CASS components from augmented inspections. Therefore, additional screening criteria must be considered as discussed in the following sections.

4.2 Screening Based on Fluence

As stated previously, the threshold for the synergistic effect of fluence is defined to be $1x10^{17}$ n/cm^2 (E>1MeV) in the GALL report [5, 15]. Below this value, additional evaluation to consider irradiation effects is not needed (over and above that for thermal aging). However, the fluence estimates for BWR CASS internals shows that almost all the components exceed the threshold for fluence regardless of which threshold value (1 x 10^{17} n/cm^2 in Reference [5], 2 x 10^{20} n/cm^2 in Reference [10] or 3 x 10^{20} n/cm^2 value recommended in this report) is selected. The only component that can be screened out is the control rod guide tube base and the core spray sparger elbows, as shown in Table 3-3. All other components must be considered for more detailed evaluation.

4.3 Screening Based on Toughness

The GALL report [15] also provides an alternative screening criterion employing fracture toughness. It is based on the recommendation that a fracture toughness value of 255 kJ/m^2 (1,450 in-lb/in.^{2}) at a crack depth of 2.5 mm (0.1 in.) can be used to differentiate between CASS materials that are non-susceptible and those that are subject to thermal aging embrittlement [4]. As stated earlier, the CF-3 and CF-8 materials used in the BWR CASS internals are not susceptible to thermal aging, but still must be evaluated for the combination of thermal aging and neutron embrittlement.

There is very little data on the fracture toughness of irradiated CASS, let alone data on CASS material that is both aged at temperature and subjected to irradiation. One way of estimating toughness is to use irradiated toughness data on stainless steel weldments [8][1]. Much of this data is derived from specimens taken from operating plants, so the data considers the synergistic effect of thermal aging and irradiation. Two methods are used to estimate the J values at a crack depth of 2.5 mm. This will be used to screen the different BWR CASS components against the NRC approved criterion of 255 kJ/m^2 (1,450 in-lb/in.^{2}) at a crack depth of 2.5 mm.

The bounding fluence (see Table 3-3) is [redacted] for all other CASS components. Therefore, the fracture toughness assessment will be based on a fluence of [redacted].

Table 4-2 and Figure 4-2 from Reference [8] show test data for irradiated stainless steel subjected to different fluence levels. The J-R curves were fit using the equation $J = C\ (\Delta a)^n$ and the table provides values for C and n for the different tests. It can be seen that the limited weld data exceeds 225 kJ/m^2 in the 4.0-4.8 x 10^{20} n/cm^2 fluence range. Only two curves are available for the appropriate fluence level. These points, irradiated to 6.3 x 10^{20} n/cm^2 and 6.4 x 10^{20} n/cm^2, respectively, are listed in Table 4-3. Both are for wrought material with the irradiation temperature being 536°F (280°C), which is close to the irradiation temperature for BWR CASS components). The test temperatures were 390°F (199°C) and 480°F (249°C) respectively. Table 4-3 shows the calculated J values at a crack depth of 2.5 mm. The second data point (638 kJ/m^2) in Table 4-3 for 6.4 x 10^{20} n/cm^2 fluence is more appropriate since the test temperature was at 480°F, which is closer to the operating temperature of most BWR internals. However, the data in Table 4-3 is valid for base (wrought) material. Toughness data for weldments under similar conditions is expected to be lower. One way of accounting for this is to use the Z-factors in Appendix C of Section XI [18] to reduce the toughness so that the equivalent toughness for weldments can be estimated. The basis for the Z-factors is described in Reference [19]. The methodology involved the determination of multipliers to be applied on the stress so that the fracture mechanics (J-T stability) results are represented by limit load evaluation using the stresses multiplied by the Z-factors. Essentially the Z-factors represent a way of accounting for the lower toughness of weld metal when applied to wrought material data. Z-factors are presented for both SMAW and SAW welds (flux welds). Z-factors need not be applied for non-flux (e.g. GTAW) welds since they have toughness values comparable to wrought materials. The Z-factor for SAW welds are higher and are given as a function of pipe diameter by the following expressions [19]:

$$Z_{SAW} = 1.30[1+0.010(OD-4)]$$

where OD is in inches and Z is the factor for SAW.

[1] The data in Reference [8] is mostly from wrought stainless steel with a limited amount of data points from welds.

Table 4-2
Experimental J-R data from [8] for irradiated stainless steel

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Figure 4-2
Experimental J-R curve data from reference [8]

Table 4-3
J at Δa=2.5 mm calculated values at 6.3×10^{20} n/cm^2 fluence (taken from Table 4-2)

Spec ID	Material	Type	Irradiation Temperature °F (°C)	Test Temperature °F (°C)	Fluence (n/cm^2)	C (kJ/m^2)	n	J at 2.5 mm (kJ/m^2)
2, CT	Type 304	Base	536 (280)	390 (199)	6.3×10^{20}	516	0.4	744
2, CT	Type 304	Base	536 (280)	480 (249)	6.4×10^{20}	426	0.44	638

The diameter of most of the CASS components is less than 12 inches, so a conservative estimate for Z is 1.3[1+0.01(12-4)] = 1.404. This is the multiplier on stress. Since J is dimensionally proportional to K^2 or σ^2, a conservative penalty on J is to divide the wrought toughness by Z^2. Based on this, the estimated weld toughness is $638/1.404^2 = 324$ kJ/m^2. This is in excess of the required value of 255 kJ/m^2 (1,450 in-lb/in.^{2}). Based on this, it can be concluded that all the CASS components can be considered not susceptible to thermal or fluence embrittlement and no augmented inspections are warranted.

An alternate way to estimate the toughness is to use the recommended lower bound properties in Reference [8]. A power law fit was used to construct a line that bounds the available data for C as a function of fluence. The power law fit for n was defined as a function of fluence so that when it is used in combination with the bounding relationship for C, the resulting predicted J-R curves match or are conservative compared to the experimental J-R curves. The data and corresponding power law fits for C and n as a function of fluence are shown in Figures 4-3 and 4-4, respectively.

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Figure 4-3
J-R curve coefficient C as a function of fluence for stainless steel base, HAZ and weld materials [8]

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Figure 4-4
J-R curve parameter n as a function of fluence for stainless steel base, HAZ and weld materials [8]

For the fluence value of the value of n is 0.48 and the value of C is 190 kJ/m^2. Since this is the lower bound for base, heat affected zone (HAZ) and weld material, no additional correction is needed for the weldment toughness. Substituting these values, the lower bound J at 2.5 mm crack extension is 295 kJ/m^2. As in the previous case, this is in excess of the required value of 255 kJ/m^2 (1,450 in-lb/in.^{2}). Based on this, it can be concluded that all the CASS components can be considered not susceptible to thermal or fluence embrittlement and no augmented inspections are warranted.

4.4 Screening Based on Stress

NUREG-1801 allows as an alternative, the use of component-specific evaluation, including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or less than 5 ksi (34.5 MPa) and thus low enough to preclude fracture, then supplemental inspection of the component is not required. NUREG-1801 does not specify whether the limit on applied stress is intended for the membrane stress, membrane + bending or peak stress. Considering that the final application is for a fracture assessment, it is reasonable to assume that the 5 ksi (34.5 MPa) stress limit applies to membrane stress or membrane + bending stress. The specification of the same stress limit (5 ksi or 34.5 MPa) regardless of the conditions (Levels A, B, C or D) is somewhat inconsistent with the generally accepted ASME Code philosophy where the allowable values are higher for low probability events. In general, stress analysis of reactor internals is based on conservative assumptions and most components might not meet the conservative 5 ksi (34.5 MPa) limit, especially when the stresses during low probability events such as seismic and accident conditions are compared with the stress limit. Nevertheless, Section 5 will discuss the stress analysis of BWR CASS internals.

4.5 Screening Based on Current BWRVIP Inspections

The requirement to address the enbrittlement issue is based on the interpretation that augmented inspections are required unless the screening criteria described above are met. However, BWR internals are already being inspected under the different BWR Vessel and Internals Project (BWRVIP) Inspection and Evaluation Requirements. BWRVIP-41, Rev. 1 [17] specifies inspection requirements for the jet pump assembly. Similarly, BWRVIP-18, Rev. 1 [20] requires inspections of the core spray sparger nozzle elbows, BWRVIP-42-A [21] specifies inspections for the LPCI coupling and BWRVIP-47-A [22] specifies inspections for the control rod guide tube base and fuel support alignment pin.

5
STRESS EVALUATION

This section summarizes the results of the stress analysis for some of the different CASS internals.

5.1 Orificed Fuel Support

The orificed fuel support (OFS) is part of the lower plenum components covered by BWRVIP-47-A [19]. Figure 5-1 shows the schematic of the orificed fuel support. The stress analysis of the OFS is described in Reference [23]. The main steady state loading on the component is due to internal pressure differences. The ΔP values and the associated stresses are shown in Table 5-1.

In addition to the pressure stresses, the stresses for seismic loading were determined by testing due to the complex geometry of the OFS. The testing methodology included the use of simulated horizontal and vertical loads to represent the seismic loading expected for the OFS. Under normal operating conditions, the OFS does not see significant loads (only differential pressure, and the weight of the fuel). Under OBE and SSE conditions, the vertical component of the seismic load is transferred from the fuel through the OFS to the Control Rod Guide Tube (CRGT). The horizontal seismic component is transferred through the OFS, the CRGT, and then to the core plate. For the OFS, the stress intensity for the upset seismic event (OBE) was [redacted] The stress intensity for the faulted seismic event (SSE) was [redacted] MPa). There is no information on the actual tensile stress (rather than the stress intensity) but the stress intensity values were high enough that the NRC limit of [redacted] is most likely exceeded.

Table 5-1
Orificed fuel support ΔP values and associated stresses

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Figure 5-1
Schematic of the orificed fuel support assembly

5.2 Control Rod Guide Tube Base

The control rod guide tube base is excluded from augmented inspection because of the low fluence ($<1 \times 10^{17}$ n/cm^2) and the fact that it is made of CF-3 or CF-8 with ferrite number less than 12. Based on this, evaluation or augmented inspection for thermal aging or neutron embrittlement is not needed. Therefore, evaluation based on stress is not necessary. The control rod guide tube base is part of the lower plenum components covered by BWRVIP-47-A [22].

5.3 Core Spray Sparger Nozzle Elbows

The core spray sparger nozzle elbows are not subjected to any stresses except during core spray injection. The core spray ΔP during injection can range from . Based on the stresses reported in a several flaw evaluation handbooks, the Pm+Pb value for the core spray sparger pipe is estimated to be less than 5 ksi (34.5 MPa). The stress in the elbow is expected to be less than that in the pipe (because of the much lower diameter). Based on this stress assessment, it is clear that the stress is below the 5 ksi (34.5 MPa) threshold and evaluation or augmented inspection for thermal aging or neutron embrittlement is not needed. Inspections of the core spray piping and sparger are covered by BWRVIP-18, Rev. 1 [20].

5.4 Jet Pump Assembly

Inspections of the jet pump assembly components are covered by BWRVIP-41, Rev. 1 [17]. The CASS components in the jet pump assembly include the transition piece, the inlet mixer assembly and the diffuser collar. Using the designations in BWRVIP-41, Rev. 1, the weld locations of interest are locations RS-3, the weld between the transition piece and riser pipe, IN-1 through IN-5 in the inlet elbow/nozzle, locations MX-1 through MX-7 in the inlet mixer and locations DF-1 through DF-3 in the diffuser. Figure 5-2 shows the different jet pump assembly weld designations, with the CASS weld locations in bold type. Since the jet pump assembly was not analyzed in detail in the original design, the stresses in the jet pump assembly were taken from a typical flaw evaluation handbook for a BWR/4 and a BWR/6 plant [24]. Although not all CASS component evaluations are included, Table 5-2 shows that the stresses at the different locations exceed the 5 ksi (34.5 MPa) limit.

Table 5-2
Example stresses at different jet pump assembly locations

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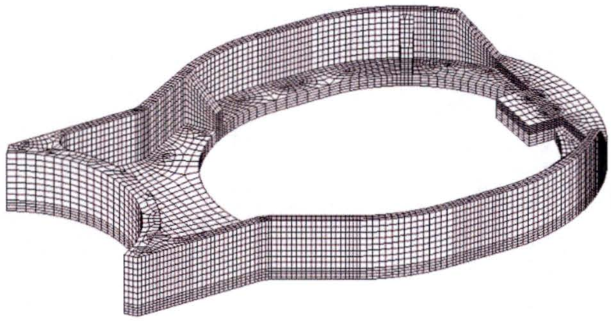
Figure 5-2
Jet pump assembly weld designations

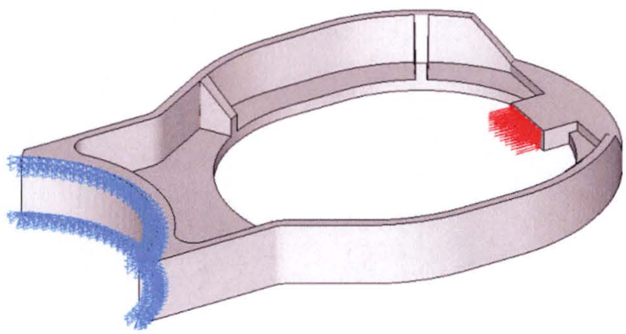
5.5 Jet Pump Restrainer Bracket

The restrainer bracket merely serves to provide point support to the inlet mixer and does not experience any significant stress due to pressure or thermal loading. The only source of stress in the restrainer bracket is due to vibratory loading on the restrainer bracket pad. For nominal set screw gaps (<5 mils) the loading on the restrainer bracket pad due to vibratory loading is estimated to be 1120 lb (508 kg). Since there was no available stress analysis of the restrainer

bracket, a finite element analysis was performed to determine the stresses in the bracket. Figure 5-3 shows the finite element model and the boundary conditions in the model. Figure 5-4 shows the stress analysis results. It is seen that the stresses are generally low everywhere except for the corner location of the restrainer bracket pad. The linearized stress, Pm+Pb is and exceeds the stress threshold.



Analysis model



Supported at riser welds
load applied as pressure load

Figure 5-3
Restrainer bracket finite element model

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Figure 5-4
Restrainer bracket stresses

5.6 LPCI Coupling

Inspections of the LPCI coupling components are covered by BWRVIP-42-A [21]. There are two types of LPCI couplings: the straight through coupling used in BWR/4-5 plants (Figure 5-5) and the offset style coupling (Figure 5-6) used in BWR/6 plants.

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Figure 5-5
Typical BWR/4-5 straight through LPCI coupling design

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Figure 5-6
BWR/6 design LPCI coupling

BWRVIP-42-A describes the details and the inspection/evaluation requirements for the two coupling designs. In the BWR/4-5 design, the sleeve is made of CASS material whereas in the BWR/6 design, both the sleeve and the collar are made of CASS material. The LPCI coupling has a calculated end-of-license renewal period of [redacted] which is similar to other CASS components in the BWR internals. Stress analysis information for the BWR/6 design [25] is summarized below. The analysis focused on the region of the weld between the LPCI ring (attached to the shroud) and the pipe extension (attached to the elbow). The highest stresses are in the region of the ring to elbow extension attachment, in the elbow and strut weld area. Only the high stress locations were reported, and therefore, there is limited information on the stresses in the CASS components (sleeves or collars).

The main sources of loading on the sleeves and collars are the pressure differences ΔP as shown in Table 5-3:

Table 5-3
Pressure differences on sleeve and collar under various conditions

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Assuming a diameter of 10 in. (254 mm) and a wall thickness of 0.3 in. (7.6 mm) the maximum pressure stress corresponding to the faulted condition is [redacted] which is well below the 5 ksi (34.5 MPa) limit. Thermal and seismic stresses are not expected to be significant in the sleeve and collar because of the slip joint. Nevertheless the limiting primary stresses from Reference [25] are shown in Table 5-4.

Table 5-4
Limiting thermal and seismic stresses in the LPCI coupling

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The limiting stresses in the above table were obtained in the strut, at the point where the strut attaches to the shroud. The loading on the sleeve and collar is relatively low; however, with no specific stresses on the sleeve and collar, the conservative approach is to assume that the LPCI coupling stresses exceed the 5 ksi (34.5 MPa) limit. Although the BWR/4-5 design was not specifically evaluated, the stresses are also assumed to exceed the 5 ksi (34.5 MPa) limit.

6 ASSESSMENT OF CASS COMPONENTS

This section describes the assessment of the need for additional analysis or augmented inspection based on the NRC criteria outlined in [5] and [15]. In addition, the BWRVIP inspection requirements outlined in BWRVIP-41, Rev. 1 [17], BWRVIP-18, Rev. 1 [20], BWRVIP-42-A [21] and BWRVIP-47-A [22] will be considered in the recommendations on augmented inspections. As described in Section 3, all the BWR CASS internals are made of CF-3 and CF-8 the average ferrite number is around 10. Therefore, thermal embrittlement alone is not an issue. The synergistic effects of neutron and thermal embrittlement are the primary focus.

6.1 Assessment of Orificed Fuel Supports

The stress due to pressure differences are within the 5 ksi (34.5 MPa) limit, but the seismic stresses exceed the limit. Therefore the stress criterion cannot be used to screen out the orificed fuel support (OFS). The fluence exceeds the 3×10^{20} n/cm^2 threshold. However, the OFS does meet the fracture toughness threshold value of 255 kJ/m^2 (1,450 in-lb/in.^{2}) at a crack depth of 2.5 mm. Thus, additional evaluation or augmented inspection is not needed. Although the OFS is included in the scope of BWRVIP-47-A, no inspections are required. In light of the fact that the OFS meets the toughness criterion and the fact that the steady state stresses are low, augmented inspections are not warranted.

6.2 Assessment of Control Rod Guide Tube Base

The control rod guide tube base is excluded from augmented inspection because of the low fluence (<1E17 n/cm^2) and the fact that it is made of CF-3 or CF-8 with ferrite number less than 12. Based on this, evaluation or augmented inspection for thermal aging or neutron embrittlement is not needed. Therefore, augmented inspections are not warranted.

6.3 Assessment of Core Spray Sparger Nozzle Elbows

The core spray sparger nozzle elbows meet both the 5 ksi (34.5 MPa) stress limit and the fluence limit. Per BWRVIP-18, Rev.1, the nozzle welds (S3 see Figure 6-1) are inspected with VT-1. If no cracking is detected, the inspection is complete. If cracking is detected, an EVT-1 of the cracked location is performed to better determine crack length for flaw evaluation. Reinspection includes a rotating sample. Since VT-1 or EVT-1 of the weld is already required, it is likely that the elbow itself would be visible as part of the inspection. In light of the fact that the stress threshold is met and the fact that inspections of the elbow welds are already being performed, no augmented inspections are warranted.

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Figure 6-1
Core spray sparger nozzle weld

6.4 Assessment of the Jet Pump Assembly

Although the CASS components in the jet pump assembly (the transition piece including the elbow, the inlet mixer and the diffuser collars) do not meet the 5 ksi (34.5 MPa) stress limit, the CASS jet pump assembly components do meet the fracture toughness threshold value of 255 kJ/m^2 (1,450 in-lb/in.^{2}) at a crack depth of 2.5 mm. Therefore, augmented inspections are not warranted.

BWRVIP-41, Rev. 1 already specifies some inspections of CASS welds. For example, the inlet elbow locations (IN-1 through IN-5), the mixer locations (MX-1 through MX-7) and diffuser collar locations (DC-1 through DC-4 and DF-1) are all covered by BWRVIP-41, Rev. 1. Figures 6-2 through 6-4 show the different inspection locations for CASS components. Most of the CASS weld inspections (where specified) are confined to the stainless steel welds that attach to the CASS material. For example, for the inlet elbows, the baseline inspection includes EVT-1 of 100% of HAZs on the mixer side of the weld. Similar inspections are required for the mixer weld locations. It should be noted, however, that the most likely source of degradation (however unlikely) for the CASS jet pump components would be IGSCC. Since the inspections cover the wrought side of the welds (which are considered more susceptible to IGSCC), the inspections would detect potential flaws before they become significant. Thus, the current inspections will provide information on the integrity of the CASS components and no augmented inspections are warranted.

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Figure 6-2
Inlet elbow inspection locations

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Figure 6-3
BWRVIP-41, Rev. 1 inlet mixer inspection locations

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Figure 6-4
Diffuser collar inspection locations (DC-1 through DF-4 and DF-1)

6.5 Assessment of the Jet Pump Restrainer Bracket

The jet pump restrainer bracket does not meet the 5 ksi (34.5 MPa) stress limit, but does meet the 255 kJ/m^2 (1,450 in-lb/in.^{2}) toughness requirement. Per BWRVIP-41, Rev. 1 inspection requirements, the restrainer bracket pad is frequently inspected for detection of wedge wear and/or rod wear from above and below the restrainer bracket. Thus, any cracking in the restrainer bracket would be detected. Consequently, no augmented inspections are warranted.

6.6 Assessment of the LPCI Coupling

The LPCI assembly CASS components (sleeve and collar) may have lower stresses, but based on the limited information in the stress report (which shows that the limiting locations exceed the 5 ksi (34.5 MPa) limit and does not provide stress information on the sleeve or the collar), it is assumed that the CASS components exceed the 5 ksi (34.5 MPa) limit. However, given that the CASS portions of the LPCI coupling are above the jet pumps and outside of the shroud, the fluence at this location is less than 5 x 10^{20} n/cm^2 (E>1 MeV). Since the fluence is less than the values of 6 x 10^{20} n/cm^2 (E>1 MeV) used to determine compliance with the 255 kJ/m^2 (1,450

in-lb/in.^{2}) toughness requirement, the CASS portions of the LPCI couplings are demonstrated to have adequate toughness. Based on this, the need for additional evaluation or augmented inspection is not necessary.

6.7 Summary of Assessment

Table 6-1 summarizes the assessment of the BWR CASS internals. As substantiated by the review of CASS CMTRs, the BWR materials have average ferrite levels of 10% with very high confidence that the levels are below 20%. The inspection recommendations given are based on the following criteria: (1) fluence <3 x 10^{20} n/cm^2 (E>1 MeV); (2) adequate toughness (>255 kJ/m^2) and (3) applied stresses (<5 ksi). If a component meets any one of the three criteria, then no augmented inspections are required. As shown in Table 6-1, the CASS materials have been demonstrated to meet at least one of the criteria, and no augmented inspections are recommended for CASS components. All CASS components either experience fluence levels of no concern, or the CASS material at the maximum fluence has adequate toughness. Several of the CASS components also experience stresses for which brittle fracture would not be an issue under any condition.

Note: In order to apply the inspection requirements for the CASS components shown in Table 6-1, (i.e., no additional inspections, beyond those currently required in BWRVIP I&E Guidelines, are needed to manage the aging effect of loss of fracture toughness of BWR reactor vessel internal components constructed of CASS), the utility must demonstrate by plant-specific fluence assessment, that the projected neutron fluence for their plant is bounded by the maximum fluence stated in Table 3-3.

Table 6-1
Assessment of BWR CASS internals

Component	Material	Ferrite Content <20% ?	Fluence (n/cm^2)	Toughness Requirement of 255 kJ/m^2 (1,450 in-lb/in.^{2}) met?	*Stress <5 ksi (34.5 MPa)?	Evaluations Results
Orificed Fuel Support	CF-8	Yes		Yes	No	No augmented inspections are warranted.
Control Rod Guide Tube Base	CF-3 or CF-8	Yes		Yes	Yes	No augmented inspections are warranted.
Core Spray Sparger Nozzle Elbows	CF-8	Yes		Yes	Yes	No augmented inspections are warranted.
Jet Pump Assembly	CF-8	Yes		Yes	No	No augmented inspections are warranted.
Jet Pump Restrainer Bracket	CF-8	Yes		Yes	No	No augmented inspections are warranted.
LPCI Assembly	CF-3 or CF-8	Yes		Yes	No	No augmented inspections are warranted.

* Augmented inspection is not required if either the toughness criteria or the stress limit requirements are met.

7
CONCLUSIONS

NUREG-1801 Rev. 1, Section XI.M13 states that a Section XI VT-3 examination is required to be performed of reactor internal components. In addition, the GALL report specifies that for the license renewal period these inspections shall be augmented by an aging management program to address the synergistic effects of thermal aging and neutron embrittlement in cast austenitic stainless steels (CASS). This aging management program consists of (a) identifying susceptible components; and (b) either performing additional inspections of these components, or performing a component-specific evaluation to confirm that the stresses (tensile loading) in the components are sufficiently low such that augmented inspections are not warranted.

This report provides an evaluation of cast austenitic stainless steel components in BWR internals, and determines where additional evaluation or augmented inspections of CASS components are needed. The evaluation covers the orificed fuel support, the control rod guide tube base, the core spray sparger nozzle elbow, the CASS components in the jet pump assembly (inlet elbow, inlet mixer, diffuser collar and restrainer bracket) and the LPCI coupling.

The evaluation includes assessment of ferrite, fluence estimates, stress analysis and fracture toughness estimation. The results are compared with the NRC approved criteria – ferrite number, stress limit (5 ksi or 34.5 MPa) and toughness (255 kJ/m^2 or 1,450 $in\text{-}lb/in.^2$) – to determine whether component-specific evaluation or augmented inspections are needed. In addition to the NRC criteria, the current BWRVIP inspections were also evaluated to determine whether they already meet the intent of the NUREG-1801 requirement. Three components were identified as having fluence levels $>3 \times 10^{20}$ n/cm^2 sufficient to warrant further evaluation: (1) orificed fuel support, (2) the CASS components in the jet pump assembly (inlet elbow, inlet mixer and diffuser collar and the restrainer bracket and (3) the LPCI coupling. Based on stress and fracture toughness considerations, no augmented inspections of the CASS components are warranted.

Note: In order to apply the inspection requirements for the CASS components shown in Table 6-1, (i.e., no additional inspections, beyond those currently required in BWRVIP I&E Guidelines, are needed to manage the aging effect of loss of fracture toughness of BWR reactor vessel internal components constructed of CASS), the utility must demonstrate by plant-specific fluence assessment, that the projected neutron fluence for their plant is bounded by the maximum fluence stated in Table 3-3.

8 REFERENCES

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A
MATERIAL DATA

Table A-1
Evaluated CMTR data and calculated δ ferrite

Content Deleted -
EPRI Proprietary Information

Table A-1
Evaluated CMTR data and calculated δ ferrite (continued)

Content Deleted -
EPRI Proprietary Information

Table A-1
Evaluated CMTR data and calculated δ ferrite (continued)

Content Deleted -
EPRI Proprietary Information

B
NRC REQUEST FOR ADDITIONAL INFORMATION DATED SEPTEMBER 29, 2011

BWRVIP 2011-175A

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

September 29, 2011

Mr. David Czufin, Chairman
Exelon Generation
Chairman, BWR Vessel and Internals Project
Electric Power Research Institute
3420 Hillview Avenue
Palo Alto, CA 94304-1395

SUBJECT: ACCEPTANCE FOR REVIEW AND REQUEST FOR ADDITIONAL INFORMATION, FOR BWRVIP-234: "BWR VESSEL AND INTERNALS PROJECT: THERMAL AGING AND NEUTRON EMBRITTLEMENT EVALUATION OF CAST AUSTENITIC STAINLESS STEEL FOR BWR INTERNALS" (TAC NO. ME5060)

Dear Mr. Czufin:

By letter dated September 10, 2010 (Agencywide Documents Access and Management System Accession No. ML102570749), the Boiling Water Reactor (BWR) Vessel and Internals Project (BWRVIP) submitted to the U.S. Nuclear Regulatory Commission (NRC) staff for review Technical Report (TR) "BWRVIP-234: BWR Vessel and Internals Project: Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steels for BWR Internals." Per your request, the Office of Chief Financial Officer granted the fee exemption (ADAMS Accession No. ML102990404) on October 26, 2011.

Upon detailed review of the provided information, the NRC staff determined that additional information is needed to complete the review. During the conference call on August 3, 2011, Mr. Bob Carter and Mr. Larry Steinert agreed with the NRC staff to provide your response to the Request for Additional Information (RAI) questions within 6 months.

If you have questions regarding this matter, please contact Andrew Hon at (301) 415-8480.

Sincerely,

John R. Jolicoeur, Chief
Licensing Process Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 704

Enclosure: RAI Questions

REQUEST FOR ADDITIONAL INFORMATION ON BWRVIP-234,

THERMAL AGING AND NEUTRON EMBRITTLEMENT EVALUATION OF CAST AUSTENITIC

STAINLESS STEEL FOR BOILING WATER REACTOR INTERNALS (TAC NO. ME5060)

RAI 1
For Figure 2-1, the absorbed energy saturation values for the CF-8M castings are between 10 and 20 ft-lbs. The saturation is reached in about 10^5 hours at the temperature of 300° C. These values are independent of the delta ferrite content. Is this result consistent with other alloys like CF-8 and CF-3?

RAI 2
Please clarify your conclusion in Section 2.1.1 on the differences between the thermal embrittlement (TE) of cast austenitic stainless steel (CASS) components in a pressurized water reactor versus a boiling water reactor (BWR).

RAI 3
Figures 2-4 and 2-5 document the toughness as a function of delta ferrite content. The aging conditions for these test results should be included for our review. If these results represent the saturation of the TE effects, please discuss why these results are different from Figure 2-1 where the TE effects are not a function of delta ferrite content.

For the purposes of this report, are the saturation values for TE of CASS components within the scope of the assessments in this report?

RAI 4
Discuss why room temperature Charpy absorbed energy is an appropriate way to quantify the extent of TE. The upper shelf energy (USE) that you show in Figure 2-5 could be a more valuable parameter because USE could be related directly to the service temperature, similar to what is done for ferritic reactor vessel properties.

RAI 5
Please provide clear and consistent definitions for "screening" and "threshold" that can be incorporated in the approved version of BWRVIP-234.

RAI 6
Welding or weld repairs and how they might affect the properties of a CASS component in the BWR environment were discussed. Welding of CASS components can increase the delta ferrite content in the heat affected zone of the weld (Mimura et al. Welding Journal, 1998, pp 350s-360s). Please discuss the impact that welding and/or weld repairs would have on the component screening.

ENCLOSURE

RAI 7
Typically, the measured delta ferrite values are not reported on the certified material test record. In Section 3.4, "Ferrite Content," the Ni and Cr equivalent equations from Hull are used to calculate the delta ferrite content.

Please discuss how calculated values compare with measured values for CASS components to demonstrate the level of confidence one can place on the calculations. Provide additional discussion in Section 4.1 as to how the uncertainty in the calculations affects the screening process.

RAI 8
In Section 4.1, "Screening Based on Ferrite Content," for the discussion about fatigue for CASS jet pump components, BWRVIP-234 assumes that BWRVIP-41 inspections are sufficient to eliminate concern for any augmented inspections for fatigue of CASS jet pump components. BWRVIP-41, Revision 2, recommends eliminating CASS components from the scope of inspections because castings are considered immune to intergranular stress corrosion cracking. Please summarize the recommendations in BWRVIP-41, Revision 2 and revise this section as-needed to demonstrate how the BWRVIP-41 inspections will impact the screening of CASS components for fatigue in BWRVIP-234.

RAI 9
In Section 4.3, "Screening Based on Toughness," the alternative method to estimate the J-R curve parameters from Reference 8 was developed for core shroud welds. The delta ferrite content of core shroud welds is typically lower than the delta ferrite content of the CASS components in the BWR fleet. Given the uncertainty in the calculation of delta ferrite (RAI 7) and the potential for an increase in the % delta ferrite due to welding/weld repairs (RAI 6), discuss the effect that a higher delta ferrite content would have on the methodology to estimate toughness.

C
BWRVIP RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION DATED SEPTEMBER 18, 2012

ELECTRIC POWER RESEARCH INSTITUTE

2012-148 BWR Vessel & Internals Project (BWRVIP)

September 18, 2012

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

Attention: Joseph Holonich

Subject: Project No. 704 – BWRVIP Response to NRC Request for Additional Information on BWRVIP-234

Reference: Letter from John R. Jolicoeur (NRC) to David Czufin (BWRVIP Chairman), "Acceptance for Review and Request for Additional Information, for BWRVIP-234: BWR Vessel and Internals Project: Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals (TAC NO. ME5060," dated September 29, 2011.

Enclosed are five (5) copies of the BWRVIP response to the NRC Request for Additional Information (RAI) on the BWRVIP report entitled "BWRVIP-234: BWR Vessel and Internals Project, Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steels for BWR Internals." The RAI was transmitted to the BWRVIP by the NRC letter referenced above.

Please note that the enclosed response contains proprietary information. A letter requesting that the response be withheld from public disclosure and an affidavit describing the basis for withholding this information are provided as Attachment 1. The response includes yellow shading to indicate the proprietary information. The proprietary information is also marked with the letters "TS" in the margin indicating the information is considered trade secrets in accordance with 10CFR2.390A.

Two (2) copies of a non-proprietary version of the BWRVIP response to the RAI are also enclosed. This non-proprietary response is identical to the enclosed proprietary response except that the proprietary information has been deleted.

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

Sincerely,

Dennis Madison
Southern Nuclear
Chairman, BWR Vessel and Internals Project

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[redacted]

Request for Additional Information on BWRVIP-234: Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals

[redacted]

[redacted]

Request for Additional Information on BWRVIP-234: Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals

Each item from the NRC Request for Additional Information (RAI) is repeated below verbatim followed by the BWRVIP response to that item.

RAI 1

For Figure 2-1, the absorbed energy saturation values for the CF-8M castings are between 10 and 20 ft-lbs. The saturation is reached in about 10^5 hours at a temperature of 300°C. These values are independent of the delta ferrite content. Is this result consistent with other alloys like CF-8 and CF-3?

BWRVIP Response to RAI 1:

The purpose of Figure 2-1 was to show that CF-8M is very susceptible to thermal embrittlement (TE). In fact it is the least resistant of the CASS materials to TE. This is consistent with results reported in NUREG/CR-4513, Revision 1 (Reference 13 in BWRVIP-234) and Reference 5 of BWRVIP-234.

NUREG/CR-4513, Revision 1, Section 3.2.1, states that the saturation room temperature (RT) impact energy of a specific cast stainless steel is indeed a function of the chemical composition <u>and</u> the ferrite content of the material. Page 3 of NUREG/CR-4513, Revision 1 states the following:

"The decrease in RT Charpy–impact energy during thermal aging at 400°C (752°F) of various heats of cast stainless steel[4–6,15,19,21] is shown in Fig. 1. The results indicate that all the materials reach a "saturation" RT impact energy, i.e., a minimum value that would be achieved by the material after long–term aging. The actual value of saturation RT impact energy for a specific cast stainless steel is independent of aging temperature but depends strongly on the chemical composition of the steel; it is lower for the Mo–bearing CF–8M steels than for the Mo–free CF–3 or CF–8 steels, and decreases with an increase in ferrite content or the concentration of C or N in the steel."

Reference 3 of BWRVIP-234 also provides an excellent review of the available data on the issue. Figure 2 below shows the effect of delta ferrite content, and aging temperature and time, and the impact energy at room temperature. This shows as ferrite content increases, the impact energy at long aging times decreases.

The saturation values of Charpy-impact energy for CF-3 and CF-8 materials used in BWRs are above CF-8M castings. Furthermore, Reference 5 of BWRVIP-234 states that "CF8M shows the greatest susceptibility to thermal aging of any of the other SA-351 grades considered in the screening criteria."

[redacted]

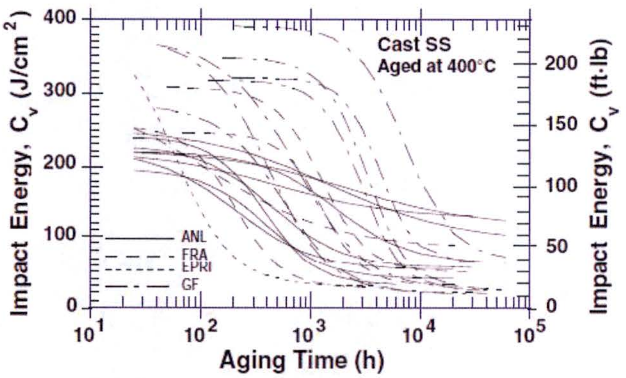

Figure 1. Decrease in Charpy-impact energy for various heats of cast stainless steels aged at 400°C

Reference: NUREG/CR-4513, Revision 1, May 1994.

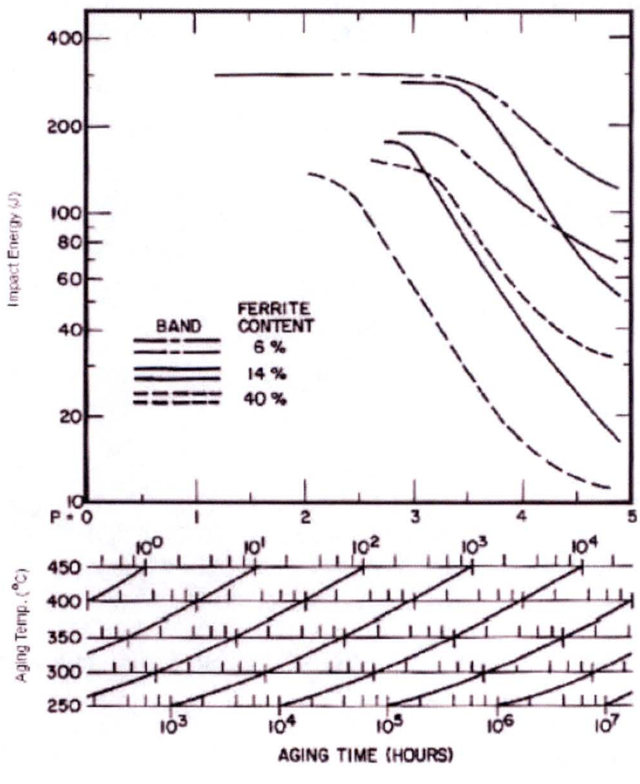

Figure 2. Influence of Ferrite Content on the Embrittlement of Cast Austenitic Stainless Steel as a Function of the Aging Parameter P (Reference 3 of BWRVIP-234)

[redacted]

RAI 2

Please clarify your conclusion in Section 2.1.1 on the differences between the thermal embrittlement (TE) of cast austenitic stainless steel (CASS) components in a pressurized water reactor versus a boiling water reactor (BWR).

BWRVIP Response to RAI 2:

The conclusion in Section 2.1.1 stated the following:

"In other words, the higher the aging temperature, the greater is the deterioration with time. Based on this, one can conclude that the CASS aging embrittlement effects in a BWR are significantly lower than components in a PWR."

The BWRVIP proposes to revise the statement as follows:

"Based on this, the higher the aging temperature, the faster the aging effect reaches saturation. Since PWRs operate at a higher temperature than BWRs, thermal aging embrittlement effects for CASS components in a BWR are expected to occur later in life than for CASS components in a PWR."

RAI 3

Figures 2-4 and 2-5 document the toughness as a function of delta ferrite content. The aging conditions for these test results should be included for our review. If these results represent the saturation of the TE effects, please discuss why these results are different from Figure 2-1 where the TE effects are not a function of delta ferrite content.

For the purposes of this report, are the saturation values for TE of CASS components within the scope of the assessments in this report?

BWRVIP Response to RAI 3:

Figure 2-4 was developed based on the equations contained in Section 3 of NUREG/CR-4513, Revision 1. Excerpts of EPRI report 1016236 (Reference 1 in BWRVIP-234) are contained in Attachment A to provide the staff with background regarding determination of the J-R curves.

The results show that J versus crack extension is a function of the delta ferrite; increasing ferrite content decreases the J-da curve.

For CF-3 and CF-8 materials, the saturation RT impact energy CVsat is determined using the following equations:

$\log_{10}CVsat = 1.15 + 1.36\exp(-0.035\Phi)$,

where the material parameter Φ is expressed as

[redacted]

$\Phi = \delta c(Cr + Si)(C + 0.4N)$,

and from

$\log10CVsat = 5.64 - 0.006(\delta c) - 0.185Cr + 0.273Mo - 0.204Si + 0.044Ni - 2.12(C + 0.4N)$.

Equations for Cvsat for CF8M are also provided in NUREG/CR-4513, Revision 1.

The saturation fracture toughness J–R curve for a specific cast stainless steel can be estimated from its RT impact energy at saturation. This is described in Section 3.2.2 of NUREG/CR4513, Revision 1.

There was no intent to apply the RT saturation values of impact energy shown in Figure 2-1 to an evaluation of BWR components. The important parameter used to evaluate component integrity is fracture toughness. For this evaluation, the lower bound value J = 255 kJ/mm^2 has been used as approved by the NRC.

Figure 2-5 is based on the following reference.

O.K. Chopra and M. H. Chung. "Initial Assessment of the Processes and Significance of Thermal Aging in Cast Stainless Steels," presented at the 16th Water Reactor Safety Information Meeting, November 1988.

Examination of the data summarized by Chopra indicates that the only material exhibiting an upper shelf energy level at temperatures below 50 ft. lb. is a very high ferrite level (35%) CASS exposed for 1000 hours at 752°F. At 662°F this level of embrittlement is reached in 10,000 hours. Using the data from Chopra for the worst embrittlement encountered, a plot was constructed of delta ferrite as a function of the observed upper shelf energy from the CVN values measured at elevated temperatures. Thus, the data indicates a clear relationship between upper shelf energy and % ferrite and provides additional justification for selecting 20% ferrite as a threshold for susceptibility.

RAI 4

Discuss why room temperature Charpy absorbed energy is an appropriate way to quantify the extent of TE. The upper shelf energy (USE) that you show in Fig. 2-5 could be a more valuable parameter because USE could be related directly to the service temperature, similar to what is done for ferritic reactor vessel properties.

[redacted]

BWRVIP Response to RAI 4:

Room temperature Charpy absorbed energy is not suggested in BWRVIP-234 as a means to quantify the extent of TE. The intent of any discussions in BWRVIP-234 regarding Charpy energy and upper shelf energy was to show that ferrite plays a strong role in toughness, i.e., as ferrite decreases the fracture toughness increases and vice versa. Regardless, the criteria for fracture toughness is based on Reference 5 of BWRVIP-234.

RAI 5

Please provide clear and consistent definitions for "screening" and "threshold" that can be incorporated in the approved version of BWRVIP-234.

BWRVIP Response to RAI 5:

"Screening" and "threshold" criteria are used to determine the susceptibility of a component to address both TE and neutron embrittlement. In this case, one single criterion is neither sufficient nor appropriate to eliminate the issue from consideration. Therefore, to evaluate the synergistic effects of both TE and neutron embrittlement it was necessary to extend the screening/evaluation criteria to address ferrite content, fluence, available fracture toughness, applied stress and current inspection practice. For example, even if the ferrite level was less than 20% for a low Mo content material, that fact in and of itself, does not mean that the material is not susceptible. The other criteria defined above should also be considered to determine the overall susceptibility of the component. Only in this way, can a meaningful and justifiable conclusion regarding susceptibility of CASS components be determined.

Therefore, the screening methodology discussed in Section 4 of BWRVIP-234 contains several aspects of a TE and neutron embrittlement assessment which when coupled, provide the overall criteria to assess susceptibility. Therefore, it is the opinion of the BWRVIP that the description of the screening criteria discussed in Section 4 of BWRVIP-234 is technically acceptable for determination of augmented inspection requirements.

RAI 6

Welding or weld repairs and how they might affect the properties of a CASS component in the BWR environment were discussed. Welding of CASS components can increase the delta ferrite content in the heat affected zone of the weld (Mimura et al. Welding Journal, 1998, pp 350s-360s). Please discuss the impact that welding and/or weld repairs would have on the component screening.

[redacted]

BWRVIP Response to RAI 6:

In general and depending on the extent of casting defects, weld repairs would be required to correct such defects. However, light welding on CF-3 and CF-8 does not appreciably affect corrosion resistance and therefore minor weld repairs are not expected to impact the susceptibility of the component. It is expected that if significant defects were discovered the casting would be rejected or repaired and then re-solution annealed.

The Mimura paper [1] discusses two test heats of material, both of which are moderately high carbon and Mo. Additionally, one of the heats is very high in ferrite relative to what is usually reported for BWR castings. This material (CF-8M) is known to be much more susceptible to TE. CF3 and CF8 are much less prone to TE in general, and especially at BWR temperatures when the ferrite is 20% or less. Consequently, the BWRVIP believes that the effects of welding on BWR CASS components as reported by Mimura would not affect the screening methodology contained in BWRVIP-234.

Ferrite in castings is beneficial for SCC resistance and any increases in ferrite due to welding of BWR CASS internals to wrought stainless steel is not expected to be significant. Futhermore, there has been no evidence of any SCC in CASS components in BWRs to date. Therefore, welding of CASS components is not expected to impact the screening criteria and assessment of TE.

RAI 7

Typically, the measured delta ferrite values are not reported on the certified material test record. In Section 3.4, "Ferrite Content," the Ni and Cr equivalent equation from Hull are used to calculate the delta ferrite content.

Please discuss how calculated values compare with measured values for CASS components to demonstrate the level of confidence one can place on the calculations. Provide additional discussion in Section 4.1 as to how the uncertainty in the calculations affects the screening process.

BWRVIP Response to RAI 7:

Measured values of ferrite were not available from CMTRs. Consequently, Hull's equivalent factors were used to determine ferrite. Per Reference 5 of BWRVIP-234, this is recommended by the NRC when actual values are not available. Thus, it is not possible to compare calculated and measured values of ferrite content.

As shown in Table A-1 of BWRVIP-234, the Mo content was not measured for all heats collected and thus, there is some uncertainty regarding the calculated ferrite. The variation of ferrite content can be examined as a function of Mo content. Using the data in Table A-1, the Mo content was varied for each heat of material from 0.1 wt.% to 0.5 wt.% in increments of 0.1 wt.%. A limit of 0.5% wt.% was used because this is the upper bound value for CF-3 and CF-8

[redacted]

materials as given in Reference 5 of BWRVIP-234 and SA-351. The results of the analysis are shown in the following table.

Mo (wt.%)	Average % Ferrite	% Ferrite Calculated from Average Chemistry
0.1		
0.2		
0.3		
0.4		
0.5		

TS

Even when assuming all heats are a maximum Mo content of 0.5 wt.%, only three heats are calculated to be slightly greater than the 20% ferrite limit TS Therefore, given the relatively small change in percent ferrite shown above from that reported in BWRVIP-234, the uncertainty in Mo in not judged to affect the ferrite screening process for CF-3 and CF-8 materials.

RAI 8

In Section 4.1, "Screening Based on Ferrite Content," for the discussion about fatigue for CASS jet pump components, BWRVIP-234 suggests that BWRVIP-41 inspections are sufficient to eliminate concern for any augmented inspections for fatigue of CASS jet pump components. BWRVIP-41, Rev. 2, recommends eliminating CASS components from the scope of inspections because castings are considered immune to intergranular stress corrosion cracking.
Please summarize the latest recommendations in BWRVIP-41, Revision 2 and revise this section as-needed to demonstrate how the BWRVIP-41 inspections will impact the screening of CASS components for fatigue in BWRVIP-234.

BWRVIP Response to RAI 8:

BWRVIP-234, page 4-3 states the following relative to fatigue of BWR CASS components.

TS

BWRVIP-41, Revision 2 continues to state that cast components in jet pumps have not exhibited any cracking to date and thus are considered to have a high resistance to IGSCC in BWR core environments. It also states that it is important to note that TE does not in itself cause cracking to

[redacted]

occur. It reduces the structural margin of a material in resisting propagation of cracks due to other initiators like IGSCC or fatigue.

[redacted] Due to the field of view **TS** using typical EVT-1 methods, cracking of any significance on the casting side of the weld will likely be detected should it occur and thus, would be reported. To date, no such cracking has occurred.

Therefore, the BWRVIP believes that the current inspection strategy will provide information on cracking in the HAZ of the casting side of the weld should it manifest itself.

RAI 9

In Section 4.3 "Screening Based on Toughness," the alternative method to estimate the J-R curve parameters from Reference 8 was developed for core shroud welds. The delta ferrite content of core shroud welds is typically lower than the delta ferrite content of the CASS components in the BWR fleet. Given the uncertainty in the calculation of delta ferrite (RAI 7) and the potential for an increase in the % delta ferrite due to welding/weld repairs (RAI 6), discuss the effect that a higher delta ferrite content would have on the methodology to estimate toughness.

BWRVIP Response to RAI 9:

As stated in the responses to RAI 6 and RAI 7, the uncertainty in delta ferrite and potential for a increase in delta ferrite due to welding and weld repairs is not expected to be significant. In RAI 6 it was shown that even when assuming the maximum Mo content of 0.5 wt.% the average increase in ferrite was relatively small. This marginal increase in ferrite would have a slight effect on reducing the toughness of the material (since an increase in ferrite results in a decrease in toughness). However, for the range of exposure conditions that would be experienced for CASS components in a BWR, the toughness values are expected to be well above recommended lower bound toughness of 255 kJ/m^2. Therefore, the BWRVIP believes that use of the lower bound toughness is sufficiently conservative and consequently, would not affect the methodology to estimate the toughness for irradiated conditions.

References:

1. H. Mimura, et al, "Thermal Embrittlement of Simulate Heat-Affected Zone in Cast Austentic Stainless Steels," Welding Journal, 1998, pp 350s-360s

[redacted]

Additional Information Provided by the BWRVIP

During preparation of this RAI response it was discovered that second bullet on page 2-4 of BWRVIP-234 is incorrect.

Currently, the second bullet states the following:

- Statically-cast, low-molybdenum material (CF-8) with relatively high δ ferrite content (> 20%) could be screened out from further evaluation.

The second bullet should read as follows:

- Statically-cast, low-molybdenum material (CF-8) with relatively high δ ferrite content (≤ 20%) could be screened out from further evaluation.

In summary, "(>20%)" should be "(≤ 20%)" to be consistent with Table 4-1. This change will be made in a revision to the report.

[redacted]

Attachment A, Excerpts from EPRI Report 1016236, Section 3

TS

[redacted]

TS

D

NRC REQUEST FOR ADDITIONAL INFORMATION DATED APRIL 24, 2013

BWRVIP 2013-060A

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

April 24, 2013

Dennis Madison
Southern Nuclear
Chairman, BWR Vessel and Internals Project
3420 Hillview Avenue
Palo Alto, CA 94304-1395

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE BOILING WATER REACTOR (BWR) VESSEL INTERNALS PROJECT (BWRVIP)-234, "THERMAL AGING AND NEUTRON EMBRITTLEMENT EVALUATION OF CAST AUSTENITIC STAINLESS STEEL FOR BWR INTERNALS" (TAC NO. ME5060)

Dear Mr. Madison:

By letter dated September 29, 2011, the U.S. Nuclear Regulatory Commission staff transmitted Request for Additional Information (RAI) questions (Agencywide Document Access and Management System (ADAMS) Accession No. ML112630638) for the BWRVIP-234, "Thermal Aging And Neutron Embrittlement Evaluation Of Cast Austenitic Stainless Steel For BWR Internals." On September 18, 2012, the BWRVIP provided its responses to the RAIs (ADAMS Accession No. ML12265A078).

The NRC staff completed its review of the responses to the RAI questions, and has identified additional areas for which information is needed to complete the review. The additional RAI questions are enclosed.

On March 21, 2013, Mr. Larry Steinert, representing BWRVIP, and I agreed that the NRC staff will receive your response to the enclosed RAI questions by May 31, 2013. If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-7297.

Sincerely,

Joseph J. Holonich, Senior Project Manager
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 704

Enclosure:
RAI Questions

SECOND REQUEST FOR ADDITIONAL INFORMATION ON BWRVIP-234,

"THERMAL AGING AND NEUTRON EMBRITTLEMENT EVALUATION OF CAST AUSTENITIC

STAINLESS STEEL FOR BWR INTERNALS" (TAC NO. ME5060)

The U.S. Nuclear Regulatory Commission (NRC) staff is in the process of reviewing the September 18, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12265A078) responses to the first set of Request for Additional Information (RAI) questions. Based on the review conducted to date, the NRC staff has developed a second set of RAI questions that includes two new RAI questions (RAI 10 and RAI 11), a revised RAI 7-a, and a follow-up RAI 9-a.

RAI 7-a
Typically, the measured delta ferrite values are not reported on the certified material test record. In Section 3.4, "Ferrite Content," the Ni and Cr equivalent equation from Hull are used to calculate the delta ferrite content. In the May 19, 2000, letter from Christopher Grimes, NRC, to Mr. Douglas Walters, Nuclear Energy Institute (NEI), the calculated ferrite content from Hull's equations represents the mean value with a significant uncertainty (± 6%).

Justify why 6 percent should not be added to the calculated ferrite values based on chemistry to represent an upper-bound to the ferrite content, which is based on chemistry. Provide additional discussion in Section 4.1 as to how the uncertainty in the calculations affects the screening process.

RAI 9-a
In the September 18, 2012, RAI response, the BWRVIP stated the following:

> As stated in the responses to RAI 6 and RAI 7, the uncertainty in delta ferrite and potential for an increase in delta ferrite due to welding and weld repairs is not expected to be significant. In RAI 6 it was shown that even when assuming the maximum Mo content of 0.5 wt.% the average increase in ferrite was relatively small. This marginal increase in ferrite would have a slight effect on reducing the toughness of the material (since an increase in ferrite results in a decrease in toughness). However, for the range of exposure conditions that would be experienced for cast austenitic stainless steel (CASS) components in a BWR, the toughness values are expected to be well above recommended lower bound toughness of 255 kJ/m^2. Therefore, the BWRVIP believes that use of the lower bound toughness is sufficiently conservative and consequently, would not affect the methodology to estimate the toughness for irradiated conditions.

The NRC staff has reviewed the technical bases for the RAI responses and compared the methodology used in BWRVIP-234 to predict fracture toughness as a function of neutron exposure to a previously reported prediction from NUREG/CR-6960. The comparison is shown in Figure 1 along with the range of predicted J values at 2.5 mm crack extension for unirradiated CF-8 with no thermal embrittlement (TE) and maximum TE (no irradiation) for CF-8 with > 15 percent delta ferrite (Section 3.1.1 of NUREG/CR-4513, Rev. 1).

ENCLOSURE

The NRC staff is concerned that the BWRVIP-100 model for CASS may be inadequate for the following reasons:

- The data base for development of the BWRVIP-100 model to predict toughness does not include any CASS materials, just welds, heat affected zone material, and base metal. Further, the number of welds in the database is limited, and the delta ferrite content for these welds is unknown. Welds typically contain 7 to 10 percent ferrite while CASS materials can have between 5 and 25 percent ferrite. Therefore, the toughness predicted by the BWRVIP-100 model may not be appropriate or conservative for CASS materials.

- The NUREG/CR-6960 model is based on data from CASS materials, welds, and wrought materials; therefore, the NRC staff believes it more conservatively represents the fracture toughness of irradiated CASS materials. At fluence values > 0.3 dpa, the toughness predicted by the NUREG/CR-6960 curve is below the value of 255 kJ/m^2 used as the basis for the screening based on toughness in BWRVIP-234.

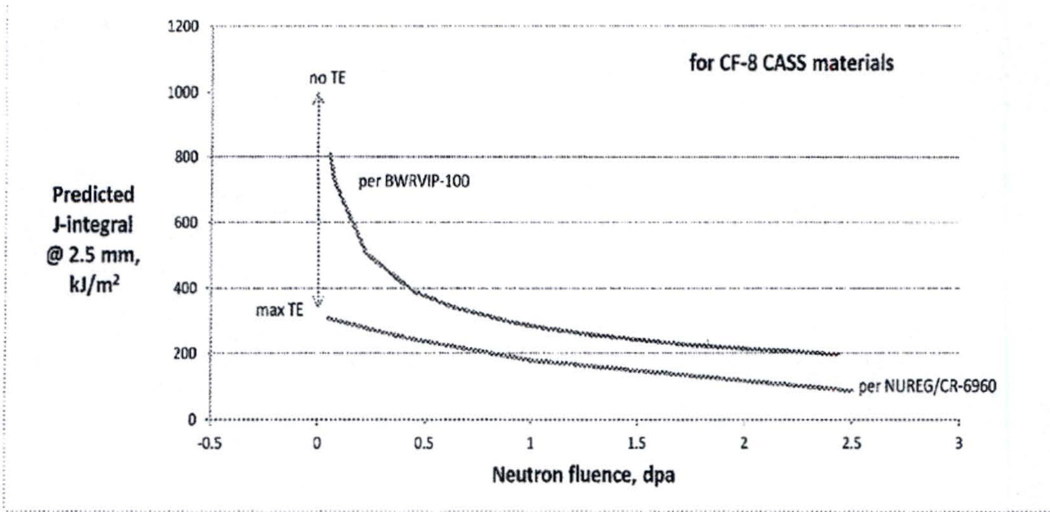

Figure 1. Plot of predicted toughness (J-integral value at 2.5 mm crack extension) from BWRVIP-100 and NUREG/CR-6960 as a function of neutron fluence. The predicted lower-bound toughness of CF-8 due to TE alone is shown for reference.

The NRC staff requests that the BWRVIP review all available data on fracture toughness of irradiated CASS materials, and the associated uncertainties, and to either revise its methodology to predict the lower-bound toughness of CASS materials at reactor operating temperatures after approximately 60 years of operation, or provide further justification that BWRVIP-100 is sufficiently conservative.

New, RAI 10
The BWRVIP-234 report has considered only the properties of CASS materials at operating temperatures. The NRC staff requests that the BWRVIP assess the structural integrity of CASS materials at typical leak test temperatures.

New, RAI 11
The nondestructive examination methods of the existing examinations discussed in Section 6, "Assessment of CASS Components," were justified because they were capable of detecting degradation other than loss of fracture toughness due to the combined effects of thermal and neutron embrittlement. Given the lower-bound toughness for a CASS component, which could have significantly higher delta ferrite than core shroud welds and is subject to both thermal and neutron embrittlement, justify the adequacy of the existing BWRVIP inspections to detect subcritical flaws as discussed in the GALL report, Rev. 2, XI.M9 paragraph 4 -- *Detection of Aging Effects.*

E
BWRVIP RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION DATED MAY 23, 2014

ELECTRIC POWER RESEARCH INSTITUTE

2014-086 BWR Vessel & Internals Project (BWRVIP)

May 23, 2014

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

Attention: Joseph Holonich

Subject: Project No. 704 – BWRVIP Response to NRC Request for Additional Information on BWRVIP-234

Reference: Letter from Joseph J. Holonich (NRC) to Dennis Madison (BWRVIP Chairman), Request for Additional Information for the Boiling Water Reactor (BWR) Vessel Internals Project (BWRVIP)-234, "Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals" (TAC NO. ME5060)," dated April 24, 2013.

Enclosed are five (5) copies of the BWRVIP response to the NRC Request for Additional Information (RAI) on the BWRVIP report entitled "BWRVIP-234: BWR Vessel and Internals Project, Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals". The RAI was transmitted to the BWRVIP by the NRC letter referenced above.

Note that in addition to responses to the NRC's specific RAI on BWRVIP-234, the enclosed RAI response includes a "supplemental information" section. As recommended by the NRC and agreed to by the industry, the "supplemental information" section provides the industry's proposed common (i.e., for both BWRs and PWRs) approach to screen cast austenitic stainless steel (CASS) reactor internal components for both thermal embrittlement (TE) and irradiation embrittlement (IE).

If you have any questions on this subject please call Ron DiSabatino (Exelon, BWRVIP Assessment Committee Technical Chairman) at 717.456.3685.

Sincerely,

A. O. McGehee   Dennis Madison

Andrew McGehee, EPRI, BWRVIP Program Manager
Dennis Madison, Southern Nuclear, BWRVIP Chairman

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Response to NRC Request for Additional Information (RAI) on
BWRVIP-234: BWR Vessel and Internals Project, Thermal Aging and Neutron
Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals

ENCLOSURE

BWRVIP Response to NRC Request for Information (RAI) on BWRVIP-234

Each Request for Additional Information (RAI) received from the NRC is repeated verbatim below, followed by the BWRVIP Response.

RAI 7-a

Typically, the measured delta ferrite values are not reported on the certified material test record. In Section 3.4, "Ferrite Content," the Ni and Cr equivalent equation from Hull are used to calculate the delta ferrite content. In the May 19, 2000, letter from Christopher Grimes, NRC, to Mr. Douglas Walters, Nuclear Energy Institute (NEI), the calculated ferrite content from Hull's equations represents the mean value with a significant uncertainty (± 6%).

Justify why 6% should not be added to the calculated ferrite values based on chemistry to represent an upper-bound to the ferrite content, which is based on chemistry. Provide additional discussion in Section 4.1 as to how the uncertainty in the calculations affects the screening process.

BWRVIP Response to RAI 7-a

Two figures from NUREG/CR-4513 (ANL-93/22) [1] illustrate the issue of delta ferrite prediction uncertainty that led to the ± 6% range cited in the Grimes letter [2]. Figure 1 shows the measured and corresponding predicted delta ferrite content of various heats of cast stainless steel from various investigators. The measurements and the predictions are shown in volume percent of delta ferrite, with a band on either side of the 45-degree line representing perfect agreement between measured and predicted delta ferrite that is parallel-displaced by 6 percent volume. Figure 2 compares two different methods for predicting or estimating the delta ferrite based on the Certified Material Test Report (CMTR) or similar material constituent measurements, which in this case is the Hull equations versus the method adopted in ASTM A800-01 [3]. Figure 2 illustrates the problem of systematic under-prediction for ASTM A800-01, whereas Figure 1 does not display any systematic under-prediction by Hull's equations when compared to measured delta ferrite.

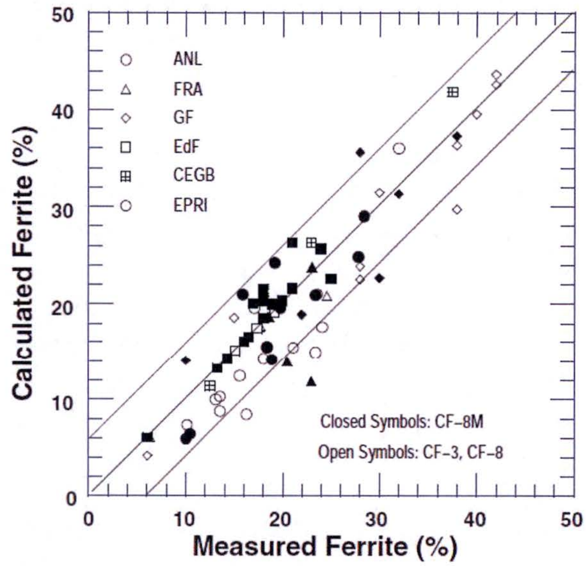

Figure 1. Measured Versus Calculated Delta Ferrite (Figure 5 from NUREG/CR-4513 [1]), Showing Results for 48 Heats of Material

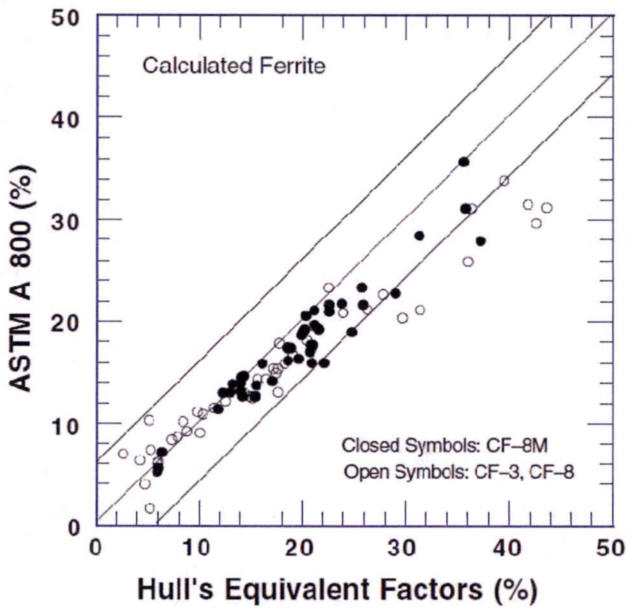

Figure 2. Comparison of Delta Ferrite Calculated by Hull's Equivalent Factors and by ASTM A800-01 (Figure 10 from NUREG/CR-4513 [1]), Also Referred to as WRC-1992

In order to study the issue further, available data on various direct measurement techniques and predictive methods for estimating the volume percentage of delta ferrite in stainless steel weldments and castings have been reviewed and assessed by the industry. From that review and assessment, it should be noted that no precise method for determining delta ferrite content exists. A destructive direct measurement technique such as metallographic point counting, if very carefully applied, might be accurate to within ± 1% delta ferrite, but the method is both destructive and time consuming. Other non-destructive direct measurement techniques, such as the Magne-Gage and the Feritscope® instruments, have an accuracy of about ± 2% delta ferrite. Various estimation or predictive methods have been proposed over the years, based on material chemical composition. Of those methods, Hull's equations have been evaluated several times (e.g., Reference 4) and compared to both measured and other predictive methods, showing that the standard predictive error using Hull's equations is of the order of 3% delta ferrite, without the persistent under-prediction characteristic of ASTM A800-01.

From the industry review and assessment, the following conclusions can be drawn:

1. Adding 6% to the delta ferrite values predicted by Hull's equations is not reasonable, since the available data clearly show that Hull's equations do not systematically under-predict measured values, which in themselves may be in error by 1 or 2% delta ferrite.
2. Since there is no evidence that a prediction based on Hull's equations has systematically and erroneously under-predicted the delta ferrite content by 6%, and since the standard error in the Hull's equations predictions is of the order of 3% delta ferrite, or perhaps as low as 2% delta ferrite, that level of standard error provides a reasonable estimate of the uncertainty in the delta ferrite prediction.
3. Since the level of standard error in Hull's equations predictions is roughly of the same order as the accuracy of non-destructive measurements of delta ferrite, and since some degree of uncertainty was included during the process of establishing the delta ferrite screening limits, there is no need to alter the screening methodology in order to explicitly include any additional measure of standard error uncertainty.

In addition to the delta ferrite prediction uncertainty, the RAI requested additional discussion in Section 4.1 of BWRVIP-234 as to how the uncertainty in the delta ferrite predictions affects the screening process. In this respect, it should be noted that a consensus was reached with respect to the delta ferrite screening levels of 14% and 20% based primarily on the data analyses in Reference 1, as corroborated by the industry review of the same data documented in Reference 5. The industry review included comparisons of fracture toughness data for CASS materials with predicted delta ferrite within ±3% of the proposed screening values, in order to assure that no precipitous changes in fracture toughness were observed (e.g., the measured and predicted J-R curves shown in Figures 13, 14, 15, and 16 from Reference 1 were

examined carefully – particularly for heats with delta ferrite content near the proposed screening values, to assure relatively smooth toughness transitions as a function of that delta ferrite content).

As a result of that industry review, and the consensus reached with the NRC staff on the screening thresholds, it is concluded that uncertainty of the order of ±3% in delta ferrite prediction has been taken into account relative to its effect on the fracture toughness of the CASS materials, such that the thermal embrittlement screening criteria are sufficiently robust to remain as previously stipulated.

<u>Background for BWRVIP Response to RAI 7-a</u>

In their review of BWRVIP-234 [6], the NRC staff questioned whether the uncertainty in estimation or prediction of delta ferrite in CASS reactor internals components required the addition of 6% to delta ferrite calculations based upon material chemical composition [7]. That RAI also asked that Section 4.1 of BWRVIP-234 be modified to discuss how the uncertainty in the delta ferrite calculations would affect the component screening process. It should be pointed out that typically, the measured delta ferrite values are not reported on the CMTR. Therefore, as described in Section 3.4 of Reference 1, "Ferrite Content," the Ni and Cr equivalent equations from Hull are used to calculate the delta ferrite content. For such calculations, as documented in Reference 2, the calculated ferrite content from Hull's equations represents the mean value with a significant uncertainty (± 6%).

The uncertainty in the estimation of delta ferrite content for cast austenitic stainless steel components, based upon CMTR chemical composition measurements, has been a technical issue for both PWR and BWR nuclear power plant licensees for the past two decades, since the delta ferrite content provides key screening criteria for determining elements of thermal aging embrittlement management programs. This delta ferrite estimation uncertainty issue is amplified in importance by a stipulated requirement stated in Reference 2, "*Note that calculated δ-ferrite should use Hull's equivalent factors or a method producing an equivalent level of accuracy (± 6% deviation between measured and calculated values).*" This stipulated requirement for components with estimated delta ferrite content in the 15 to 25% range means that the uncertainty in delta ferrite leads to even greater uncertainty in the selection of aging management program elements, in particular for upward adjustments of the estimated delta ferrite from relatively benign values of 15 or 16% upward to 21 or 22%.

The major source of the concern about delta ferrite uncertainty can be traced to Figure 1, which shows the measured and corresponding predicted delta ferrite content of various heats of cast stainless steel from various investigators. The measurements and the predictions are shown in percent of delta ferrite, with a band on either side of the 45-degree line representing perfect

agreement between measured and predicted delta ferrite that is parallel-displaced by 6%. The sources of the data are shown in the legend, ranging from ANL-generated data to other sources, such as Framatome, Georg Fisher, Electricite de France (EDF), the Central Electricity Generating Board (CEGB) of the United Kingdom, and the Electric Power Research Institute (EPRI).

Figure 1 displays the definitive data supporting the contention of the staff of the NRC that *any predicted or estimated* value of delta ferrite must be considered to have a potential under-predicted error of 6% such that, for screening purposes, the predicted or estimated delta ferrite must be adjusted upward by that potential error. In other words, for an estimated or predicted delta ferrite of 18% by volume, which would put that component material below a possible aging management program criteria value of 20%, the actual percent to be used for screening purposes would have to be adjusted upward to 24% by volume. It should be noted that the "calculated" values of percent delta ferrite plotted in Figure 1 are based upon the nominal chemical composition of the individual stainless steel castings studied and a set of predictive equations known as the Hull equations, where the percent of delta ferrite is derived from the Hull factors for equivalent chrome content, Cr_{eq},

$$Cr_{eq} = (\%Cr) + 1.21(\%Mo) + 0.48(\%Si) - 4.99$$

and equivalent nickel content, Ni_{eq},

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

and the estimated delta ferrite, in percent, is then calculated from the following equation

$$\delta(\%) = 100.3(Cr_{eq}/Ni_{eq})^2 - 170.72(Cr_{eq}/Ni_{eq}) + 74.22$$

The other relevant figure from Reference 1 is Figure 2, which compares two different methods for predicting or estimating the delta ferrite based on the nominal chemical composition of the material, in this case the Hull equations versus ASTM A800-01 [3], which uses a figure called the Schoefer diagram and expressions for the equivalent chrome content, Cr_{eq},

$$Cr_{eq} = (\%Cr) + 1.5(\%Si) + 1.4(\%Mo) + (\%Cb) - 4.99$$

and the equivalent nickel content, Ni_{eq},

$$Ni_{eq} = (\%Ni) + 30(\%C) + 0.5(\%Mn) + 26(\%N - 0.02\%) + 2.77$$

As illustrated by Figure 2, the predicted or estimated percent delta ferrite values from the ASTM procedure "*are ≈20% lower than those obtained from Hull's method for ferrite levels*

>12% and are comparable for lower ferrite level," as cited in Reference 1. As a result, Reference 1 also states that delta ferrite estimates "*determined by the ASTM method for cast stainless steels with >12% ferrite may yield nonconservative estimates of fracture properties.*" Such nonconservative estimates were confirmed in this study by creating a spreadsheet covering all 48 heats of material listed in Table 1 of Reference 1, and by comparing the ASTM A800-01 predictions with the measured values for >12% delta ferrite.

This demonstrates a systematic under-prediction of delta ferrite content compared to measured values, in particular within the range of 12 to 25% where such predictions are needed. In addition, according to Reference 1, an alternative predictive method based on Hull's equations illustrates unacceptable uncertainty if bounding uncertainty is used as the measure. Prudence dictates that the findings in Reference 1 need to be confirmed and that other alternatives need to be investigated, in particular within the delta ferrite range of 15 to 25%. In addition, if the Hull equation estimates can be shown not to systematically under-predict delta ferrite content, an argument can be made that average error – as opposed to bounding error – should be the basis for measuring the level of accuracy of the method.

Appendix A provides the review of the existing data on delta ferrite measurement and estimation.

References

1. O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR-4513 (ANL-93/22), Revision 1, Argonne National Laboratory, Argonne, Illinois (August 1994).
2. Letter, Christopher I. Grimes, Chief, License Renewal and Standardization Branch, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, to Douglas J. Walters, Nuclear Energy Institute, "License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components," May 19, 2000 (ADAMS Accession No. ML003717179).
3. ASTM A800/A800M, *Standard Practice for Steel Casting, Austenitic Alloy, Estimating Ferrite Content Thereof*, Annual Book of ASTM Standards, Volume 01.02, American Society for Testing and Materials, West Conshohocken, PA, 2013.
4. L. S. Aubry, P. F. Wieser, W. J. Pollard, and E. A. Schoefer, "Ferrite Measurement and Control in Cast Duplex Stainless Steels," in: Stainless Steel Castings, ASTM STP 756, V. G. Behal and A. S. Melilli, Editors, American Society for Testing and Materials, pp. 126-164, 1982.
5. R. E. Nickell and M. A. Rinckel, "Evaluation of Thermal Aging Embrittlement for Cast Austenitic Stainless Steel Components," Report No. EPRI TR-106092, Revision 1, Electric Power Research Institute, Palo Alto, California, December 2000.

6. BWRVIP-234: BWR Vessel and Internals Project. "Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steels for BWR Internals," Electric Power Research Institute, Palo Alto, California, December 2009. 1019060.
7. Letter, Joseph J. Holonich, Senior Project Manager, Licensing Processes Branch, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, to Dennis Madison, Chairman, BWR Vessel and Internals Project, Second Request for Additional Information on BWRVIP-234, "Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel For BWR Internals," (TAC No. ME5060), April 24, 2013.

Appendix A to BWRVIP Response to RAI 7-a

Review of Delta Ferrite Measurement and Calculation Methods

In order to evaluate alternative and perhaps more accurate delta ferrite estimation methods, it should be noted that delta ferrite measurement and estimation for austenitic stainless steel weldments and castings has been the subject of extensive research for at least six decades. The first comprehensive review of the methodology was provided by W. T. DeLong, for stainless steel weldments ***only***, as the Adams lecturer at the 55th annual meeting of the American Welding Society (AWS) in 1974 [A.1]. Additional progress for weldments was summarized by D. J. Kotecki [A.2] in 1997. In addition, an excellent comparative review of both measured and predicted delta ferrite for cast duplex stainless steels was provided by Aubrey et al. [A.3] in 1982. Some of the major findings by Aubrey et al. are described in the following paragraphs.

First, in order to establish a measurement baseline, the accuracy of various direct delta ferrite measurement techniques needed to be established. In this case, Figure A-1 shows that direct measurement by metallographic point counting, which is covered by ASTM E562-11, has very little error, provided that reasonable care is taken with the procedure.

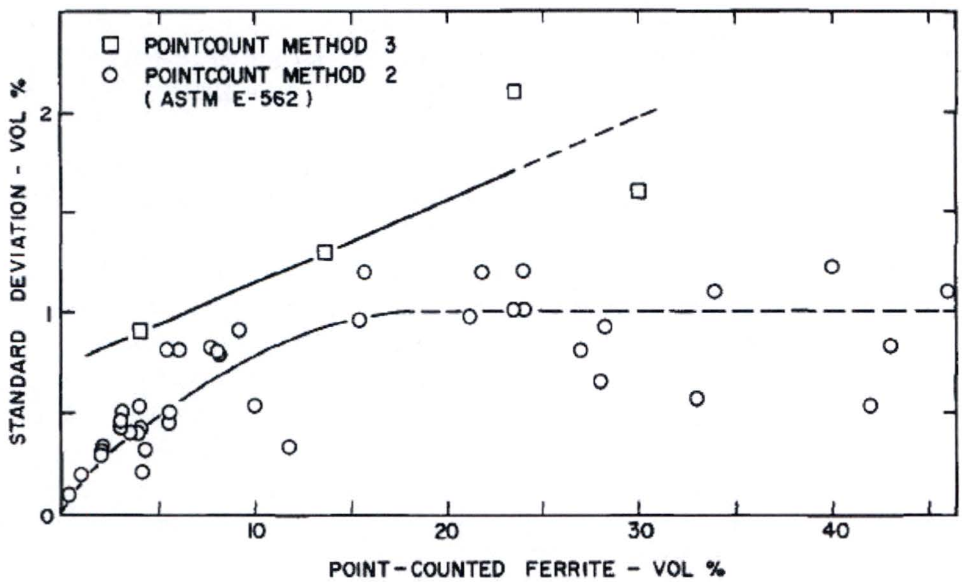

Figure A-1. Accuracy Demonstration for Metallographic Point Counting (Figure 4 from Reference A.3)

It is important to note that there is essentially no change in the standard deviation as the percentage of delta ferrite increases above 10%, with a root mean squared (RMS) error of less

than 0.5%. Considering the uncertainty in the material chemical composition measurements, metallographic point counting, using careful procedures, can be treated as a sufficiently accurate benchmark against which to compare other direct measurement techniques and various prediction methods.

Therefore, an important conclusion from the work by Aubrey et al. [A.3] is that metallographic point counting, otherwise known as quantitative metallography or optical metallography – essentially visually verifying the amount of delta ferrite in a destructive sample by etching and polishing, possibly including a staining procedure to accentuate and distinguish the various metallurgical structures – is the most accurate of the various direct measurement techniques for determining delta ferrite content. In addition, the processes of etching, polishing, staining, and counting are covered by ASTM E562-11 (Standard Test Method for Determining Volume Fraction by Systematic Manual Point Count) or ASTM E1245-03 (Standard Practice for Determining the Inclusion or Second-Phase Constituent Content of Metals by Automatic Image Analysis) and their supporting standards. Perhaps the only difficulty is the degree of accuracy for stainless steel welds, where the relatively small amount of delta ferrite in some cases may result in ferrite islands with delta ferrite shapes that are difficult to count. However, the method – although a destructive method – is thought to be excellent for castings with somewhat higher ferrite content and with connected ferrite islands. The ability to use large numbers of grids with statistical averaging tends to improve accuracy and enable metallographic point counting to be the baseline delta ferrite value against which other measurement and estimation methods are usually compared. The accuracy of this destructive method is generally assumed to be in the range of ± 1 FN (ferrite number[1]) or less.

However, in spite of its accuracy, the need for simpler, non-destructive, field-applicable magnetic measurement methods has been driving developments in the field for several decades. The two most widely used are the Magne-Gage, a device that measures the delta ferrite content through the force required to "tear" a magnet with calibrated properties away from the sample ferro-magnetic substrate, and the Feritscope®, which induces a magnetic field in the sample ferro-magnetic substrate and then measures the resulting magnetic field strength in order to back out the magnetic permeability. Neither of these methods is considered to be as accurate as metallographic point counting, but the hand-held Feritscope® – in particular – is recognized as a field-applicable device with sufficient accuracy to be useful as an engineering tool. Aubrey et al. [A.3] showed in Figure A-2 that the standard deviation for Feritscope® measurements in their studies was less than ± 1 FN, but with some measurements exceeding that variation as FN approaches 30.

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[1] Ferrite number (FN) is the currently accepted designation for ferrite measurement and refers to a magnetically determined scale of ferrite measurement. It is related to ferrite volume (%) as shown in the constitution diagram relating nickel equivalent and chromium equivalent values (see ASME Code, Section III). A FN of 10 is approximately 9.2% ferrite by volume.

omewhat similar results were observed by Aubrey et al. [A.3] with respect to direct .easurements made with the Magne-Gage device, as shown in Figure A-3. In this case, one .ust recognize that the Magne-Gage block dial reading has to be divided approximately by a ıctor of ten in order to correspond to an FN measurement.

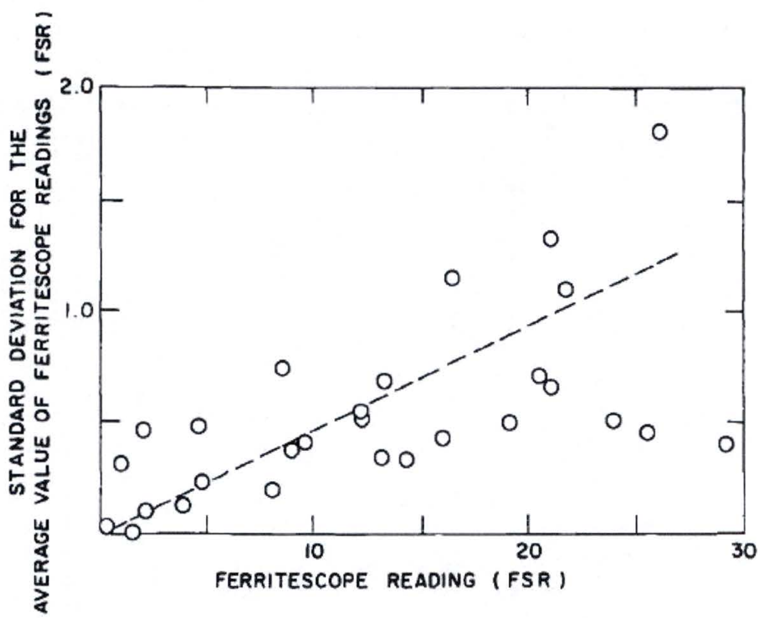


Figure A-2. Accuracy Demonstration for Feritscope® Measurements (Figure 7 from Reference A.3)

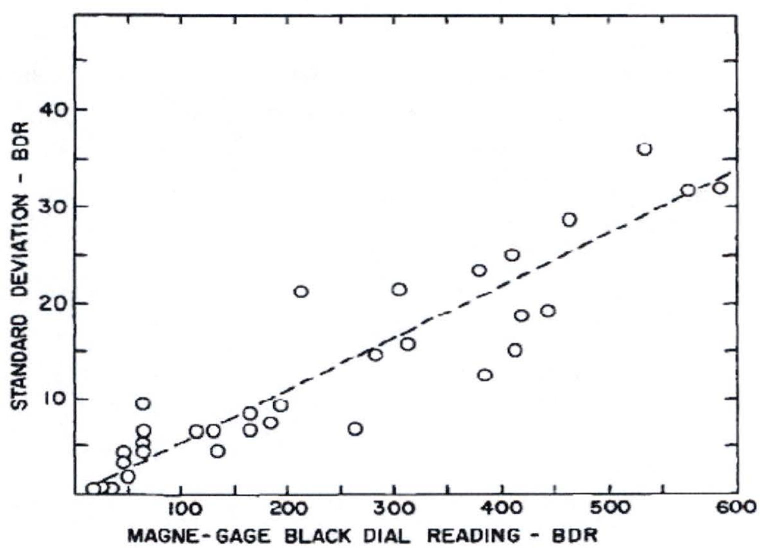
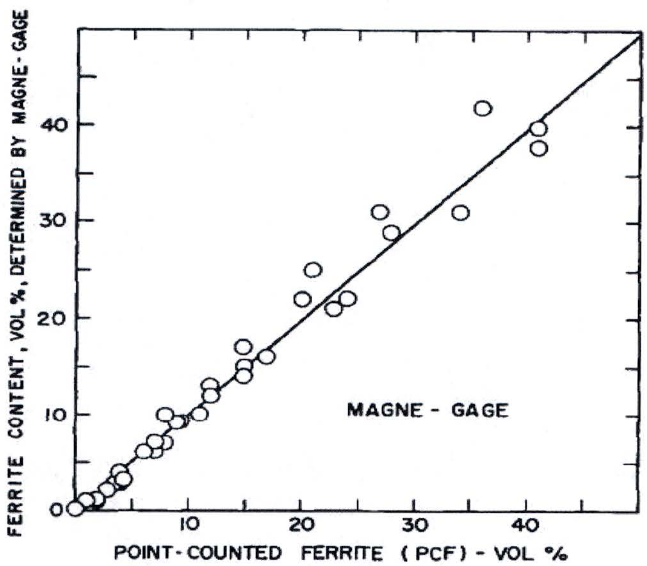


Figure A-3. Accuracy Demonstration for the Magne-Gage Device (Reference A.3)

The accuracy of both the Feritscope® and the Magne-Gage can also be assessed by plotting the readings from the two magnetic devices against the results from metallographic point counting, which is considered to be the most accurate of the direct measurement methods. Figures A-4 and A-5 show that the Magne-Gage percent measurements are almost exactly identical to point counting up to ferrite content of about 18%, and within 2 to 3% of the point counting measurement above that value. Figure A-5 shows that the Feritscope® measurements are perhaps even more accurate (or slightly over-measured) up to ferrite content nearing 25%.

From these comparisons it can be concluded that field measurements using hand-held devices seem to be capable of determining delta ferrite percent within of the order of ± 1%, or over-measuring slightly, within the crucial delta ferrite range of 15 to 25%.



Figures A-4. Comparison of the Accuracy of Magne-Gage Measurements with Metallographic Point Counting (Figure 11 from Reference A.3)

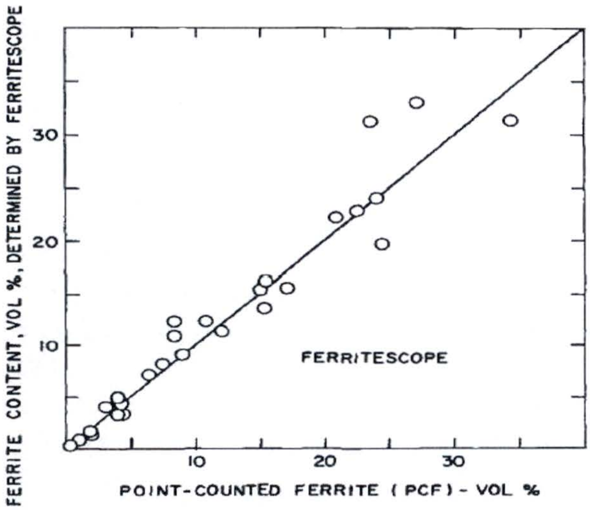


Figure A-5. Comparisons of the Accuracy of Feritscope® Measurements with Metallographic Point Counting (Figure 12 from Reference A.3)

In addition to evaluating the most widely used direct measurement techniques, Aubrey et al. [A.3] also evaluated several of the estimation or predictive methods that are based upon measured chemical composition, such as Hull's equations and the Schoefer equations that form the basis for the standard prediction method in ASTM A800. Those equations are shown as Model 1 and Model 4 in Figure A-6 (Table 8 from Reference A.3).

These equations have also been entered into a spreadsheet for this study and for comparison with the values shown in Table 1 of NUREG/CR-4513.

Note that Model 4 is the basis for a standard practice contained in ASTM A800-01 that covers guidance for all of the various methods – both measurement and prediction – that can be used to estimate the ferrite number. The particular prediction method adopted in the ASTM standard is that developed by Schoefer [A.4] which, like the Hull method and most of the other existing predictive methods, is based on an austenite stabilizer measure, called the nickel equivalent, Ni_{eq}, and a ferrite stabilizer measure, called the chrome equivalent, Cr_{eq}.

Table A-1 (adapted from Table 6 from Reference A.3) provides a comparison of the three direct measurement techniques (point counting, Magne-Gage, and Feritscope®), along with Hull (Model 1) and Schoefer (Model 4) predictions based on chemical composition for 45 different heats of material. Since more than half of the heats had percent delta ferrite less than or equal to 15%, Table 6 from Reference A.3 has not been completely reproduced here. Instead, Table A-1 shows the comparison for the heats of material with measured or estimated

delta ferrite within the range of interest. Note that, for the heats with measured delta ferrite between 15 and 25% (10 heats), the three direct measurement techniques typically vary by only 1%, if at all, while the two predictive models differ by 1 to 3%. Table 8 from Reference A.3 shows that the Hull equation model was the best of the various models tested, but still had a standard error of estimation of about 3%.

The conclusion that can be reached from the work in Reference A.3 is that direct measurement techniques, including both destructive laboratory methods and non-destructive field-deployed methods, are accurate within a typical error of 1% delta ferrite or less, while the very best of the predictive models are accurate within a typical error of 3% or less. If an additional standard deviation is added to that error to increase confidence, the direct measurement techniques might have an inherent error of 1% to 2%, while the predictive models would have a potential error of 5% to 6%.

Table A-1. Comparison of Ferrite Content Determined by Measurement or Estimation (Adapted from Table 6 of Reference A.3)

Specimen Number	Point-Count Measurement	Magne-Gage Measurement	Feritscope® Measurement	Hull's Equations Estimations	Schoefer Estimations
4-1491	11	10	12	9	8
19	12	12	11	16	14
14	15	15	16	21	17
6872	15	14	13	18	13
SB-1	16	17	16	16	14
3496	17	16	17	16	21
45S-10840	25	20	20	.....	.....
30	20	22	.....	21	22
15	21	25	.....	24	27
3485	22	23	22	24	23
3491	21	21	22	23	21
3494	24	23	24	22	25
45S-20643	24	22	22	25	21
SB-8	27	32	33	30	30

TABLE 8—*Results obtained with four models for predicting the ferrite content (Vol. %) from chemical composition.*

Model	Model Type	Equation	Correlation Coefficient	Standard Error of Estimation
1	composition ratio with Hull's equivalent factors [*16*]	$\%F = 100.30\left(\frac{Cr_e}{Ni_e}\right)^2 - 170.72\left(\frac{Cr_e}{Ni_e}\right) + 74.22$ $Cr_e = \%Cr + 1.21\%Mo + 0.48\%Si - 4.99$ $Ni_e = \%Ni + 0.11\%Mn - 0.0086Mn^2 + 18.4\%N + 24.5\%C + 2.77$	0.954	3.03
2	composition ratio with an equivalent factor of 1.4 for molybdenum (this investigation)	$\%F = 55.84\left(\frac{Cr_e}{Ni_e}\right)^2 - 87.87\left(\frac{Cr_e}{Ni_e}\right) + 35.39$ $Cr_e = \%Cr + 1.4\%Mo + 1.5\%Si - 4.99$ $Ni_e = \%Ni + 30\%C + 0.5\%Mn + 26(N - 0.02) + 2.77$	0.953	3.08
3	Hull's [*16*]	$\%F = 0.599\theta^2 + 1.715$ $\theta = -(\%Ni + 0.11\%Mn - 0.0086\%Mn^2 + 18.4\%N + 24.5\%C + 4.7 - 0.94\%Cr - 1.14\%Mo - 0.45\%Si)$	0.943	3.35
4	Schoefer composition ratio [*13*]	$\%F = 66.00\left(\frac{Cr_e}{Ni_e}\right)^2 - 106.13\left(\frac{Cr_e}{Ni_e}\right) + 44.30$ $Cr_e = \%Cr + \%Mo + 1.5\%Si - 4.99$ $Ni_e = \%Ni + 30\%C + 0.5\%Mn + 26(N - 0.02) + 2.77$	0.940	3.48

Figure A-6. Results Obtained from Four Models for Predicting the Ferrite Content (Vol. %) from Chemical Composition (Table 8 from A.3)

Two other more recent studies should be mentioned, beginning with the review for both weldments and castings provided in 1999 by Carl Lundin and his colleagues at the University of Tennessee [A.5], in a literature review for the Steel Founders' Society of America (SFSA) and the Department of Energy (DOE), and the paper by Bermejo [A.6] in the Welding Journal in 2012. Lundin et al. [A.5] reviewed each measurement and estimation technique available at the time, including relatively unproven methods – such as x-ray diffraction, magnetic saturation, and the Mossbauer effect – that were determined to be useful in the laboratory only, pending additional development, while emphasizing the methods with more practical and efficacious application – such as metallographic point counting, estimation based on material constitution information, and various magnetic (permeability and attraction) instrumentation methods. Bermejo [A.6] has summarized the available measurement and predictive methods while also providing an extremely detailed historical record of the maturation of the various predictive techniques, including the neural network approach developed by Vitek and his colleagues at the Oak Ridge National Laboratory in the late 1990s and early 2000s [A.7, A.8]. In addition, Bermejo developed an alternative predictive methodology that, based upon his doctoral dissertation work, he recommends as being much more accurate than the existing methods. That alternative predictive method will be discussed in detail subsequently.

Recent Accuracy Studies

The accuracy of the Schoefer predictive method and of eighteen other proposed predictive methods has been evaluated by Martorano et al. [A.9] in a recent paper in the Iron and Steel Institute of Japan (ISIJ) International journal. For their study the authors prepared sixteen stainless steel castings with measured delta ferrite (by Feritscope®) ranging from 0 to approximately 12%, with varying casting thicknesses. The Feritscope® measurements varied slightly with thickness, but in general were identical with less than 1% standard deviation. The highest delta ferrite casting had the highest standard deviations – of the order of 10.6%±0.6% for a thickness of 10 mm, 11.0%±1.0% for a thickness of 20 mm, 12.0%±1.0% for a thickness of 30 mm, and 11.0%±2.0% for a thickness of 40 mm.[2] The global measurements showed a value of 11.0%±1.0%. With respect to the various predictive methods, Martorano et al. used four measures of error: (1) average relative error, in percent; (2) error; (3) standard deviation; and (4) critical error interval. Of these different measures, the authors pointed out that the smallest relative errors were found for a prediction method called the Siewert diagram, followed by a predictive method called Hull-Schaeffler. Both of these methods had standard deviations of the order of 2.4 delta ferrite percentage. However, it should be pointed out that a large number of the castings had very low delta ferrite content, so that the study has limited applicability.

The work of Vitek et al. [A.7, A.8] using neural networks was carried out in the late 1990s and early 2000s, demonstrating the improved accuracy of the neural network models in comparison to the standard at that time – the prediction method that was adopted by the Welding Research Council in 1992, and that is often referred to as WRC-1992. Figures A-7 and A-8 illustrate the improvement in prediction in comparison to measured values. Figure A-7 shows the results for the neural network model, which Vitek labels as FNN-1999, which gives very good agreement with experimentally determined ferrite numbers at the lower ferrite numbers, which are typical for stainless steel welds. However, the neural network model predictions tend to be less accurate at the intermediate ferrite numbers ($15 < FN < 25$), which are more typical for cast components. On the other hand, Figure A-8 shows the results for the WRC-1992 prediction model, which systematically under-predicts the experimentally determined ferrite number by 3 or 4 FN at lower ferrite numbers that are typical of stainless steel welds, with gradual improvement as FN approaches 15 and above. The WRC-1992 model appears to have a worst-case error of about 3 FN in the range $15<FN<25$. This result is in general agreement with the results of Aubrey et al. [A.3].

More recent neural network predictive modeling by Vasudevan and his colleagues [A.10] at the Indira Gandhi Centre for Atomic Research (IGCAR) fast reactor establishment in India

[2] The Feritscope® measurements with a hand-held, field device indicate that delta ferrite can be measured in the field to within an accuracy of ±1%.

claims moderate improvement over the work of Vitek et al. Figure A-9 illustrates the level of improvement, with root-mean-squared errors that are less than FN 2 reported.

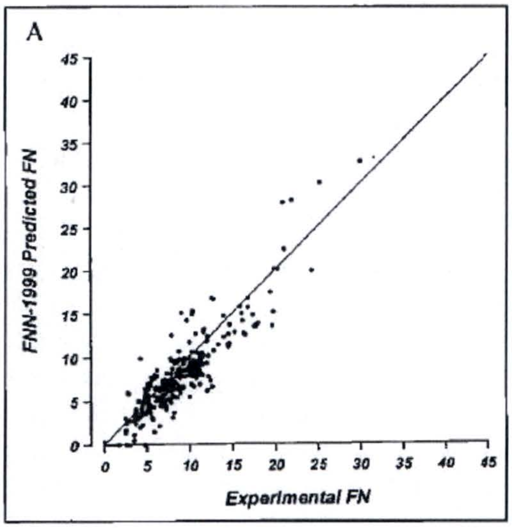


Figure A-7. (Figure 4A from Reference A.8)

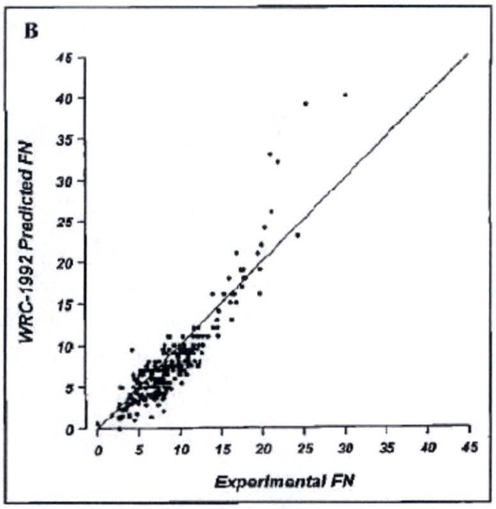


Figure A-8. (Figure 4B from Reference A.8)

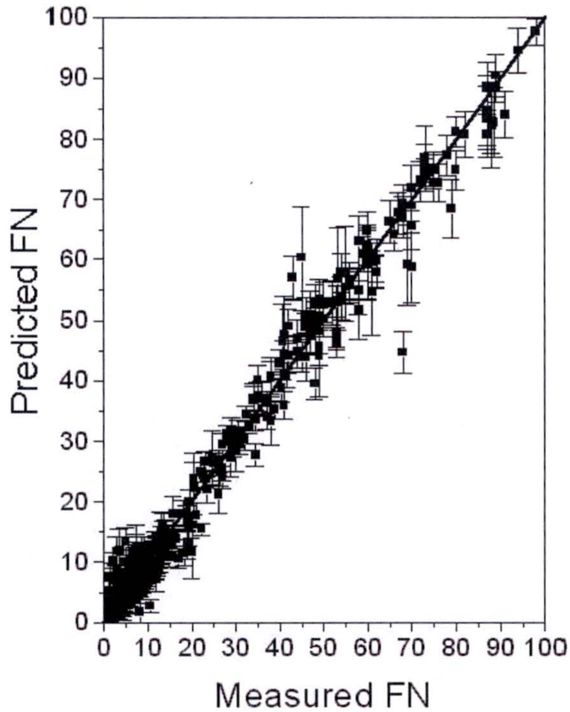


Figure A-9. Figure Taken from Paper by Vasudevan et al. (Reference A.10)

Potential Improved Delta Ferrite Predictive Model

In very recent years attempts have been made to optimize the delta ferrite predictive models in a manner similar to the adaptive learning and Bayesian approach used by the neural network modelers. Perhaps the most ambitious and perhaps the most successful of these approaches is based on the doctoral dissertation work of Bermejo, which he summarized and compared to other predictive methods in Reference A.6. The comparison also includes his own delta ferrite predictive method, based upon optimal data fitting, which he refers to as the New Method. Bermejo [A.6] describes the results as follows:

"*It is shown that the WRC-1992 underestimates the FN value throughout all the composition ranges while the FNN-1999 makes an accurate forecast for samples with FN<10. However, similar to WRC-1992, that method also underestimates the values for samples with FN>15. It is clear that the general expression provides better matches than do WRC-1992 and FNN-1999 for these 87 samples.*"

Bermejo [A.6] stated earlier in the paper that, through comparison with the 279 samples in the database with chemical composition within the austenitic ranges, "*Statistical processing confirms that the expected error (in FN prediction) is + 1.01 with a ± 1.06 FN confidence interval for a probability of 68% and a ± 2.12 FN confidence interval for a probability of 95%. Therefore, the general expression provides FN estimation with an error of 1.01 FN ± 2.12 FN and a probability of 95%.*"

The "general expression" is given as

$$Cr_{eq} = (\%Cr) + 1.37(\%Mo)$$

and

$$Ni_{eq} = (\%Ni) + 22(\%C) + 14.2(\%N) + 0.31(\%Mn),$$

with the ferrite number, FN, given by

$$\begin{aligned} FN = {} & 54.22 - 126.26\,(Cr_{eq} + Ni_{eq}) \\ & + [-48.11 + 37.14\,(Cr_{eq} + Ni_{eq})]\,(Cr_{eq} / Ni_{eq}) \\ & + [-0.23 + 61.95\,(Cr_{eq} + Ni_{eq})]\,(Cr_{eq} / Ni_{eq})^2 \end{aligned}$$

The Bermejo [A.6] equations have been programmed into a spreadsheet as part of this study and used to compare with the predictions and measurements reported in Table 1 of NUREG/CR-4513. In particular, the Bermejo [A.6] equations and the Hull equations were compared for accuracy for 20 of the 48 heats evaluated in NUREG/CR-4513 that had measured delta ferrite content between 15 and 25%. For Hull's equations, the root mean square (RMS) error over that range was 3.96% delta ferrite. However, when the most egregious measured outlier (a measured data point that was grossly and consistently different from any predictive method no matter which method was used) was removed from the data set, the RMS error dropped to 3.57% delta ferrite, and when the two most egregious measured outliers were removed, the RMS error dropped to 3.15% delta ferrite. Such a change with only two apparent outliers removed from a population of 20 data points is extremely significant, implying that a much larger data set with reliable measurements against which to compare should tend toward an RMS error band of approximately 3% delta ferrite over the range from 15 to 25% delta ferrite, in agreement with previous studies.

Conclusions

From this limited exercise of the data set given in NUREG/CR-4513, it can be concluded that, over the important delta ferrite range of 15 to 25%, Hull's equations can be shown to provide a predictive capability with a standard error of about 3% delta ferrite for a limited data set of measured values for comparison. Evidence from more comprehensive studies with a much larger data set of measured values for comparison indicates that the standard error can be

reduced down to 2.5%, and potentially down to 2%. In addition, while there is considerable evidence that the ASTM A800 standard systematically under-predicts delta ferrite content over the range of $15\% < \delta < 25\%$, the predictions using Hull equations and potential alternatives, such as the Bermejo equations, do not display such systematic under-prediction tendencies. At least two of the alternative methods – the neural network approach developed by Vitek and his colleagues at ORNL and the optimized regression data fitting approach suggested by Bermejo – have been shown to predict delta ferrite content with a standard error of the order of ±2%, with a sufficient robust data set of measured values, over the range of $15\% < \delta < 25\%$ delta ferrite content.

As pointed out in this Appendix A, Hull's equations have been evaluated several times and compared to both measured and other predictive methods, showing that the standard predictive error using Hull's equations is of the order of 3% without the persistent under-prediction characteristics of the ASTM standard method. The various studies are all in general agreement with each other. Therefore, based on this review and assessment, the following conclusion can be drawn:

> Adding 6% to the delta ferrite values predicted by Hull's equations is not necessary, since the available data clearly show that Hull's equations do not systematically under-predict measured values. There should be no requirement to assume that a prediction based on Hull's equations has systematically and erroneously under-predicted the delta ferrite content by 6%; instead, the standard error in the Hull's equations predictions is of the order of 3%, and that level of standard error provides a reasonable estimate of the uncertainty in the delta ferrite prediction in the absence of an under-predictive trend.

To implement this conclusion, the guidance in Reference A.11 should be changed to read "*Note that calculated δ-ferrite should use Hull's equivalent factors or a method producing an equivalent level of accuracy (±3% standard predictive error between measured and calculated values).*" Standard predictive error, rather than bounding error, should be used in this guidance, since Hull's equivalent factors do not systematically under-predict measured values of delta ferrite. In the absence of systematic under-prediction, such as that shown by the WRC-1992 standard, there is no need to require 95% or 99% confidence in the calculation.

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A.10. M. Vasudevan, A. K. Bhaduri, B. Raj, and K. P. Rao, "Delta Ferrite Prediction in Stainless Steel Welds Using Neural Network Analysis and Comparison with Other Prediction Methods," Journal of Materials Processing Technology, Volume 142, No. 1, pp. 20 to 28, 10 November 2003.

A.11. Letter, Christopher I. Grimes, Chief, License Renewal and Standardization Branch, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, to Douglas J. Walters, Nuclear Energy Institute, "License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components," May 19, 2000 (ADAMS Accession No. ML003717179).

RAI 9-a

In the September 18, 2012, RAI response, the BWRVIP stated the following:

> As stated in the responses to RAI 6 and RAI 7, the uncertainty in delta ferrite and potential for an increase in delta ferrite due to welding and weld repairs is not expected to be significant. In RAI 6 it was shown that even when assuming the maximum Mo content of 0.5 wt.% the average increase in ferrite was relatively small. This marginal increase in ferrite would have a slight effect on reducing the toughness of the material (since an increase in ferrite results in a decrease in toughness). However, for the range of exposure conditions that would be experienced for cast austenitic stainless steel (CASS) components in a BWR, the toughness values are expected to be well above recommended lower bound toughness of 255 kJ/m^2. Therefore, the BWRVIP believes that use of the lower bound toughness is sufficiently conservative and consequently, would not affect the methodology to estimate the toughness for irradiated conditions.

The NRC staff has reviewed the technical bases for the RAI responses and compared the methodology used in BWRVIP-234 to predict fracture toughness as a function of neutron exposure to a previously reported prediction from NUREG/CR-6960. The comparison is shown in Figure 1 along with the range of predicted J values at 2.5 mm crack extension for unirradiated CF-8 with no thermal embrittlement (TE) and maximum TE (no irradiation) for CF-8 with > 15 percent delta ferrite (Section 3.1.1 of NUREG/CR-4513, Rev. 1).

The NRC staff is concerned that the BWRVIP-100-A model for CASS may be inadequate for the following reasons:

- The data base for development of the BWRVIP-100-A model to predict toughness does not include any CASS materials, just welds, heat affected zone material, and base metal. Further, the number of welds in the database is limited, and the delta ferrite content for these welds is unknown. Welds typically contain 7% to 10% ferrite while CASS materials can have between 5% and 25% ferrite. Therefore, the toughness predicted by the BWRVIP-100-A model may not be appropriate or conservative for CASS materials.

- The NUREG/CR-6960 model is based on data from CASS materials, welds, and wrought materials; therefore, the NRC staff believes it more conservatively represents the fracture toughness of irradiated CASS materials. At fluence values > 0.3 dpa, the toughness predicted by the NUREG/CR-6960 curve is below the value of 255 kJ/m^2 used as the basis for the screening based on toughness in BWRVIP-234.

The NRC staff requests that the BWRVIP review all available data on fracture toughness of irradiated CASS materials, and the associated uncertainties, and to either revise its methodology to predict the lower-bound toughness of CASS materials at reactor operating temperatures after approximately 60 years of operation, or provide further justification that BWRVIP-100-A is sufficiently conservative.

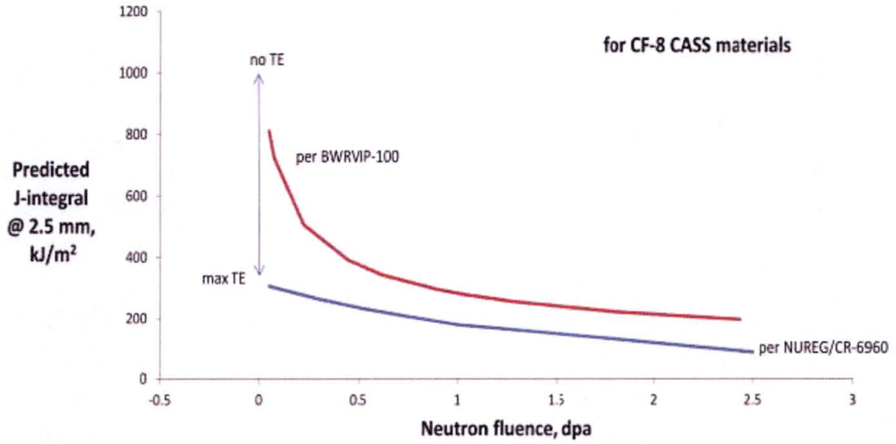

Figure 1. Plot of predicted toughness (J-integral value at 2.5 mm crack extension) from BWRVIP-100-A and NUREG/CR-6960 as a function of neutron fluence. The predicted lower-bound toughness of CF-8 due to TE alone is shown for reference.

<u>BWRVIP Response to RAI 9-a</u>

This response addresses the requests by the NRC staff that BWRVIP: (1) review all available data on fracture toughness of irradiated CASS materials, and the associated uncertainties; and (2) either revise its methodology to predict the lower-bound toughness of CASS materials at reactor operating temperatures after approximately 60 years of operation, or provide further justification that BWRVIP-100-A is sufficiently conservative.

In order to address the first request, the industry has documented efforts to review all of the available data on fracture toughness of irradiated CASS materials, along with the associated uncertainties. That review is represented by MRP-276 [1], which includes discussions of the data contained in NUREG/CR-6960 [2], but does not address more recent test data, such as those in Reference 3, that have been published since the issuance of MRP-276. In addition to that discussion, the industry shares the regulatory concerns that some sub-size compact tension

(CT) specimen test data on thermally-aged and irradiated CF-8M material with relatively high delta ferrite have been obtained with measured fracture toughness that falls below the fracture toughness screening criterion of 255 kJ/m^2 at a crack extension of 2.5 mm (0.1 in.). However, evidence in the open literature shows that miniature CT specimens may lead to substantially lower upper shelf elastic-plastic fracture toughness measurements, in comparison to full-size CT specimens. Therefore, the industry urges caution in the literal interpretation of the miniature specimen test results, especially since an upward adjustment of the miniature specimen data – as suggested in the open literature [4] – would lead to relatively consistent results with irradiated SAW and SMAW fracture toughness data, such as those used to support the BWRVIP-234 [5] fracture toughness estimation model that shows the lower bound J at 2.5 mm crack extension is 295 kJ/m^2. Specifically, the test results from 18 1TC(T) and 20 miniature CT specimens of RPV and low-alloy steel tested at different temperatures clearly show that miniature CT samples exhibit lower fracture toughness properties, both in terms of initiation of ductile tearing (according to various test standards) and in terms of resistance to ductile crack propagation (J-R curve). The reduction of tearing resistance might be attributed to work hardening prevailing over loss of constraint in the uncracked ligament for a side-grooved specimen, or to the inadequacy of J-integral to represent ductile crack extension in very small specimens [4].

With respect to the second request regarding the methodology for estimating CASS bounding fracture toughness in BWRVIP-234 [5], the NRC staff has raised two concerns. First, the data base for the development of the BWRVIP predictive model for CASS fracture toughness reduction does not include any actual CASS material data, and only reflects fracture toughness data for austenitic stainless steel welds, heat-affected zone (HAZ) material, and base metal. Such data limit the amount of delta ferrite that might be contained in the materials and therefore perhaps might be non-conservative with respect to fracture toughness reductions as a function of embrittlement. Second, the NRC staff is more inclined to rely on the data generated in Reference 2, especially on the data in that reference, such as the sub-size CT specimen data, that show reductions in fracture toughness well below the 255 kJ/m^2 J-value at 2.5 mm (0.1 in.).

However, the process used in BWRVIP-234 to estimate the bounding fracture toughness is much more conservative than described in the NRC staff RAI, since the irradiated fracture toughness data for the welds, heat-affected zones, and base metal have been further reduced through the use of the square of the Z factors that take into account the alleged differences in delta ferrite between CASS and stainless steel welds. This conservative approach is described in some detail in Section 4.3 of BWRVIP-234, and will only be summarized here for relative completeness.

The first thing to note about that procedure is that, other than the use of J-R curve data contained in BWRVIP-100-A [6], the methodology used in BWRVIP-234 is independent of

BWRVIP-100-A. The second thing to note about the methodology is that, since the J-R curves only extended out to 1.5 to 1.6 mm of crack extension, curve fitting of the data using the C and n parameters was necessary in order to extrapolate the J-R curves out to 2.5 mm of crack extension. Third, while two wrought stainless steel J-R curves with appropriate neutron irradiation exposure were used for this curve fitting and extrapolation, Z factors were then used to adjust the extrapolated J-R values at 2.5 mm downward to reflect flux weld fracture toughness properties, in accordance with ASME Code Section XI, Appendix C practice. This process is known to be conservative.

The validity of the BWRVIP-234 methodology in this regard is confirmed by comparing the derived SAW and SMAW fracture toughness approximation with actual irradiated weld data, even though that actual weld data might have been obtained on material with a much higher fluence. The irradiated SAW and SMAW fracture toughness data can also be compared with the irradiated and thermally-aged fracture toughness data in Figure 2, which may help to understand why the effects of using sub-size CT test specimens may have significantly influenced the fracture toughness results.

The validity confirmation starts with Reference 2 which, in addition to the CASS data, also provides some very useful fracture toughness data on SAW and SMAW heat-affected zone material that has been subjected to accumulated neutron irradiation fluence of 1.44×10^{21} n/cm^2 (about 2 dpa). Figure 2, taken from Figure 72a in Reference 2, and Figure 3, taken from Figure 72b in Reference 2, are shown below for comparison. From these figures, the J-value at 2.5 mm (0.1 in.) is above 300 kJ/m^2 for the SMAW heat-affected zone, and slightly below 250 kJ/m^2 for the SAW heat-affected zone. For the wrought materials used as the starting point in BWRVIP-234, the J-R curve values at 2.5 mm of crack extension (see Table 4-3 of Reference 5) were 744 and 638 kJ/m^2, respectively. When these wrought values are adjusted downward by Z^2 factor, which is approximately 3, the agreement with the actual fracture toughness data is quite good, confirming the approach used in BWR-234.

Based on these findings, the industry recommends that the existing fracture toughness screening criterion for toughness of 255 kJ/m^2 at a crack extension of 2.5 mm (0.1 in.) remain in place.

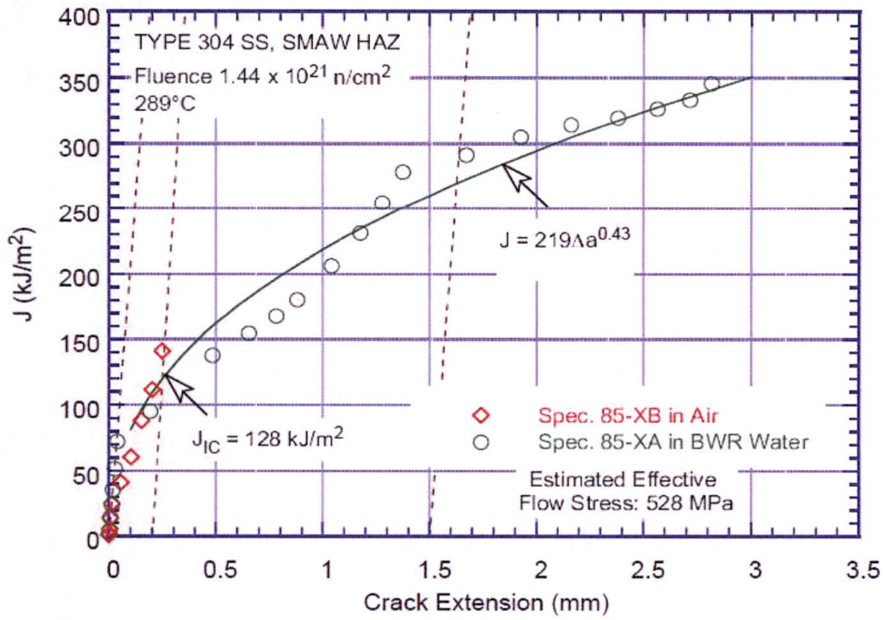

Figure 2. J-R Curve for Irradiated SMAW Heat-Affected Zone Material at 289°C Tested in Both Air and BWR Water (From Reference 2)

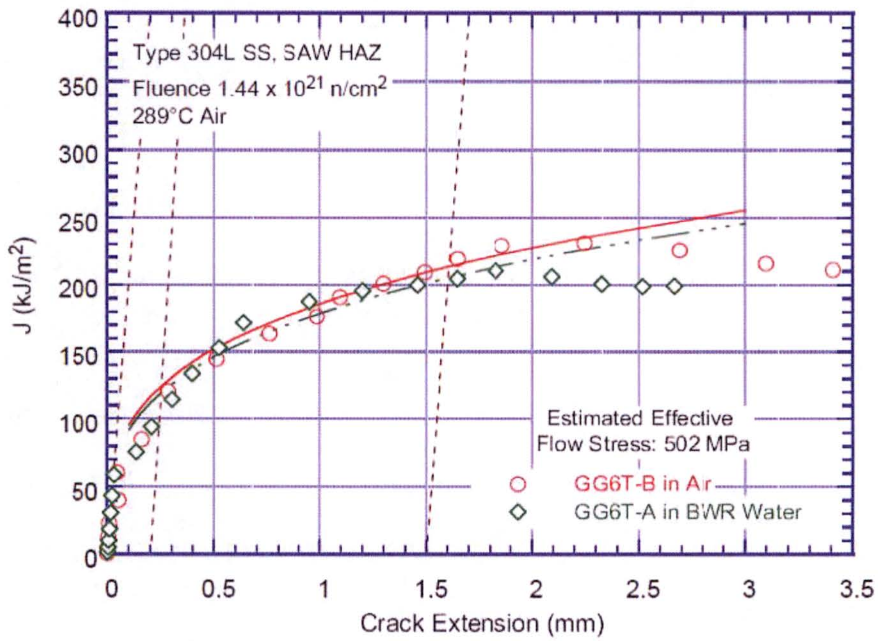

Figure 3. J-R Curve for Irradiated SAW Heat-Affected Zone Material at 289°C Tested in Both Air and BWR Water (From Reference 2)

It is recognized that the delta ferrite content for these HAZ materials is somewhat below the delta ferrite content for the CASS piping, pump body, and valve body materials studied in much of the literature, and with which the staff may be most familiar. However, the delta ferrite content for the reactor internals materials of concern is typically much lower, often comparable with the delta ferrite content of austenitic stainless steel welds and heat-affected zones. Therefore, the comparisons of fracture toughness data for these HAZ materials roughly parallel the comparisons observed previously, and – with the exception of the sub-size CT specimen data reported in Reference 2 – the bounding behavior of the HAZ materials data continues to hold true. Based upon previous work and the data in Reference 2, the J-R curve data for irradiated CASS material would be expected to be bounded from below by the irradiated SAW and SMAW J-R curve data. If the irradiated and thermally-aged J-R curve data in Figure 3 were to be adjusted upward to reflect observations made by other sub-size CT specimen investigators testing for upper shelf elastic-plastic fracture toughness behavior, the final results would be comparable. Therefore, the BWRVIP-100-A methodology is sufficiently conservative for determining the lower-bound toughness of CASS materials in BWR internals at reactor operating temperatures after approximately 60 years of operation.

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Appendix to BWRVIP Response to RAI 9-a

Technical information is provided in this appendix to support the response to the requests by the NRC staff that the BWRVIP: (1) review all available data on fracture toughness of irradiated CASS materials, and the associated uncertainties; and (2) either revise its methodology to predict the lower-bound toughness of CASS materials at reactor operating temperatures after approximately 60 years of operation, or provide further justification that BWRVIP-100-A is sufficiently conservative.

Industry Review of Irradiated CASS Data

With respect to the first request by the NRC staff, the industry has documented its efforts to review all of the available data on fracture toughness of irradiated CASS materials, with an evaluation of the associated uncertainties. That review is represented by MRP-276 [A.1], which includes discussions of the data contained in NUREG/CR-6960 [A.2], but does not include a review of the information from ANL-12/56 [A.3] or the information in even more recent documents. While the MRP-276 review was primarily aimed at PWR internals fabricated from CASS materials, the results of that review are considered reasonably applicable to BWR internals fabricated from CASS materials, as well.

Based upon the MRP-276 review and associated discussion, the industry shares the regulatory concerns that some sub-size compact tension (CT) specimen test data on thermally-aged and irradiated CF-8M material with relatively high delta ferrite have been obtained with measured fracture toughness that falls below the fracture toughness screening criterion of 255 kJ/m^2 at a crack extension of 2.5 mm (0.1 in.). Those data were originally published in Reference A.2 and reproduced in Reference A.1, and are provided here as Figures A-1 and A-2. Those figures show J-R fracture toughness curves for what appears to be a limiting CASS material – CF-8M – thermally aged at 400°C for 10,000 hours and subsequently irradiated to an accumulated fluence of 1.63×10^{21} n/cm^2 (2.46 dpa). The value of J at 2.5 mm (0.1 in.), for both specimens, is in the neighborhood of 120 to 135 kJ/m^2, well below the presumed lower bound of 255 kJ/m^2 derived from the SAW and SMAW data cited in References A.4a and A.4b. However, these data deserve more scrutiny, especially with respect to the effect of using sub-size test specimens.

First, note that the data from Figure A-2 are also shown in Figure A-1, albeit in a compressed fashion. Second, note that the J-R curve results were obtained from ¼-thickness compact tension (CT) specimens for all of the open symbol data, compared with full-thickness CT specimens for the closed symbol data. Third, note that all of the specimens were presumably taken from the presumed limiting material – CF-8M – with delta ferrite content of the order of 28%. Fourth, while the data with open symbols shown in both Figures A-1 and A-2 represent specimens that have been both thermally aged and irradiated to 2.46 dpa, the remaining data with open symbols and the data with closed symbols are based upon the same heat of thermally-aged CASS material, but not irradiated.

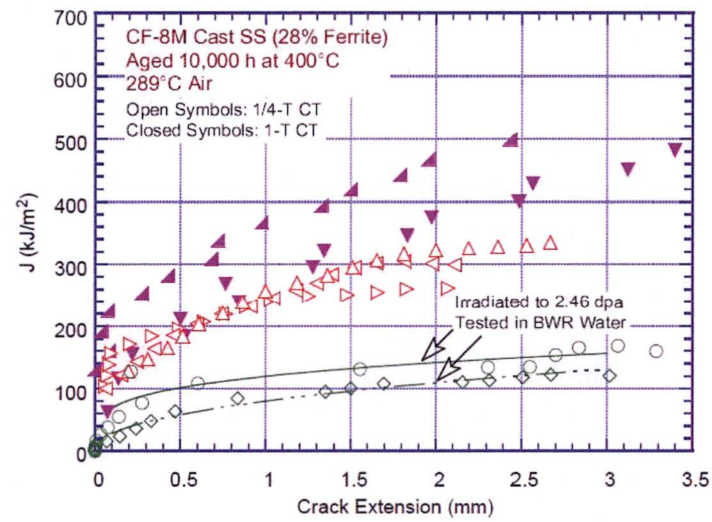

Figure A-1. J-R Fracture Toughness Curves for Thermally-Aged and Irradiated CF-8M Material (From Reference A.2)

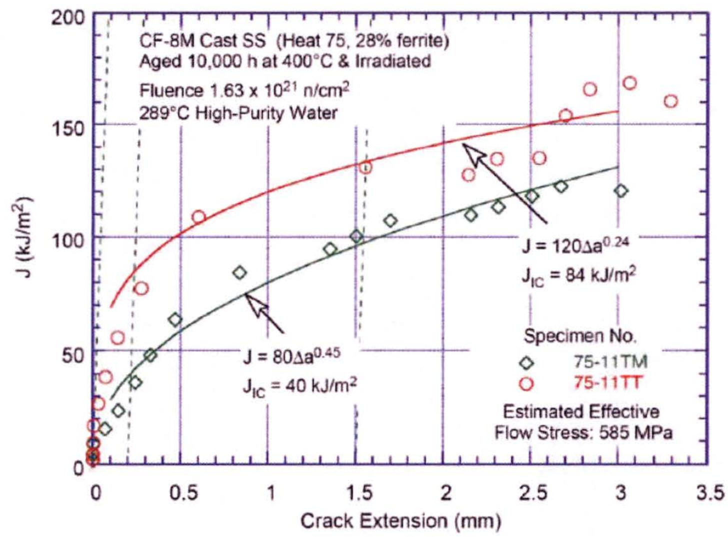

Figure A-2. J-R Curves for Thermally-Aged and Irradiated CF-8M Material Tested in Simulated BWR Water Environments (From Reference A.1)

A number of possible conclusions can be reached from examining Figure A-1. First, the two higher J-R curve data sets – one with open triangular symbols and the other with closed triangular symbols – give quite different results, even though the materials are from the same heat with presumably identical thermal aging. The only difference seems to be that one set was

based on full-size compact tension test (CT) specimens, while the other set was based on ¼-thickness CT test specimens, with the ¼-thickness specimens displaying in most cases substantially lower toughness. Second, if those differences are neglected for the time being, the effect of the 2.46 dpa neutron irradiation exposure is very significant, as noted in the discussion of the irradiated and thermally-aged data in Reference A.2. Finally, since the closed symbol data is based upon unirradiated material, note that the value of J at 2.5 mm (0.1 in.) for the full-size CT test specimens ranges from 400 to 500 kJ/m^2, well above 255 kJ/m^2. The presumably unirradiated but thermally-aged ¼-thickness CT test specimen data exhibit J values at 2.5 mm (0.1 in.) that are considerably lower, in some cases of the order of 250 kJ/m^2, indicating some question about the influence of the sub-size CT specimens on the resulting fracture toughness data.

Essentially this same effect has been observed by other investigators [A.5, A.6a, A.6b] attempting to use miniature CT test specimens to measure fracture toughness in ductile materials, such as ferritic steel materials at upper shelf temperatures. The reduced values of elastic-plastic fracture toughness in such cases, as represented by J-R curves, are very typical of those seen in Figure A-1.

Therefore, based upon evidence in the open literature that miniature CT specimens may lead to substantially lower upper shelf elastic-plastic fracture toughness measurements, in comparison to full-size CT specimens, the industry urges caution in the literal interpretation of the miniature specimen test results. This caution is well placed, since an upward adjustment of the miniature specimen data – as suggested in References A.6a and A.6b – would lead to relatively consistent results with irradiated SAW and SMAW fracture toughness data. (Such SAW and SMAW fracture toughness data were used to support the BWRVIP-234 fracture toughness estimation model that shows the lower bound J at 2.5 mm crack extension is 295 kJ/m^2). Specifically, the test results reported in Reference A.5 for 18 1TC(T) and 20 miniature CT specimens of RPV and low-alloy steel tested at different temperatures clearly show that miniature CT samples exhibit lower fracture toughness properties, both in terms of initiation of ductile tearing (according to various test standards) and in terms of resistance to ductile crack propagation (J-R curve).

This observed reduction of tearing resistance might be attributed to work hardening prevailing over loss of constraint in the uncracked ligament for a side-grooved specimen, or to the inadequacy of the J-integral to represent ductile crack extension in very small specimens [A.5]. Irrespective of the potential source of the error, the effect on the upper shelf elastic-plastic behavior for irradiated sub-size low-alloy ferritic steel test specimens is essentially identical to the elastic-plastic behavior for irradiated sub-size austenitic stainless steel test specimens. Based on these findings, the industry recommends that the existing fracture toughness screening criterion for toughness of 255 kJ/m^2 at a crack extension of 2.5 mm (0.1 in.) is still valid for use in evaluating BWR vessel internals, pending further evaluation of the effect of using sub-size CT specimens to determine fracture toughness in the upper shelf or near upper shelf regions.

In summary, Reference A.1 provides the industry data review requested by the NRC staff. Note that Reference A.1 is primarily aimed at PWR reactor internals components fabricated from wrought austenitic stainless steels, and the data for cast reactor internals component materials is not singled out for emphasis. In addition, rather than evaluating the latest irradiated CASS materials fracture toughness data (other than even more recent data, such as those contained in Reference A.3), the major technical point made in Reference A.1 was that no synergistic effect between thermal-aging and neutron irradiation embrittlement was observed for either wrought or cast austenitic stainless steels. The major recommendation in that report was that, for the purposes of flaw evaluations, for irradiation exposure less than 0.3 dpa, fully-thermally-aged fracture toughness data would be sufficiently conservative to cover combinations of aging effects on reduction in fracture toughness, while for irradiation exposure greater than 0.3 dpa, the reduction in fracture toughness from neutron irradiation exposure alone was sufficient to provide bounding results.

The purpose of reviewing those data again here is quite different, specifically to examine the available fracture toughness data for CASS materials subject to either thermal aging embrittlement, neutron irradiation embrittlement, or combinations of the two, in order to determine whether the currently accepted lower bound fracture toughness of 255 kJ/m^2 should be revised downward.

Based upon the limited data available for the presumably limiting material – CF-8M – no such downward revision is justified solely on the basis of sub-size CT specimen testing. Either some form of upward adjustment of the data is needed, following the efforts tried in References A.6a and A.6b, or additional testing efforts are needed to generate data on full-size CT specimens to verify the validity of the existing data.

BWRVIP-234 Methodology

With respect to the second request regarding the methodology for estimating CASS bounding fracture toughness in BWRVIP-234, the NRC staff has raised two concerns. First, the data base for the development of the BWRVIP predictive model for CASS fracture toughness reduction does not include any actual CASS material data, and only reflects fracture toughness data for austenitic stainless steel welds, heat-affected zone (HAZ) material, and base metal. Such data limit the amount of delta ferrite that might be contained in the materials and therefore perhaps might be non-conservative with respect to fracture toughness reductions as a function of embrittlement. Second, the NRC staff is more inclined to rely on the data generated in Reference A.2, especially on the data in that reference, such as the sub-size CT specimen data, that show reductions in fracture toughness well below the 255 kJ/m^2 J-value at 2.5 mm (0.1 in.).

The NRC staff has expressed concerns previously about the fracture toughness screening value of 255 kJ/m^2, which has its origins in the letter from Christopher I. Grimes to Douglas J. Walters, dated May 19, 2000 [A.7], that cited an acceptable screening criterion that could be used to

differentiate between non-significant and potentially significant reductions in fracture toughness for thermally-aged cast austenitic stainless steel (CASS) components. The J-R crack growth resistance value of 255 kJ/m^2 at a crack depth of 2.5 mm (0.1 in.) was selected based on several studies conducted by EPRI [A.4a, A.4b, and A.8]. However, it should be noted that, at the time this screening criterion was selected and (later) accepted by NRC, the impact of possible additional reductions in fracture toughness from neutron irradiation embrittlement was only partially considered. Therefore, the applicability of the criterion was limited with respect to reactor internals, and was more specifically focused at that time on piping, pump, and valve bodies.

With respect to this screening value and the methodology used in BWRVIP-234 to estimate the bounding fracture toughness, it should be pointed out that the methodology is much more conservative than indicated by the NRC staff. For example, the irradiated fracture toughness data for the welds, heat-affected zones, and base metal have been reduced through the use of the square of the Z factors that take into account the alleged differences in delta ferrite between CASS and stainless steel welds. This conservative approach is described in some detail in Section 4.3 of BWRVIP-234, and will only be summarized here for relative completeness.

The first thing to note about that procedure is that, other than the use of J-R curve data contained in BWRVIP-100-A [A.9], the methodology used in BWRVIP-234 is independent of BWRVIP-100-A. The second thing to note about the methodology is that, since the J-R curves only extended out to 1.5 to 1.6 mm of crack extension, curve fitting of the data using the C and n parameters was necessary in order to extrapolate the J-R curves out to 2.5 mm of crack extension. Third, while two wrought stainless steel J-R curves with appropriate neutron irradiation exposure were used for this curve fitting and extrapolation, Z factors were then used to adjust the extrapolated J-R values at 2.5 mm downward to reflect flux weld fracture toughness properties, in accordance with ASME Code Section XI, Appendix C practice. This process is known to be conservative.

A confirmation of the validity of the BWRVIP-234 methodology in this regard is to compare the derived SAW and SMAW fracture toughness approximation with actual irradiated weld data, even though those actual weld data might have been obtained on material with a much higher fluence. In the process of confirming that validity, the irradiated SAW and SMAW fracture toughness data can also be used to compare with the irradiated and thermally-aged fracture toughness data in Figure A-1, which may help to understand why the effects of using sub-size CT test specimens may have significantly influenced the fracture toughness results.

The validity confirmation starts with Reference A.2 which, in addition to the CASS data, also provides some very useful fracture toughness data on SAW and SMAW heat-affected zone material that has been subjected to accumulated neutron irradiation fluence of 1.44×10^{21} n/cm^2 (about 2 dpa). Figure 2, taken from Figure 72a in Reference A.2, and Figure 3, taken from Figure 72b in Reference A.2, are shown above for comparison. From these figures, the J-value at 2.5 mm (0.1 in.) is above 300 kJ/m^2 for the SMAW heat-affected zone, and slightly

below 250 kJ/m^2 for the SAW heat-affected zone. For the wrought materials used as the starting point in BWRVIP-234, the J-R curve values at 2.5 mm of crack extension (see Table 4-3 of BWRVIP-234) were 744 and 638 kJ/m^2, respectively. When these wrought values are adjusted downward by Z^2 factor, which is approximately 3, the agreement with the actual fracture toughness data is quite good, confirming the approach used in BWRVIP-234.

Again, it is recognized that the delta ferrite content for these HAZ materials is somewhat below the delta ferrite content for the CASS reactor internals materials of concern. However, the comparisons of fracture toughness data for these HAZ materials roughly parallel the comparisons observed previously, and – with the exception of the sub-size CT specimen data reported in Reference A.2 – the bounding behavior of the HAZ materials data continues to hold true. If the irradiated and thermally-aged J-R curve data in Figures A-1 and A-2 were to be adjusted upward to reflect observations made by other sub-size CT specimen investigators testing for upper shelf elastic-plastic fracture toughness behavior, the final results would also be comparable. Therefore, the BWRVIP-100-A methodology is sufficiently conservative for determining the lower-bound toughness of CASS materials in BWR internals at reactor operating temperatures after approximately 60 years of operation.

Original Technical Basis for the 255 kJ/m^2 Fracture Toughness Screening Value

The original technical basis for the 255 kJ/m^2 fracture toughness screening value was explained in some detail in Section 2.4 of Reference A.4b, with a section title of "Comparisons to Fracture Toughness of Weldments." That discussion is repeated below for background purposes and to provide potentially useful supplemental information.

> *"A comparison of fracture toughnesses for aged CASS material and those for austenitic stainless steel weld metal is instructive. Mills [13] has offered a microstructural explanation for the lower toughness of submerged-arc welds (SAW) and shielded-metal-arc welds (SMAW), relative to gas-tungsten-arc welds (GTAW), that explains the microstructural changes in CASS materials during thermal aging embrittlement, as well. He observed that the failure mode for all of the welds was a dimple rupture mechanism, but that the SAW and SMAW failures were initiated by a combination of decohesion of second-phase particles of manganese silicide and local rupture/decohesion of delta ferrite particles. Using a composite plot of fracture toughness data, as shown in Figure 23 (Figure 11 from Reference 13), he suggested lower-bound J_c values for SAW, SMAW and GTAW at 427°C (800°F) to 538°C (1000°F) of 40, 70, and 230 kJ/m^2 (228, 400, and 1300 in-lb/in^2), respectively. At 24°C (75°F), the recommended lower-bound values of J_c were 100 kJ/m^2 (571 in-lb/in^2) for SAW and SMAW, and 350 kJ/m^2 (2000 in-lb/in^2) for GTAW.*

Jaske and Shah [14], in their review of available fracture toughness data for aged CASS materials, point out that very few of the data fall below the very conservative component end-of-life lower-bound limit recommended by Framatome of 100 kJ/m^2 (571 in-lb/in^2), even for fully-aged conditions. A more direct comparison can be obtained by plotting the lower-bound J-R curve data for statically-cast and centrifugally-cast CF-8M material with delta ferrite ranging from 15 % to about 28 %, as tabulated by Chopra and Shack [15], against J-R curve data for Type 316 SAW metal, as given in Reference 16. Figure 24 shows that statically-cast CF-8M material with delta ferrite less than 10 % has considerably greater crack growth resistance at reactor operating temperature than does SAW metal, while statically-cast CF-8M material with delta ferrite levels between 10 and 15 % has a resistance to crack growth that is quite similar, but slightly greater, to that of SAW metal at reactor operating temperature.

Only statically-cast CF-8M material with delta ferrite greater than 15 % displays a crack growth resistance below that for SAW metal, and then only for very large crack extensions. Figure 25 shows that, even for centrifugally-cast CF-8M material with delta ferrite greater than 15 %, the crack growth resistance is similar, but slightly greater, than that for SAW metal at reactor operating temperature. Even though the fracture toughness data for CF-8M material are limited to a delta ferrite content of about 28 %, these comparisons are likely to be valid for materials with higher delta ferrite content, based upon fracture toughness trend curves as a function of delta ferrite content for low-molybdenum material. The latter data extends out to the 40 % delta ferrite range. This extrapolation is also supported by Figure 12, which illustrates the saturation effect of an aging parameter that includes delta ferrite content. Such favorable comparisons justify the use of existing weld metal acceptance criteria for flaws detected and sized during the inservice inspection of CASS components, as discussed in Section 3."

The references to the work by Mills, by Jaske and Shaw, and by Chopra and Shack given in Reference A.4b are shown here as References A.10, A.11, and A.12, respectively.
The discussion in Section 2.4 of Reference A.4b led to the following recommendations in Section 4 of Reference A.4b, with a section title of "Proposed Evaluation Procedure." The first and last paragraphs of those recommendations are repeated below for background purposes.

The proposed procedure for evaluating the effects of thermal aging embrittlement effects (e.g., reduction of fracture toughness) in Class 1 reactor coolant system and primary pressure boundary cast austenitic stainless steel components is composed of two parts: (1) screening to determine whether or not the effects of thermal aging embrittlement are potentially significant to the continued function of a particular

CASS component during the license renewal term; and (2) when the effects of thermal aging embrittlement are found to be potentially significant for CASS components, an aging management program based upon periodic inservice inspection and flaw evaluation criteria that provides the basis for demonstrating aging management during the license renewal term. This same approach, with minor modifications, also applies to Class CS internals components.

In summary, the effects of thermal aging embrittlement on Class 1 CASS reactor coolant system and primary pressure boundary components, and for Class CASS CS internals components are found to be either not significant for the license renewal term, based upon material chemical composition and casting type screening criteria, or, if the effects are potentially significant, can be managed adequately through the license renewal term by the periodic volumetric, surface, and visual inservice inspection program elements specified in the ASME Code Section XI, Subsection IWB, or the alternative inservice examination and flaw tolerance evaluation procedures of ASME Nuclear Code Case N-481. When conditions are detected during these inservice inspections that exceed the allowable limits given in Table IWB-3518-2, engineering evaluations of either detected or postulated flaws shall be carried out using material properties and acceptance criteria applicable to SAW and SMAW metal per the evaluation procedures presented in IWB-3640. More favorable material properties and acceptance criteria may be justified, on a case-by-case basis, using the fracture toughness data in Reference 15."

The Reference 15 cited in the closing paragraph of the recommendations is shown here as Reference A.12, and the term Class CS refers to ASME Code core support structures. Therefore, the purpose of the comparisons to SAW and SMAW weld metal fracture toughness was not to establish a criteria or screening value of fracture toughness, but instead to provide a justification for using the ASME Section XI IWB-3640 allowable weld defect sizes for CASS components, with the clearly stated alternative of using more favorable fracture toughness data where the use of such data could be justified. This original technical basis continues to be valid.

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RAI-10

The BWRVIP-234 report has considered only the properties of CASS materials at operating temperatures. The NRC staff requests that the BWRVIP assess the structural integrity of CASS materials at typical leak test temperatures.

BWRVIP Response to RAI-10

An industry review of existing impact energy data on high delta ferrite CASS materials shows a minimal change of the temperature transition from upper shelf behavior to lower shelf behavior as a function of thermal aging embrittlement. This minimal change is most noticeable for the high molybdenum-bearing CASS grades, but can be detected even for the low molybdenum-bearing grades. In the very worst cases, the change is noticeable at temperatures lower than about 150°F, in the range of BWR leak testing temperatures. However, the effect of this change in the transition to lower shelf behavior is not evident in elastic-plastic fracture toughness measurements, as shown by fracture toughness J-R curves.

In order to evaluate the effects of this minimal change in the transition to lower shelf behavior and the even less significant reduction in fracture toughness, the industry review considered the differences in service loads during normal operation and the loads during leak testing. The review found that the reduction in service loads during leak testing will more than compensate for any relatively minimal reduction in fracture toughness at leak test temperatures.

Evaluation of CASS Internals at Leak Test Temperatures

Reference 3 illustrates the effect of concern, with examples shown in Figures 2-1 and 2-2 of Reference 3 (for CASS materials with the designation CF-3 and CF-8, respectively) and Figure 2-3 of Reference 3 (for CASS materials with the designation CF-8M); these figures are reproduced below as Figures 1, 2, and 3. Note that the characteristics of the impact energy temperature dependence are very similar to the brittle-to-ductile temperature transition behavior exhibited by ferritic steels, albeit with a lower shelf that only becomes apparent at test temperatures well below nuclear reactor primary pressure boundary operating temperatures. Such behavior is to be expected for the relatively high delta ferrite for the materials being tested, but may be slightly exaggerated for the much lower delta ferrite in typical BWR reactor internals CASS materials. Note also that, similar to the effects of irradiation embrittlement on ferritic steels, the upper shelf of the transition region is lowered as a function of the thermal embrittlement time, very noticeably for the CF-8M material in Figure 3 that already reflects reduced upper shelf characteristics as the result of its higher molybdenum content.

It should also be noted that all three of the heats whose impact data is shown in the figures have very high delta ferrite content, which provides the primary explanation for the transition

temperature behavior shown. CASS materials with lower delta ferrite content display much less pronounced transition temperature behavior because of the reduced delta ferrite content. This can be illustrated by examining Figure 6 from Reference 4, reproduced below as Figure 4, which compares the unaged and thermally-aged impact energy results for a stainless steel weldment with a much lower delta ferrite content – in the range of 5 to 10%. Note that, while the impact energy throughout the temperature transition is reduced, the upper shelf of that transition region extends down to room temperature and somewhat lower. The upper shelf in Figures 1 and 2 for high delta ferrite CF-3 and CF-8 extend almost to room temperature, but do exhibit some slight reduction below about 100°F, while the slight reduction for the molybdenum-bearing grade CF-8M in Figure 3 in the aged condition is noticeable at about 200°F. It is this slight reduction in impact energy that could be of concern relative to fracture toughness at leak test temperatures.

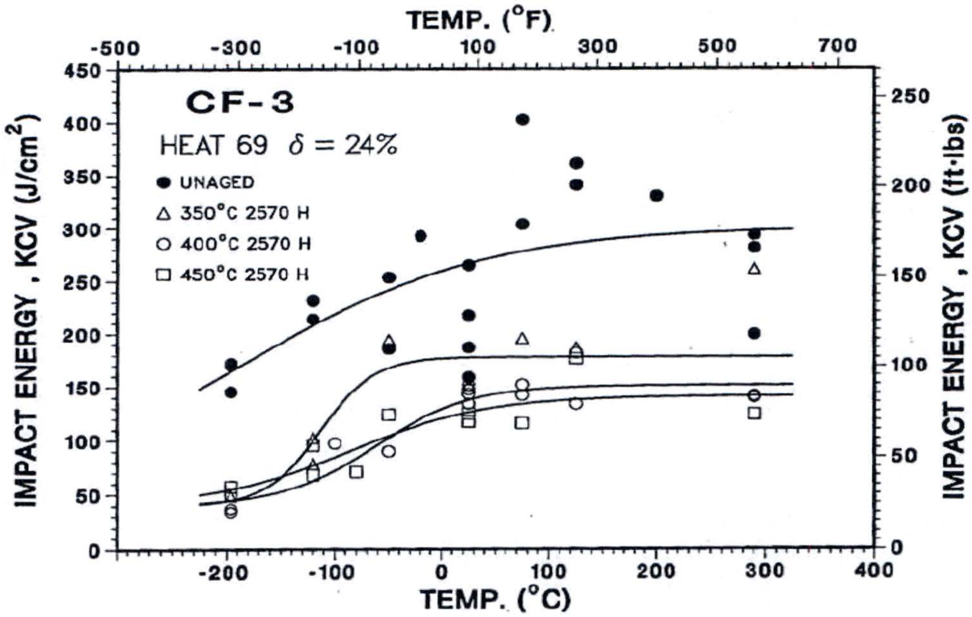

Figure 1. Impact Energy as a Function of Temperature for Heat 69 of CF-3 Material for Both Unaged and Various Thermal Aging Conditions

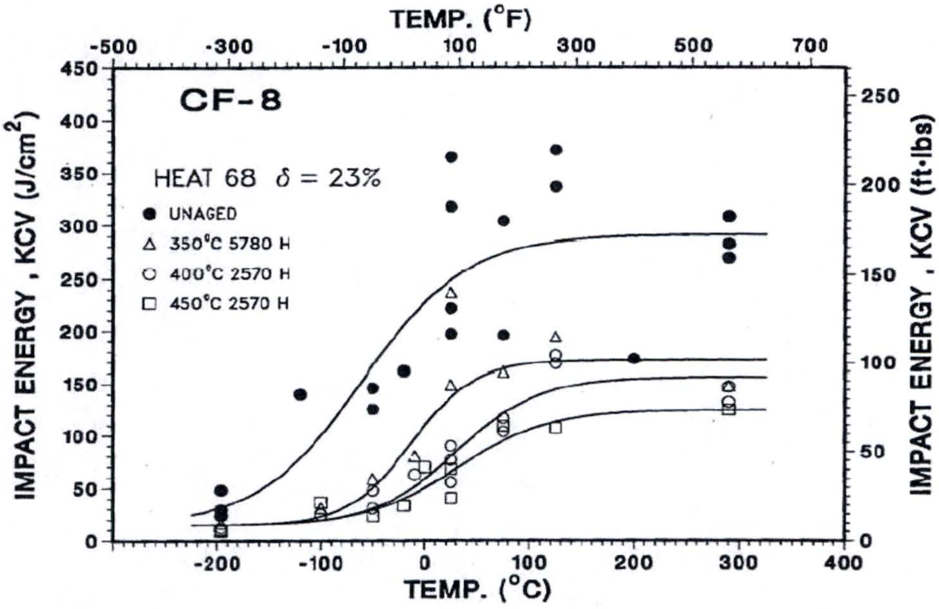

Figure 2. Impact Energy as a Function of Temperature for Heat 68 of CF-8 Material for Both Unaged and Various Thermal Aging Conditions

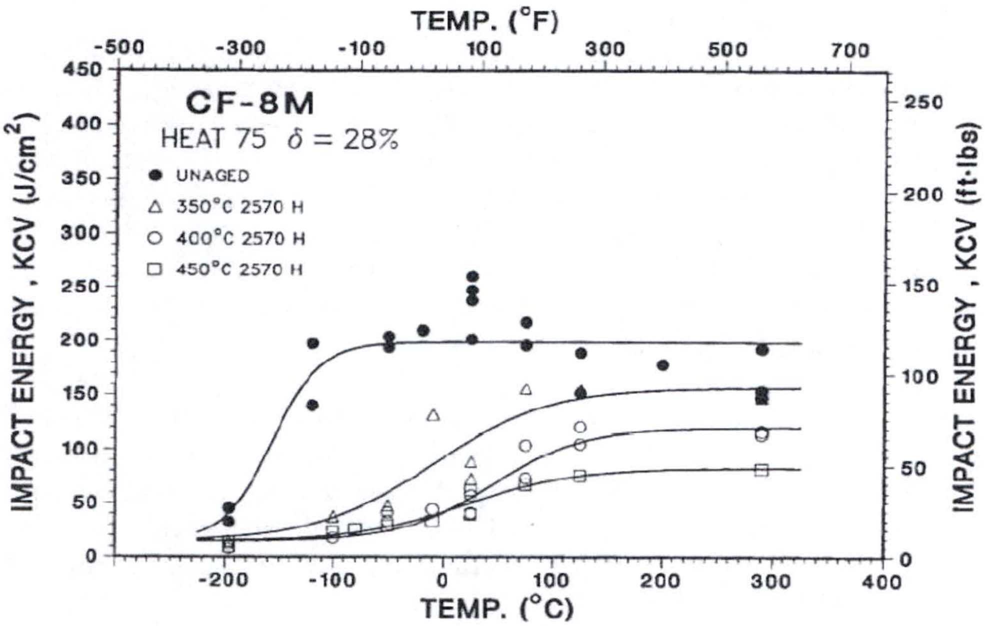

Figure 3. Impact Energy as a Function of Temperature for Heat 75 of CF-8M Material for Both Unaged and Various Thermal Aging Conditions

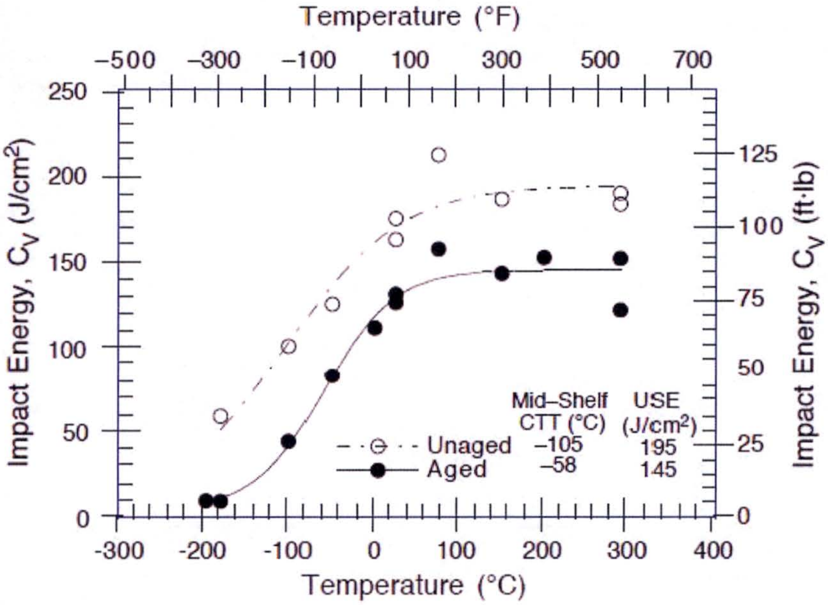

Figure 4. Impact Energy as a Function of Temperature for Stainless Steel Weldment in Both Unaged and Aged Conditions

<u>**Fracture Toughness Data**</u>

While the reduction in impact energy for high delta ferrite grades of CASS, in particular for the molybdenum-bearing grades, has been measured and is well known, similar effects on the actual fracture toughness of CASS materials are less well defined. Examining the effects first for stainless steel weldments, Figures 5 and 6 (Figures 13a and 13b from Reference 4) show that, while the fracture toughness in the aged condition at room temperature is below the unaged fracture toughness, and the fracture toughness at 290°C is essentially the same in both unaged and aged conditions, the fracture toughness in the aged condition at room temperature is virtually the same as that for the aged condition at 290°C.

The situation is roughly the same for the high delta ferrite CASS grades CF-3, CF-8, and CF-8M, as shown in Figure 7 (Figure 13 from Reference 4). In all three cases, the fracture toughness J-R curves at room temperature in the aged condition are significantly lower than the fracture toughness J-R curves in the unaged condition; however, fracture toughness J-R curves in the aged condition at room temperature are not significantly different (or lower) than the fracture toughness J-R curves in the aged condition at 290°C.

From this quick review, we conclude that, while the impact energy measurements as a function of temperature indicate that transition to lower shelf behavior for aged CASS materials is noticeable at temperatures lower than about 150°F, in particular for the higher molybdenum-bearing grades, the effect of this transition to lower shelf behavior is not evident in elastic-plastic fracture toughness measurements, as shown by fracture toughness J-R curves. Any such reduction in fracture toughness is considered minimal and likely non-existent for BWR leak test conditions. Even with a minimal reduction in fracture toughness, the reduction in service loads (i.e., reduced recirculation pump flows, thermal loading due to heatup, etc.) during leak testing will more than compensate for any relatively small reduction in fracture toughness at leak test temperatures.

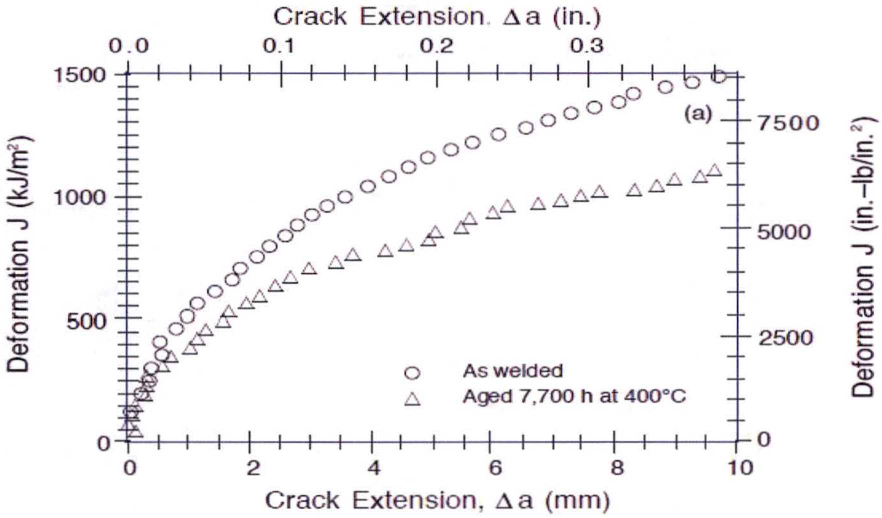

Figure 5. Fracture Toughness J-R Curve at Room Temperature for Stainless Steel Weldment in Both Unaged and Aged Conditions

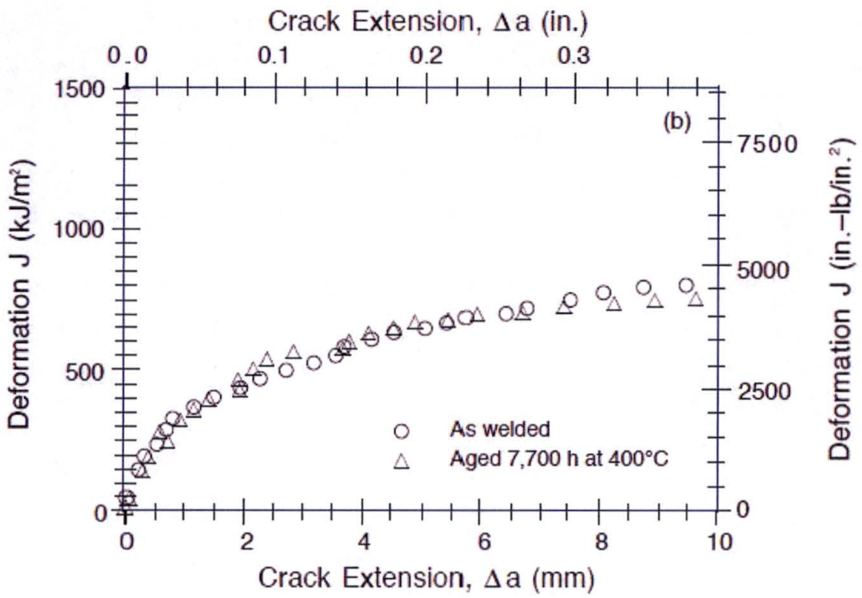

Figure 6. Fracture Toughness J-R Curve at 290°C for Stainless Steel Weldment in Both Unaged and Aged Conditions

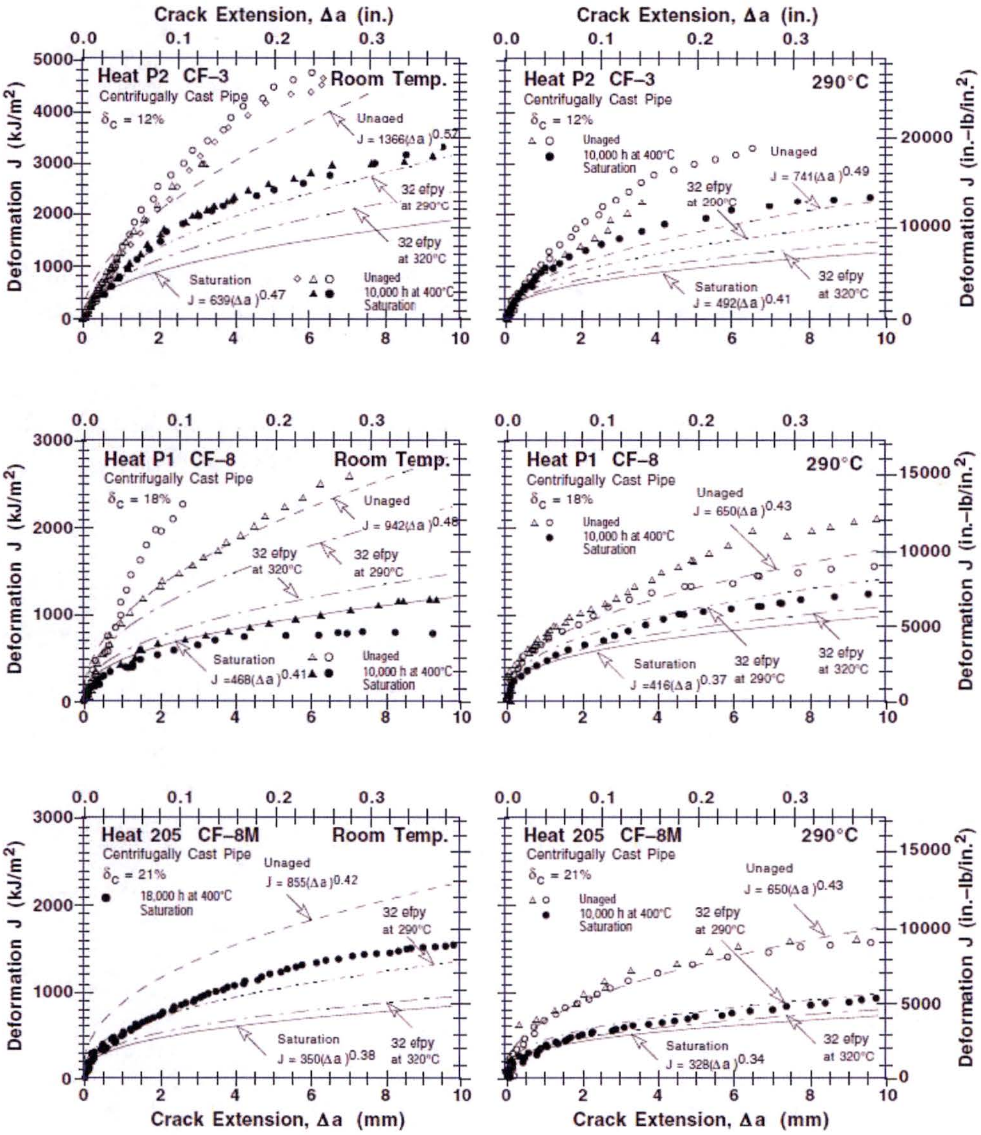

Figure 7. Fracture Toughness J-R Curves at Room Temperature and 290°C for Grades CF-3, CF-8, and CF-8M in Both Unaged and Aged Conditions

References

1. BWRVIP-234: BWR Vessel and Internals Project. "Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steels for BWR Internals," EPRI, Palo Alto, California, December 2009. 1019060.
2. Letter, Joseph J. Holonich, Senior Project Manager, Licensing Processes Branch, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, to Dennis Madison, Chairman, BWR Vessel and Internals Project, Second Request For Additional Information on BWRVIP-234, "Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel For BWR Internals," (TAC No. ME5060), April 24, 2013.
3. D. J. Garenda, W. F. Michaud, T. M. Galvin, W. F. Burke, and O. K. Chopra, "Effects of Thermal Aging on Fracture Toughness and Impact-Energy Strength of Stainless Steel Pipe Welds," NUREG/CR-6428 (ANL-95/47), Argonne National Laboratory, Argonne, Illinois (May 1996).
4. O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR-4513 (ANL-93/22), Revision 1, Argonne National Laboratory, Argonne, Illinois (August 1994).

<u>RAI-11</u>

The NDE methods of the existing examinations, discussed in Section 6, "Assessment of CASS Components," were justified because they were capable of detecting degradation other than loss of fracture toughness due to the combined effects of thermal and neutron embrittlement. Given the lower-bound toughness for a CASS component, which could have significantly higher delta ferrite than core shroud welds and is subject to both thermal and neutron embrittlement, justify the adequacy of the existing BWRVIP inspections to detect subcritical flaws as discussed in the GALL report, Rev. 2, XI.M9 paragraph 4 -- *Detection of Aging Effects*.

<u>BWRVIP Response to RAI-11</u>

While some BWR internal components might exhibit fracture toughness that approaches the lower bound value, this fact does not impede nor compromise the ability of the existing examinations to detect cracking before it reaches a critical flaw size. The BWRVIP has been utilizing, demonstrating, and constantly improving both visual and ultrasonic techniques to detect flaws well before component failure is predicted. The inspection frequencies proposed by the BWRVIP for detection of flaws have been accepted by the NRC. These techniques and procedures have been reviewed and approved by the NRC [1, 2, 3, 4]. Furthermore, it is noteworthy that the NRC has recognized the value of and endorsed the BWRVIP inspection methodologies in NUREG-1800, Rev. 2, XI.M9.

It is the BWRVIP's position that nothing has changed that would alter NRC's stated position in the GALL or the referenced Safety Evaluations. The existing inspection regime for BWR internals is such that flaws can be detected prior to any of them reaching a critical size. Thus, the aging effect of reduced toughness is still managed indirectly by inspections.

It is also worth noting that: (1) wrought material is more likely to crack than CASS and inspecting the wrought materials serves as a leading indicator, and (2) inspection of wrought to CASS weld joints would also reveal any flaw in the CASS portion of the weld joint [5]. The applicable and NRC-approved BWRVIP guidelines recommend more stringent inspections, such as EVT-1 examinations or ultrasonic methods of volumetric inspection, for certain selected components and locations. The nondestructive examination (NDE) techniques appropriate for inspection of BWR vessel internals, including the uncertainties inherent in delivering and executing NDE techniques in a BWR, are described in BWRVIP-03.

Even when comparing applied crack-driving forces to lower bound fracture toughness, there are no BWR internals that are discussed in Section 6 of BWRVIP-234 that will fail due to an undetectable flaw and thus challenge the integrity of the component. Cracks would have to be visible before they are large enough to challenge integrity. Consequently, the aging effects due

to thermal and irradiation embrittlement are considered by the BWRVIP to be adequately managed.

In conclusion, it is the BWRVIP's position that the NRC has accepted the aging management approach for BWR internals and no new experience has been observed to date that challenges that position. Based on the above discussions, the BWRVIP believes that the NRC accepts the inspection methodologies for BWR components as capable of performing their intended function of detecting flaws in the reactor vessel internals and detecting cracks of concern associated with non-pressure boundary RPV internals.

References

1. BWRVIP Letter 95-313A, "Evaluation of BWR Core Shroud and Inspection Evaluation Guidelines, GENE-523-113-0894, Revision 1, March 1995, and BWRVIP Core Shroud NDE Uncertainty & Procedure Standard, November 22, 1994," June 16, 1995.
2. BWRVIP Letter 98-259, "Safety Evaluation of BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03)" (TAC NO. M95369), June 8, 1998.
3. BWRVIP Letter 99-218A, "Final Safety Evaluation of BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03) Revision 1," (TAC NO. M95369), July 15, 1999.
4. BWRVIP Letter 2010-186, "NRC Safety Evaluation of BWRVIP-03, Revision 6 (RPV and Internals Examination Guidelines)," July 11, 2008.
5. BWRVIP Letter 2001-062, "Final Safety Evaluation of BWR Vessel and Internals Project, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41)," (TAC NO. M99870), February 22, 2001.

<u>Supplemental Information Provided by the BWRVIP and MRP</u>

In June 2013, the BWRVIP and the PWR Materials Reliability Program (MRP) formed an industry working group with the objective of developing a common screening methodology and hierarchy for thermal and irradiation embrittlement of CASS components. These issues arose from NRC's review of BWRVIP-234 and MRP-227-A, as well as from numerous meetings with the staff in 2012 and 2013. The following is hereby submitted by the BWRVIP to the NRC staff on behalf of industry for review. The following discussion represents a consensus approach for BWRs and PWRs to address the screening of thermal embrittlement (TE) and irradiation embrittlement (IE). Following the staff's approval, numerous revisions to the BWRVIP-234 report will be required to incorporate this approach. However, the conclusions currently stated in BWRVIP-234 will remain unchanged; that is, no components fabricated of CASS require inspection through 60 years of operation.

Hierarchy of TE and IE Screening Criteria for CASS Used in the Reactor Vessel Internals of PWRs and BWRs

Section 1: Hierarchy of TE and IE Screening

Status/Issues:
Recommendations for assessing the potential for in-service embrittlement of cast austenitic stainless steels (CASS) were initially issued by the NRC. More recently the BWRVIP [1] and the MRP [2, 3] have proposed recommendations for the treatment of CASS. Of these different approaches, the NRC guidance in the communication often referred to as the Grimes letter [4] is by far the most restrictive both in approach and in the neutron irradiation screening threshold.

Industry (BWRVIP and MRP) approaches to aging management of reactor vessel internals, based upon industry sponsored research and testing as well as regulatory sponsored programs have considered thermal embrittlement (TE) and irradiation embrittlement (IE) of CASS as essentially separate mechanisms. As such, these mechanisms are controlled and limited by the material microstructure and its potential susceptibility to embrittlement within the separate phases. Regulatory guidance, provided in the Grimes letter has considered that irradiation processes act in concert with thermal embrittlement to result in a loss of toughness at fluences well below those that would be considered damaging to wrought austenitic stainless steels. The regulatory guidance directed the licensee to recognize and address that both thermal and irradiation effects would lead to a synergistic loss of toughness resulting not only in acceleration of loss of toughness but also in exacerbation of loss of toughness for reactor vessel internals components, i.e., the loss of toughness in the presence of both TE and IE would be greater than either effect alone, even at "saturation" of the individual effects.

On this basis, the Grimes letter concluded that even at fluence levels as low as 1×10^{17} n/cm^2 (E > 1MeV)[3], synergistic effects could occur. The 1×10^{17} n/cm^2 threshold appears to have been identified by analogy with low-alloy steels, i.e., based on the lowest fluence values that had been observed to develop the onset of embrittlement in RPV steels and their weldments. Essentially, the Grimes letter proposes that fluences at which embrittlement of the ferrite phase may occur can synergistically enhance thermal embrittlement in CASS such that all CASS structures are potentially susceptible to embrittlement, i.e., not just those that would be identified by virtue of their material chemical composition and microstructure as being potentially susceptible to thermal embrittlement. Using this approach, the Grimes letter

[3] All quoted fluence data in n/cm^2 will be for E > 1MeV. For brevity, the qualifier regarding neutron energy will be omitted in further references to neutron fluence levels. For comparison with other characterizations of irradiation exposures, 1dpa = 6.7×10^{20} n/cm^2 for neutrons of the specified energies.

dictates that the hierarchy of screening for TE and IE is that IE screening must be performed before TE screening.

The approach of the Grimes letter leads to issues regarding both its fundamental assumptions and its practical implementation. Firstly, the Grimes letter greatly expands the types and compositions of CASS that must be considered susceptible to thermal embrittlement in the presence of irradiation. It is well known that while ferritic steels frequently embrittle during thermal exposures, austenitic steels do not. Thus, in CASS only the ferrite phase undergoes thermal embrittlement. This phase will embrittle at relatively low fluence, i.e. around 1×10^{17} n/cm^2. The addition of this mechanism over and above thermal embrittlement may be expected to accelerate the onset of embrittlement of the overall composite structure. It is not, however, reasonable to consider that the additional mechanism would markedly increase the embrittlement of the CASS beyond that of the same material in the fully thermally embrittled condition, i.e. with fully embrittled ferrite. Conversely, for other CASS compositions that do not undergo thermal embrittlement, by virtue of their low ferrite content, embrittlement in response to irradiation exposure requires much greater neutron fluences such as are needed to produce embrittlement in the austentitic phase of the steels. Such fluences are similar to those that are necessary to embrittle fully austenitic (i.e. wrought) stainless steels.

More practically, the requirement of the Grimes letter regarding the hierarchy of screening, i.e., irradiation before thermal – requires extensive fluence calculation to be performed to model very large sections of the reactor internals. This is especially true if the fluence criterion is set at the low level of 1×10^{17} n/cm^2 that was recommended in the Grimes letter. Irradiation screening is a much more involved process than thermal screening. Irradiation fields are time and location varying, and extensive efforts are required to identify lifetime fluences for all relevant locations for CASS materials used in RV internals. Conversely, at least within the reactor vessels, thermal fields are essentially constant and well known. Because the thermal exposures are so extensive and uniform, complete components can be readily "screened in" for sufficient thermal exposures to induce the potential for embrittlement using relatively simple criteria. Subsequently, screening for potential susceptibility to irradiation embrittlement would only need to be performed on components that have not already screened in for thermal embrittlement.

Proposed Resolution:
The proposed revised screening hierarchy, replacing that directed in the Grimes letter, would consider first screening for thermal embrittlement and then subsequently screening for irradiation embrittlement. This process would operate on a "screening in" basis. It would identify those components that screen in for thermal effects and then, for the regions that had not screened in already, re-screen for irradiation effects.

The first phase of the screening would identify those components that would screen in based on potential susceptibility to thermal embrittlement by considering all of the identified CASS components. The thermal screening would employ the same criteria as those identified in the Grimes letter, i.e., chemistries and processing parameters that have already been associated with the potential for susceptibility to thermal embrittlement.

The second phase of the screening would be applied to those components that had not been screened in for potential susceptibility to thermal embrittlement. It would identify the components and regions of components that would have been exposed to lifetime fluences sufficient to induce potential susceptibility to irradiation embrittlement. This screening would identify regions of components that would be exposed to irradiation fluences above a critical criterion value. The definition of the most appropriate value for this fluence level is discussed in Section 2; however, it is noted here that this criterion is related to the fluence required to produce irradiation embrittlement in the austenite phase of the CASS. This value is much higher than the fluence value of 1×10^{17} n/cm^2 assigned by the Grimes letter. On this basis the irradiation screening process called for by the revised hierarchy is expected to be much less extensive and less complex than that required by the Grimes letter's guidance.

Those components that did not screen in under either criterion, thermal then irradiation, would not be susceptible to thermal and irradiation embrittlement.

Rationale for Proposed Resolution:
The rationale for the selected approach is that the thermal fields are so extensive and effectively uniform in LWR reactor internals that the potential for thermal embrittlement of CASS components is much easier to assess than the potential for irradiation embrittlement. Moreover, this screening is applied to entire components, whereas irradiation screening calls for the ability to discriminate between different sections of components. To assess the potential for thermal embrittlement, comparisons of the CASS alloy composition, processing practices, and expected delta-ferrite composition would be made with the criteria as published in the Grimes letter and reproduced in Table 1.

Of additional importance for the proposed revision of the screening hierarchy is that it is simpler and easier to apply than the original guidance of the Grimes letter and that it is also based in an understanding of the mechanisms and causes of irradiation and thermal embrittlement in the separate phases that constitute the duplex CASS structures.

Table 1. Thermal Embrittlement Screening Criteria For CASS Material

Molybdenum (Wt. %)	Casting Method	Delta-Ferrite (%)	Susceptibility Determination
High 2.0-3.0	Static	>14%	Potentially Susceptible to TE
		≤14%	Not Susceptible to TE
	Centrifugal	>20%	Potentially Susceptible to TE
		≤20%	Not Susceptible to TE
Low 0.5 max	Static	>20%	Potentially Susceptible to TE
		≤20%	Not Susceptible to TE
	Centrifugal	All	Not Susceptible to TE

CASS alloy compositions and processing conditions utilized during component manufacturing are generally well known, so assessments of the potential for susceptibility to thermal embrittlement can be readily made by comparison with the criteria given in the Grimes letter. (In the case that actual alloy compositions and other such details are not available, conservative assumptions regarding potential alloy compositions and processing would be made.) In the event that components screen in for potential thermal embrittlement there is then no need to perform the more complex assessments needed for potential irradiation embrittlement. Inspection or evaluation plans would be required for those components that screen in for thermal embrittlement.

The NRC guidance for screening CASS components for thermal and irradiation embrittlement was put forward in the Grimes letter. The primary focus of the Grimes letter was thermal aging, and title of the guidance letter was "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components" [4]. This document invoked the work at Argonne National Laboratory [referenced here as 5, 6, and 7] and considered that the embrittling effect of the ferrite under thermal conditions was "to be severe enough to make the materials susceptible to fracture if the ferrite forms a continuous phase surrounding the grain boundaries in the microstructure."

As pointed out by Scott in the "Expert Panel Report on Proactive Materials Degradation Assessment," NUREG/CR-6923 [8], CASS structures may be considered to fall into two categories, one in which fracture is controlled by the ferrite phase and the other in which fracture is controlled by the austenite phase. Reference 8 points out that it is not necessary for the ferrite phase to provide a continuous brittle fracture path for ferrite embrittlement to be the governing process in CASS fracture. Even though the appearance of the fracture surfaces of embrittled ferrite CASS may reflect ductile rupture processes, the governing process can be

brittle fracture of the ferrite particles if it subsequently facilitates the initiation of voids within the austenite that then promote the ductile rupture process. Conversely, embrittlement of CASS by embrittlement of the austenite can only occur by irradiation effects, which would result in channel fracture rather than ductile rupture. The controlling parameter that differentiates between the two types of structures is the amount and distribution of the ferrite phase. High-volume fractions of ferrite (>20%), distributed in contiguous particles, cause the CASS fracture properties to be ferrite controlled. Low-volume fractions of ferrite and separation of the ferrite particles cause the properties of the CASS to be austenite controlled. Table 2 identifies the manifestations of the differences in behaviors between the CASS structures that are ferrite controlled and those that are austenite controlled.

Table 2. Effect of Controlling Phase on Embrittlement Behaviors of Duplex CASS Structures

Controlling Phase	Thermal Embrittlement Susceptibility	Irradiation Embrittlement Susceptibility
Ferrite	Susceptible to Thermal Embrittlement	Susceptible to IE (severe) if Fluence > Φ_{crit}^{α}
Austenite	Not Susceptible to Thermal Embrittlement	Susceptible to IE (moderate) if Fluence > Φ_{crit}^{γ}

Where Φ_{crit}^{α} is the critical fluence required to embrittle ferrite and Φ_{crit}^{γ} is the critical fluence required to embrittle austenite.

The NRC guidance noted that the degree of embrittlement strongly depended on the amount and distribution of the ferrite and the chemical composition of the steel, particularly its molybdenum content and also its manner of casting (i.e., static vs. centrifugal). Under thermal conditions the ferrite phase can undergo the microstructural phase changes that lead to hardening and embrittlement. The Grimes letter identified the types of structures (dependent on the component route) and the amount of the ferrite phase in the CASS microstructure that would be required to induce susceptibility to thermal embrittlement. These criteria were given in Table 1, and the ferrite content can be calculated from the alloy composition using Hull's equivalent factors [9]. As a practical matter, it is therefore straightforward to determine which CASS structures and components should be screened in for susceptibility to thermal embrittlement.

This approach is considered to be adequate for the first stage of the screenings since it identifies those CASS structures in which the fracture is controlled by the ferrite and it conservatively assumes that the ferrite has been fully embrittled during service. It inherently allows for acceleration of thermal embrittlement by irradiation processes acting synergistically within the ferrite phase. It assumes the saturation hardening of the ferrite is produced by the

simultaneous irradiation and thermal processes. Since the end-point embrittlement is already considered to have induced the maximum effects of loss of toughness, this process does not require the consideration of whether simultaneous thermal and irradiation effects produce even greater effects of loss of toughness (i.e., exacerbation of the effects). It should be noted that the thermal screening process employs exactly the same criteria as those recommended by the Grimes letter. The thermal screening phase of the process is given by Table 3.

Table 3. Strategy for CASS Embrittlement Screening Using Screening for Thermal Embrittlement as the First Stage of Screening

Molybdenum (Wt. %)	Casting Method	Delta-Ferrite (%)	Thermal Embrittlement Susceptibility	Irradiation Embrittlement Susceptibility
High 2.0-3.0	Static	>14%	Potentially Susceptible to TE	*No need for IE screening*
		≤14%	*Not Susceptible to TE*	Need for IE Screening
	Centrifugal	> 20%	Potentially Susceptible to TE	*No need for IE screening*
		≤20%	*Not Susceptible to TE*	Need for IE Screening
Low 0.5 max	Static	>20%	Potentially Susceptible to TE	*No need for IE screening*
		≤20%	*Not Susceptible to TE*	Need for IE Screening
	Centrifugal	All	*Not Susceptible to TE*	Need for IE Screening

The second phase of the revised hierarchy addresses the irradiation screening for those CASS compositions that have been assessed and found to be not susceptible to thermal embrittlement, resulting in the process outlined in Table 4.

Table 4. Strategy for Screening of CASS for Potential Thermal and Irradiation Embrittlement Using the Proposed Revised Screening Hierarchy

Molybdenum (Wt.%)	Casting Method	Delta-Ferrite %	Thermal Embrittlement Susceptibility	Irradiation Embrittlement Susceptibility
High 2.0-3.0	Static	>14%	Potentially Susceptible to TE Conduct inspection or component specific evaluation	*No need for IE screening*
		≤14%	Not Susceptible to TE	Susceptible to IE if Fluence > Φ_{crit}^{γ}
	Centrifugal	>20%	Potentially Susceptible to TE Conduct inspection or component specific evaluation	*No need for IE screening*
		≤20%	Not Susceptible to TE	Susceptible to IE if Fluence > Φ_{crit}^{γ}
Low 0.5 max	Static	>20%	Potentially Susceptible to TE Conduct inspection or component specific evaluation	*No need for IE screening*
		≤20%	Not Susceptible to TE	Susceptible to IE if Fluence > Φ_{crit}^{γ}
	Centrifugal	All	Not Susceptible to TE	Susceptible to IE if Fluence > Φ_{crit}^{γ}

Since the compositions and structures that will undergo embrittlement at low fluence have already been identified and screened in for thermal embrittlement, the second stage of the screening needs only to consider screening for irradiation that may embrittle the austenite phase of the CASS structures. Screening for irradiation embrittlement should then be performed using a fluence level that is consistent with embrittlement of the austenite phase. It is well known that austenitic steels do not embrittle at the low fluences that are sufficient to embrittle ferritic steels [10, 11, 12, 13]. The appropriate fluence for screening for embrittlement of austenite is, therefore, recognized to be much higher than the value specified

in the Grimes letter. The value of this revised IE screening criterion is discussed in Section 2. However, because this screening fluence would be significantly greater than the 1×10^{17} n/cm^2 of the Grimes letter, it would remove many components and regions of the reactor internals from the consideration for IE.

<u>Section 2: Screening Criteria for IE and TE of CASS</u>

Status/Issues:

The NRC guidance document for consideration of the irradiation embrittlement (IE) of CASS, the Grimes letter, calls for all CASS components that are exposed to neutron fluences greater than 1×10^{17} n/cm^2 to be considered potentially susceptible to IE. The NRC guidance letter sets the screening level at 1×10^{17} n/cm^2 to account for potential synergistic interactions between thermal and irradiation embrittlement. In contrast, industry investigations by the BWRVIP and the MRP have identified much higher levels of fluence, specifically 3×10^{20} n/cm^2 and 6.7×10^{20} n/cm^2, for screening of CASS compositions for potential susceptibility to irradiation embrittlement. Moreover, the guidance of the expert panel on proactive materials degradation assessment [8] also aligns with this approach. The lower fluence screening level identified by the Grimes letter requires much more extensive calculation of the radiation fields and accumulated fluence than do the criteria identified by the industry groups.

The key difference between the Grimes letter and the industry approaches is that the Grimes letter considered that irradiation induced a synergistic effect in materials that were already considered to be susceptible to thermal embrittlement, irrespective of materials structure, constitution, or chemical composition. The approach taken in the Grimes letter was also to initially screen all CASS for potential effects of irradiation embrittlement. On that basis, it would be appropriate to set a screening level that would account for IE of the ferrite that would occur at low fluence.

Proposed Approach:
The proposed approach for irradiation screening of CASS components in reactor internals is to employ the 6.7 x 10^{20} n/cm^2 (1 dpa) screening value proposed by the MRP-175 document [2] coupled with the screening hierarchy proposed in Section 1. Under this approach the potential for TE is considered prior to the potential for IE. In screening for TE first, the process is identifying and "screening in" structures whose fracture behaviors are controlled by the ferrite phase. Thus, this approach effectively also automatically screens in CASS structures that would be susceptible to irradiation embrittlement at low fluence, i.e., those structures that experience a loss of fracture toughness by the IE of the ferrite.

By identifying the compositions that are susceptible to TE and IE at low fluence in the first stage of screening, the second stage of screening needs only to identify those components (from those that have not yet screened in) that can be susceptible to IE at the higher fluence level that will induce IE of the austenite phase. In these components, essentially, the austenite phase in the CASS structures is of a sufficiently high-volume fraction and sufficiently extensive as to contain and restrict the fracture influence of any potentially embrittled ferrite phase. In this case embrittlement due to irradiation can only occur at fluences that are sufficient to produce embrittlement in the austenite phase itself. The fluence required to effect this embrittlement is known to be in the 6.7 x 10^{20} to 6.7 x 10^{21} n/cm^2 regime (i.e., the 1 to 10 dpa range) [2, 3]. Using this selected screening criterion for IE and the proposed revised screening hierarchy, the overall screening process becomes that given by Table 5.

Table 5. Proposed Revised Hierarchical Process for Assessment of Potential Thermal and Irradiation Embrittlement of CASS

Moly (Wt.%)	Casting Method	Delta-Ferrite %	Primary Screening Thermal Embrittlement Susceptibility	Secondary Screening Irradiation Embrittlement Susceptibility
High 2.0-3.0	Static	>14%	Potentially Susceptible to TE – Conduct inspection or component specific evaluation	*No need for IE screening*
		≤14%	Not Susceptible to TE	Fluence > 6.7 x 10^{20} n/cm^2 (1dpa) Potentially susceptible to IE – Conduct inspection or component specific evaluation Fluence < 6.7 x 10^{20} n/cm^2 (1dpa) *Not potentially susceptible to IE – screen out, no additional measures*
	Centrifugal	>20%	Potentially Susceptible to TE – Conduct inspection or component specific evaluation	*No need for IE screening*
		≤20%	Not Susceptible to TE	Fluence > 6.7 x 10^{20} n/cm^2 (1dpa) Potentially susceptible to IE – Conduct inspection or component specific evaluation Fluence < 6.7 x 10^{20} n/cm^2 (1dpa) *Not potentially susceptible to IE – screen out, no additional measures*
Low 0.5 Max	Static	>20%	Potentially Susceptible to TE – Conduct inspection or component specific evaluation	*No need for IE screening*
		≤20%	Not Susceptible to TE	Fluence > 6.7 x 10^{20} n/cm^2 (1dpa) Potentially susceptible to IE – Conduct inspection or component specific evaluation Fluence < 6.7 x 10^{20} n/cm^2 (1dpa) *Not potentially susceptible to IE – screen out, no additional measures*
	Centrifugal	All	Not Susceptible to TE	Fluence > 6.7 x 10^{20} n/cm^2 (1dpa) Potentially susceptible to IE – Conduct inspection or component specific evaluation Fluence < 6.7 x 10^{20} n/cm^2 (1dpa) *Not potentially susceptible to IE – screen out, no additional measures*

Rationale for Proposed Resolution:
The rationale for the proposed screening criterion for IE of CASS compositions that are not susceptible to TE is that in these structures IE can only occur by embrittlement of the austenite phase. IE of the ferrite phase is only effective in inducing potential susceptibility to embrittlement for the whole structure if fractures in such locally embrittled regions can link up and control cracking of the overall structure. In the proposed revised screening hierarchy these structures have already been screened in for thermal embrittlement, and there is no need to consider the conditions required to produce IE in them. The fluence required to embrittle austenite is, however, significantly greater than that required to embrittle ferrite. This difference derives from the differences in crystal structure, mechanisms of (dislocation slip driven) deformation, and the potential for second phase hardening in the two materials.

IE of austenite occurs at much higher fluence than IE of ferrite because of the different mechanisms of embrittlement in two materials of different crystal structure. IE of ferrite occurs because of hardening due to spinodal decomposition and G-phase precipitation, which in the body-centered cubic (bcc) structure of the ferrite induces restricted cross-slip and embrittlement [10]. IE of austenite in stainless steels does not occur as a result of phase effects, since decomposition and precipitation hardening do not occur in austenitic stainless steels. IE of fully austenitic stainless steel occurs because of hardening by the production of lattice defects, which restricts homogeneous slip deformation and promotes localized shear, concentrated in narrow bands – so-called "channel slip" – which in turn produces a reduced ductility, embrittled mode of fracture [10, 13]. The lattice damage required to restrict homogenous slip in the austenitic structure requires very high fluence to provide for the accumulation of the required production and accumulation of radiation-induced lattice defects in the face-centered cubic (fcc) structure.

Investigations of irradiation damage and embrittlement of wrought stainless steels, welds, and cast austenitic stainless steels have shown that such structures can withstand irradiation exposure well into the 6.7×10^{20} to 6.7×10^{21} n/cm^2 regime (1.0 to 10 dpa) before the onset of significant embrittlement can be discerned. Table 6 lists the most relevant observations and proposed criteria for screening for the onset of irradiation embrittlement of wrought and cast austenitic stainless steels.

Table 6. Observations and Estimations of the Neutron Irradiation Fluence Required to Produce IE in Wrought and (Non-Thermally Embrittling) Cast Austenitic Stainless Steels

Source	Wrought Stainless Steel	CASS
NUREG/CR-6960 Full Embrittlement [9]	2 to 6.7 x 10^{21} n/cm^2 (3-10 dpa)	~6.7 to 33 x 10^{20} n/cm^2 (1-5 dpa)
NUREG/CR-6960 Estimated Onset of Embrittlement [9]	3.35 x 10^{20} n/cm^2 (0.5 dpa)	2.0 x 10^{20} n/cm^2 (0.3 dpa)
MRP Recommended Screening Value [2, 3]	1 x 10^{21} n/cm^2 (1.5 dpa)	6.7 x 10^{20} n/cm^2 (1.0 dpa)
BWRVIP-234 Screening Value [1]	Not Applicable	3 x 10^{20} n/cm^2 (0.45 dpa)
Grimes Letter Thermal + Irradiation Synergistic Embrittlement [4]	Not Applicable	1 x 10^{17} n/cm^2 (1.5 x 10^{-4} dpa)

Figure 1 summarizes the fracture toughness data compiled by the Argonne National Laboratory investigations into the loss of toughness of austenitic steels resulting from neutron exposures in test and power generating reactors [10, 11, 14]. The authors of reference 11 noted that for initially very ductile wrought 316 stainless steels, the sharp drop in toughness occurred in the 2.0 x 10^{21} to 6.7 x 10^{21} n/cm^2 regime (3 dpa to 10 dpa regime). Generically they considered that wrought austenitic stainless steels showed little change or no change in toughness below 3.35 x 10^{20} n/cm^2 (0.5 dpa). The authors considered that less ductile materials displayed the onset of embrittlement somewhat earlier. Welds and CASS were considered to display adequate toughness, up to at least 2.0 x 10^{20} n/cm^2 (0.3 dpa). It should be noted, however, that these authors employed a somewhat elevated criterion for the onset of embrittlement since they considered that the materials irradiated to less than the threshold dose would have a fracture toughness J_{Ic} of at least 135 kJ/m^2. This value is actually quite high for materials that are not expected to perform as pressure boundary components.

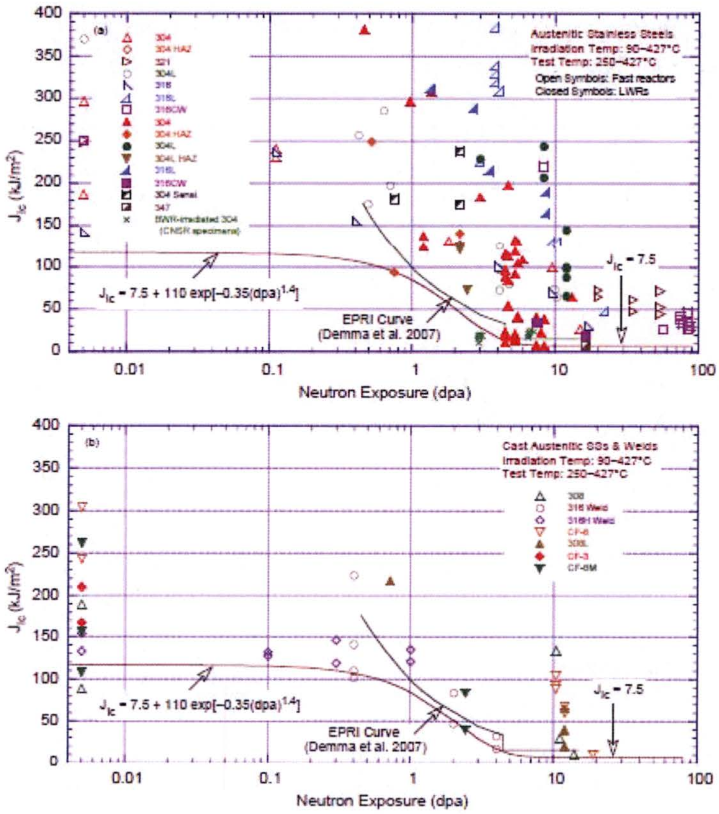

Figure 1. IE Fracture Toughness Data for Irradiated Wrought Stainless Steels, Stainless Steel Welds, and Cast Austenitic Stainless Steels (from Chopra et al. [10, 11, 14]

The MRP considered the data presented by the ANL studies in determining the proposed screening values for IE assessment of irradiation embrittlement of wrought and cast austenitic stainless steels. Because this program wished to identify the screening level where embrittlement effects would become sufficiently significant that plant aging management steps would be required, the program utilized the region of the data in Figure 1 where the toughness was observed to demonstrate the onset of a significant decrease in the loss of toughness. From the lower bound curve of Figure 1, the MRP identified the value of about 3.35 x 10^{21} n/cm^2 (5 dpa) as the regime where wrought austenitic stainless steels suffered significant onset of loss of toughness. In order to provide a margin of safety, the threshold value was set to approximately one-third of this fluence, at 1.0 x 10^{21} n/cm^2 (1.5 dpa). The expected minimum toughness can be calculated from the equation in the figure. For CASS, the MRP (in References 2 and 3) conservatively established that the screening fluence should be discounted

from the value established for the wrought materials by about one-third to account for the greater scatter observed in unirradiated CASS and weld materials. (Note that, in Figure 1, at the highest fluences, the toughnesses of the CASS and weld materials are similar to those of the wrought materials. However, at lower fluences, the CASS and weld materials tend to exhibit lower toughness. This is true even in the unirradiated state and reflects the well-known inferior ductility of cast materials versus their wrought counterparts.) It is considered that there is no need to include a further safety factor since the process of developing the screening fluence already included a factor of three in developing the screening value for the wrought materials.

According to the lower bound curve of Figure 1, even at 6.7 x 10^{20} n/cm^2 (1 dpa), irradiated austenitic materials including CASS and welds (that do not thermally embrittle) should be expected to display a minimum toughness of about 85 kJ/m^2. Not only is this toughness expected to be adequate to provide functionality in internals applications, but as can be seen from the data points in the plot, the CASS and weld data are tending to become grouped with the wrought data.

Materials from BWR reactors have been assessed to determine the effect of irradiation on welded austenitic stainless steels at BWR reactor fluences and temperatures. By extension, because CASS materials are comprised of the same mixtures of austenite and ferrite as welds, the observations and measurements of BWR irradiated austenitic welds provide a basis for prediction of the value of the fluence that would produce significant embrittlement of CASS in BWR conditions. BWRVIP-100-A [15] developed toughness curves for wrought and welded austenitic stainless steels as a function of the calculated irradiation fluence in BWR internals. Toughness measurements were obtained for fractures in weld heat-affected zones as well as base metal and weld metals. Predictions of the effect of irradiation on fracture toughness were then obtained using the equation:

$$J = \Delta a^{n}$$

Using curve fits to the upper bound values of the pre-exponential term C and lower bound values to the exponent n taken as a function of fluence, a conservative set of J vs. crack extension curves was developed. On the basis of these curves, adequate toughness was identified to be retained in BWR shroud materials to at least 3 x 10^{20} n/cm^2 (0.45dpa). The toughness predicted to be remaining in the materials irradiated to this level is, however, predicted to be quite high, e.g., J for Δa of 2.5mm is well above 250 kJ/m^2, which as noted earlier is much more than is required for service in internals. Moreover, the approach taken to develop the value involved conservative bounding curves for the prediction of the constants. Thus, this predicted toughness is expected to be a lower bound value.

from the value established for the wrought materials by about one-third to account for the greater scatter observed in unirradiated CASS and weld materials. (Note that, in Figure 1, at the highest fluences, the toughnesses of the CASS and weld materials are similar to those of the wrought materials. However, at lower fluences, the CASS and weld materials tend to exhibit lower toughness. This is true even in the unirradiated state and reflects the well-known inferior ductility of cast materials versus their wrought counterparts.) It is considered that there is no need to include a further safety factor since the process of developing the screening fluence already included a factor of three in developing the screening value for the wrought materials.

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The data presented by previous investigations therefore indicate that an appropriate screening fluence for the onset of significant embrittlement for irradiation effects should be above the 2.0×10^{20} n/cm^2 (0.3 dpa) and 3.0×10^{20} n/cm^2 (0.45 dpa) fluence levels identified in the ANL and BWRVIP documentation, since these are, respectively, the fluence below which no onset of loss of toughness was identified and a lower bound conservative estimate for BWR shroud material performance. At the fluences investigated, the BWRVIP data also indicate very little difference between weld behavior and base metal and HAZ behaviors. This similarity would indicate that the criteria for IE screening for weld/CASS materials should not be markedly different than that for wrought stainless steels. This is in agreement with the approach of the MRP, which proposed discounting the screening value of fluence for the wrought stainless steels to obtain the appropriate screening value for the CASS and weld materials.

As the data point scatter in Figure 1 indicates, conclusive data clearly delineating the onset of IE and supporting the selection of specific screening fluence levels for CASS and wrought materials are sparse. Moreover, the effect of the CASS structure on the interactions of IE and TE are not clear. In NUREG/CR-6960 [10], while it was recognized that significant effects on the austenite phase would not occur below 3.35×10^{20} n/cm^2 (0.5 dpa), the potential for synergy could not be ruled out. The effect of synergy was noted to be acceleration of thermal embrittlement for fluences at which the ferrite would be subject to thermal embrittlement, i.e., the 1×10^{17} n/cm^2 (0.1 mdpa) level, and above the level of 2.0×10^{20} n/cm^2 the recommendation for the toughness was proposed to be the lower of the effect of thermal exposure or the effect of irradiation fluence predicted by the lower bound curve for all stainless steels given in Figure 1. Since the MRP approach utilized the lower bound curve of Figure 1 to identify the 6.7×10^{20} n/cm^2 (1 dpa) level as the appropriate screening value, this value would appear to be a reasonable screening value for both accommodating potential synergistic interactions of IE and TE and also identifying the level at which the onset of loss of toughness can be considered to be significant.

More recently developed data support the acceptance of a fluence screening value significantly higher than that recommended in the Grimes letter. Both Chen et al. [16] and Kim et al. [17] have tested CASS materials that had been exposed to relatively high fluences. Both examinations determined fracture toughness J-R curves as a function of thermal and irradiation exposures. The data from these investigations are reproduced in Tables 7 and 8, respectively. Both sets of data support the conclusion that significant toughness remains after even very high irradiation exposures. The Chen et al. data indicate very high toughness at the highest irradiation exposures of 1.35×10^{19} n/cm^2 (0.02 dpa). The Kim et al. data display lower toughness values; however, even the lowest toughness values were still measured on materials that had been irradiated to very high fluences (6, 10, and 12 dpa). These toughnesses compare well with the data plotted in Figure 1, demonstrating that screening at 1 dpa would have identified material that still had considerable remaining toughness, i.e., at least of the order of 120 kJ/m^2. This toughness should be sufficient to withstand significant incipient cracking

under internals loading conditions and is significantly above that identified in the criteria for acceptance for internals in MRP-175, Appendix F, Section F.4. Thus, both sets of data should to be taken to support the argument that the screening level for IE should be significantly greater than the 1×10^{17} n/cm^2 threshold put forward in the Grimes letter.

The data in Tables 7 and 8 also highlight the possible interactions and potential synergy between IE and TE. Synergy can accelerate the effect of degradation due to one mode in the presence of another. Synergy can also exacerbate the effect of degradation due to one form in the presence of the other. The guidance of the Grimes letter and some interpretations of the documents from Argonne National Laboratory consider that synergy can both accelerate and exacerbate the loss of toughness during service irradiation and thermal exposure. In contrast, the approach inherent in the revised screening hierarchy and the elevated IE screening criterion, while recognizing the accelerating effect of synergy, does not account for potential exacerbation of effects based on the following arguments.

Table 7. Toughness Data for Thermally Aged and Irradiation Aged High Ferrite Number CASS from Chen et al. [16]

Grade	Heat	Ferrite %	Heat Treatment	Irradiation	J_q	$J=C(\Delta a)^n$		J at 2.5mm
					KJ/m^2	C	n	KJ/m^2
CF3	69	24	Unaged	0	700	756	0.31	1004
CF3	69	24	Fully Aged	0	167	296	0.51	472
CF3	69	24	Unaged	0.02 dpa	204	430	0.64	773
CF3	69	24	Fully Aged	0.02 dpa	116	362	0.85	789
CF-8	68	23	Unaged		753	783	0.27	1003
CF-8	68	23	Fully Aged		242	396	0.51	632
CF-8	68	23	Unaged	0.02 dpa	183	359	0.57	605
CF-8	68	23	Fully Aged	0.02 dpa	171	372	0.62	657
CF-8M	75	28	Unaged		437	583	0.45	881
CF-8M	75	28	Fully Aged		156	274	0.46	418
CF-8M	75	28	Unaged	0.02 dpa	145	336	0.66	615
CF-8M	75	28	Fully Aged	0.02 dpa	106	259	0.64	466

Table 8. Toughness Data for Thermally Aged and Irradiation Aged Low Ferrite Number CASS from Kim et al. [17]

Material	FN	Thermal	Thermal Categorization	Fluence	J at 2.5mm KJ/m^2	J_q KJ/m^2
Aged CF-8 CASS	16.7	20,000 hrs (2.28 yrs) at 325°C	Almost "Fully TE"	0	578	94
CF-8 CASS 6.3 dpa	16-20	1 yr in Bor at 325°C	Partially TE (1/3 time to saturation)	6.3	175	74
CF-8 CASS 12 dpa	16-20	1 yr in Bor at 325°C	Partially TE (1/3 time to saturation)	12	105	71
Semi aged CF-8 CASS 10.4 dpa	16-20	100 hrs at 400°C + 1 yr in Bor at 325°C	Partially TE	10.4	143	103
Aged Cass CF-8 6.3 dpa	16-20	950 hrs at 400°C + 1 yr in Bor at 325°C	"Fully TE"	6.3	210	129
Aged CF-8 CASS 12 dpa	16-20	950 hrs at 400°C + 1 yr in Bor at 325°C	"Fully TE"	12	123	57
WEC Fuel Nozzle Clamp (CF3), Service aged	12	3 year at 325°C	"Fully TE"	0.08	876	529
308 TIG mockup Weld Irradiated at Bor-6	8-15	1 year at 330°C in Bor	Partially TE	10.4	525	164

The concept that synergy will increase the rate of embrittlement is not in question; low levels of irradiation will harden and embrittle the ferrite phase. In the case that the behavior of the CASS is ferrite controlled, this will result in acceleration of the embrittlement process. In the case that the behavior of the CASS is austenite controlled, hardening of the ferrite will result in somewhat higher internal stresses but it is not clear that this process would enhance the production of channel slip. Thermal effects would accelerate embrittlement of the ferrite during radiation exposures of the austenite and ferrite. However, at the fluence levels needed to embrittle the austenite it would be expected that the accelerated hardening of the ferrite would not have a significant effect on the overall behavior of the CASS.

The data of Kim et al. [17] appear to demonstrate that CASS compositions that are not susceptible to thermal embrittlement retain toughness to very high irradiation fluences and also demonstrate minimal effect of TE in combinations with IE. The minimum toughness values

are driven by IE, and this toughness dataset seems to support the proposition that there is no exacerbation effect under the combined actions. By comparing the toughness of materials given no thermal aging, at partial thermal aging, and at full thermal aging plus irradiations up to 12 dpa, Kim et al. demonstrated that a CASS composition that was immune to thermal embrittlement was also not immune to irradiation embrittlement at low fluences. These data appear to support the argument that CASS compositions that are not susceptible to thermal embrittlement should also not be susceptible to IE at fluences significantly below those which would embrittle wrought austenitic stainless steels. These values of the remaining toughness even after 6 dpa exposure support the selection of a very high fluence for IE screening of materials that are not susceptible to TE. The proposed screening value of 6.7 x 10^{21} n/cm^2 (1 dpa) is significantly below the 6 dpa level at which significant remaining toughness was found by Kim et al. in their irradiated low ferrite CASS.

To date there are really no data that unequivocally point to any exacerbation effect due to synergy of the combined thermal and irradiation effects. The end point of embrittlement saturation seems, from the data given in Figure 1, to be effectively independent of the constitution of the steel microstructure. Moreover, recently Chopra and Rao [14] have reported that under LWR conditions CASS embrittlement of the austenite phase did not occur below 0.5 dpa. This value is again well above the fluence expected for embrittlement of the ferrite phase and significantly above the 1 x 10^{17} n/cm^2 of the Grimes letter. Note that while the data of Chen et al. [16] may display some greater loss of toughness on thermal plus irradiation exposures than on irradiation or thermal exposure alone, these data do not necessarily reflect exacerbation. The IE measurements were only made after 0.02 dpa of irradiation exposure with the remaining toughness J_q values around 200 kJ/m^2 and $J_{2.5mm}$, of the order of 400 to 800 kJ/m^2, reflecting very significant residual toughness. These data appear to be indicative that saturation loss of toughness had not occurred in these materials and that combined IE and TE would have induced some acceleration of the loss of toughness process. Note that, although these materials did display good toughness even when exposed to 0.02 dpa fluence, the compositions of these materials indicate that they would have been considered to be screened in for thermal embrittlement if they were found in the internals of a PWR under the proposed revised screening hierarchy. This factor seems to indicate that the proposed revised screening hierarchy and elevated IE screening criterion approach is still a conservative one.

At issue for the proposed fluence screening level is whether the proposed delta ferrite screening criteria are sufficiently conservative to assure that fluences below 1 dpa, in fully embrittling the ferrite phase, and will not reduce the fracture toughness of the composite structure to the point that structural integrity is threatened. It is the industry's opinion that a CASS internals component that has been screened out for TE, because the delta ferrite is $\leq$ 20%, will not be subject to an unacceptable reduction in fracture toughness as the result of accelerated delta ferrite embrittlement at a fluence < 1 dpa. The data relevant to this issue have been published in References 15 and 16, and a comparison of the data has been previously

developed by the industry and published in MRP-276 [3]. Because of this previous detailed discussion, only a summary discussion will be presented here.

Reference 16 examined the effects of low-level irradiation exposure (0.08 dpa) on the fracture toughness, as measured by the J-R crack growth resistance curve, for both unaged and thermally-aged CF-3, CF-8, and CF-8M CASS materials, with all three materials having delta ferrite in the 24 to 28% range. All three of the materials would have been screened in for TE using the current screening criteria. In none of the cases were the results from unirradiated thermally-aged material compared to the thermally-aged results irradiated to 0.08 dpa. For the CF-3 and CF-8 materials, the effect of the small amount of irradiation was not observable within experimental data scatter. However, for the CF-8M material, the difference between the unaged, irradiated fracture toughness and the aged, irradiated fracture toughness was quite noticeable, with the curve-fitted J value at 2.5 mm of crack growth equal to 615 kJ/m^2 for the unaged, irradiated material and 466 kJ/m^2 for the aged, irradiated material. However, Figure 15 of Reference 5 provides an experimental J-R curve for the same material aged 10,000 hours at 400°C and tested at essentially the same temperature (290°C versus 320°C). The J value at 2.5 mm is estimated to be about 500 kJ/m^2, so the irradiation at 0.08 dpa appears to have little or no effect, since the fracture toughness reduction is accounted for essentially by the thermal aging.

Therefore, from the results of Reference 16, even for delta ferrite content well beyond the 20% screening level, an irradiation level of 0.08 dpa seems to have little or no effect on the screening sequence, and fluences well above 0.08 dpa would be needed to have an effect on the austenite phase. MRP-276 [3] has carried out a similar evaluation of the same data, with the same conclusion, citing other work in the literature that 0.3 dpa exposure would be needed before any effects of irradiation would be noticeable, and that 1.0 dpa exposure would be a reasonable threshold for significant effects. Reference 3 also examined the effects of irradiation on welds containing 5 to 15% delta ferrite, drawing the same conclusions. From these studies, the current delta ferrite screening criteria are sufficiently robust to demarcate between ferrite-controlled and austenite-controlled phenomena, without any need to adjust for levels of irradiation exposure below 1 dpa.

<u>Summary</u>

A revised screening hierarchy based on first screening complete components for thermal embrittlement is proposed. The second stage of screening addresses those components and materials that have not already "screened in" for thermal embrittlement, and it identifies those components and regions of components that should be screened in for irradiation embrittlement.

The thermal screening is based on the same thermal screening criteria recommended by the Grimes letter. The IE screening criterion is based on a higher value of 6.7 x 10^{20} n/cm^2 (1 dpa) to correspond to screening for IE of the austenite phase.

The proposed revised screening hierarchy is based on the differences between the embrittlement of high-ferrite-containing CASS, which will be susceptible to thermal embrittlement and will undergo irradiation embrittlement at relatively low fluences, and the embrittlement of low-ferrite-containing CASS, which will not be susceptible to thermal embrittlement and will only undergo irradiation embrittlement at much higher fluences.

The identification of the IE screening fluences is in agreement with the IE screening value identified in MRP-175 (Appendix F) for PWR reactor internals. It is higher than the lower bound conservative values identified in BWRVIP-234 and BWRVIP-100-A for acceptable remaining toughness in BWR shroud structures and also higher than the fluence identified for the onset of loss of toughness in CASS by NUREG/CR-6960. However, this screening value is utilized to identify the loss of toughness that can be considered significant to the functional performance, whereas the other values represent conservative values for predicting the onset of some loss of toughness.

Experimental CASS toughness measured data after specific thermal and irradiation exposures support the hypothesis that screening for TE will identify materials that are also potentially susceptible to IE at low fluences and that use of the proposed revised screening hierarchy will allow for the use of a significantly higher, austenite embrittlement related IE screening criterion.

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F

SUPPLEMENTAL INFORMATION SUBMITTED BY BWRVIP ON MARCH 9, 2015

ELECTRIC POWER RESEARCH INSTITUTE

2015-025 BWR Vessel & Internals Project (BWRVIP)

March 09, 2015

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

Attention: Joseph Holonich

Subject: Project No. 704 – Summary of Industry Position on Screening Criteria for Thermal and Irradiation Embrittlement for PWR and BWR Reactor Internals Fabricated of Cast Austenitic Stainless Steel

References:
1. Letter from D. Madison (BWRVIP Chairman) and A. McGehee (BWRVIP Program Manager) to J. Holonich (NRC), Project No. 704 – BWRVIP Response to NRC Request for Additional Information on BWRVIP-234, dated May 23, 2014. ADAMS - ML14174A841.
2. Letter from J. Holonich (NRC) to A. Mendiola (NRC), Summary of the July 15, 2014, Meeting with the Electric Power Research Institute on Items Related to Cast Austenitic Stainless Steel and the Materials Reliability Program-227-A, "Pressurized Water Reactor Internals Inspection And Evaluation Guidelines," September 9, 2014. ADAMS - ML14127A077.
3. "NRC Position on Aging Management of CASS Reactor Vessel Internal Components," June 23, 2014, ADAMS - ML14174A719.
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5. NUREG/CR-4513, ANL-93/22, Revision 1, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," Argonne National Laboratory, May 1994.

The purpose of this letter is to provide the status of the Industry (PWR and BWR) activities associated with developing and proposing a generic screening criteria to address thermal and irradiation embrittlement of RPV internals fabricated of cast austenitic stainless steel (CASS).

Reference 1 was provided to the NRC in response to a Request for Additional Information (RAI) regarding BWRVIP-234. The RAI response contained supplemental information that provided the technical bases for an Industry screening criteria for CASS internals.

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A meeting was later held on July 15, 2014 at NRC offices in Washington, DC to discuss the Industry and NRC positions (References 2, 3). Based on that meeting, the Industry committed to investigate additional information where there was disagreement with the NRC regarding the screening criteria and provide said information to the NRC at a later date. That information was informally submitted for NRC's consideration (Reference 4) and is contained in Attachments A through D of this letter. A conference call was then held on November 20, 2014 with the NRC to discuss this information.

Following this conference call the NRC indicated that they were not willing to deviate from their position documented in Reference 3.

Despite the NRC's current stance on the matter, the following comments regarding a generic TE and IE criteria for evaluation of reactor internals are important in forming the basis for the industry position:

- The industry recognizes that the NRC is intending to revise its original IE position in the Grimes letter from $1x10^{17}$ n/cm^2 (0.00014 dpa) to a position in the region of 0.5 to 1.5 dpa. Industry agrees that this position is more appropriate and has proposed a conservative value of 1 dpa.
- It is not appropriate (for the staff) to penalize all non-Mo containing CASS (such as CF3 and CF8) because of the lower properties associated with the Mo-containing CF8M materials that are included in the ANL/NRC database, i.e., impose a lower bound.
- There is no substantive data that demonstrates a synergistic effect for CASS materials that are typical of reactor internals (such as CF3 and CF8). As such the industry has proposed criteria that do not combine TE and IE. Each mechanism is considered distinct and separate.
- The introduction of a new criteria set for the category of materials with ferrite content between 15% to 20% having a lower proposed screening value for IE of 0.45 dpa is unnecessary and technically unfounded. The introduction of this category of ferrite content is significantly burdensome to licensees since it will require more complex and potentially more error-prone assessments of CASS material components by virtue of having more categories. Industry maintains that the categories of materials and associated ferrite levels contained in the Grimes letter are appropriate.
- The criteria proposed by the industry (20% Ferrite and 1 dpa) have been shown to project significant margin on toughness reduction and therefore safety when compared to measured embrittlement behavior for CF3 and CF8 materials.

Regardless, this letter serves to formally submit this additional information to the NRC for review and consideration in developing a regulatory position for the screening criteria associated with TE and IE for CASS internals and to inform revisions to NUREG-4513, Rev. 1 (Reference 5). The PWR and BWR Industry position for TE and IE screening criteria is shown in Attachment D.

BWRVIP 2015-025

Irrespective of the staff's decision and future promulgation of a regulatory position on TE and IE for CASS internals, the BWRVIP formally requests the NRC to evaluate and resolve the RAI responses associated with BWRVIP-234 and issue a Safety Evaluation.

If you have any questions on this subject please call Ron DiSabatino (Exelon, BWRVIP Assessment Committee Technical Chairman) at 717.456.3685.

Sincerely,

A. O. McGehee Tim Hanley

Andrew McGehee, EPRI, BWRVIP Program Manager
Tim Hanley, Exelon Corp., BWRVIP Chairman

Industry Response to July 15, 2014 Meeting with NRC on CASS

On July 15, 2014, representatives from the BWRVIP/MRP Working Group on Cast Austenitic Stainless Steels (CASS) met with the NRC staff to discuss the thermal and irradiation embrittlement screening criteria for cast austenitic stainless steel. While the working group and NRC staff were in general agreement on the thermal embrittlement (TE) screening criteria, there was a marked difference of opinion on the irradiation embrittlement (IE) screening criteria, particularly for materials with delta ferrite contents of 15-20% (the NRC proposed 0.45 dpa, while the industry proposed 1 dpa).

Agreement was reached on the TE screening criteria in part because the data on low-molybdenum and high-molybdenum material are considered separately, resulting in separate TE screening criteria for these materials. This distinction is made because high-molybdenum materials show a distinctly greater loss of fracture toughness from TE and IE effects than low-molybdenum materials. This is reviewed and discussed further in Attachment A

The separate consideration of low- and high-molybdenum materials test data was not carried over into the IE domain. The data set used to determine the NRC's position on IE was obtained from NUREG-7027, which considered irradiated high- and low-molybdenum materials together, along with data on irradiated welds. Because these materials were considered together, the resulting lower bound curve is judged to be overly conservative relative to the low-molybdenum material. A reexamination of this data (see Attachment B), shows that by considering the low-molybdenum materials test data independently of the other materials it can be concluded that there is a substantial safety margin in the industry's proposed 1 dpa screening criteria.

These screening criteria are being implemented to ensure that CASS components in reactor vessel internals maintain adequate fracture toughness during the period of extended operation. The screening level to determine whether this requirement was met was a J value of 255 KJ/m^2 at a crack extension of 2.5 mm. This value was originally determined for pressure boundary components (i.e., large diameter piping, etc.) and was expected to be highly conservative for application to reactor vessel internals. Additional calculations (see Attachment C) demonstrate the level of this conservatism and show that reactor vessel internal components can safely operate with fracture toughness values much lower than the 255 KJ/m^2 value specified in the Grimes letter.

In summary, the actions taken by the CASS industry working group and summarized in Attachments A, B and C support the proposed screening criteria contained in Attachment D. These criteria provide a significant amount of margin for screening of TE and IE. It is concluded that that the use of the proposed screening criteria will allow for continued safe and reliable operation of the LWR fleet.

Attachment A

Discussion on Low Molybdenum vs High Molybdenum CASS Grades

Introduction

A meeting was held on July 15, 2014 at the NRC Office in Washington, DC between the staff of the U.S. Nuclear Regulatory Commission (NRC) and the BWRVIP/MRP Working Group on Cast Austenitic Stainless Steels (CASS). The topic of discussion focused on the thermal aging and neutron irradiation embrittlement screening criteria (separately) proposed by the industry and the NRC staff. The proposals included criteria for low-molybdenum CASS grades, such as CF-3 and CF-8, and high-molybdenum CASS grades, such as CF-8M. There was agreement between industry and the NRC staff regarding the differences in such measures of embrittlement as elastic-plastic crack growth resistance (J-R curve) for the low-Mo materials versus the high-Mo materials, at least where thermal aging embrittlement effects are dominant. In fact, screening criteria proposed by the staff are identical to the industry-proposed screening criteria for the case of thermal aging embrittlement.

However, the areas of agreement between the industry and the staff did not carry over to combined environments where the effects of neutron irradiation embrittlement begin to approach the same level as those from thermal aging embrittlement. In order to address these differences in criteria, the available data for CF-3, CF-8, and CF-8M materials in the literature are reviewed and assessed, with a recommendation to the NRC staff for a change to their proposed criteria to more closely agree with the criteria proposed by the industry.

Review and Summary of Thermal Aging Embrittlement Data

The significant differences in thermal aging embrittlement behavior of the low-molybdenum grades of cast austenitic stainless steel, such as CF-3 and CF-8, versus the thermal aging embrittlement behavior of the high-molybdenum grades, such as CF-8M, have been known for at least two decades, with extensive data and associated interpretations of said data documented in NUREG/CR-4513, Revision 1 [1]. The lower-bound thermal aging embrittlement estimates shown in Figures 3 (statically-cast steels) and 4 (centrifugally-cast steels) from Reference 1 with varying amounts of delta ferrite provide an excellent illustration of those significant differences, in this case with the measure of significance being the crack-growth resistance J-R curve. In particular, the lower-bound formulas cited in Section 3.1.1 of Reference 1 (and plotted in Figures 3 and 4) can be used to compare the estimated crack growth resistance values at 2.5 mm of crack extension for the CF-3, CF-8, and CF-8M steels with >15% delta ferrite. Note that Figure 3 provides lower-bound thermal aging estimates for static-cast steels, while Figure 4 provides lower-bound thermal aging estimates for centrifugally-cast steels.

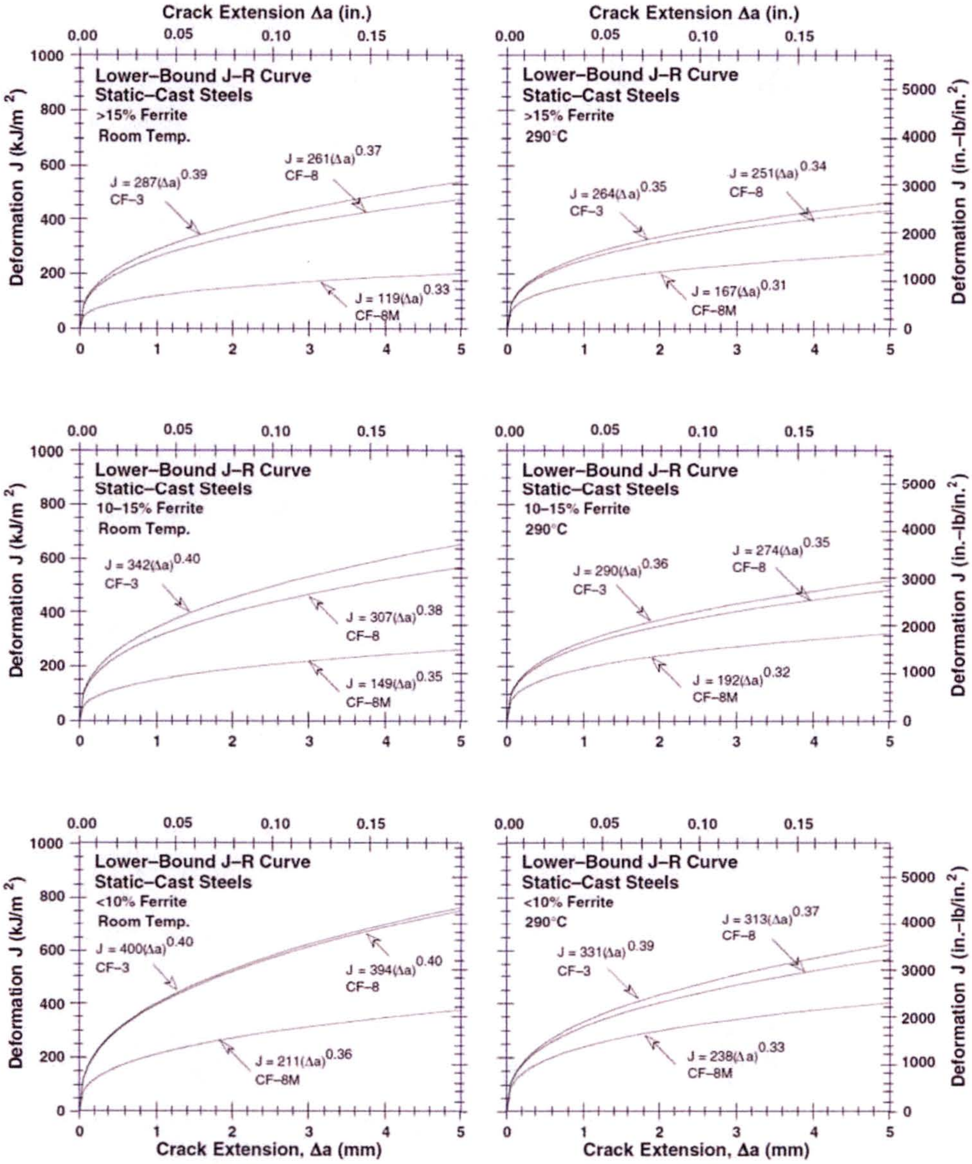


Figure 3. Predicted lower-bound J-R curves at RT and 290°C for static-cast SSs with ferrite contents >15, 10–15, or <10%

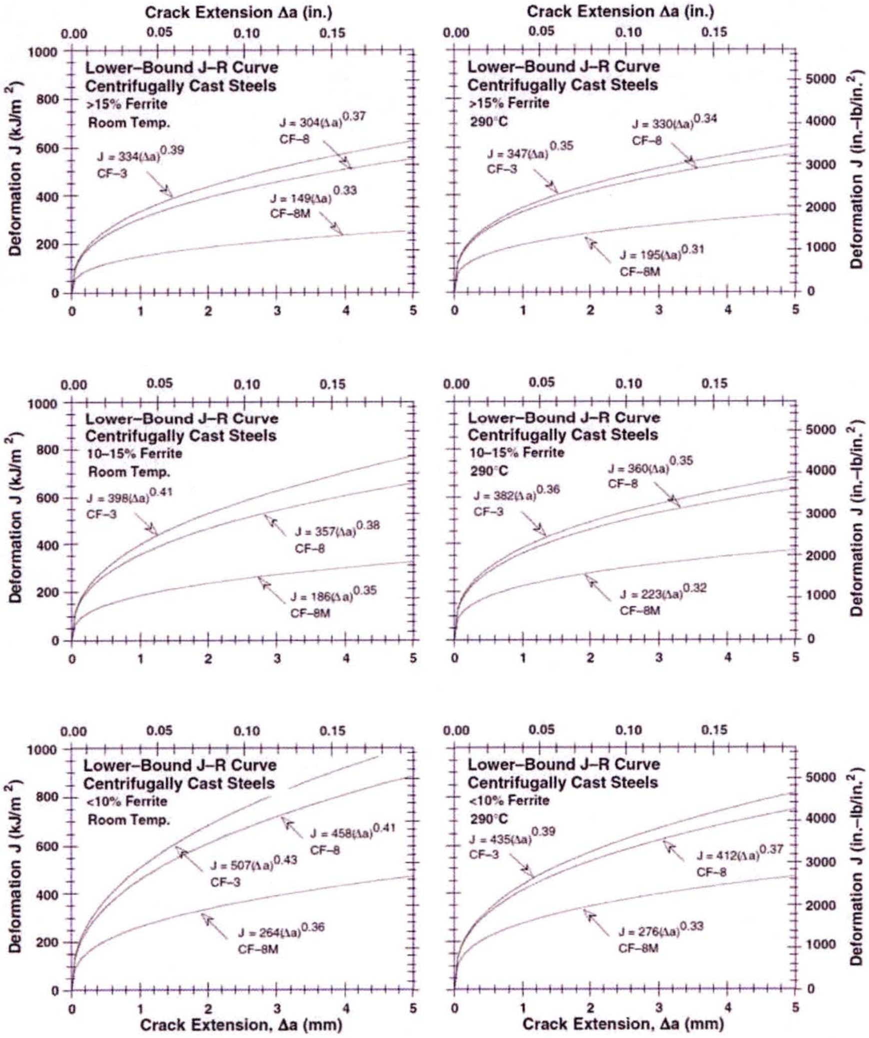


Figure 4. Predicted lower-bound J-R curves at RT and 290°C for centrifugally cast SSs with ferrite contents >15, 10-15, or <10%

For CF-3, the lower bound crack growth resistance, for statically- and centrifugally-cast materials, varies between 364 and 478 kJ/m^2, and for CF-8, between 343 and 451 kJ/m^2. On the other hand, the CF-8M values vary between 161 and 259 kJ/m^2. When the data are examined very closely, there is no significant difference between the static-cast lower-bound estimates for CF-3 and CF-8 in Figure 3 and the centrifugally-cast lower-bound estimates for CF-3 and CF-8 in Figure 4. The data also show that these insignificant differences are roughly identical to the differences between CF-3 and CF-8 estimates within the static-cast and centrifugally-cast populations. Therefore, based on this examination of lower-bound estimates, the CF-8M data exhibit substantially lower fracture toughnesses and are therefore unsuitable as a lower-bound estimate for the low-molybdenum grades. Lower bound estimates must be conservative, but they must also remain representative of the data. Table 1, shown below, succinctly summarizes the data comparisons extracted from Figures 3 and 4 from Reference 1.

Table 1. Lower Bound Fracture Toughness Estimates

CASS Steels	J-values at 2.5 cm of Crack Growth (kJ/m^2)	Comments
CF-3 Steels	364 to 478	Lower bound above the threshold value of 255 kJ/m^2
CF-8 Steels	343 to 451	Lower bound above the threshold value of 255 kJ/m^2
CF-8M Steels	161 to 259	Lower bound might be less than threshold values of 255 kJ/m^2

Table 1 clearly shows that lower bound fracture toughness values of CF-3 and CF-8 steels are much higher than CF-8M steels, and that only the fracture toughness of CF-8M steels fall below the threshold value of 255 kJ/m^2. Therefore, the screening criteria for CF-8M (high molybdenum) steels should be considered separately from CF-3 and CF-8 (low molybdenum) steels.

Thus far, the data evaluation has been based on lower-bound fracture toughness estimates derived from Charpy impact correlations. However, these lower-bound thermal aging embrittlement estimates can be confirmed by examining actual crack extension resistance curve measurements for various heats of CF-3, CF-8, and CF-8M with delta ferrite content in the range of 20%. For example, Figure 14 from Reference 1 shows J-R curve measurements, along with thermal aging saturation estimates for two heats of CF-3 material (Heat 69 and Heat I) and one heat of CF-8 material (Heat 68), while Figure 15 from Reference 1 shows similar results for three heats of CF-8M material (Heats 74, 75, and 758).

In this case, it should be noted that the actual fracture toughness data are based on thermal aging saturation conditions and are therefore somewhat higher than the lower-bound fracture toughness

estimates. Also note that the thermal aging saturation fracture toughness data are based entirely on static-cast data for materials with delta ferrite in the range of 15% to 25%.

The crack extension resistance value at 2.5 mm of crack extension for the thermal aging saturation conditions for CF-3 and CF-8 materials ranges between 681 and 820 kJ/m^2 at room temperature, with values between 516 and 584 kJ/m^2 at operating temperature, while the values for CF-8M material are less than half of the low-molybdenum material values.

Therefore, the findings from the examination of the data in Figures 14 and 15, and the derived J values at 2.5 mm of crack extension, are completely consistent with the lower-bound thermal aging embrittlement estimates cited previously for both static-cast and centrifugally-cast material. In other words, the development of screening criteria for the low-molybdenum grades should be treated separately from the development of screening criteria for the high-molybdenum grades.

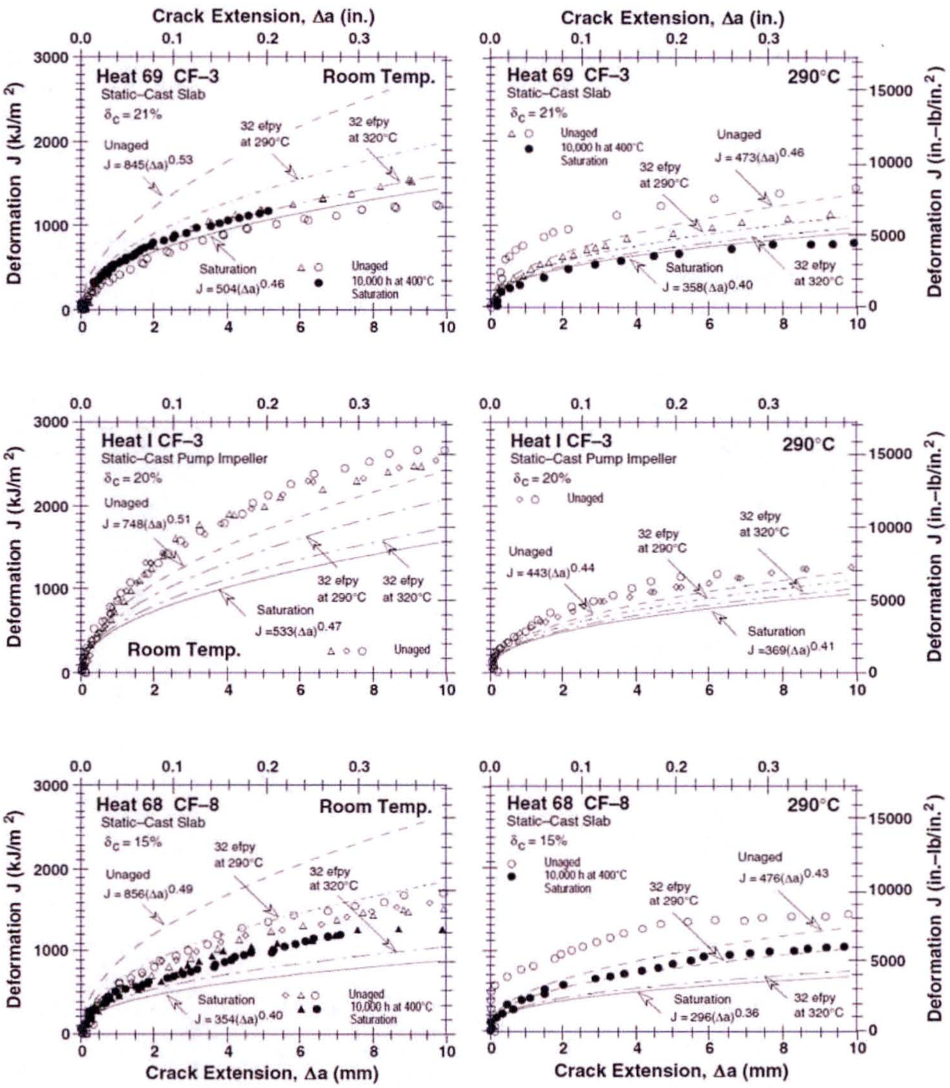


Figure 14. Saturation fracture toughness J-R curves at RT and 290°C, estimated from the chemical composition of static-cast CF-3 and CF-8 steels, and determined experimentally

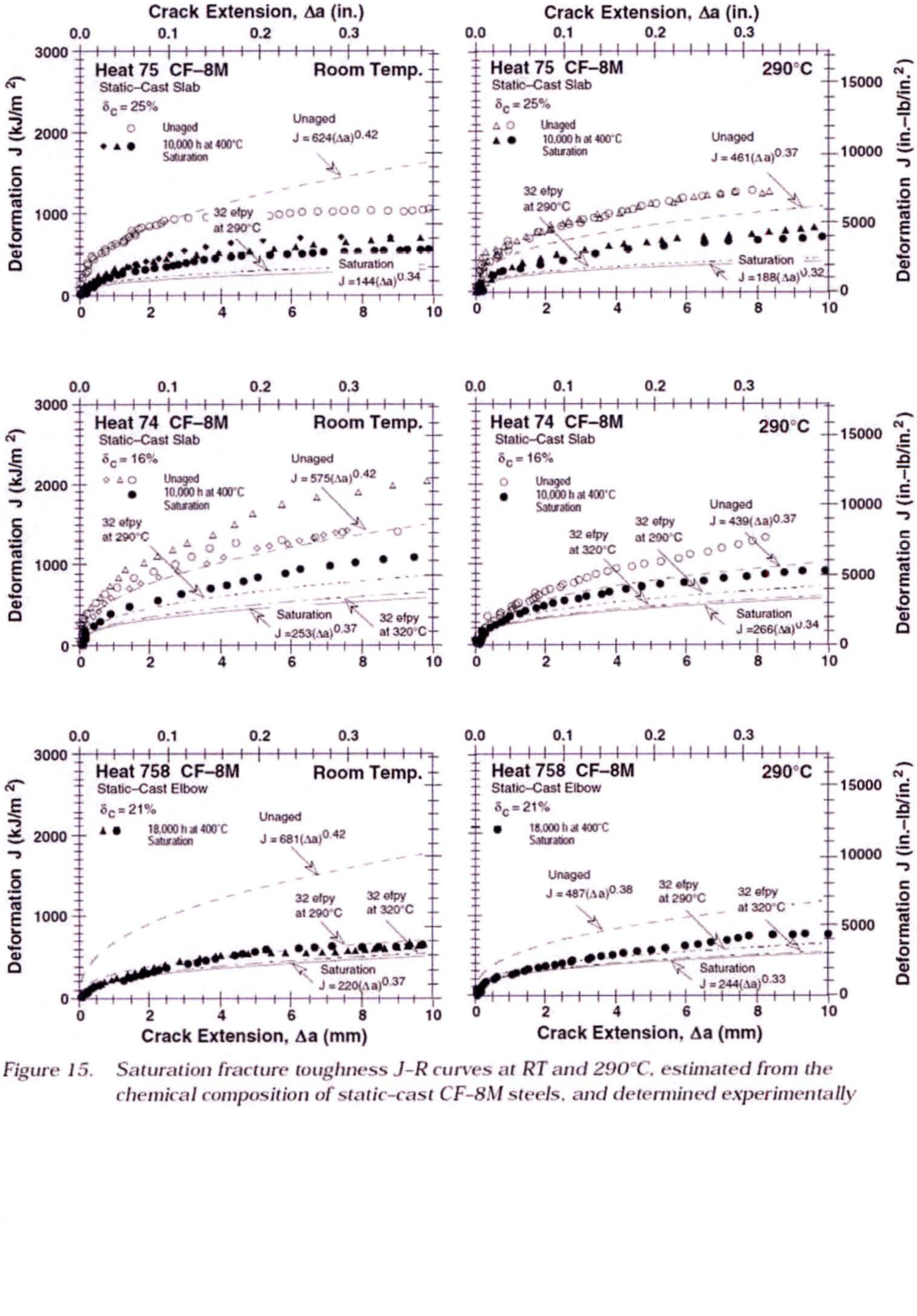


Figure 15. Saturation fracture toughness J-R curves at RT and 290°C, estimated from the chemical composition of static-cast CF-8M steels, and determined experimentally

Review and Summary of Combined TE and IE Data

The available data for purely thermally-embrittled CASS materials is quite extensive. However, that is not the case for a combination of thermal aging and neutron irradiation embrittlement. In spite of the limited data, the evaluation of available crack growth resistance data for CF-3, CF-8, and CF-8M material is not altered by including information on material subjected to both thermal aging and neutron irradiation embrittlement, as shown in Figures 23, 41, 49, and 58 from ANL-12/56 [2]. The results for Specimen B-1 (static-cast CF-3 material taken from Heat 69, with approximately 24% delta ferrite, thermally aged for 10,000 hours at 400°C, and then subjected to a neutron irradiation dose of 0.08 dpa) are shown in Figure 23 – taken from Reference 2. The extrapolated J value at 2.5 mm of crack extension is 789 kJ/m^2, very similar to the saturated, thermally-aged result, as shown in Figure 14 for Heat 69 from Reference 1.

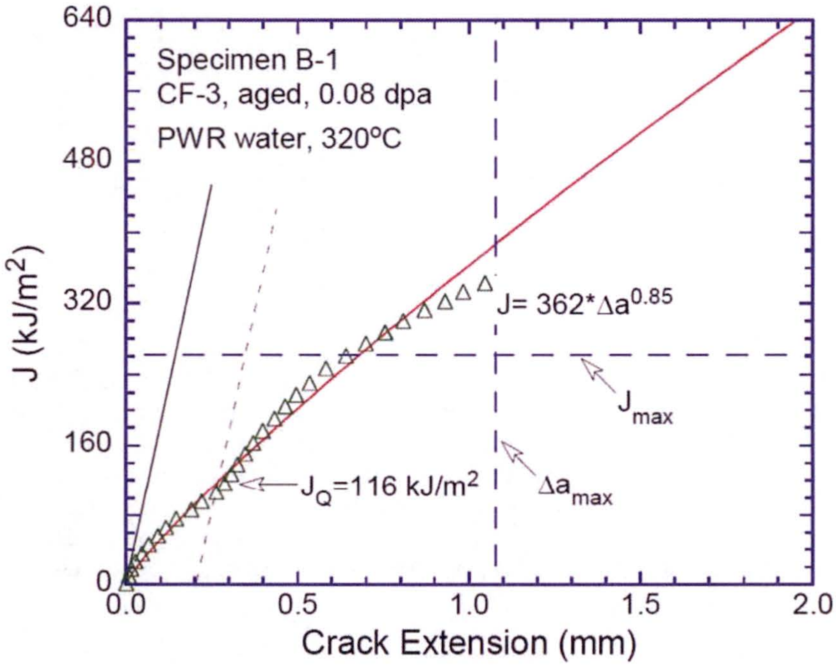


Figure 23. The J-R curve for specimen B-1.

Similarly, the results for Specimen F-1 (static-cast CF-8 material taken from Heat 68, with approximately 23% delta ferrite, thermally aged for 10,000 hours at 400°C and then subjected to a neutron irradiation dose of 0.08 dpa) are shown in Figure 41 – also taken from Reference 2. The extrapolated J value at 2.5 mm of crack extension is 657 kJ/m^2, very similar to the saturated, thermally-aged result, as shown in Figure 14 from Reference 1 for Heat 68.

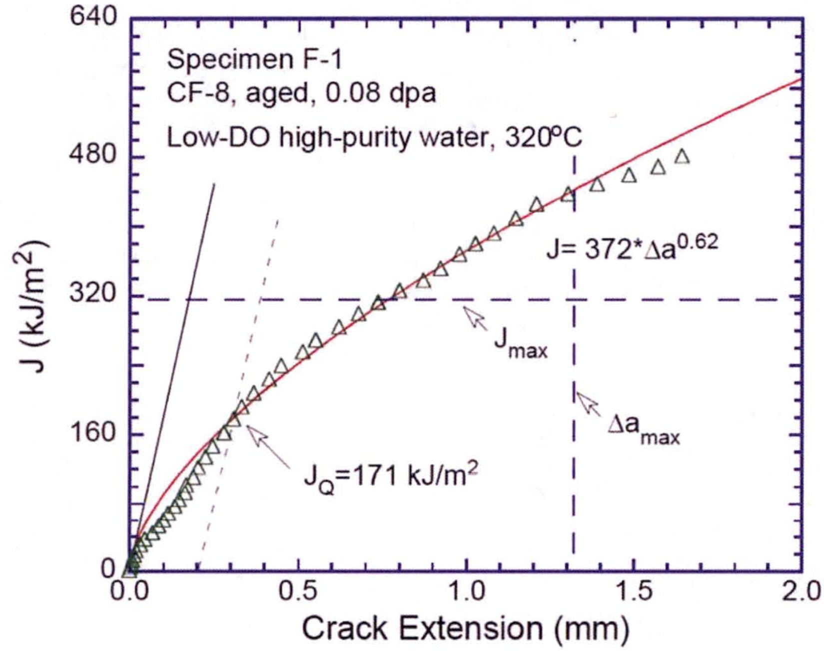


Figure 41. The J-R curve of specimen F-1.

Finally, the results for Specimens I-1 (CF-8M material taken from Heat 75, with approximately 28% delta ferrite, unaged but subjected to a neutron irradiation dose of 0.08 dpa) and J-1 (CF-8M material taken from Heat 75, with approximately 28% delta ferrite, thermally aged for 10,000 hours at 400°C, and then subjected to a neutron irradiation dose of 0.08 dpa) are shown in Figures 49 and 58 – also taken from Reference 2. In this case the extrapolated J value at 2.5 mm of crack extension for Specimen J-1 is 466 kJ/m^2, which can be compared to the results shown in Figure 15 from Reference 1 for Heat 75 that appear to be of the order of 300 to 400 kJ/m^2 at room temperature and operating temperature.

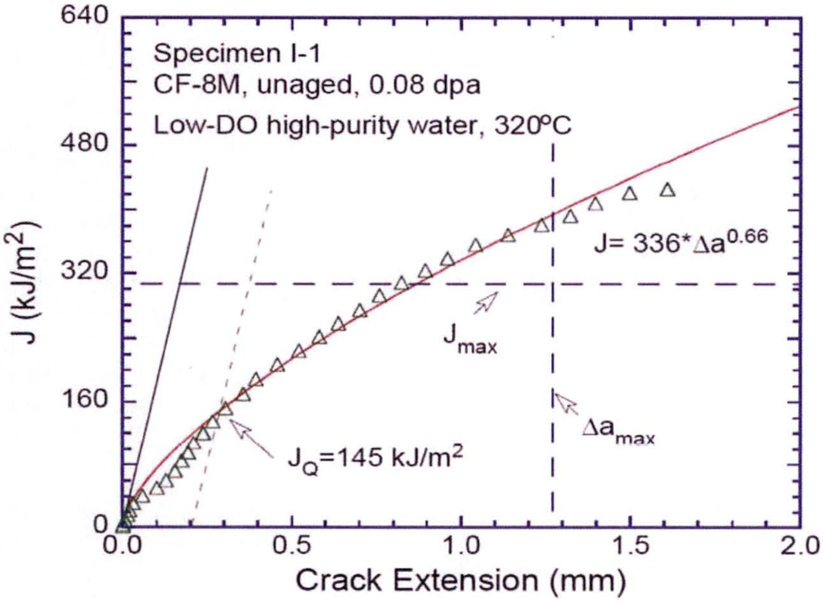


Figure 49. The JR curve of specimen I-1.

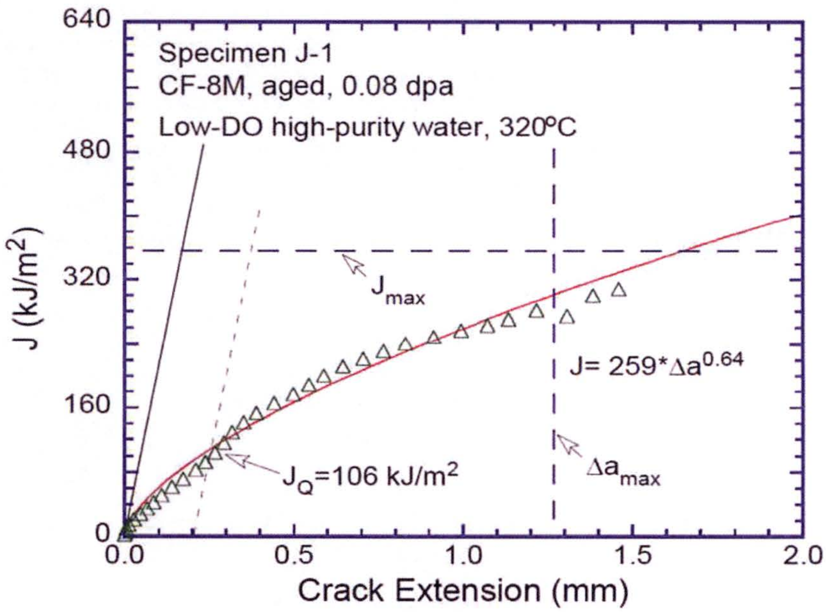


Figure 58. The JR curve of specimen J-1.

Conclusions and Recommendation

This review and summary of available data on CF-3, CF-8, and CF-8M CASS materials subjected to both individual and combined effects of thermal aging and neutron irradiation embrittlement demonstrated that:

- Thermal aging embrittlement effects on high-Mo grades are significantly more pronounced than the effects on the low-Mo grades, by roughly a factor of two or more based upon lower-bound crack growth resistance estimates. For CF-3, the lower-bound crack growth resistance values at 2.5 mm of crack extension for > 15% delta ferrite vary between 364 and 478 kJ/m^2, and for CF-8, between 343 and 451 kJ/m^2. On the other hand, the CF-8M values vary between 161 and 259 kJ/m^2 which are significantly lower compared to CF-3/8. These results are consistent with the results obtained by examining actual crack growth resistance value measurements at 2.5 mm of crack extension for the thermal aging to saturation for CF-3 and CF-8 materials (low-Mo) that range between 681 and 820 kJ/m^2 at room temperature, with values between 516 and 584 kJ/m^2 at operating temperature, while the values for CF-8M material (high-Mo) are less than half of the low-molybdenum material values.
- There is no significant difference between the static-cast lower-bound estimates and the centrifugally-cast lower-bound estimate for CF-3 and CF-8 materials, and those relatively small differences are roughly identical to differences between CF-3 and CF-8 lower-bound estimates. Using lower-bound crack growth resistance values at 2.5 mm of crack extension as the comparative measure, the CF-3 static-cast values of 364 and 410 kJ/m^2 compare very well to the 477 and 478 kJ/m^2 values for centrifugally-cast material, while the CF-8 static-cast values of 343 and 366 kJ/m^2 compare favorably with the 427 and 451 kJ/m^2 values for centrifugally-cast material. The differences between static-cast and centrifugally-cast values, and between CF-3 and CF-8 values are both inconsequential in comparison to the differences between low-Mo and high-Mo materials. Therefore these two grades may be considered as a combined category for screening purposes.
- The added effect of neutron irradiation embrittlement does not change the conclusions drawn with respect to low-Mo versus high-Mo fracture toughness. Finally, since the conclusions are drawn primarily from static-cast fracture toughness data, a recommendation to the NRC staff to maintain the screening threshold for low-Mo static-cast grades at 20% is supported by the data.

Therefore, this review shows that the CF-8M data provide an extremely conservative and thus unrepresentative lower bound to the data for the low-molybdenum grades. As such, the CF-8M lower bound is separable and distinct from the low-molybdenum grade lower bounds, which therefore require their own screening category and fracture toughness estimates. Thus, the delta ferrite levels stated in the BWRVIP-234 RAI response (from the Grimes letter) is further supported by this evaluation.

Finally, this review and the associated quantitative comparisons utilize the data sets as delineated in extensive documentation in NUREG/CR-4513, Revision 1 [1] and other relevant publically available references.

References

[1]. O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR-4513 (ANL-93/22), Revision 1, Argonne National Laboratory, Argonne, Illinois (August 1994).

[2]. Y. Chen, B. Alexandreanu, and K. Natesan, "Crack Growth Rate and Fracture Toughness Tests on Irradiated Cast Stainless Steels," Report No. ANL-12/56, Argonne National Laboartory, Argonne, Illinois (November 2012).

Attachment B

Conclusions Regarding Screening Criteria for Irradiation Embrittlement of CASS Materials

The screening criteria for TE and IE in CASS components proposed and discussed with the NRC [1] appears to be based on materials testing data that includes a high-molybdenum material, grade CF-8M, presented in Figure 64 in NUREG 7027 [2], with the following conclusions on page 79:

1. CF-8M materials represent a 'worst case' for thermal embrittlement
2. Thermal aging does not seem to lower the toughness below that expected for irradiation alone at the dose levels examined.

In regards to the use of this data, it is important to note that:

- CF-8M materials are not present in LWR core internal components.
- Moreover, the embrittlement identified in NUREG-7027 is conservative because many of the heats examined had delta ferrite contents greater than 20% and as high as 42% [3]. It is also important to emphasize that in LWR reactor internals, delta ferrite content does not typically exceed 20% as compared to the materials assessed in the NUREG database [2, 3].
- Loss of fracture toughness due to thermal embrittlement is driven by the amount of delta ferrite in a given material, thus the use of high delta ferrite material data is also conservative.
- ANL results comparing loss of fracture toughness measurements on unaged, thermally aged only, irradiation aged only and sequentially thermally aged and irradiation aged CF-3 and CF-8 materials show significant loss of toughness on ***all*** forms of aging. Measured toughnesses of thermally aged, irradiation aged and sequentially thermal plus irradiation aged materials are similar. Given the variances that are inherent in estimating J values as material characteristics, the differences in loss of toughness cited in Reference 4 cannot be taken to unequivocally demonstrate that irradiation and thermal aging are significantly additive effects to the loss of toughness in CF-3 and CF-8 cast austenitic stainless steels.
- The data summarized in Figure 64 of NUREG-7027 includes many high ferrite content and Mo-containing chemical compositions. The correlations developed in analyses based on this database are therefore expected to be very conservative with regard to the irradiation and thermal response of low ferrite content and low Mo containing chemical compositions.

The NRC is concerned that interaction between thermal and irradiation embrittlement mechanisms may lower the fracture toughness of CASS material below levels expected from the effect of either mechanism alone. Based on the potential for a synergistic effect and taking into account the NUREG-7027 data, the NRC has concluded that a very low level of irradiation would reduce the fracture toughness of a CASS component below the 255 KJ/m^2 criteria. This led to the NRC proposing a neutron irradiation screening criteria of 0.45 dpa for low molybdenum static CASS components with 15-20% delta ferrite. By reexamination of the data in NUREG-7027 a considerable amount of conservatism can be removed from the analysis while still providing reasonable assurance that CASS components in the LWR environment will retain adequate fracture toughness through the period of extended operations.

The irradiated materials data from NUREG-7027 were digitized from the plots of NUREG-7027 and reanalyzed to produce a trend curve for irradiated CF-3 and CF-8 materials only as shown in Figure 1 below. Note that the exposures of these materials not only represent irradiation effects but also some degree of thermal embrittlement. These data for CF-3 and CF-8 materials are most relevant to the behavior of CASS in LWR internals, as these are the materials that are principally known to be present in LWR reactor internals. Thus the correlation and proposed screening criteria are appropriately developed for these materials.

As an outcome of this analysis a plot of fracture toughness against neutron exposure was created, and a 'best fit' line was developed and the resulting curve was calculated to have a strong correlation with the data (R^2=0.94). The best fit line was then modified using conservative engineering judgment and standard curve offsetting techniques to bound all of the irradiated CF-3 and CF-8 material measured toughness values. (i.e., the pre-exponential constant multiplier was reduced by approximately 25% and the exponential factor was increased by 35% in order to bound every data point above the red-line exponential curve fit.) A constant of 30 kJ/m^2 was also included as a saturated fracture toughness value.

This saturated fracture toughness value is proposed by Chopra in NUREG-7027, and the overall form of the equation is consistent with irradiation hardening models [6]. This results in a curve fit where all data points are above the curve, with resulting curve fit of:

$$J_{2.5mm} = 30 + 520\frac{KJ}{m^2} \times e^{-0.25 \times dpa}$$

This bounding line is shown as a red line in Figure 1.

By using the red (lower bound) line as a gauge for how CF-3 and CF-8 materials will age in the LWR environment, it can be shown that these materials will not encroach upon the 255 KJ/m^2 fracture toughness limit until the fluence level experienced by these components approaches 3.3 dpa. The industry proposed use of 1 dpa for neutron exposure screening criteria, provides more than a 2 dpa margin on neutron exposure between a CASS component at 1 dpa and the 255

KJ/m^2 fracture toughness limit. At 1 dpa, the fracture toughness predicted by the bounding curve in Figure 1 is 435 KJ/m^2, which results in a fracture toughness margin of 180 KJ/m^2.

Therefore, the industry's proposed criterion of 1 dpa has substantial margin in both the fluence and fracture toughness domains. This analysis and argument supports the use of the irradiation screening criteria of 1 dpa proposed by the industry for low molybdenum CASS components with ferrite content less than 20 percent.

Additionally, a study designed to investigate the potential interaction between thermal and irradiation embrittlement mechanisms concluded that irradiation can reduce the extent of thermal aging effects [5], suggesting an antergistic effect between the two mechanisms rather than a synergistic one.

This information indicates that the NRC's position is highly conservative, and that the industry's position on the screening criteria of CASS components, while also conservative, is justified.

References:

1. U.S. NRC, "NRC Staff Compiled Comments on Industry CASS Screening Position," July 18, 2014 (NRC ADAMS Accession Number ML14198A282.

2. NUREG/CR-7027, "Degradation of LWR Core Internal Materials due to Neutron Irradiation," December, 2010 (NRC ADAMS Accession Number ML102790482.

3. NUREG/CR-4513, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," U.S. Nuclear Regulatory Commission, August 1994 (NRC ADAMS Accession No. ML052360554).

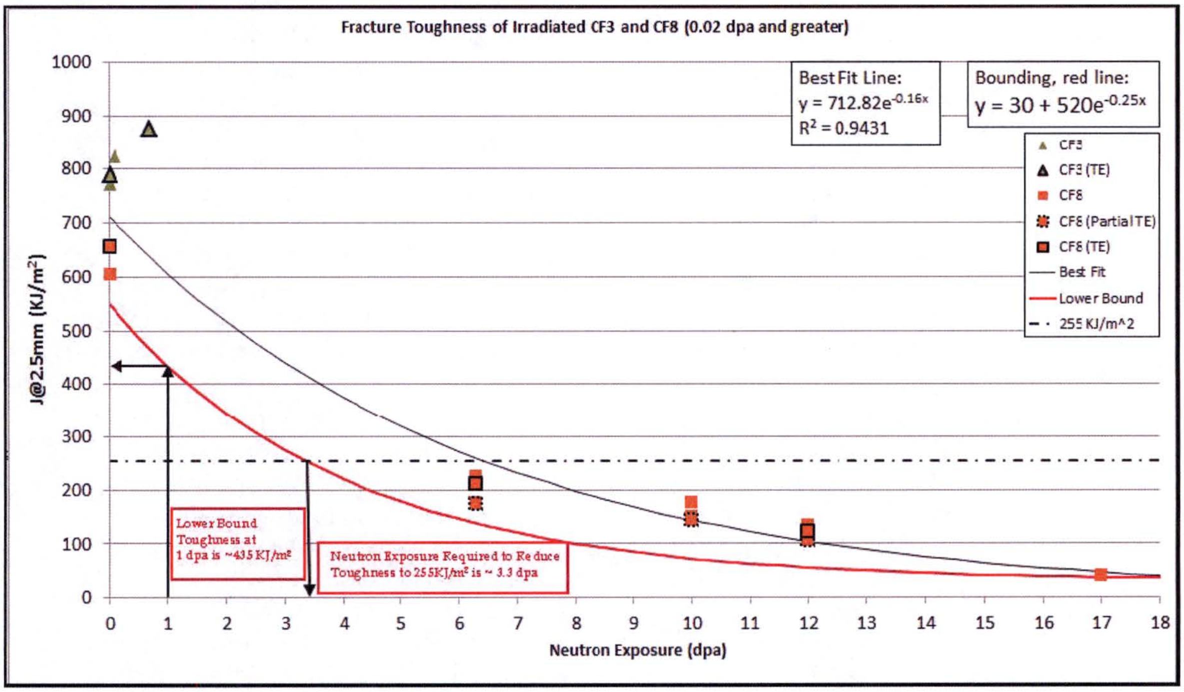
4. ANL-12/56, "Crack Growth and Fracture Toughness Tests on Irradiated Cast Stainless Steels," Argonne National Lab., November 2012.

5. K Fuji, K. Fukuya, "Effects of Radiation on Spinodal Decomposition of Ferrite in Duplex Stainless Steel," Journal of Nuclear Materials, May 2012. Presented at the NuMat 2012 Conference, pages 613 to 616, October 22-25, 2012, Osaka, Japan.

6. Was, Gary S., "Fundamentals of Radiation Materials Science: Metals and Alloys," Springer-Verlag, 2007.

7. Kim, C., R. Lott, S. Byrne, M. Burke, and G. Gerzen, "Embrittlement of Cast Austenitic Stainless Steel Reactor Internals Components," Proc. 6th Intl. Symp. On Contribution of Materials Investigations to Improve the Safety and Performance of LWRs, Fontevraud 6, French Nuclear Energy Society, SFEN, Fontevraud Roayal Abbey, France, September 18-22, 2006.

Figure 1 – Irradiated CF-3 and CF-8 Fracture Toughness Data from Literature (Refs. 2, 4 and 7)



Attachment C

Reactor Internals CASS Flaw Tolerance

In the draft Interim Staff Guidance (ISG) on Aging Management of CASS Reactor Vessel Internal Components, issued in June 2014, the NRC technical position observed that "The fracture toughness screening value of 255 kJ/m^2 specified in the Grimes Letter is based on a generic flaw tolerance evaluation for piping, and may be overly conservative for RVI CASS components that are subject to mainly compressive stresses during operation, and are part of a population of redundant components where failure of individual components can be tolerated." The technical position went on to add that "therefore, the staff applies the 255 kJ/m^2 value for screening purposes with the knowledge that there likely is additional conservatism present in this screening for non-pressure boundary RVI components." In an effort to provide further evidence with respect to the NRC staff technical position, some industry efforts since the July 15, 2014 meeting with the NRC staff have been directed toward this issue. The results are summarized in the following paragraphs.

The industry selected three different geometries that can be related to typical BWR and PWR CASS reactor internals in order to estimate flaw tolerance capability: (1) a large-diameter cylinder with a through-wall vertical flaw located in a longitudinal seam weld, subjected to combined membrane and bending stress; (2) an edge-cracked beam-column subjected to combined membrane and bending stress; and (3) a six-inch-diameter, Schedule 40 pipe with a through-wall longitudinal flaw subjected to internal pressure. The internal pressure for the third geometry was selected such that the circumferential tensile stress in the pipe was identical to the membrane tensile stress level chosen for the first two geometries. In all three case studies, the initial flaw size (length for the first and third geometries, and depth for the second geometry) was selected to be consistent with reactor vessel internals fabrication workmanship standards. No flaw growth criteria were applied to the initial flaw sizes, although the initial flaw sizes were increased to some extent in order to determine the rate at which the crack driving force increased as a function of the increase in flaw size.

For the first geometry (large-diameter cylinder with through-wall vertical flaw), the diameter was selected to be 452 inches with a wall thickness of either one or two inches, the membrane tensile stress was selected to be 5 ksi, and the bending stress was selected to be 7.5 ksi. The total initial flaw length was selected to be 0.226 inches (about ¼-inch in length), but was increased in increments up to 4.294 inches. The applied stress intensity factors were found using linear-elastic fracture mechanics (LEFM), and were then converted to J-crack driving forces. For the largest initial flaw lengths examined, the typical LEFM stress intensity is of the order of 25 $ksi\sqrt{in}$, which converts to an elastic-plastic crack driving force of about 4 kJ/m^2. This implies a very large margin relative to the screening value of 255 kJ/m^2.

For the second geometry (edge-cracked beam column), the solid column diameter was selected to be three inches, and the membrane and bending stresses were selected to be identical to those selected for the first geometry. The smallest initial flaw depth was selected as 0.1 inches, and was incremented by 0.1 inch up to a maximum depth of 1.7 inches. The applied stress intensity factors were found using linear-elastic fracture mechanics (LEFM), and were then converted to

J-crack driving forces. For the largest initial flaw depth examined (1.7 inches), the LEFM stress intensity was 42 ksi√in, which converts to an elastic-plastic crack driving force of about 10 kJ/m^2. Again, this calculation implies a very large margin relative to the screening value of 255 kJ/m^2.

For the third geometry (six-inch diameter Schedule 40 pipe with a through-wall longitudinal flaw subjected to internal pressure), the internal pressure was selected such that the circumferential stress was approximately 5 ksi, matching the membrane tensile stress used for the first and second geometries. Because of the thin pipe wall, this case was found to be the most critical of the three geometries. The initial flaw length was selected to be 0.1 inches, and was then incremented in 0.1-inch increments up to a maximum flaw length of 2.0 inches. For this case, the applied stress intensity factors were found using linear-elastic fracture mechanics (LEFM), and were then converted to J-crack driving forces. In addition, this case was also analyzed using elastic-plastic fracture mechanics, in order to compare derived crack-driving forces with directly calculated elastic-plastic crack-driving forces. For an initial flaw length of 2.0 inches, the LEFM stress intensity factor was about 54 ksi√in, which converts to a crack driving force of about 96 $in\text{-}lb/in^2$ or about 16 kJ/m^2. The directly-calculated elastic-plastic crack driving force is almost exactly the same, but very slightly lower. When the initial flaw length was doubled – to about 4.0 inches, the applied LEFM stress intensity was found to be 120 ksi√in, which converts to a crack driving force of about 476 $in\text{-}lb/in^2$ or about 83 kJ/m^2. Even for this very severe example, the calculations show a sizable margin relative to the screening value of 255 kJ/m^2.

From this exercise, CASS reactor internals components subjected to nominal stress levels, even in the presence of initial flaws that are well beyond fabrication workmanship acceptance criteria, are extremely flaw tolerant, with margins against flaw instability of the order of a factor of five to 10 relative to the fracture toughness screening criterion of 255 kJ/m^2 specified in the Grimes letter. When this flaw tolerance is coupled with the additional margin inherent in the separation of CF-3/CF-8 screening data from CF-8M screening data, the conservatism of the industry technical position is further confirmed.

Attachment D

Table 1. Revised % Ferrite Ranges (Industry Position)

Molybdenum (wt. %)	Casting Method	% Ferrite	TE Screening Susceptibility Result	IE Screening Susceptibility Result
High 2.0-3.0% (CF-3M & CF-8M)	static	< 10	No	≥ 1 dpa
		≥ 10 - ≤ 14	No	≥ 0.45 dpa
		> 14	Yes	N/A
	centrifugal	≤ 20	No	≥ 1 dpa
		> 20	Yes	N/A
Low 0.5% max (CF-3 & CF-8)	static	≤ 20	No	≥ 1 dpa
		> 20	Yes	N/A
	centrifugal	All	No	≥ 1 dpa

G
RECORD OF REVISIONS

BWR-234-A	Information from the following documents was used in preparing the changes included in this revision of the report. 1. *BWR-234, BWR Vessel and Internals Project: Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steels for BWR Internals*, EPRI, Palo Alto, CA: 2009 1019060. 2. Letter from John Jolicoeur (NRC) to David Czufin (BWRVIP Chairman) "ACCEPTANCE FOR REVIEW AND REQUEST FOR ADDITIONAL INFORMATION, FOR BWRVIP-234: "BWR VESSEL AND INTERNALS PROJECT: THERMAL AGING AND NEUTRON EMBRITTLEMENT EVALUATION OF CAST AUSTENITIC STAINLESS STEEL FOR BWR INTERNALS" (TAC NO. ME5060), dated September 29, 2011. 3. Letter from Dennis Madison (BWRVIP Chairman) to Document Control Desk, U.S. Nuclear Regulatory Commission, Attention Joseph Holonich, "Project No. 704 - BWRVIP Response to NRC Request for Additional Information on BWRVIP-234, dated September 18, 2012. 4. Letter from Joseph Holonich (Senior Project Manager, Office of Nuclear Regulatory Regulation, U.S. Nuclear Regulatory Commission) REQUEST FOR ADDITIONAL INFORMATION FOR THE BOILING WATER REACTOR (BWR) VESSEL AND INTERNALS PROJECT BWRVIP-234, "THERMAL AGING AND NEUTRON EMBRITTLEMENT EVALUATION OF CAST AUSENITIC STAINLESS STEEL FOR BWR INTERNALS" (TAC NO. ME5060), dated April 24, 2013. 5. Letter from Dennis Madison (BWRVIP Chairman) to Document Control Desk, U. S. Nuclear Regulatory Commission, Attention Joseph Holonich, "Project No. 704 – BWR Response to NRC Request for Additional Information on BWRVIP- 234", dated May 23, 2014. 6. Letter from Tim Hanley (BWRVIP Chairman) to Document Control Desk, U. S. Nuclear Regulatory Commission, Attention Joseph Holonich, "Project No. 404 – Summary of Industry Position on Screening Criteria for Thermal and Irradiation Embrittlement for PWR and BWR Reactor Internals Fabricated of Cast Stainless Steel", dated March 9, 2015. 7. Letter from Kevin Hsueh, (Chief, Licensing Processes Branch, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission) to Tim Hanley (BWRVIP Chairman), "FINAL SAFETY EVALUATION OF THE BWRVIP -234: THERMAL AGING AND NEUTRON EMBRITTLEMENT OF CAST AUSENITIC STAINLESS STEEL FOR BWR INTERNALS (TAC NO. ME5060)", dated June 22, 2016. Details of the revisions can be found in Table G-1.

Table G-1
Revision details

Required Revision	Source of Requirement for Revision	Description of Revision Implementation
Add NRC Safety Evaluation (SE) behind title page	NRC request	Added NRC SE after Disclaimer page
Revise Section 3.5 of the report to reflect NRC SE Condition: "The licensee's plant specific fluence assessment of the six generic CASS components must demonstrate that the projected neutron fluence is bounded by the maximum fluence stated in Table 3-3 of the TR."	NRC SE Topical Report Condition	The following paragraph added to Sections 3.5, 6.7 and 7.0 of the report: "Note: In order to apply the inspection requirements for the CASS components shown in Table 6-1, (i.e., no additional inspections, beyond those currently required in BWRVIP I&E Guidelines, are needed to manage the aging effect of loss of fracture toughness of BWR reactor vessel internal components constructed of CASS), the utility must demonstrate by plant-specific fluence assessment, that the projected neutron fluence for their plant is bounded by the maximum fluence stated in Table 3-3."
Revise conclusion in Section 2.1.1 to state: "Based on this, the higher the aging temperature, the faster the aging effect reaches saturation. Since PWRs operate at a higher temperature than BWRs, thermal aging embrittlement effects for CASS components in a BWR are expected to occur later in life than for CASS components in a PWR."	BWRVIP commitment made in response to NRC RAI No. 2 (BWRVIP Correspondence File No. 2012-148)	Revised conclusion in Section 2.1.1 to state: "Based on this, the higher the aging temperature, the faster the aging effect reaches saturation. Since PWRs operate at a higher temperature than BWRs, thermal aging embrittlement effects for CASS components in a BWR are expected to occur later in life than for CASS components in a PWR."
Correct 2nd bullet on page 2-4 to state: "Statically-cast, low-molybdenum material (CF-8) with relatively high δ ferrite content ($\leq$ 20%) could be screened out from further evaluation."	Correction to original report (BWRVIP-234) as documented on page 9 of NRC RAIs (BWRVIP Correspondence File No. 2012-148)	Correct 2nd bullet on page 2-4 to state: "Statically-cast, low-molybdenum material (CF-8) with relatively high δ ferrite content ($\leq$ 20%) could be screened out from further evaluation."

Table G-1
Revision details (continued)

Required Revision	Source of Requirement for Revision	Description of Revision Implementation
Added remainder of NRC correspondence in Appendices B-F	NRC request	Added new Appendices B-F
End		

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