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Materials Reliability Program: Thermal Aging and Neutron Embrittlement Assessment of Cast Austenitic Stainless Steels and Stainless Steel Welds in PWR Internals (MRP-276)

LICENSED MATERIAL

Materials Reliability Program: Thermal Aging and Neutron Embrittlement Assessment of Cast Austenitic Stainless Steels and Stainless Steel Welds in PWR Internals (MRP-276)

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REPORT SUMMARY

This report summarizes the results of a review of the component items that could be affected by a synergy between thermal aging and irradiation embrittlement for the PWR internals of the Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse designs.

Background

The component items fabricated mainly from cast austenitic stainless steel (CASS) material or containing structural welds are listed in Materials Reliability Program (MRP) report MRP-189-Rev. 1 and MRP-191. Although these items were originally evaluated under the irradiation embrittlement limits set by MRP-175 (the MRP screening limit is 6 x 10^{20} n/cm^2, E > 1.0 MeV), the U.S. Nuclear Regulatory Commission (NRC) has proposed a more stringent screening criterion in the current generic aging lessons learned (GALL) report (1 x 10^{17} n/cm^2, E > 1.0 MeV), which might require the evaluation of additional items. The lower fluence screening value proposed in the GALL report is the result of concern about a potential synergistic effect of thermal and irradiation aging embrittlement in CASS.

Objectives

- To summarize testing results and assessments on the synergistic effects of thermal aging and irradiation on the fracture properties of CASS materials and austenitic stainless steel welds to date. A summary of the basis for the embrittlement screening criteria developed in MRP-175 will also be included. A list of reactor components that are below the MRP-175 fluence screening criterion of 6.7 x 10^{20} n/cm^2 (E > 1.0 MeV) but exceed the criterion of 1 x 10^{17}n/cm^2 (E > 1.0 MeV) suggested by the NRC will also be compiled.

- To assess the requirements for fracture property results to analyze the additional locations between the GALL criterion and the MRP-175 criterion. Aging times and irradiation conditions for CASS materials and welds identified will be estimated. In addition, structural requirements including operating stresses and any condition requiring fracture analysis will be identified.

- To identify any additional test data or structural analyses that are needed to fill the existing gaps in the available information. A *gap* is defined as a property requirement (identified in the previous objective) that cannot be adequately addressed using the information available from the first objective. A suggested test and analysis matrix will be developed.

Approach

As part of the preparation of this report, existing evaluations of thermal aging in CASS materials and stainless steel welds were supplemented with a summary of available assessment and testing data on both irradiated CASS materials and irradiated austenitic stainless steel welds. Next, a list

of reactor internals items that exceed the NRC criterion was developed. After potential requirements were identified for fracture toughness data on CASS and stainless steel welds in PWRs to complete the analysis for an aging management program, any gaps between available test data and analysis requirements were identified and prioritized. Finally, future testing and structural analysis recommendations—including test conditions—were made to close the identified gaps, and structural evaluations for these CASS components and for weld materials were proposed.

Results

The findings and recommendations are summarized in Section 6 of this report. It is clear from the available data that the NRC-recommended fluence screening criterion for a synergistic effect (that is, 1×10^{17} n/cm^2; E > 1.0 MeV) is overly conservative and that the MRP-175 screening criteria (that is, 6×10^{20} n/cm^2; E > 1.0 MeV) is sufficiently conservative. The conclusions of this study support a recommendation to withdraw GALL AMP XL.M13 to allow the establishment of requirements for aging management of CASS PWR internals items based on MRP-227 in GALL AMP XI.M16. Based on the findings of this study, it is evident that implementation of the MRP-227-Rev. 0 Guidelines provides appropriate aging management for irradiated CASS. The austenitic stainless steel welds represent a significant gap because of the lack of test data. The most significant gap in the available test data is the lack of data from specimens exposed to low (less than ~1 dpa) radiation doses and long thermal aging times at PWR operating temperatures. It is highly recommended that all plans for reactor decommissioning or component replacement be monitored to identify potential sources of CASS—especially austenitic stainless steel weld materials that would expand the current range of available fracture toughness data.

EPRI Perspective

Based on the results of this project, the MRP will decide if further work is needed, in particular, testing of irradiated CASS and welds. Recommended future structural analyses will be addressed in the Assessment Issue Task Group (ITG), as needed; recommended future testing will be addressed in the Mitigation and Testing ITG, as needed. The MRP will monitor all plans for reactor decommissioning or component replacement to identify potential sources of CASS, especially austenitic stainless steel weld materials, for the recommended future testing.

Keywords

PWR internals
Aging management
Irradiation embrittlement
Thermal embrittlement
Cast austenitic stainless steel (CASS)
Austenitic stainless steel welds

LIST OF ACRONYMS

AMP – Aging Management Program

ANL – Argonne National Laboratory

ASME – American Society of Mechanical Engineers

B&PV – Boiler and Pressure Vessel (B&PV) Code

B&W – Babcock & Wilcox

CASS – Cast Austenitic Stainless Steel

CRGT – Control Rod Guide Tube

dpa – Displacements Per Atom (a measurement of the accumulated dose)

E – Expansion, I&E Guidelines Category

EPRI – Electric Power Research Institute

EPFM – Elastic-Plastic Fracture Mechanics

FA – Fuel Assembly

FCA – Flux Cored Arc Weld

FN – Ferrite Number

GALL – Generic Aging Lessons Learned

GMA – Gas Metal Arc Weld

GTA – Gas Tungsten Arc Weld

HAZ – Heat-Affected Zone

HFR – High Flux Reactor

I&E Guidelines – Inspection and Evaluation Guidelines

IASCC – Irradiation-Assisted SCC

IE – Irradiation Embrittlement

IC – Irradiation-Enhanced Creep

IGSCC – Intergranular SCC

ISR – Irradiation-Enhanced Stress Relaxation

JOBB – Joint Owners Baffle Bolt Program

LEFM – Linear Elastic Fracture Mechanics

MMA – Manual Metal Arc Weld

MRP – Materials Reliability Program

MTIG – Manual Tungsten Inert Gas Arc Weld

N – No Additional Measures, I&E Guidelines Category

NRC – Nuclear Regulatory Commission

P – Primary, I&E Guidelines Category

PWR – Pressurized Water Reactor

RV – Reactor Vessel

SCC – Stress Corrosion Cracking

SA – Submerged Arc Weld

SMA – Shielded Metal Arc Weld

STIG – Semi-Automatic Tungsten Inert Gas Arc Weld

TE – Thermal Aging Embrittlement

VT – Visual Testing (Nondestructive Examination Technique)

X – Existing, I&E Guidelines Category

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INTRODUCTION

1.1 Purpose

The purpose of this report is to summarize the current state-of-knowledge of neutron irradiation-induced fracture toughness property changes in cast austenitic stainless steel (CASS) and austenitic stainless steel weld materials, principally Types CF-8, CF-3, CF-8M, and CF-3M cast materials and Types 308 and 308L weld metals. These grades are the most commonly used CASS and weld metals in the PWR internals.

This report will provide the data and the technical basis for the age-related embrittlement mechanisms (thermal embrittlement, irradiation embrittlement1, and any potential synergistic effects between these mechanisms) relative to fracture toughness associated with Pressurized Water Reactor (PWR) internals component items. This report was prepared under the direction and sponsorship of the EPRI Materials Reliability Program (MRP).

This report is a key element in an overall strategy for managing the effects of aging in PWR internals using knowledge of internals design, materials and material properties, and applying screening methodologies for the known age-related degradation mechanisms. Related MRP documents include the following:

- Framework and Strategy for Managing Aging Effects in PWR Internals [1-1]
- Inspection and Flaw Evaluation Strategies for Managing Aging Effects in PWR Internals [1-2]
- PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values [1-3]
- Pressurized Water Reactor Internals Inspection and Evaluation Guidelines [1-4]

1.2 Background

CASS and austenitic stainless steel weld materials in PWR vessel internals can experience hardening and embrittlement due to extended periods of exposure to elevated temperature and neutron fluence. This embrittlement must be recognized and monitored as part of age-related degradation management programs for the internals as an operating PWR unit enters the license renewal period. Embrittlement by itself is not failure or loss of functionality. The loss of fracture

[1] Irradiation Embrittlement is also referred to as Radiation Embrittlement by some authors, but both terminologies refer to the same degradation phenomenon.

toughness due to embrittlement cannot be detected by existing non-destructive examination techniques. However, embrittlement does cause a reduction in a material's resistance to cracking, thus lowering the flaw size that can be tolerated under normal or upset conditions. There is also concern that materials that experience thermal embrittlement and irradiation embrittlement concurrently have the potential to experience an increased level of embrittlement due to postulated synergistic effects between the two mechanisms.

The Nuclear Regulatory Commission (NRC) included requirements for thermal and irradiation embrittlement of CASS in NUREG-1801: "Generic Aging Lessons Learned (GALL) Report" [1-5]. Appendix A contains a summary of the GALL report requirements for CASS materials relative to thermal and irradiation embrittlement, which ultimately refer to acceptable aging management programs (AMP) described in Sections XI.M12 and XI.M13 of the GALL report. Section XI.M12 of the GALL report provides a program description of an AMP for thermal embrittlement of CASS materials in the reactor coolant system. Section XI.M13 of the GALL report provides a similar program description for CASS materials that incorporates potential effects of both thermal and irradiation embrittlement, which specifies that:

> *"The reactor vessel internals receive a visual inspection in accordance with ASME Section XI, Subsection IWB, Category B-N-3. This inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) reactor vessel internals. This aging management program (AMP) includes (a) identification of susceptible components determined to be limiting from the standpoint of thermal aging and susceptibility (i.e., ferrite and molybdenum contents, casting process, and operating temperature) and/or neutron irradiation embrittlement (neutron fluence), and (b) for each "potentially susceptible" component, aging management is accomplished though either a supplemental examination of the affected component based on the neutron fluence to which the component has been exposed as part of the applicants 10 year ISI program during the license renewal term, or a component-specific evaluation to determine its susceptibility to loss of fracture toughness."*

The GALL report provides the following guidelines as to the CASS components covered by the Section XI.M13 AMP:

> ***"Parameters Monitored/Inspected:*** *The program specifics depend on the neutron fluence and thermal embrittlement susceptibility of the component. The AMP monitors the effects of loss of fracture toughness on the intended function of the component by identifying the CASS materials that either have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) or are determined to be susceptible to thermal aging embrittlement. For such materials, the program consists of either supplemental examination of the affected component based on the neutron fluence to which the component has been exposed, or component-specific evaluation to determine the component's susceptibility to loss of fracture toughness."*

The current GALL report does not contain a similar requirement for thermal aging and irradiation embrittlement of austenitic stainless steel welds whose microstructure and susceptibility to thermal and irradiation embrittlement are similar to CASS. However, in MRP-175 the MRP has taken the conservative position to evaluate potential embrittlement of these welds in the PWR internals using the same metrics as used for CASS.

Castings having low operational stresses or compressive stresses are unlikely to crack in the PWR operating environment and will have a high tolerance for flaws. The NRC staff in NUREG-1801, Section XI.M13 acknowledges this by not requiring augmented inspections (beyond ASME Section XI, B-N-3) if castings can be shown to operate under compressive stresses or under low tensile stresses (< 5 ksi) during ASME Code Level A, B, C and D conditions by a component-specific evaluation. This exemption due to stress consideration is only for CASS components exceeding the 1017 n/cm2 (E>1.0 MeV) that are not susceptible to thermal aging embrittlement (i.e., it does not apply to materials that exceed the NUREG-1801, Section XI.M12 susceptibility requirements.). Such a definitive threshold stress value of less than 5 ksi by a component-specific evaluation is not provided by Section XI.M12 or Section XI.M13 for the following CASS components using a component-specific evaluation approach:

- CASS components potentially susceptible to thermal embrittlement, or
- CASS components potentially susceptible to thermal embrittlement and irradiation embrittlement (> 10^{17} n/cm^2, E>1.0 MeV)

Additionally, for austenitic stainless steel welds, there is presently more uncertainty. MRP-175 [1-3] notes that there is a paucity of welding material studies, especially in the PWR environment. And in contrast to CASS materials, some austenitic stainless steel welds are in higher fluence regions of the PWR internals (e.g., core barrel welds) and higher residual stresses exist as a result of the welding process. As a result, two concerns remain regarding the long-term performance of austenitic stainless steel welds. First, if there is a synergistic effect between irradiation and thermal aging embrittlement in these welds, the fracture toughness at low fluence levels may be below the prediction curves based on shorter-term irradiation data (<15 dpa). Second, when extended license period are considered, there are no existing high fluence (>15 dpa) fracture data to confirm that the lower bound fracture toughness for austenitic stainless steel welds will saturate (level off) in a similar way as the wrought austenitic stainless steel materials.

Chapter 2 provides a summary of the technical basis behind the screening criteria for thermal and irradiation embrittlement developed in MRP-175 for CASS and austenitic stainless steel welds. Chapter 3 provides a summary of recently available fracture toughness data and the results of recent assessments on the potential synergistic effects of thermal aging embrittlement and irradiation on the fracture properties of CASS and austenitic stainless steel weld materials. Chapter 4 provides an assessment of the requirements for fracture property results to analyze CASS and austenitic stainless steel welds in reactor internals of B&W, CE, and Westinghouse PWR designs. Chapter 5 contains data and assessment gap analyses for each of the vendor designs. Chapter 6 contains a summary of the results and conclusions from this effort and any recommendations for future testing. Appendix A contains a summary of current GALL requirements.

1.3 References

[1-1] Materials Reliability Program: Framework and Strategies for Managing Aging Effects in PWR Internals (MRP-134) - EPRI Report 1008203, 2005.

[1-2] Materials Reliability Program: Inspection and Flaw Evaluation Strategies for Managing Aging Effects in PWR Internals (MRP-153) – EPRI Report 1012082, 2005.

[1-3] Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175) – EPRI Report 1012081, 2005.

[1-4] Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Rev. 0) – EPRI, Palo Alto, CA: 2008. 1016596.

[1-5] Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Rev. 1, September 2005.

2
MRP-175 THERMAL AGING AND IRRADIATION EMBRITTLEMENT SCREENING CRITERIA AND BASIS

2.1 General Description of Thermal Aging Embrittlement

Thermal aging embrittlement (TE), sometimes simply known as thermal embrittlement, is a time and temperature dependent process whereby a material undergoes microstructural changes leading to decreased ductility, toughness, and impact properties. MRP-80 [2-1] is a document that was prepared for the MRP to summarize the available TE data for materials utilized in PWRs and to identify those materials potentially susceptible to TE. Much of the information provided in MRP-175 is contained in that summary report.

This phenomenon is usually accompanied by an increase in yield strength, ultimate tensile strength, and hardness. For the PWR internals, CASS and austenitic stainless steel welds are among the materials potentially susceptible to TE. CASS materials typically embrittle in the temperature range of 700 to 1000°F (371 to 538°C) within a short time. [2-2] For example, Charpy impact test data show that CASS TE reaches saturation after 2,600 hours at 752°F (400°C).[2-3] Although the data indicating thermal aging embrittlement in stainless welds is limited, they have been included in this study because they are known to exhibit duplex microstructures, similar to the CASS materials.

2.1.1 CASS

The most commonly used CASS materials in PWR internals are ASME SA-351 or ASTM A 296 Grades CF-3M and CF-8. They have a duplex microstructure consisting of austenite (gamma phase) and ferrite (delta phase). The delta-ferrite phase is susceptible to TE at reactor internals operating temperatures. The volume fraction of ferrite is typically 10 to 20%, but may attain 25%.

The mechanisms of TE that cause embrittlement in CASS have been reviewed and evaluated extensively by Chopra, Chung, et al. [2-4 – 2-11] of Argonne National Laboratory (ANL). Embrittlement or loss of toughness in CASS during elevated-temperature exposure is related to (1) the formation of the Cr-rich alpha-prime phase and the Ni-rich and Ti-rich silicide (G phase) in the delta-ferrite, and (2) the precipitation of carbides at the austenite-ferrite phase boundaries. The phase-boundary carbides play a significant role in embrittlement for exposure at temperatures $\geq$ 400°C (752°F), but have less effect on the embrittlement at exposure temperatures < 400°C (752°F).

Different heats of CASS may exhibit different degrees of property degradation depending on the amount, size, and distribution of ferrite in the duplex austenitic/ferrite structure and the presence of carbides at the grain boundaries.[2-12] When the ferrite tends to be interconnected (rather than being present as isolated islands), the potential loss of toughness is increased.[2-13] Low ferrite CASS alloys typically exhibit isolated islands of ferrite and therefore show less thermal aging susceptibility.[2-14] Figure 2-1 shows examples of CASS with two different ferrite levels.

For typical PWR internals temperatures [< 350°C (662°F)], the formation of the alpha-prime phase and the G phase in the ferrite are the primary factors involved in embrittlement. Also, Chopra and Chung report that the kinetics of the formation of these phases appears to be different at temperatures < 400°C (752°F). Because of these differences in formation and precipitation behavior, the results of tests on material subjected to accelerated aging at temperatures ≥ 400°C (752°F) are recommended not to be extrapolated to the lower temperatures.

The alpha-prime phase typically forms by the process of spinodal decomposition. Spinodal decomposition refers to the reaction whereby two phases of the same crystal lattice type, but different compositions and properties, form because a miscibility gap exists in the alloy system. In the iron-chromium system, these immiscible phases are known as the iron-rich alpha phase and the chromium-rich alpha-prime phase. This phase separation process occurs at a very fine scale (on the order of only a few nanometers) in the ferrite regions of cast stainless steel, and use of an atom probe field ion microscope is required to resolve the presence of the alpha-prime phase.[2-15] After long-term aging at PWR internals temperatures, there are indications that alpha-prime phase also can form by means of a nucleation and growth process (in addition to spinodal decomposition).[2-7] Whether one or both of these mechanisms occurs seems to depend on the composition of the ferrite and the exposure temperature.

G phase forms in the ferrite by a nucleation and growth process. Its rate of formation is enhanced by increased levels of carbon and molybdenum. [2-7] When the G phase is present, it appears to mitigate the degree of embrittlement caused by the alpha-prime phase. A cast stainless steel pump cover material (CF-8) was found to be embrittled (with a room temperature Charpy impact energy of 131 J/cm^2 [77 ft-lbs]) after 8 years of service at a temperature of 284°C (543°F) in a BWR. [2-11] Annealing for one hour at 550°C (1022°F) dissolved the alpha-prime phase and restored the Charpy impact resistance to the level expected for un-aged material (232 J/cm^2 [137 ft-lbs]), but had no effect on the G phase that was present. Thus, the G phase had no significant effect on the degree of embrittlement, and the alpha-prime embrittlement was easily reversed by a short heat treatment at a moderate temperature. This annealing treatment was used in the laboratory to verify that embrittlement was primarily caused by the alpha-prime, but impractical to apply to the assemblies containing the CASS component items inside the PWR internals.

Because only the ferrite phase is embrittled by long-term service at operating temperatures, the overall TE of a CASS component item depends on the amount and morphology of the ferrite that is present. In addition, molybdenum, being influential in the amount of ferrite and G phase formed, is considered the most important chemical element to TE. Therefore, screening criteria for CASS were developed based on temperature, molybdenum content, and ferrite levels. In the case of a CASS component item that is welded, test results have been shown that the heat-affected zone (HAZ) is potentially more susceptible to TE than the unaffected base metal.[2-16]

Most CASS components in the PWR internals are not subjected to high neutron fluence, although some CASS component items located at the edge of the reactor core may reach fluence levels on the order of 10^{20} n/cm^2, E > 1.0 MeV. The Nuclear Regulatory Commission (NRC) Staff has proposed the potential existence of a synergistic[2] effect of neutron irradiation on TE. As a result, Section XI.M13 of NUREG-1801 Rev. 01 defines CASS components with fluence >10^{17} n/cm^2, E>1.0 MeV as potentially susceptible to thermal embrittlement. However, this synergistic concept or the 10^{17} n/cm^2, E>1.0 MeV fluence was not based on irradiated CASS test data. Insufficient test data were available during the preparation of MRP-175 to prove or disprove this proposal, and a screening fluence of 6.7x10^{20} n/cm^2, E>1.0 MeV was applied to CASS and welded components (see Section 2.4).

2.1.2 Austenitic Stainless Steel Welds

The austenitic stainless steel weld deposits used in PWR internals, typically Types 308 or 308L, have a similar duplex microstructure to CASS material, but with a lower volume fraction of delta-ferrite (in the range of 5 to 15%, by ASME B&PV Code specifications but typically 5 to 10%) and notably lower chromium contents. The ferrite content is beneficial in preventing hot cracking and stress corrosion cracking, but it is a potential source of TE for austenitic stainless steel weldments.

The TE of austenitic stainless steel welds has been investigated by several researchers and the results are summarized in MRP-80.[2-1] Fracture toughness of austenitic stainless steel welds is found to be dependent on the weld process, but insensitive to filler metal.[2-17] Figure 2-2 shows the room temperature J_{Ic} of several commonly used stainless steel welds by five different weld processes including gas-tungsten-arc (GTA), shielded-metal-arc (SMA), submerged-arc (SA), gas-metal-arc (GMA) and flux-cored-arc (FCA).[2-18] Welds produced by the GTA process obtain the highest toughness values while welds produced by the SA process consistently have the lowest toughness values. This is mainly due to the fact that GTA welds have the lowest inclusion density due to the inert gas protecting the molten pool from oxygen and to the absence of a flux. No statistical difference is found between J-R curves for SA and SMA welds. The J_{Ic} fracture toughness data for GMA and SMA weld processes are intermediate.

Mills[2-19] performed a study where he investigated the TE of Type 308 welds fabricated using the GTA welding process. The ferrite content of the weld metal was about 10 ferrite number (FN).[3] Mills' results indicate that the fracture toughness of the welds was not affected by aging at 427°C (800°F) for 10,000 hours. However, other research results show that thermal aging may cause a reduction in both the impact energy and fracture toughness of SMA welds. Hale and Garwood[2-20] investigated the thermal aging of Type 19-9-L austenitic welds made by a manual-metal-arc (MMA) welding process. The ferrite content of the weld metal was in the range of 5 to 9 FN. Their results show that aging at 400°C (730°F) for 10,000 to 20,000 hours

[2] The word synergistic, in this case, refers to the possibility that the effects of neutron irradiation and thermal aging could be greater than the sum of the effects from each mechanism considered individually.

[3] 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.

had little effect on the room temperature tensile properties and Charpy impact energy of these welds, but resulted in a significant increase in the ductile-to-brittle transition temperature measured at the 27-J energy level, which was increased from -158 to -75°C (-252 to -103°F), an increase of 83°C (150°F). The results also show that TE at 400°C (752°F) for up to 10,000 hours reduces the J-R fracture toughness of the weld metal. The lower bound fracture toughness, J_{IC}, measured at 300°C (572°F) was reduced from 67 to 32 kJ/m^2. Most of this reduction in fracture toughness took place in the first 1,000 hours.

Alexander et al. [2-21] also investigated the TE of Type 308 weld material fabricated using the SMA welding process. The ferrite content of these welds was 12% (by volume). Their results also show that aging of these welds at 343°C (650°F) for 20,000 hours resulted in a minimal effect on the room temperature tensile properties but caused a significant increase in the ductile-to-brittle transition temperature measured at the 68-J energy level, an increase from -25 to 60°C (572°F) or 85°C (153°F). However, O'Donnell et al. [2-18] have argued that a significant reduction in fracture toughness is likely when the ferrite volume fraction exceeds 10%. It appears that there may be a synergistic interaction between the embrittled ferrite phase and inclusions in the SMA welds. Further investigation is needed of the thermal aging behavior of SMA welds having a ferrite content representative of that in production welds.

In summary, although no significant TE is anticipated for the typical delta-ferrite levels (5-10%) in austenitic stainless steel weld deposits, a significant reduction in fracture toughness is likely if the ferrite volume fraction exceeds a volumetric level of ~10%. Unless there is a synergistic effect between neutron dose and TE, TE is unlikely to become significant for these austenitic stainless steel weld locations with relatively high fluence.

2.2 Thermal Embrittlement Screening Criteria

Temperature and time at temperature are the overriding environmental parameters controlling TE. Based on recognized industry efforts [2-22, 2-23], the following CASS alloys have been determined to be susceptible to loss of toughness and are to be screened for TE by MRP-175:

- Centrifugal castings with > 20% ferrite
- Static castings with molybdenum content ≤ 0.50% and ferrite > 20%
- Static castings with molybdenum content > 0.50% and ferrite >14%

Austenitic stainless steel weld materials show a wide range of tensile and fracture toughness properties in the as-welded and un-aged condition. Their microstructure and aging susceptibility are superior to static stainless steel castings due to lower molybdenum, lower chromium, and lower delta-ferrite content. In addition, austenitic stainless steel welds (e.g., Types 308 and 308L) are similar to statically-cast CASS with low molybdenum contents and it was suggested that the same screening criteria be used. Thus, utilizing the same screening criteria as suggested for statically-cast CASS materials, no weld metal will fall above the ferrite screening level (i.e., >20%) due to the lower ferrite content. Provided that the levels of ferrite and molybdenum contents are low in these welds, it is expected that the effects of TE will be insignificant.

Table 2-1 provides the screening criteria that were developed in MRP-175 for use by the MRP.

2.3 General Description of Irradiation Embrittlement

Irradiation embrittlement (IE) refers to the phenomenon of loss of ductility and fracture toughness from exposure to high-energy neutrons (E > 1.0 MeV). The loss of ductility and fracture toughness is usually accompanied by marked increases in yield and ultimate tensile strength. Mechanistically, irradiation embrittlement results from lattice defects from neutron bombardment. High-energy neutrons displace atoms from their normal lattice positions creating interstitial atoms and vacant lattice sites (also known as point defects). Although most point defects are annihilated by recombination, surviving point defects form various irradiated microstructures consisting of dislocations, precipitates, and cavities. Cavities, which are three-dimensional clusters of vacancies, gas atoms (bubbles), or a combination of the two, can be associated with other microstructural features such as precipitates, dislocations, and grain boundaries. These defects and precipitates from irradiation are obstacles to dislocation movement and result in increased yield and tensile strengths and decreased work hardening capacity and ductility, and loss of fracture toughness.

The detrimental effect of irradiation embrittlement has long been recognized for the low alloy steel PWR and BWR reactor pressure vessel materials. However, until recently, irradiation embrittlement had not been considered to be a concern for PWR internals. This is partly because the PWR internals are constructed of austenitic stainless steels, which possess high levels of ductility and fracture toughness even after neutron exposure levels that would have caused significant embrittlement in low alloy steel reactor vessels. However, the proximity to the core means that the neutron flux for many lower PWR internals component items is one to three orders of magnitude higher than for the low alloy steel PWR reactor vessel. A large reduction in fracture toughness of stainless steels due to neutron irradiation can significantly increase the sensitivity to flaws that are either pre-existing during PWR construction or flaws developed during service due to SCC, IASCC, or fatigue. Decreasing toughness values correlate to decreasing critical crack lengths that can be tolerated by the structure. This affects inspection requirements and procedures.

MRP-79 Rev. 1 [2-24] is a document that was prepared for the MRP to summarize the available IE data for materials utilized in PWRs and to identify those materials potentially susceptible to IE. Much of the information provided in MRP-175 is contained in that summary report. In the present discussion of IE, the affected materials are limited to austenitic stainless steel welds and CASS.

2.3.1 CASS Materials in Fast Reactors

At the time the MRP-79 technical basis was developed, the only evaluation of fracture toughness of fast reactor irradiated CASS material was reported by Mills.[2-25] A CF-8 casting was irradiated in the Experimental Breeder Reactor II (EBR-II) at 400 to 427°C (752 to 801°F) to 19 dpa ($6.0x10^{22}$, E > 0.1 MeV). For the J-R specimen test at 427°C (801°F), only one of the three data points was located within the ASTM recommended exclusion limits; the other two data points fell beyond the maximum exclusion limit of 1.5 mm. While an accurate measure of J_c could not be calculated, the results were generally consistent with the irradiated Type 308 weld data that resulted in a J_c of 11 kJ/m^2 (see Figure 2-3).

2.3.2 CASS Materials in PWRs and BWRs Reactors

At the time of publication of MRP-79, no known CASS material fracture toughness test results were available for PWR or BWR irradiation.

2.3.3 Austenitic Stainless Steel Weld Metals in Fast Reactors

A review of Types 308 and 316 austenitic stainless weld metals irradiated in fast reactors[2-19] was compiled in the MRP-79 report. It was concluded that the irradiation embrittlement of weld metals is similar to that for wrought austenitic stainless steels, i.e., fracture toughness deteriorates rapidly with the first few dpa of exposure and then gradually levels off. Exposures up to 1 dpa have no significant effect on fracture resistance. Beyond 1 dpa, fracture resistance diminishes more rapidly than in the base metal because the embrittled delta-ferrite serves as an effective microvoid nucleation site. Neutron exposures of 2-4 dpa are seen to cause a 70-90% reduction in J_{Ic}. [2-26 – 2-28] At exposures above 10 dpa, saturation J_{Ic} values range from 10-30 kJ/m^2.[2-25, 2-29]

Fracture toughness saturates after 10 dpa; however, the maximum fluence for the weld metal data points is limited to about 15 dpa. The minimum elevated temperature K_{Jc} fracture toughness was estimated to be 40 MPa√m (36 ksi√in), which is lower than the minimum fracture toughness for highly irradiated wrought austenitic stainless steels in fast reactors. Fracture surfaces for highly irradiated welds exhibit channel fracture with small microvoids superimposed on the crystallographic facets.[2-25] The microvoids are nucleated by failure of embrittled delta-ferrite particles, but they cannot develop into dimples because of restricted plastic deformation capabilities after irradiation. These small holes acting as stress concentrators prematurely nucleate channel fracture, which causes saturation J_{Ic} or K_{Jc} levels for welds to fall below base metal values. At these toughness levels, 1-2 cm (0.4-0.8 in) thick welded components possess sufficient constraint to induce plane strain fracture conditions.

2.3.4 Austenitic Stainless Steel and Weld Metals in PWRs and BWRs

The MRP-79 report also provided the then currently available elevated temperature fracture toughness data of austenitic stainless steels and welds irradiated in BWRs or PWRs. The results are summarized in Figure 2-4. Analysis shows that the reduction of fracture toughness with increasing neutron dose in BWRs and PWRs is consistent with that observed in fast reactors. However, a few of these data points at relatively high fluence (between about 2 and 18 dpa) fall below the scatter band for stainless steels irradiated in fast reactors, but no consistent trend between thermal and fast reactors at equivalent doses (expressed in dpa or n/cm^2) has been observed. The data indicate that for neutron fluence exposures less than 0.5 dpa, only a limit load evaluation is necessary in order to support continued service. For dose levels up to approximately five dpa, Elastic-Plastic Fracture Mechanics (EPFM) can be considered for design and operational analyses. For higher dose levels, it is recommended that Linear-Elastic Fracture Mechanics (LEFM) analyses be considered. When the neutron exposures are greater than five dpa and less than 15 dpa, LEFM analyses can be used with a limiting fracture toughness (K_{Ic}) of 55 MPa√m (50 ksi√in). At irradiation exposures equal to or greater than 15 dpa, a conservative lower bound fracture toughness of 38 MPa√m (35 ksi√in) is recommended. It is currently believed by many researchers that the difference may be linked to the formation of large numbers of nanometer size gas bubbles (due to the insolubility of helium in metals) during LWR neutron irradiation, which occurs to a much lesser extent in fast reactors.

2.4 Irradiation Embrittlement Screening Criteria

Significant scatter is observed in the loss of fracture toughness between 0 and 0.5 dpa for austenitic stainless steel welds and cast austenitic stainless steels. This scatter in the initial onset of irradiation embrittlement reflects the differences in initial thermomechanical conditions and heat-to-heat variations.

Fracture toughness decreases rapidly with increasing neutron dose for neutron exposures between 0 and 10 dpa as shown in Figure 2-5. The following equation [2-34] was developed to bound all available fracture toughness data from fast reactors, BWRs, and PWRs. This lower bound fracture toughness line is included in a plot of all the available data in Figure 2-4.

$$\text{Lower Bound } K_{Jc}\left(\text{MPa}\sqrt{\text{m}}\right) = 180 - 142 * \left[1 - \exp\left(-\,dpa\right)\right] \qquad \textbf{Equation 2-1}$$

References 2-30 and 2-31, also confirmed by References 2-32 and 2-33, indicate that for austenitic stainless steel component items with neutron exposures > $3x10^{21}$ n/cm^2, E > 1.0 MeV [> 4.5 dpa] LEFM (in place of EPFM) should be considered for design and operational analyses. In addition, a number of material properties (e.g., yield strength and tensile strength) plateau when exposed to doses in the range of 5-10 dpa. Thus, based on equation 2-1 and Figure 2-4, at 5 dpa ($3.3x10^{21}$ n/cm^2, E > 1.0 MeV) the corresponding lower bound fracture toughness is about 40 MPa√m (36 Ksi√in), which should be adequate toughness for functionality. Also, as shown in Figure 2-4, there is a reasonable database available to establish this minimum value. However, for added assurance in performing the screening, 30% of the 5 dpa value (or 1.5 dpa) was suggested. Therefore, for wrought austenitic stainless steel, a screening neutron exposure for irradiation embrittlement was conservatively established to be $\geq$ 1.5 dpa, or $1x10^{21}$ n/cm^2, E > 1.0 MeV.

Because the austenitic stainless steel weld metals and CASS show much greater variability in initial values, their screening neutron exposure was conservatively established to be somewhat lower at $\geq$ 1 dpa or $6.7x10^{20}$ n/cm^2, E > 1.0 MeV. In addition, this screening criterion was suggested for use in evaluation of the potential synergistic effect of dose on thermal aging embrittlement.

Table 2-2 provides the screening criteria that were developed in MRP-175 for use by the industry.

2.5 References

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Table 2-1
MRP-175 Thermal Aging Embrittlement Screening Criteria for PWR Internals Materials

<table>
<tr><th rowspan="2">Material</th><th colspan="2">Criteria</th></tr>
<tr><th>Parameter[a]</th><th>Value</th></tr>
<tr><td>CASS
(Centrifugal Castings)</td><td>Ferrite</td><td>> 20%</td></tr>
<tr><td rowspan="2">CASS
(Static Castings)</td><td>Molybdenum
and
Ferrite</td><td>≤ 0.50%
and
> 20%</td></tr>
<tr><td>Molybdenum
and
Ferrite</td><td>> 0.50%
and
> 14%</td></tr>
<tr><td rowspan="3">Austenitic SS Welds[b]</td><td>Molybdenum
and
Ferrite</td><td>≤ 0.50%
and
> 20%</td></tr>
<tr><td>Molybdenum
and
Ferrite</td><td>> 0.50%
and
> 14%</td></tr>
<tr><td colspan="2">TE is not anticipated as an issue due to ASME Code procurement requirements for low levels of ferrite (5-15%) and low Mo levels.</td></tr>
</table>

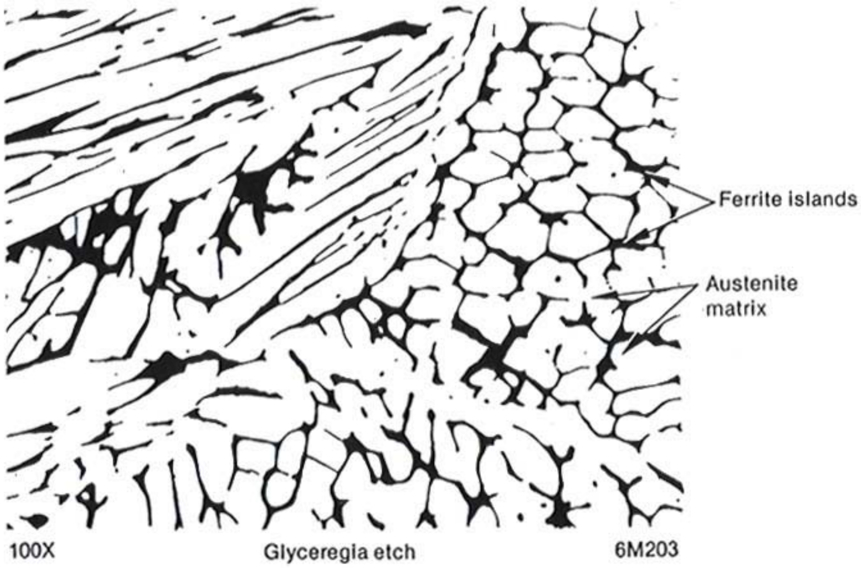
Notes:

a. Temperature of PWR internals is sufficient, so no screening value is necessary, particularly those locations subject to significant gamma heating.

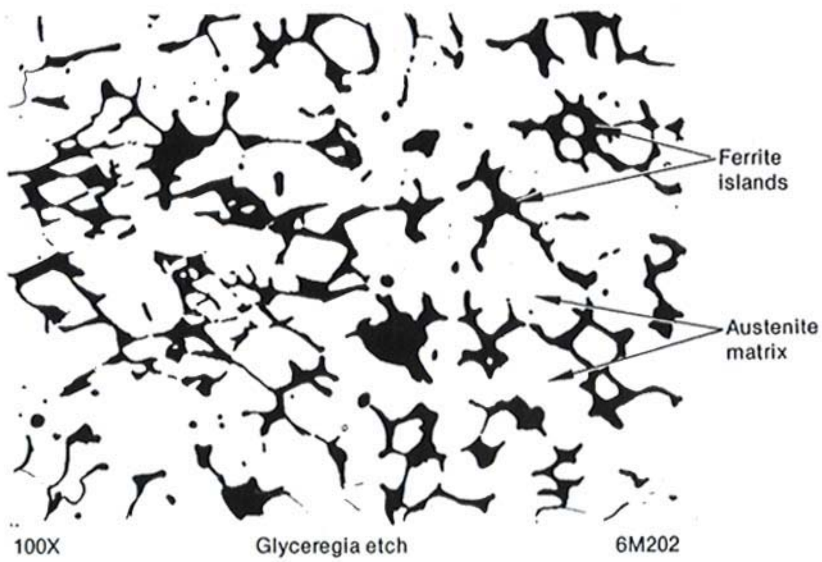
b. The same criteria suggested for static castings have been applied to austenitic stainless steel welds.

Table 2-2
MRP-175 Irradiation Embrittlement Screening Criterion for PWR Internals Materials

Material	Criterion	
	Parameter	Value
Austenitic SS Welds	Dose	$\geq 6.7 \times 10^{20}$ n/cm^2, E > 1.0 MeV [$\geq$ 1 dpa]
CASS		



a. Grade CF-8 stainless steel (16% ferrite)



b. Grade CF-8M stainless steel (30% ferrite)

Figure 2-1
Typical Microstructures of Centrifugally Cast Austenitic Stainless Steel, with Islands of Ferrite in an Austenite Matrix [2-14]

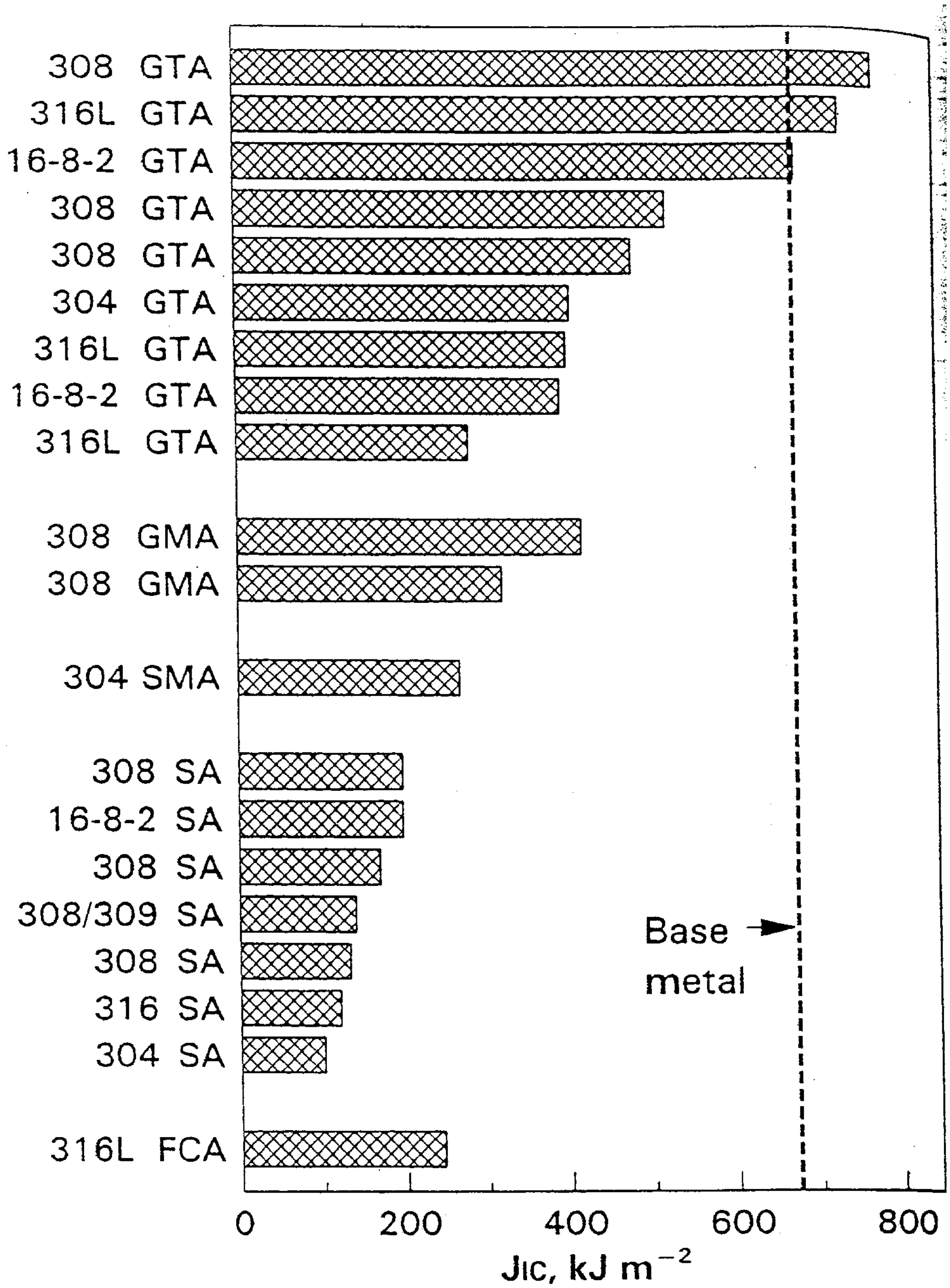
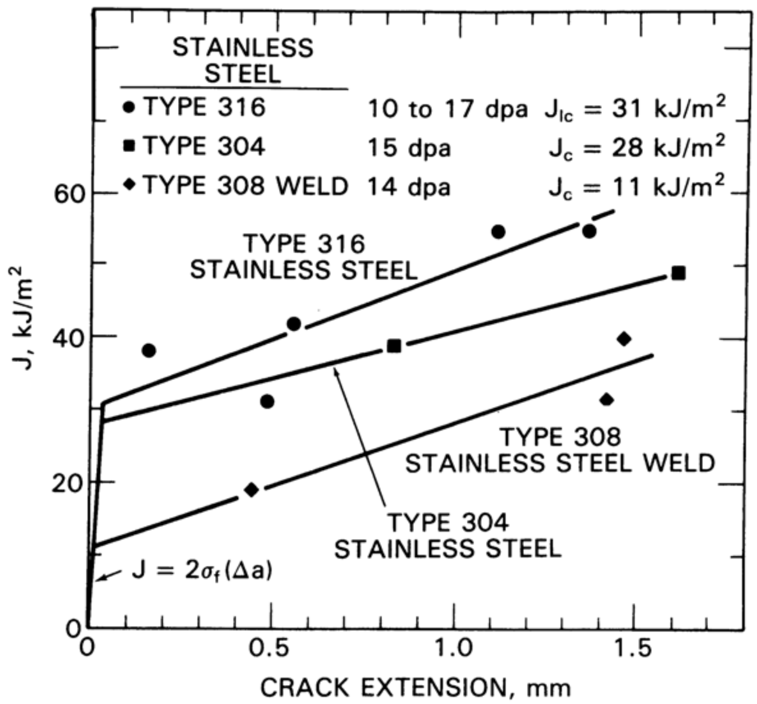
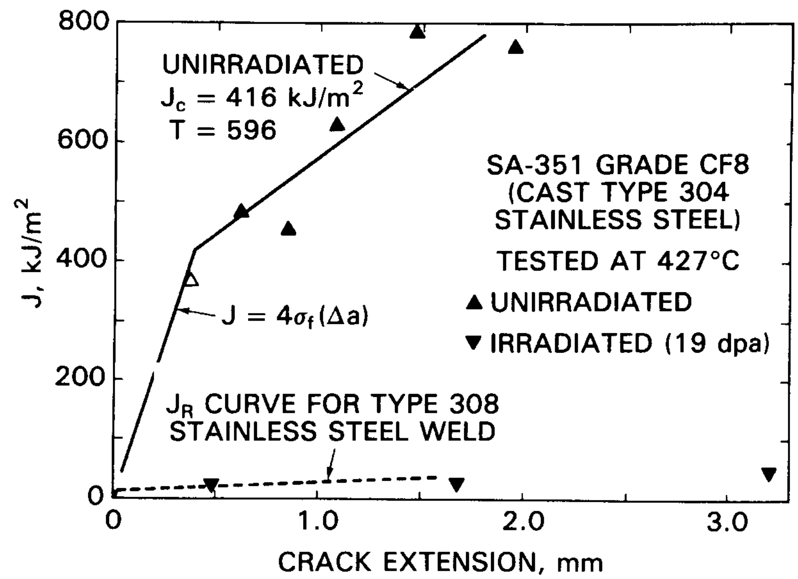


Figure 2-2
Comparison of J_{IC} of Unirradiated Stainless Steel Weld by Different Processes [2-15]

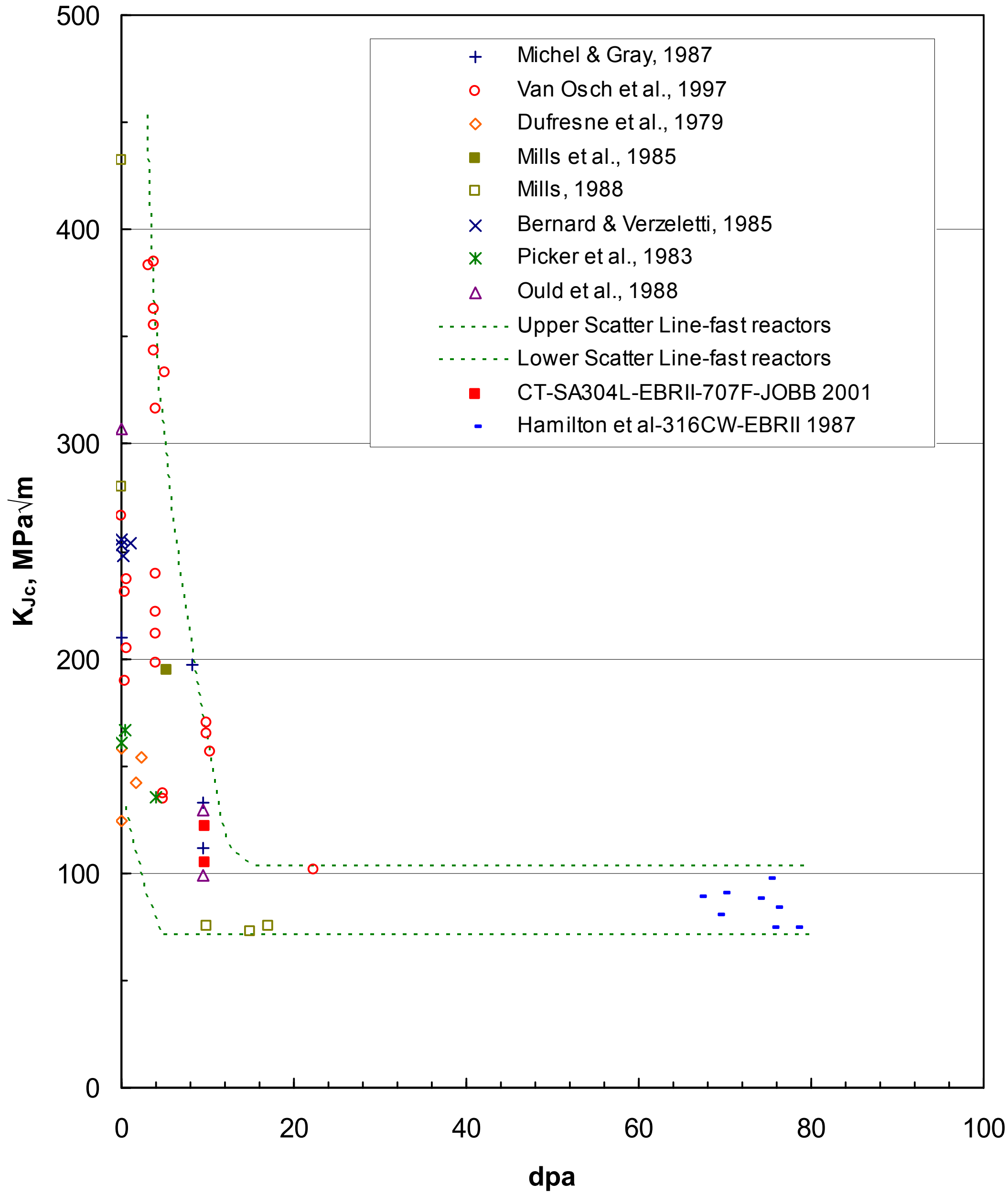


(a)



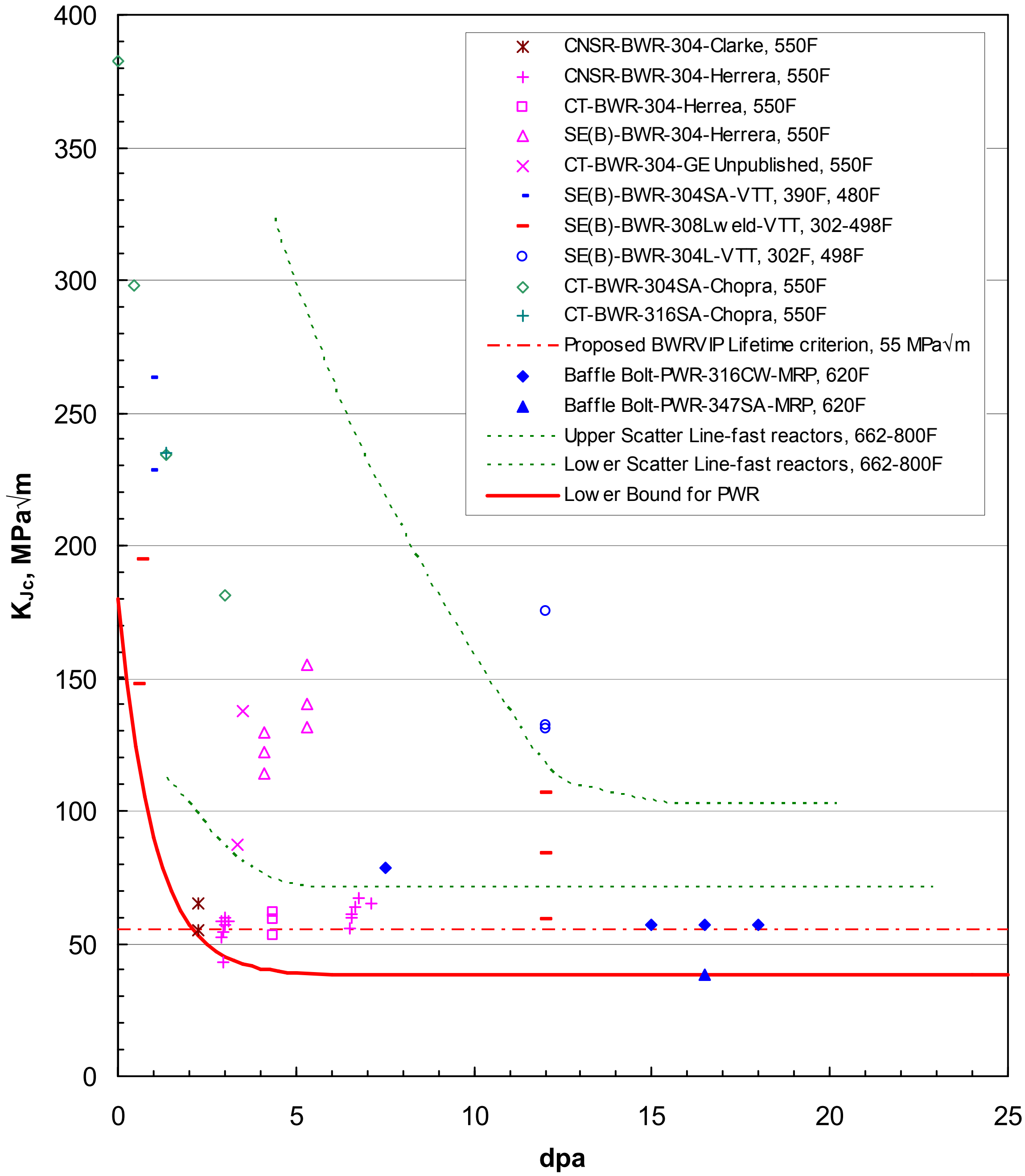
(b)

Figure 2-3
J-R Curves at 427°C (801°F) for Fast Reactor Irradiated Materials: (a) Type 308 and (b) CF-8. The cast material results are consistent with the overall fracture behavior for the Type 308 weld, as represented by the dashed JR curve in (b). [2-25]



Irradiation temperature 350–450°C (662–842°F)
Test temperature 205– 427°C (401–801°F)

Figure 2-4
Elevated Temperature Fracture Toughness K_{jc} as a Function of dpa for Type 304 and Type 316 Irradiated in Fast Reactors [2-24]



The test temperature is °F indicated.
Dashed lines represent the fast reactor scatter band
The lower bound for PWR (thick red solid line) is per Eq. 2-1.

Figure 2-5
Elevated Temperature Fracture Toughness K_{Jc} of Austenitic Stainless Steels and Welds Irradiated in BWRs or PWRs as a Function of dpa [2-34]

3 RECENTLY ACQUIRED FRACTURE TOUGHNESS DATA FOR IRRADIATED THERMALLY AGED CASS AND AUSTENITIC STAINLESS STEEL WELD MATERIALS AND SYNERGISTIC ASSESSMENT

3.1 Fracture Toughness Test Data Source 1

Recent work by Chopra et al. [3-1, 3-2] reported the fracture toughness test data of thermally aged CASS material (CF-8M) that was irradiated in a helium environment in the Halden heavy water BWR. The CF-8M material was obtained from a statically cast plate with a ferrite content of ~28%. Prior to irradiation, the material was thermally aged for 10,000 hours at 400°C (752°F), which would be expected to saturate the thermal aging propensity of this material. The material was irradiated to $1.63x10^{21}$ n/cm^2, E > 1.0 MeV [~2.46 dpa] at ~297°C (567°F). The fracture toughness tests were conducted in a simulated normal-water chemistry (NWC) BWR environment at 289°C (552°F).

Two tests were performed on the CF-8M (Heat 75) material and the J_{Ic} values were reported to be 40 and 84 kJ/m^2, indicating significant embrittlement to this thermally aged and irradiated material. The J-R curves for the two tests are shown in Figure 3-1.

The available fracture toughness data in the open literature were reviewed for austenitic stainless steels and welds. The change in J_{Ic} of CASS and weld metals is shown in Figure 3-2 as a function of neutron exposure. The fracture toughness data from both fast reactor and LWR irradiations are included in the figure. The procedures for determining J_{Ic} varied among these studies. These data indicated that the toughness of CASS and welds is lower than that observed of similarly irradiated wrought stainless steels for all fluences less than 10 dpa [$6.7x10^{21}$ n/cm^2 E > 1.0 MeV], which is considered a saturation level for fracture toughness.

Based on the data in Figure 3-2, Chopra et al. proposed that ~0.3 dpa [~$2x10^{20}$ n/cm^2, E > 1.0 MeV] can be considered a neutron dose below which irradiation has little or no effect on fracture toughness. Similarly, Chopra et al. proposed that the fracture toughness of austenitic stainless steels irradiated at less than this dose has a minimum J_{Ic} of 135 kJ/m^2 (771 in-lb/in^2).

Based on the CF-8M and weld test data and a review of literature data, Chopra et al. concluded that any potential synergistic effects on the minimum toughness would occur only for fluences greater than 0.3 dpa [~$2x10^{20}$ n/cm^2, E > 1.0 MeV]. Below 0.3 dpa, the minimum toughness can be estimated from the correlations available for thermal embrittlement of CASS. For fluences greater than 0.3 dpa, the minimum fracture toughness of CASS can be assumed to be given by the lesser of the minimum predicted toughness for thermal aging or the lower bound curves for the fracture toughness of irradiated stainless steels.

3.2 Fracture Toughness Test Data Source 2

Horsten and Belcher [3-3] conducted a test program to evaluate the effects of irradiation, thermal aging, specimen orientation, and test temperature on the fracture toughness of Type 308L stainless steel strip clad deposit material. Specimens were irradiated in the high flux reactor (HFR) at Petten (in the Netherlands) to doses of ~0.05 and ~0.1 dpa (~3.2×10^{19} and ~5.2×10^{19} n/cm^2, E > 1.0 MeV) at 295°C (562°F). Testing was performed at room temperature, 100°C (212°F), 200°C (392°F), and 295°C (562°F). The fracture properties of the clad material were unaffected by irradiation up to 5.2×10^{19} n/cm^2, E > 1.0 MeV (< 0.1 dpa), in agreement with the conclusions of Chopra (see Section 3.1).

Thermal aging [1000 hours at 400°C (752°F)] had no significant effect on the fracture behavior in either the irradiated or unirradiated conditions tested at 100°C (212°F) and 295°C (562°F) as illustrated in Figure 3-3. Clad fracture properties were also unaffected by orientation in the plane of the clad layer. Fracture toughness was reported in terms of $J_{0.2bl}$ (in accordance with the European Structural Integrity Society Procedure ESIS P2-92) [3-4].

Fracture toughness results were concluded to be independent of irradiation, thermal aging, and specimen orientation, and only dependent upon test temperature (for the conditions tested). The mean and lower bound values of $J_{0.2bl}$ (with 90 percent prediction intervals) are given in Table 3-1 and the results are shown in Figure 3-4. Therefore, it was concluded that the fracture properties of Type 308L clad material are unaffected by irradiation to approximately 0.1 dpa (~5.2×10^{19} n/cm^2, E > 1.0 MeV) at 295°C (562°F) and thermal aging for 1,000 hours at 400°C (752°F).

3.3 Fracture Toughness Test Data Source 3

Kim et al. [3-5, 3-6] reported the fracture toughness data test results from a number of sources for CASS and austenitic stainless steel weld metals. The materials investigated were the CASS grades CF-3 and CF-8, and Type 308 weld metal.

A fuel assembly component item (top nozzle clamp) fabricated from CF-3 material with approximately 10% ferrite was removed from service after two fuel cycles (3 to 4.5 years) of operation in a commercial PWR unit at 320°C. It had been exposed to a neutron fluence estimated to be in the range of 6 to 10×10^{19} n/cm^2, E > 1.0 MeV.

Additional CASS (CF-8) and weld (Type 308 gas-tungsten-arc weld) materials were obtained from the Joint Owners Baffle Bolt (JOBB) program.[3-7] These specimens were irradiated in the Boris test reactor to a neutron fluence of 6 to 12 dpa (3.7 to 7.4×10^{21} n/cm^2, E > 1.0 MeV). Because only moderate thermal aging was expected under the Boris test reactor irradiation conditions (~1 year at 325°C or 617°F), some of the JOBB specimens were thermally aged prior to irradiation. All of the pre-irradiation aging was conducted at 400°C (752°F). Two of the specimens were aged for 950 hours (aged) and three specimens were aged for 100 hours (semi-aged). The ferrite content is 16-20% for the JOBB CASS and 5-10% for the JOBB weld.

The fracture toughness results from these materials were compared to results and assessments from a number of studies conducted at Argonne National Laboratory.[3-1, 3-8, 3-9] Figure 3-5 represents the thermal aging embrittlement models developed by Chopra [3-8] and used by [3-5, 3-6] to predict the J-R curves for a CF-8 CASS with FN=15 aged for 3 years at 320°C (610°F). The bands in the figure are defined by [3-5, 3-6] to indicate varying levels of thermal embrittlement and irradiation embrittlement. The toughness values represented by the upper bands of the figure indicate that the material had only experienced thermal embrittlement. Toughness represented by the lower bands indicates that the material had been embrittled well beyond the level that could occur by thermal embrittlement alone. The middle bands indicate that the toughness decrease could be the result of a combination of the two embrittlement mechanisms.

The results of the tests described in [3-5, 3-6] were compared to these bands to determine the level of irradiation embrittlement indicated by the measured toughness. Figure 3-6 and Figure 3-7 show the results of this analysis for the CF-8 and CF-3 CASS materials, respectively.

The J-R results shown in Figure 3-6 for CF-8 material irradiated to a dose of 6 to 12 dpa indicated that this material had been embrittled beyond the expected level for thermal embrittlement alone. Fracture toughness values, J_{Ic} (reported as J_Q), range from 57-129 kJ/m^2 and are provided in Table 3-2. The loss of fracture toughness in the irradiated CF-8 material was attributed to normal reductions in toughness due to irradiation. The J-R curves for the JOBB CF-8 specimens are shown in Figure 3-8, which shows no evidence of a synergistic effect. Loss of fracture toughness correlates mainly with the irradiation dose, but does not correlate to the aging treatment conditions (no aging, semi-aging, or full aging) prior to the irradiation.

The J-R results for the CF-3 material irradiated to a dose of 0.08 dpa are shown in Figure 3-7. Fracture toughness values, J_{Ic} (reported as J_Q), range from 370-688 kJ/m^2 and are provided in Table 3-2. For this material, the results indicated that there was little to no effect of the irradiation exposure on the fracture toughness. The measured values were consistent with the predictions of the thermal embrittlement model. The absence of an irradiation effect on fracture toughness at the lower level dose of 0.08 dpa is in agreement with test results by Chopra et al. in Section 3-1 and Horsten and Belcher in Section 3-2.

It was concluded by the authors that the results of the testing indicated that there was no evidence of a synergistic effect of irradiation embrittlement and thermal aging for CASS over the range of temperatures and irradiation doses tested. The only weld metal specimen tested in the program exhibited a toughness value consistent with an irradiated stainless steel specimen. This one data point for weld material indicates that there may not be a synergistic effect for the weld metals. However, it was noted that more testing will be required to support such a conclusion on the synergistic effect. The authors indicated that the most significant gap in the data available for addressing the synergy between irradiation and thermal aging is the lack of data from specimens exposed to low (less than ~1 dpa) radiation doses and long thermal aging times. This is represented graphically in Figure 3-9. The shaded areas in Figure 3-9 define regions where data exist, while the unshaded regions are regions where little or no data exist. There is no empirical evidence of a synergistic effect of irradiation and thermal aging. Similarly, there is no empirical evidence on which to reject this hypothesis in the thermal exposure-irradiation exposure regimes represented by the unshaded regions of Figure 3-9.

3.4 Summary of Fracture Toughness Synergistic Assessments

As noted in Section 2, MRP-79[3-10] summarized the then currently available test data for irradiation embrittlement of stainless steel alloys. Figures 3-10, 3-11, and 3-12 clearly show that the fracture toughness of stainless steel alloys deteriorates rapidly in the first few dpa and then gradually levels off (saturates) after approximately 10 dpa. A similar figure can be found in the JOBB program summary,[3-7] as shown in Figure 3-13. It can also be seen that the minimum fracture toughness value for highly irradiated weld materials is approximately 10 kJ/m^2 (equivalent to a K_{Jc} value of 40 $MPa\sqrt{m}$), which is a bit lower than the minimum observed for highly irradiated wrought stainless steels in fast reactors. It was concluded in MRP-79 that fracture toughness saturation is a direct result of the saturation of irradiation defect microstructure, which is largely independent of the starting conditions.

The thermal aging and irradiation embrittlement mechanisms have been well established as discussed in Section 2. At typical PWR internals temperatures, thermal embrittlement is mainly due to formation of the alpha-prime phase from spinodal decomposition in the ferrite of CASS and austenitic stainless steel welds. Inclusions are also a significant contributing factor to the fracture toughness of austenitic stainless steel welds. On the other hand, irradiation embrittlement is due to lattice defects from neutron bombardment. There are no test data or mechanisms to suggest that the original ferrite phase amount, size, and distribution or alpha-prime precipitation in the ferrite would be significantly altered by irradiation, especially at a neutron dose level of $1x10^{17}$ n/cm^2, E > 1.0 MeV. The available test data clearly indicate that this level of fluence does not change the fracture toughness of CASS or welds that have been completely thermally aged.

Fracture of thermally embrittled CASS and welds is by brittle cleavage fracture of ferrite or separation of the ferrite/austenite phase boundaries. Inclusions also play a significant role in the fracture toughness of austenitic stainless steel welds. Fracture of highly irradiated CASS and welds on the other hand is by channel fracture, which exhibits localized plastic deformation. Therefore, prior to the formation of significant irradiation defects as evidenced by a detectable loss of fracture toughness due to irradiation embrittlement, there is no mechanism for the cleavage fracture of embrittled ferrite to be affected by irradiation defects. The minimum fluence level for a detectable fracture toughness loss due to irradiation embrittlement has been well established to be >0.5 dpa ($3.3x10^{20}$ n/cm^2, E > 1.0 MeV) [3-10] in austenitic stainless steels. This level of irradiation is much higher than the screening threshold of $1x10^{17}$ n/cm^2, E > 1.0 MeV as suggested by NRC to account for a potential synergistic effect.

The available test data have not shown any evidence to indicate a synergistic effect of thermal aging and irradiation. Given the wide scatter of fracture toughness values related to thermal and irradiation embrittlement, it has been the general practice to only use a minimum or lower bound fracture toughness in fracture mechanics based evaluations. As Chopra et al. recently concluded (see Section 3.1), the minimum fracture toughness below 0.3 dpa can be estimated by considering thermal embrittlement alone, and the minimum fracture toughness above 0.3 dpa can be estimated by using the lesser of the minimum predicted thermal embrittlement or the lower bound irradiated fracture toughness curve.

Nevertheless, there has not been sufficient testing to disprove any synergistic potential under all possible combinations of different thermal aging conditions and irradiation conditions. The most significant gap in the test data is the lack of data from specimens exposed to low (less than ~1 dpa) radiation doses and long thermal aging times as the thermal embrittlement process for laboratory testing is typically accelerated (at 400°C) above normal PWR operating temperatures. This is represented graphically in Figure 3-9. The shaded areas in Figure 3-9 define regions where data exist, while the unshaded regions are regions where little or no data exist. As shown by the information presented earlier in this chapter, there currently is no empirical evidence of a synergistic effect of irradiation and thermal aging. Similarly, there is no empirical evidence on which to reject this hypothesis in the thermal exposure-irradiation exposure regimes represented by the unshaded regions of Figure 3-9. However, it is evident from the available data presented in this chapter that the NRC recommended fluence screening criterion for a synergistic effect (i.e., $1x10^{17}$ n/cm^2, E > 1.0 MeV) is far too conservative and that the MRP-175 screening criterion (i.e., $6x10^{20}$ n/cm^2, E > 1.0 MeV) is sufficiently conservative.

The effect of low radiation doses on CASS materials thermally aged for 60 years in a reactor environment has not been determined. Available fracture toughness testing of aged CASS material has concluded that synergistic effects between thermal embrittlement and irradiation embrittlement are negligible in high fluence irradiations in a fast reactor spectrum. The proposed NRC approach extends to lower neutron fluences where irradiation embrittlement of austenitic stainless steels is not normally expected. Thus, a gap remains for investigation of this type of long-term, low dose exposure of CASS in light water reactors.

If a similar thermal radiation synergistic effect is postulated for austenitic stainless steel welds in the internals, there is a corresponding knowledge gap for welds. As with CASS materials, no data are available for weld materials at low radiation doses for extended periods of thermal aging. Thus, if there is a synergistic effect between irradiation and thermal aging in these welds, the fracture toughness at low fluences may be below the predictive curves, which are principally based on short-term irradiation data for wrought materials.

In addition, when extended service lives are considered, there are no existing data to demonstrate that stainless steel welds approach the same limiting fracture toughness as the wrought materials. In contrast to CASS materials, some stainless steel welds are present in higher fluence regions of PWR internals (e.g., core barrel and core shroud welds) and higher residual stresses are potentially present due to the welding process. As a result, concerns remain for the possible existence of a synergy between thermal aging and irradiation embrittlement and for the effect of high irradiation doses on the long-term fracture toughness of stainless steel welds.

3.5 References

[3-1] Chopra, O.K., et al., "Fracture Toughness of Irradiated Wrought and Cast Austenitic Stainless Steels in BWR Environment," 13th International Conference on Environmental Degradation of Materials in Nuclear Power Systems, Canadian Nuclear Society, August 19-23, 2007.

[3-2] Chopra, O.K. and Shack, W.J., "Crack Growth Rates and Fracture Toughness of Irradiated Austenitic Stainless Steels in BWR Environments," NUREG/CR-6960, U.S. Nuclear Regulatory Commission, March 2008.

[3-3] Horsten, M.G., and Belcher, W.P.A., "Fracture Toughness and Tensile Properties of Irradiated Reactor Pressure Vessel Cladding Material," Effects of Radiation on Materials, 20th International Symposium, ASTM STP 1405, American Society for Testing and Materials (2002).

[3-4] European Structural Integrity Society, "Procedure for Determining the Fracture Behavior of Materials," ESIS P2-92, January 1992.

[3-5] Kim, C., et al., "Embrittlement of Cast Austenitic Stainless Steel Reactor Internals Components," Proceedings of Fontevraud 6: Contribution of Materials Investigations to Improve the Safety and Performance of LWRs, French Nuclear Energy Society (SFEN), September 18-22, 2006.

[3-6] WCAP-16496-P, "Testing of Cast Austenitic Stainless steel (CASS) Material for Thermal/Irradiation Embrittlement," Westinghouse Electric Co. LLC Report, 2008.

[3-7] "Joint Owners Baffle Bolt Program: JOBB-CD Version 05.12," EPRI, Palo Alto, CA: 2005. 1012083.

[3-8] 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).

[3-9] Chopra, Gruber and Shack (NUREG/CR-6826), "Fracture Toughness and Crack Growth Rates of Irradiated Austenitic Stainless Steels," NUREG/CR-6826 (ANL-03/22), Argonne National Laboratory, Argonne, Illinois (August 2003).

[3-10] Materials Reliability Program: A Review of Radiation Embrittlement of Stainless Steels for PWRs (MRP-79) –Revision 1, EPRI Report 1008204, 2004.

Table 3-1
Fracture Toughness mean and lower bound fitted curve values for Type 308L materials [3-3]

Temperature (°C/°F)	$J_{0.2bl}$ (kJ/m^2)	
	Mean	Lower Bound
RT	309	276
100/212	270	246
200/392	229	196
295/562	187	162

Table 3-2
Fracture Toughness mean and lower bound fitted curve values for Type 308L materials [3-3] Irradiated Material Fracture Toughness Results.[3-5, 3-6]

Sample ID	Material	J_Q (kJ/m^2)
J1-5	Type 308 Weld (TIG) – 10.4 dpa	164
J8-1	Un-aged CASS – 6.3 dpa	74
J8-3	Un-aged CASS – 12 dpa	71
J4-1	Semi-aged CASS – 10.4 dpa	102
J4-2	Semi-Aged CASS – 10.4 dpa	112
J4-3	Semi-Aged CASS – 10.4 dpa	94
J9-1	Aged CASS – 6.3 dpa	129
J9-3	Aged CASS – 12 dpa	57
N03	Fuel Nozzle Clamp – 0.08 dpa	688
N11	Fuel Nozzle Clamp – 0.08 dpa	370

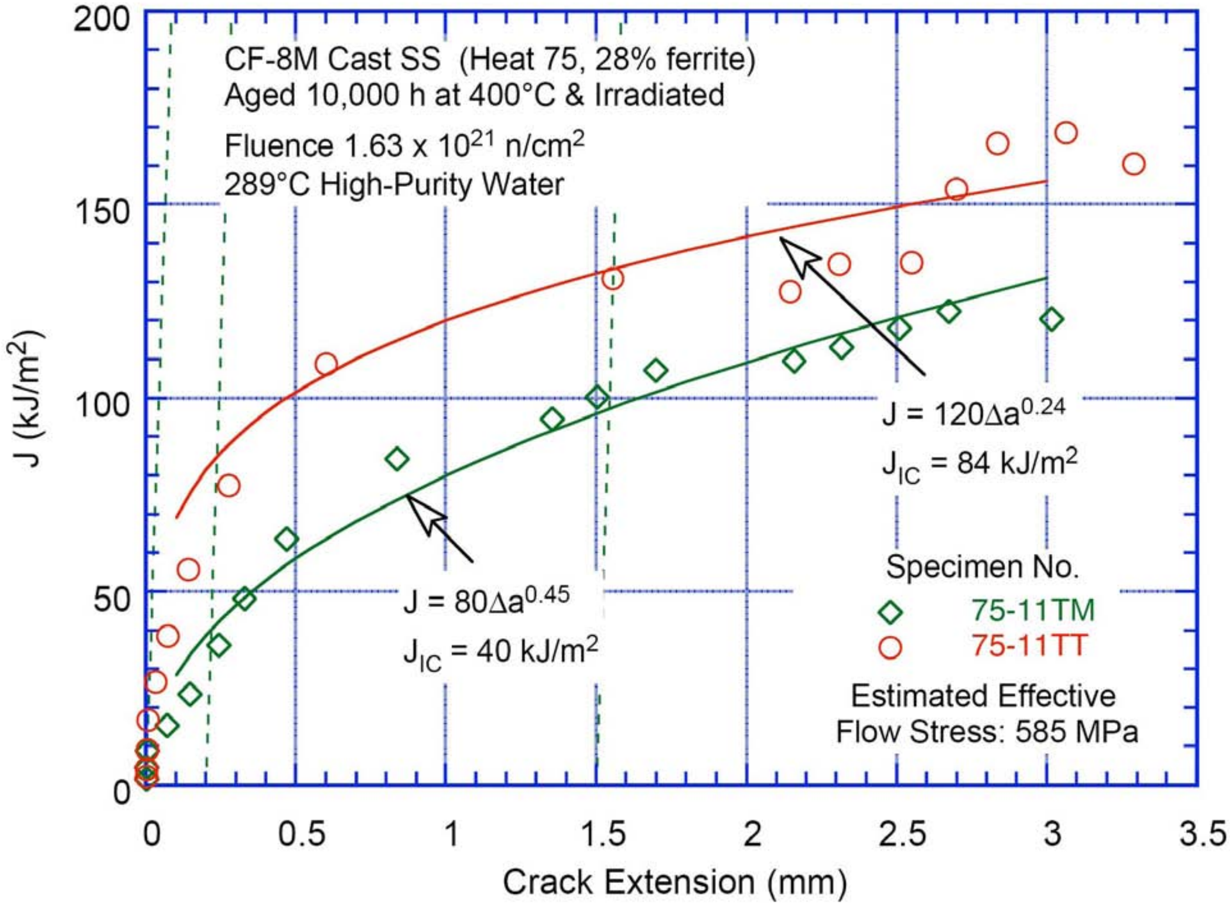


Figure 3-1
J-R curves for thermally-aged and irradiated CF-8M material and tested in simulated BWR water environments. [3-1, 3-2]

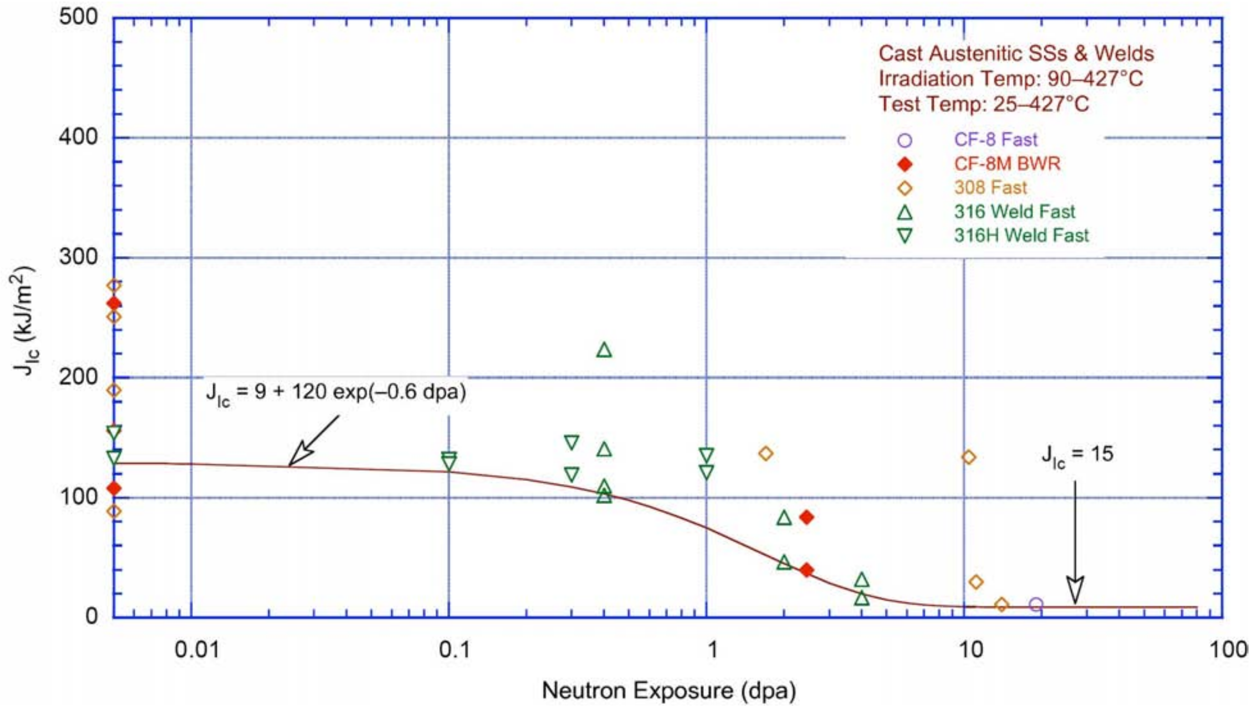


Figure 3-2
The change in initiation toughness J_{Ic} of CASS and austenitic stainless steel weld metals as a function of neutron exposure. [3-1, 3-2] (Authors' note, the label J_{Ic} = 15 by Ref [3-1, 3-2] should be J_{Ic} = 9).

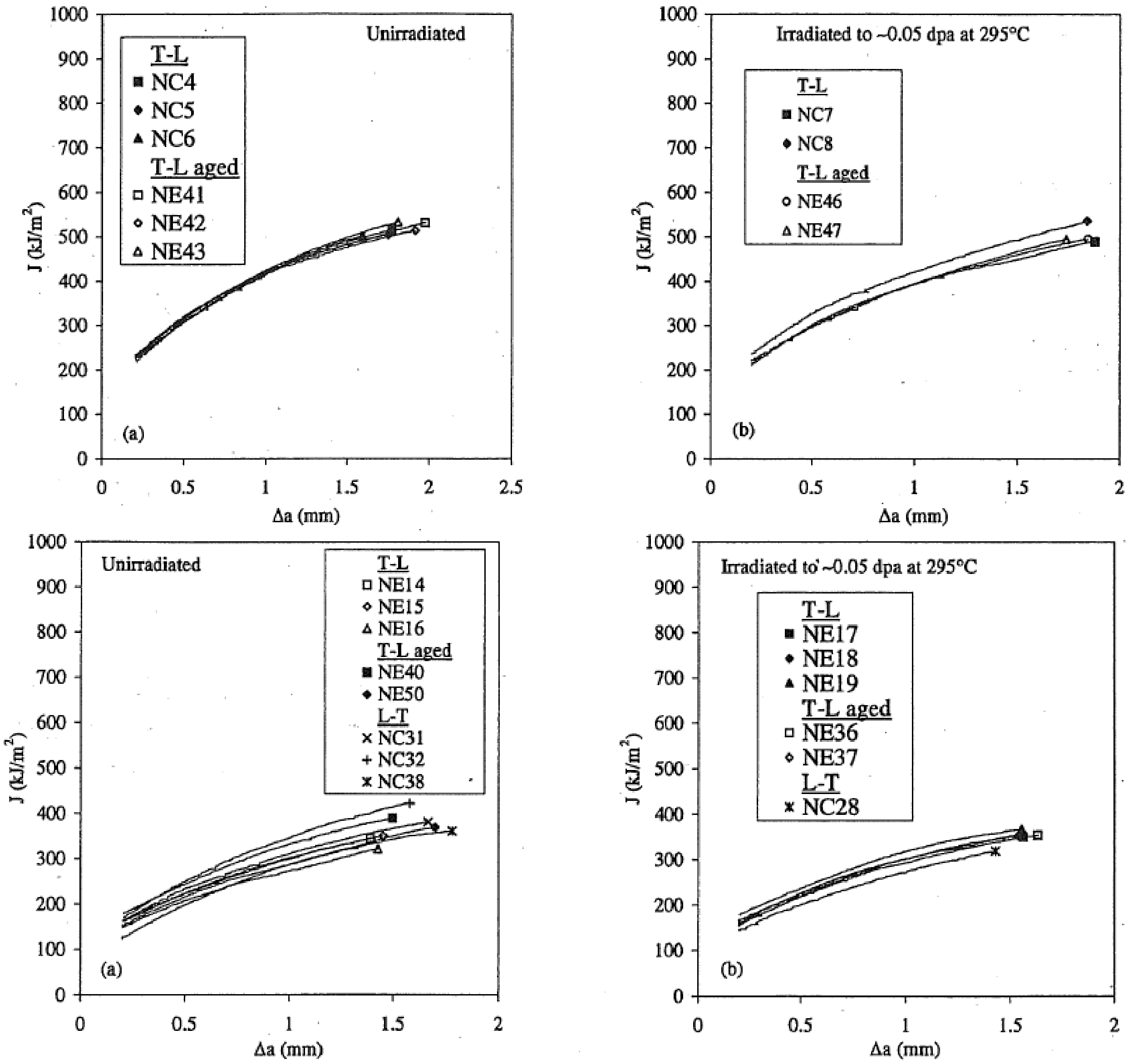


Figure 3-3
No effect of thermal aging on fracture toughness (J-R) is seen with or without irradiation on Type 308L weld metal as a function of temperature. Top, tested at 100°C; bottom tested at 295°C.[3-3]

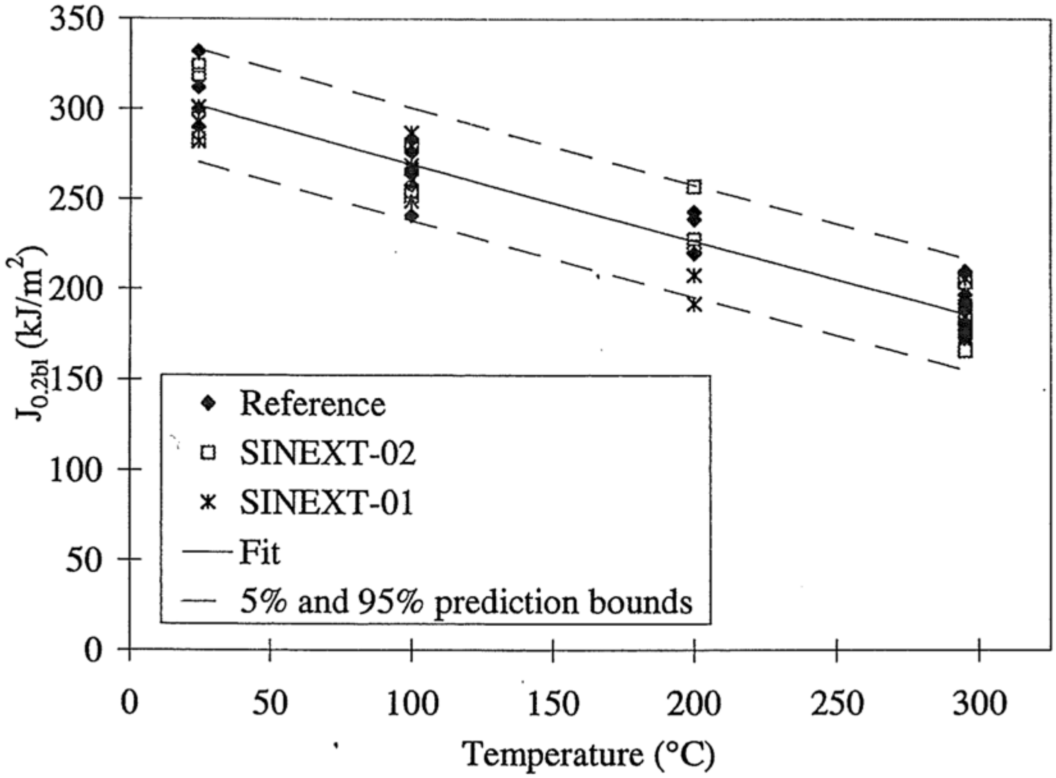


Figure 3-4
The observed change in initiation toughness (mean, upper, and lower bound) of Type 308L weld metal as a function of temperature. [3-3]

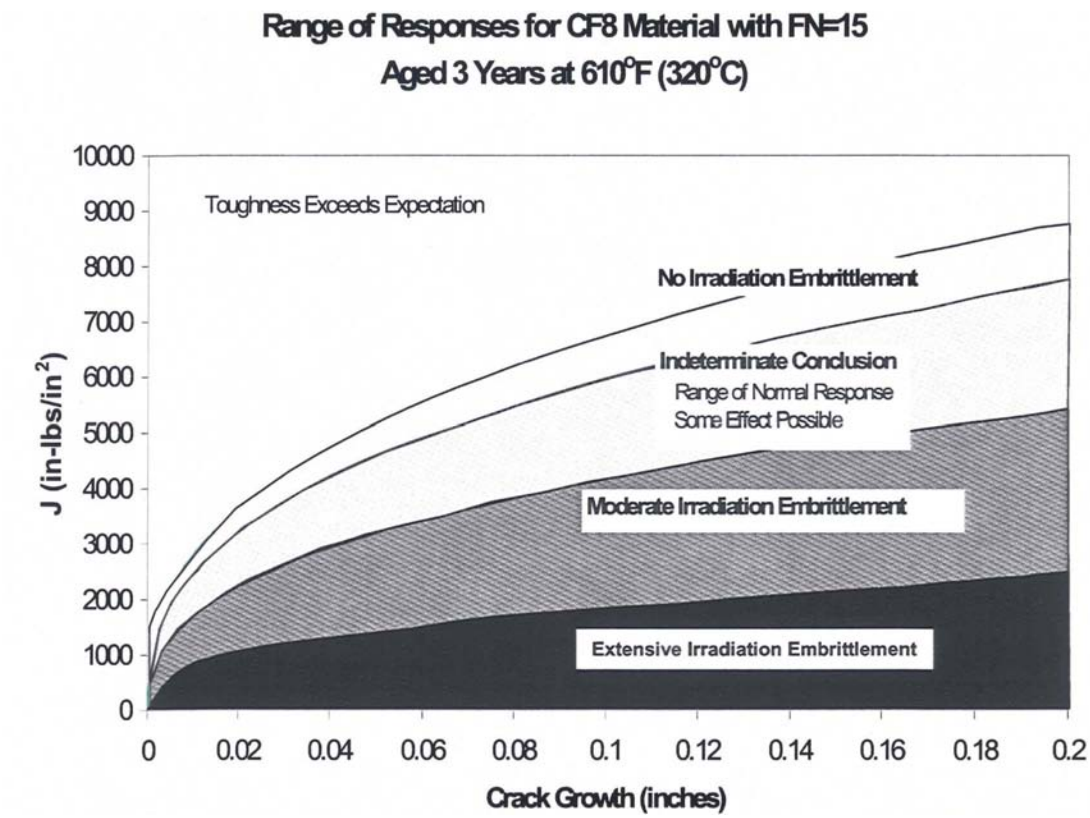


Figure 3-5
Fracture Response Map for a Typical CF-8 Material Aged at 610°F for 3 Years [3-5, 3-6]

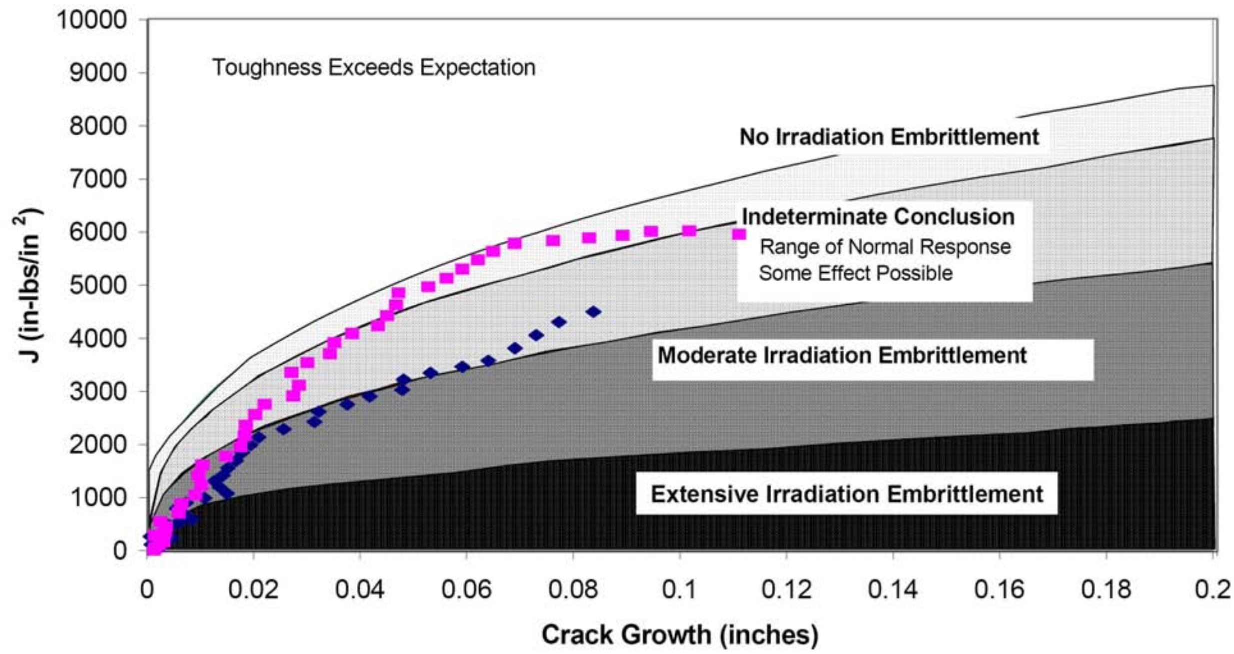


Figure 3-6
Measured J-R Curves for the JOBB CF-8 Material Irradiated to 6 and 12 dpa Compared to the Bands Predicted for Thermal Aging Alone in Figure 3-5 [3-5, 3-6]

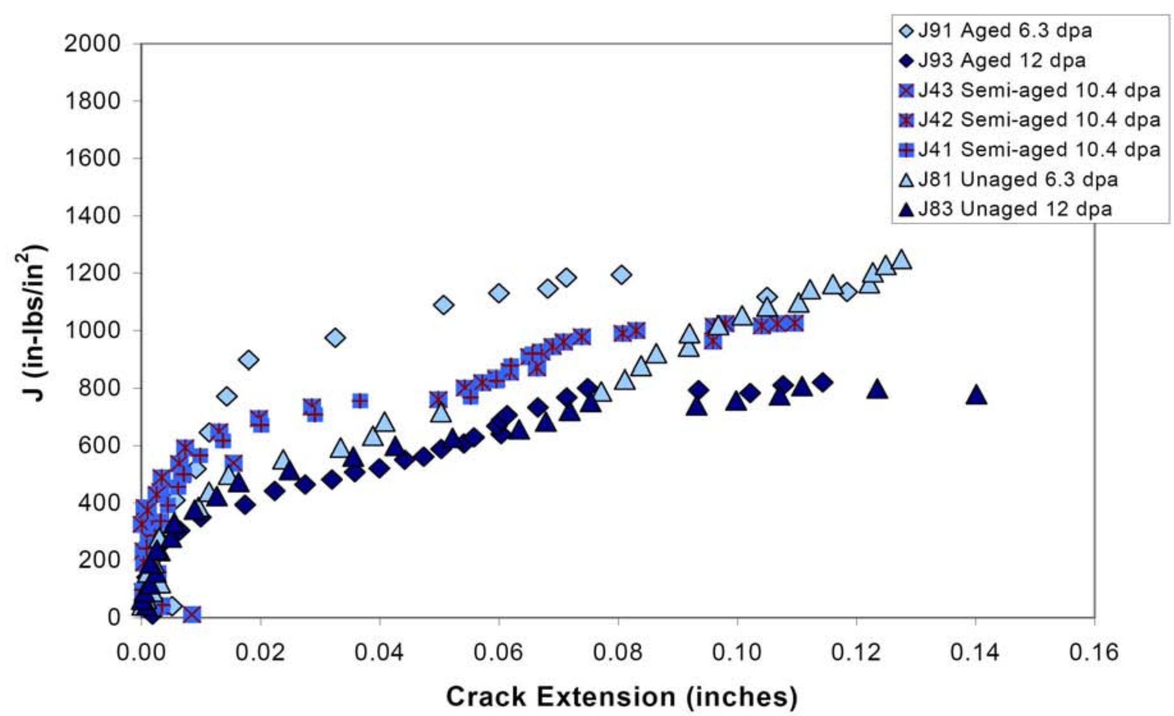


Figure 3-7
Measured J-R Curves for the JOBB CF-8 [3-5, 3-6]

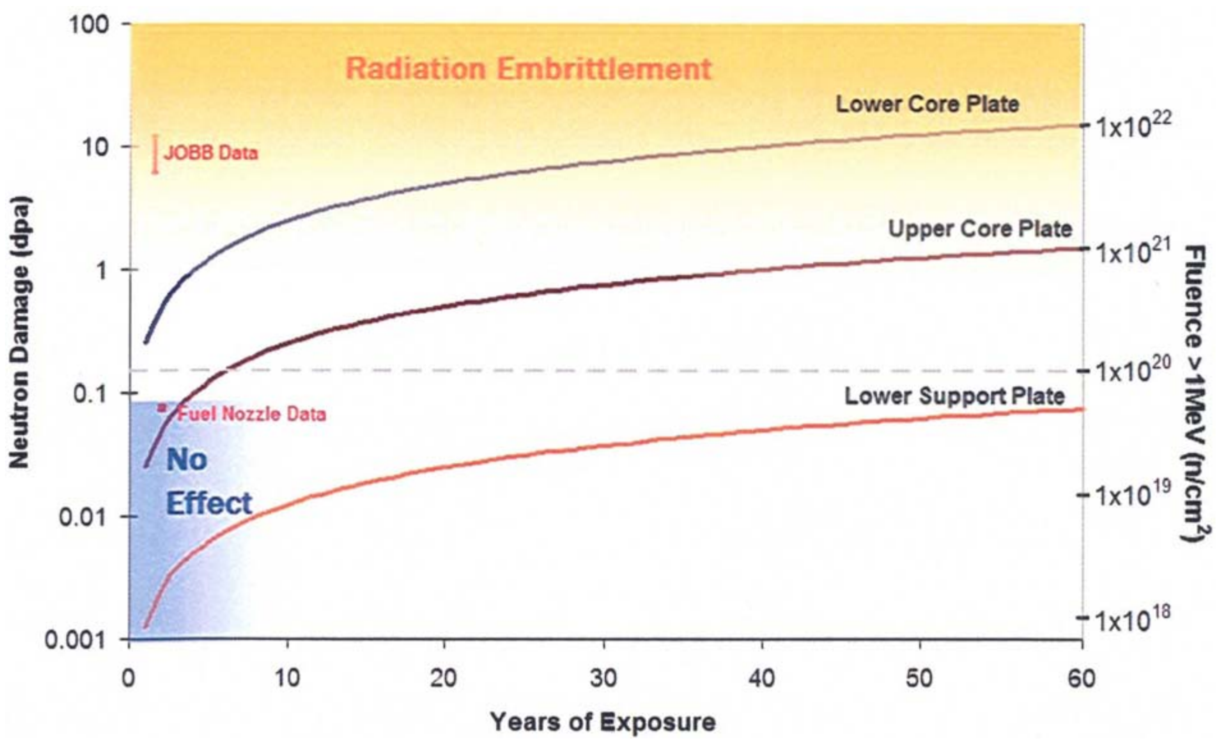
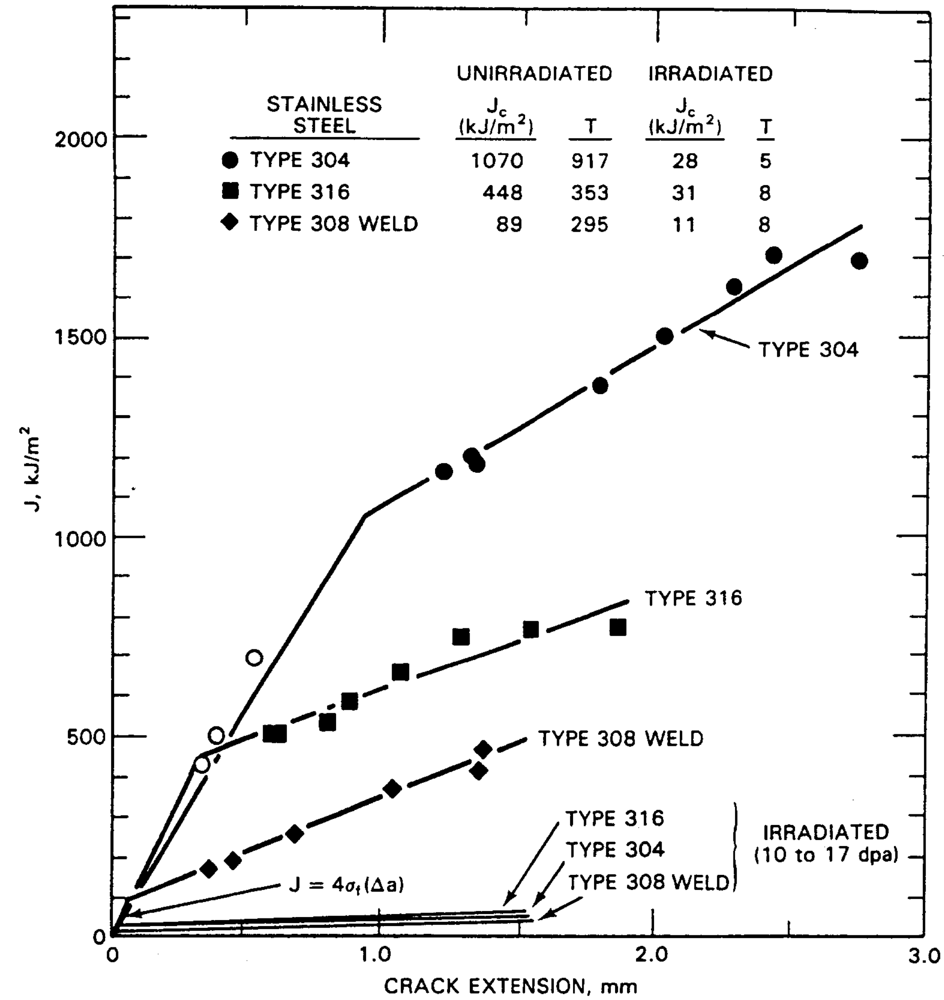


Figure 3-8
Implications of Data for PWR Internals Applications [3-5, 3-6]



6]

Tearing modulus was computed from: $T = \frac{dJ}{da}\frac{E}{\sigma_Y^2}$

where E is the elastic modulus, σ_Y is the effective yield strength or flow stress, and dJ/da is the slope of J-R curve beyond J_c.

Figure 3-9
Comparison of J-Δa curves for unirradiated and irradiated Types 304 and 316 stainless steels and Type 308 weld irradiated in EBR-II at 400-427°C (752-801°F).[3-10]

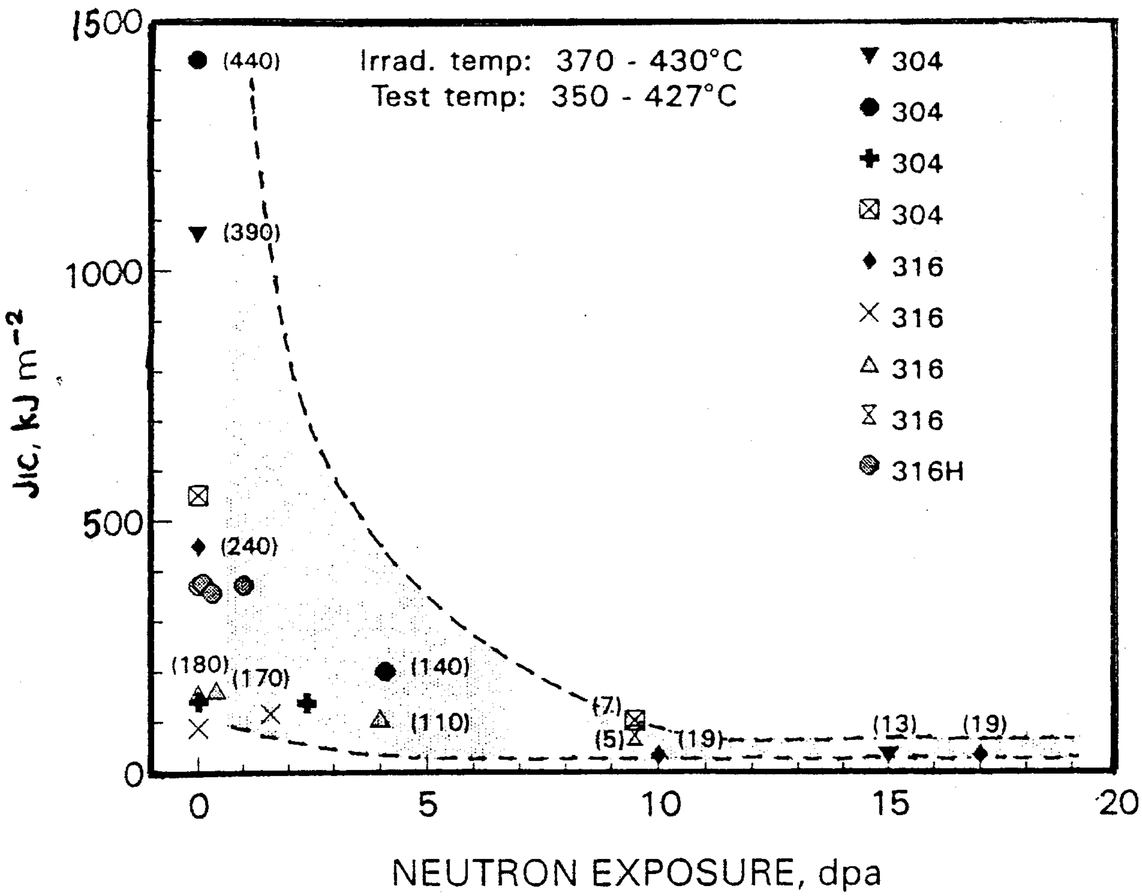


Figure 3-10
Jlc of various heats of Types 304 and 316 stainless steels as a function of neutron exposure. The irradiation temperatures were 370-430°C (698-806°F) and tested at 350-427°C (662-800°F). Values inside parentheses are dJ/da in MPa.[3-10]

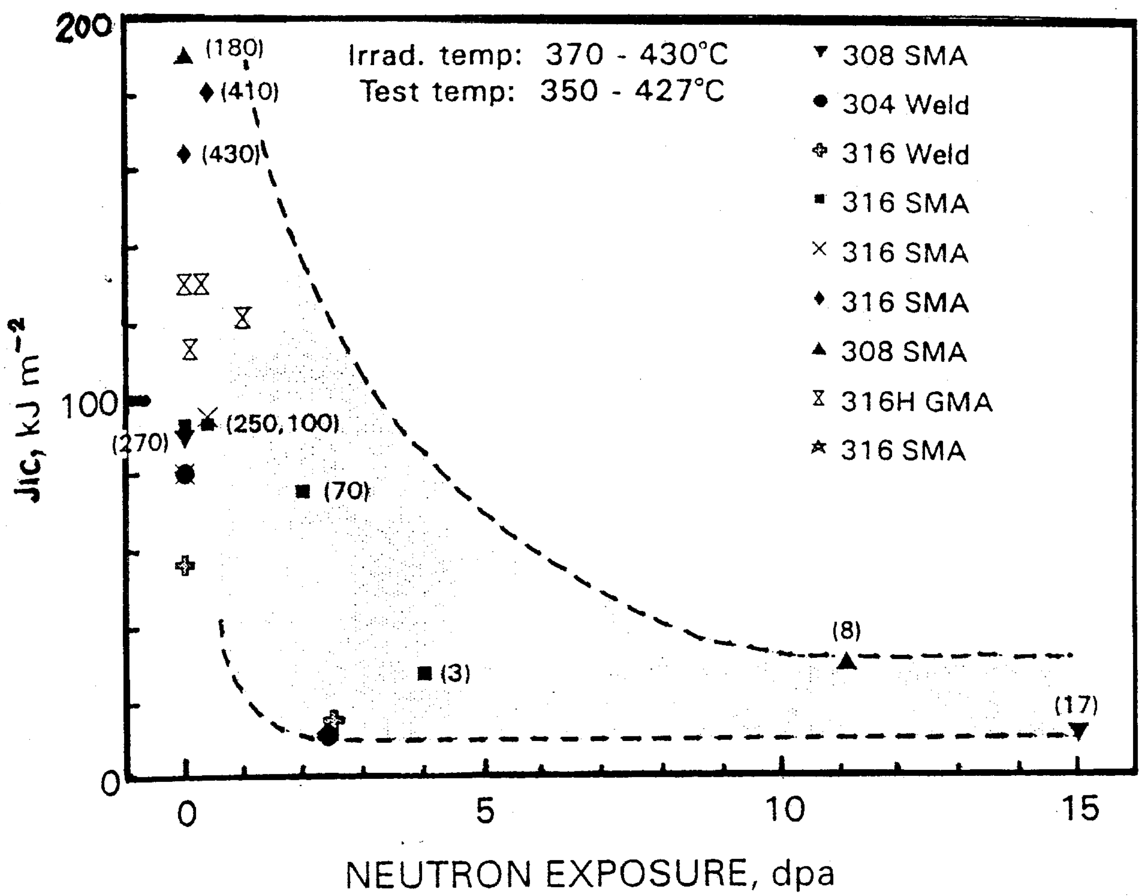


Figure 3-11
Jlc of Stainless steel welds irradiated at 370-430°C as a function of neutron exposure. Values of dJ/da in MPa are inside parentheses.[3-10]

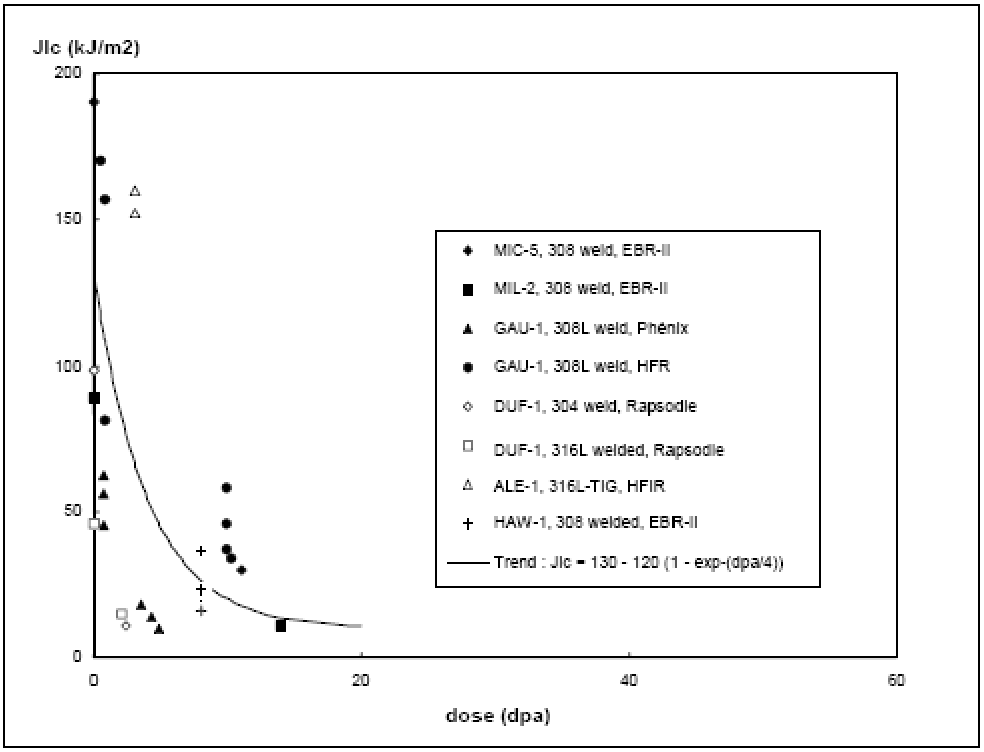


Figure 3-12
Summary of JIC values for irradiated Type 308 weld materials.[3-8]

4
FLUENCE SCREENING ASSESSMENTS FOR CASS AND AUSTENITIC WELD MATERIALS IN PWR INTERNALS

This chapter provides a listing of CASS and austenitic stainless steel weld locations in the PWR internals whose 60-year lifetime neutron fluence is below the MRP-175 [4-1] irradiation embrittlement screening criterion of 6.7×10^{20} n/cm^2, E > 1.0 MeV but exceed the criterion of 1×10^{17} n/cm^2, E > 1.0 MeV recommended for CASS materials in the GALL report [4-2] by the U.S. NRC.

Detailed descriptions of the PWR internals items for each of the NSSS vendor designs are provided in MRP-156 [4-3], MRP-157 [4-4], MRP-189-Rev. 1 [4-5], and MRP-191 [4-6].

4.1 B&W Components

For the B&W design, there are 62 items identified as being fabricated from CASS and/or welded with austenitic stainless steel material. The anticipated fluence for each of these items is assessed below relative to their potential for embrittlement over the 60-year lifetime of the PWR internals.

4.1.1 CASS Items

The following five items were fabricated using CASS material in the B&W designed PWR internals:

- Control rod guide tube (CRGT) assembly spacer castings, CF-3M
- Core support shield (CSS) assembly cast outlet nozzles, CF-8 (ONS-3 and DB only)
- CSS assembly vent valve discs, CF-8
- In-core monitoring instrumentation (IMI) guide tube assembly spider castings, CF-8
- Plenum Cylinder Reinforcement Castings, CF-8 (DB only, also containing 26 (2x13) round bars)

Note: the plenum cylinder reinforcement castings at DB were recently identified during a unit-specific records search and do not appear in any of the previously published MRP reports.

Figure 4-1 shows the approximate boundary for items that will reach a 60-year (54 EFPY) lifetime fluence of 1×10^{17} n/cm^2, E>1.0 MeV, which is based on a recent fluence calculation [4-8] for the B&W RV and internals. Table 4-1 contains a summary of the B&W design items fabricated from CASS and estimated 60-year fluence levels. Of these CASS items, only the IMI guide tube assembly spiders have been determined to exceed the MRP-175 irradiation embrittlement screening criterion of $6.7x10^{20}$ n/cm^2, E>1.0 MeV during the 60-year lifetime. The IMI guide tube assembly spiders have been classified as "Primary" for irradiation embrittlement by the "PWR Internals Inspection and Evaluation Guidelines" (MRP-227-Rev. 0 [4-7]).

In addition, the CRGT assembly spacer castings on the bottom three levels, out of a total of 10 levels, will reach a 60-year lifetime fluence between $1x10^{17}$ n/cm^2, E>1.0 MeV and $6.7x10^{20}$ n/cm^2, E>1.0 MeV. Also, the recently identified plenum cylinder reinforcement castings, which have not been evaluated or classified in the MRP documents to date, will mostly be below a fluence of $1x10^{17}$ n/cm^2, E>1.0 MeV, but the lower edge of this item will reach a fluence above this level. Finally, the 60-year lifetime (54 EFPY) fluence for the remaining CASS items will not exceed $1x10^{17}$ n/cm^2, E>1.0 MeV.

Therefore, each of the above three items that exceed a fluence of $1x10^{17}$ n/cm^2, E>1.0 MeV would have to consider a potential synergistic effect of thermal and irradiation embrittlement in development of evaluation acceptance criteria.

4.1.2 Austenitic Stainless Steel Welds

The 57 austenitic stainless steel welded items in the B&W design are listed in Table 4-2. It can be seen that the majority of these items have been screened and evaluated in MRP-189 Rev. 1 and concluded to be "A" items, which were then ultimately classified as "No Additional Measures."

Of the 57 welded items in Table 4-2, 21 have been determined to exceed the MRP-175 irradiation embrittlement screening criterion of $6.7x10^{20}$ n/cm^2, E>1.0 MeV during the 60-year lifetime. Some have been classified as "Primary" for irradiation embrittlement by the "PWR Internals Inspection and Evaluation Guidelines" (MRP-227-Rev. 0), but most remain classified as "No Additional Measures."

Based on a recent fluence calculation [4-8], there are 36 welded items that will reach a 60-year lifetime (54 EFPY) fluence between $1x10^{17}$ n/cm^2, E>1.0 MeV and $6.7x10^{20}$ n/cm^2, E>1.0 MeV. Ten of these welded items will have areas that are mostly below $1x10^{17}$ n/cm^2, E>1.0 MeV. All of these welded items have been classified as "No Additional Measures." Utilizing a lower fluence screening criterion would require that each of these items be considered for potential irradiation embrittlement and fracture toughness data requirements.

4.2 Combustion Engineering Components

For the Combustion Engineering (CE) design, there were 45 components identified as being fabricated from CASS and/or as containing structural welds. These components are potentially susceptible to embrittlement mechanisms due to their duplex (austenite/ferrite) grain structure. Thus, the effect of irradiation and temperature on fracture toughness must be considered.

Table 4-3 lists the 10 CE components containing either CASS or austenitic stainless steel structural welds that were screened in (i.e., exceeding the MRP-175 fluence criterion of $6.7x10^{20}$ n/cm^2, E>1.0 MeV) for irradiation embrittlement in MRP-191. Nine of the 10 components screened in for irradiation embrittlement were also classified as either Primary or Expansion inspection items in the MRP-227 recommendations. These recommendations included irradiation embrittlement as a potential degradation mechanism. Therefore, adoption of a lower fluence screening criterion would not change the results of the original screening process. However, if a flaw were to be discovered in any of these components, the potential synergistic effects of thermal and irradiation embrittlement would have to be considered in developing evaluation acceptance criteria. The In-Core Instrumentation (ICI) guide tube was the only structural weld component that was originally identified for irradiation embrittlement that was reassigned to Category A due to the lack of a credible damage issue.

The remainder of the CE components fabricated from CASS material or containing structural welds are listed in Table 4-4. Most of these 35 components were classified as Category A (aging effects below screening criteria) components in MRP-191, while some were classified as Primary or Expansion components in MRP-227. All of these components were below the MRP-175 irradiation embrittlement fluence screening criterion. As with the B&W items, the lower fluence criterion cited in the NRC GALL report for CASS and weld metal irradiation embrittlement would require that some of these components be considered for potential irradiation embrittlement. Additional effort would be required to determine fracture toughness data requirements for the components in Table 4-4.

Four CASS components were identified for potential thermal embrittlement in the CE PWR internals. Only one of the four CASS components was also identified for irradiation embrittlement and included in Table 4-3. The remaining three components are included in Table 4-4. The core support columns in Table 4-3 were classified as an Expansion inspection component in MRP-227. The core support system contains multiple core support columns. The evaluation methodology outlined in the generic acceptance criteria of WCAP-17096 [4-9] presumes that any observed crack implies local failure and analyzes the core support system to determine the minimum number of unfailed columns required to maintain system integrity. Therefore, no measurement or calculation of fracture toughness is required to complete this evaluation. The three components containing CASS in Table 4-4 are the control element assembly (CEA) shrouds, CEA shroud bases, and the modified CEA shroud extension shaft guides. These three components may require further analysis to address the postulated synergy between thermal and irradiation embrittlement.

The remaining 32 items in Table 4-4 are welds in the CE reactor internals that fell below the MRP-175 screening criteria. If the far more conservative screening fluence of $1x10^{17}$ n/cm^2, E>1.0 MeV suggested by the NRC for CASS were also applied to these welds, 12 of the 32 would have been identified for potential embrittlement. There is no GALL requirement for applying this fluence limit to welds. Most of the components additionally identified for embrittlement are part of the core support barrel assembly (and thermal shield where applicable). Monitoring of these assemblies is already included in the MRP-227 inspection program. The additionally identified items would not be considered Primary inspection locations for embrittlement due to the tenuous nature of applying the $1x10^{17}$ n/cm^2, E>1.0 MeV screening limit to welds.

4.3 Westinghouse Components

For the Westinghouse design, there were 31 components identified in MRP-191 that were fabricated from CASS and/or contained structural welds. As with the CE design, these components are potentially susceptible to embrittlement mechanisms unique to the duplex grain structure. If a fracture analysis of these components is required, the effect of irradiation on the fracture toughness in the duplex stainless steels will have to be considered.

The original screening process identified irradiation embrittlement as a potential degradation mechanism for the 14 Westinghouse components listed in Table 4-5. Eight of the 14 were classified in the MRP-227 recommendations as Primary, Expansion, or Existing components. Of the remaining six, three were originally classified in Category A in MRP-191 and three were classified as “No Additional Measures” in MRP-227. “Category A” components are those for which the aging effects were below the screening criteria applied in MRP-191 and for which the significance of aging degradation is minimal. The “No Additional Measures” components from MRP-227 were those for which no additional actions were required for the management of aging degradation. All of these recommendations considered irradiation embrittlement as a potential degradation mechanism, but they did not consider the extent or precise location of the embrittlement. Adoption of alternative screening criteria with a lower fluence would not change the screening result for these components as the original screening process was a binary “in/out” evaluation. However, to the extent that irradiated fracture properties are required to evaluate acceptance criteria, additional consideration of synergistic thermal and irradiation effects may be required if a flaw is detected in any of these components.

Six of the components listed in Table 4-5 were identified as possibly containing CASS. Although irradiation embrittlement was identified as a potential degradation mechanism for all six of these items, three were among those classified in Category A because there was no credible damage issue identified, and a fourth was among those classified as “No Additional Measures.” As described above, additional analysis would not change the classification of these components since they were already screened in for irradiation embrittlement, and a lower fluence level would not affect the screening results. The only CASS component classified as a Primary component in the MRP-227 inspection strategy was the lower flange weld on the control rod guide tube assembly. The other CASS component included as an Expansion component in the MRP-227 inspection strategy is the lower core support column bodies. It should be noted that cast stainless steel is listed as an allowable material for both of these components. Plant-specific fabrication records would have to be examined to determine the actual material of construction. As both of these components are parts of highly redundant systems, the evaluation methodology outlined in the generic acceptance criteria of WCAP-17096 [4-9] presumes that any observed crack implies local failure and analyzes the system to determine the minimum number of unfailed components required to maintain system integrity. There are no flaw tolerance requirements in the dispositioning of these components, so calculation or measurement of the fracture toughness is not required.

The remaining 17 Westinghouse components containing structural welds or CASS are listed in Table 4-6. While thermal embrittlement was considered to be a potential degradation mechanism for the two CASS components in Table 4-6, irradiation embrittlement was not considered to be a potential degradation mechanism for any of the 18 components because all fell below the MRP-175 irradiation embrittlement criterion. Adoption of a fluence criterion consistent with the $1x10^{17}$ n/cm^2, E>1.0 MeV criterion cited by the NRC GALL report could result in identification of irradiation embrittlement as a degradation mechanism in some additional CASS and welded components. Additional components that would be identified for irradiation embrittlement are indicated in Table 4-6.

The only two CASS components in Table 4-6 are the intermediate flanges on the control rod guide tube assembly and the lower support casting. Although the design specifications allowed the use of CASS for these components, a small, and as yet undetermined, number of plants actually used cast materials in these components. The intermediate flanges on the control rod guide tube assemblies are well removed from the core and would not exceed the 10^{17} n/cm^2 E>1.0 MeV fluence limit. Portions of the lower support casting could exceed this limit. However the component is considered to be highly flaw tolerant.

Only two of the sixteen welds listed in Table 4-6 were additionally identified for potential embrittlement by using the lower fluence limit. These welds are the welds in the core barrel outlet nozzles and the welds in the upper core plate alignment pins. The core barrel outlet nozzles are already an Expansion location in the inspection strategy for the core barrel. The alignment pins were considered an Existing in the ASME code inspections. There is no reason to expect that the reduced screening would alter any of the current MRP-227 inspection recommendations for welded components in Westinghouse plants.

4.4 References

[4-1] Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175) – EPRI, Palo Alto, CA: 2005. 1012081.

[4-2] Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Rev. 1, September 2005.

[4-3] Materials Reliability Program: Pressurized Water Reactor Issue Management Table, PWR-IMT Consequence of Failure (MRP-156). EPRI, Palo Alto, CA: 2005. 1012110.

[4-4] Materials Reliability Program: Updated B&W Design Information for the Issue Management Tables (MRP-157). EPRI, Palo Alto, CA: 2005. 1012132.

[4-5] Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189 Rev. 1). EPRI, Palo Alto, CA: 2009. 1018292.

[4-6] Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Designs (MRP-191). EPRI, Palo Alto, CA: 2006. 1013234.

[4-7] Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Rev. 0). EPRI, Palo Alto, CA: 2008. 1016596.

[4-8] 32-9025791-001, "DB1 Fluence Analysis," AREVA NP Inc. Proprietary Document.

[4-9] WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," Westinghouse Electric Co. LLC Report, 2009.

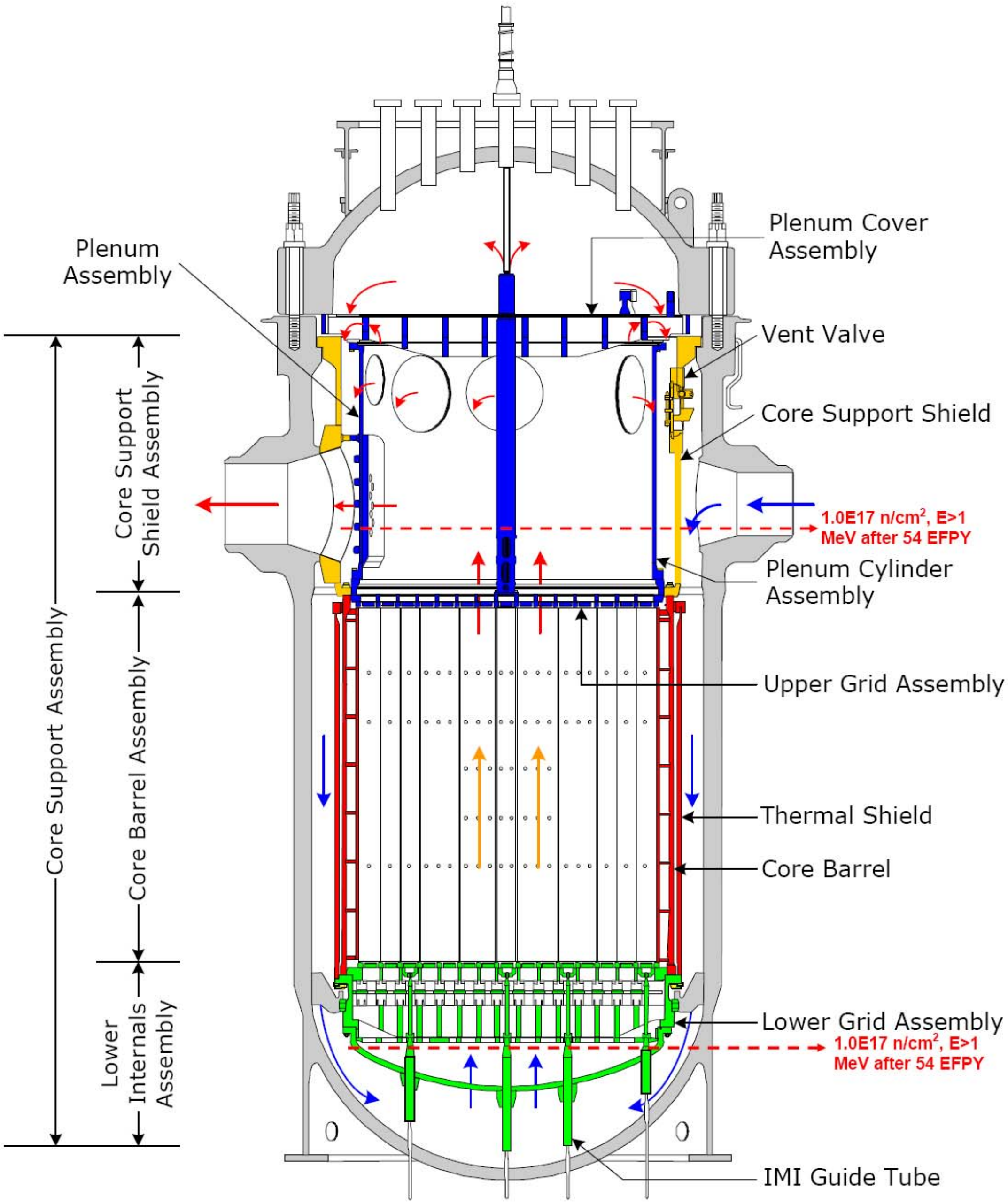


Figure 4-1
Approximate position of the $1x10^{17}$ n/cm^2, E>1.0 MeV fluence lines above and below the core after 60 years of operation (54 EFPY) in the B&W design PWR internals.

Table 4-1
Summary of CASS Items in B&W PWR Internals and the Estimated 60-Year Neutron Fluence

CASS Items	Grade	60-Yr Fluence, n/cm^2, E>1.0 MeV	MRP-227-Rev. 0 Classification
CRGT assembly spacer castings	CF-3M	6.7×10^{20}> (levels 1 to 3) >1.0×10^{17}; (levels 4 to 10) <1.0×10^{17}	Expansion due to thermal embrittlement
CSS assembly cast outlet nozzles (ONS-3 and DB only)	CF-8	<1.0×10^{17}	Primary due to thermal embrittlement (see Note 1)
CSS assembly vent valve discs	CF-8	<1.0×10^{17}	Primary due to thermal embrittlement (see Note 1)
IMI guide tube assembly spiders	CF-8	>6.7×10^{20}	Primary due to irradiation and thermal embrittlement
Plenum cylinder reinforcement castings each containing 13 round LOCA bosses (DB only, Note 2)	CF-8	6.7×10^{20}> (lower edge) >1.0×10^{17}; (most area) <1.0×10^{17}	Not listed in MRP-227 (Rev. 0), see Note 2

Notes:

1. These items were classified as “Primary” in MRP-227 (Rev. 0) due to their unknown chemical composition and ferrite content at the time. Subsequent fabrication records search was able to determine the ferrite content for the outlet nozzles (ONS-3) and vent valve discs (ONS-1, ONS-2, and ONS-3) to be below the MRP-175 screening criteria for CASS. Additional records searches for the other B&W units are planned. Therefore, any CASS items whose ferrite content can be determined to be below the MRP-175 screening criteria can be justifiably moved into the “No Additional Measures” category.
2. The two plenum cylinder reinforcement plates, each containing 13 round LOCA bosses, at Davis-Besse are made of CASS material. This CASS item was not included in MRP-227 (Rev. 0), but was recently confirmed for the PWR internals of Davis-Besse during a records search.

Table 4-2
Summary of Austenitic Stainless Steel Welds in B&W PWR Internals with 60-year fluence above $1x10^{17}n/cm^2$ (E>1.0 MeV)

Assembly	Weld ID	Weld Description	Weld Process and Weld Metal	Weld Type	60-Yr Fluence, n/cm2, E>1.0 MeV	MRP 189 (Rev. 1)/ MRP-227 (Rev. 0) Category
Core Barrel	WC-11	top and bottom core barrel cylinders to top and bottom core barrel flanges circumferential seam weld	ASA with 308L	Double U-groove	$>6.7x10^{20}$	“A “
Core Barrel	WC-12	top and bottom thermal shield cylinder vertical seam weld	ASA with 308L	Double U-groove	$>6.7x10^{20}$	“A “
Core Barrel	WC-13	top thermal shield cylinder to bottom thermal shield cylinder circumferential seam weld	ASA with 308L	Double U-groove	$>6.7x10^{20}$	“A “
Core Barrel	WC-15	core barrel-to-former plate cap screw locking pin to core barrel cylinder weld	MTIG or STIG with 308L	Fillet/Locking	$>6.7x10^{20}$	Primary
Core Barrel	WC-15	baffle-to-former bolt locking pin to baffle plate weld	MTIG or STIG with 308L	Fillet/Locking	$>6.7x10^{20}$	Primary
Core Barrel	WC-16	baffle-to-baffle bolt locking ring to baffle plate weld	MTIG or STIG with 308L	Fillet/Locking	$>6.7x10^{20}$	Primary or Expansion (Note 1)
Core Barrel	WC-20	core barrel top and bottom cylinder vertical seam weld	ASA with 308L	Double U-groove	$>6.7x10^{20}$	Expansion
Core Barrel	WC-21	core barrel top cylinder to core barrel bottom cylinder circumferential seam weld	ASA with 308L	Double U-groove	$>6.7x10^{20}$	Expansion
Core Barrel	WC-94	plug weld locking dowel flush with special flat head baffle-to-former shoulder screw	MTIG or STIG with 308L	Plug/Locking	$>6.7x10^{20}$	Primary
Core Barrel	WC-141	locking clip (for original LCB bolt) to lower grid shell forging weld	MTIG or STIG with 308L	Fillet/Locking	$>1.0x10^{17}$, $<6.7x10^{20}$	“A “
Core Barrel	WCF-220	plug dowel to thermal shield upper restraint "A" weld	MTIG or STIG with 308L	Fillet/Locking	$>1.0x10^{17}$, $<6.7x10^{20}$	“A “

Assembly	Weld ID	Weld Description	Weld Process and Weld Metal	Weld Type	60-Yr Fluence, n/cm2, E>1.0 MeV	MRP 189 (Rev. 1)/ MRP-227 (Rev. 0) Category
Core Support Shield	WCF-176	locking clip (for original UTS bolts) to thermal shield upper restraint "A" weld	MTIG or STIG with 308L	Fillet/Locking	$>1.0x10^{17}$, $<6.7x10^{20}$	"A "
Core Support Shield	WC-43	core support shield cylinder to bottom flange circumferential seam weld	ASA with 308L	Double U-groove	$>1.0x10^{17}$, $<6.7x10^{20}$	"A "
Core Support Shield	WC-46	core support shield cylinder to outlet nozzle weld	MMA with 308	J-groove	most $<1.0x10^{17}$, some $>1.0x10^{17}$, $<6.7x10^{20}$	"A "
Core Support Shield	WC-47	core support shield cylinder to outlet nozzle fillet weld (opposite WC-46)	MMA with 308	Fillet	most $<1.0x10^{17}$, some $>1.0x10^{17}$, $<6.7x10^{20}$	"A "
Core Support Shield	WC-51	core support shield cylinder vertical seam weld	ASA with 308L	Double U-groove	most $<1.0x10^{17}$, some $>1.0x10^{17}$, $<6.7x10^{20}$	"A "
Core Support Shield	WC-52	round (LOCA) bars to core support shield cylinder ID surface weld	MMA with 308	Fillet	most $<1.0x10^{17}$, some $>1.0x10^{17}$, $<6.7x10^{20}$	"A "
Core Support Shield	WC-53	flow deflector (U-baffle) to core support shield cylinder weld	MTIG root with 308L, MMA balance with 308L	Fillet	most $<1.0x10^{17}$, some $>1.0x10^{17}$, $<6.7x10^{20}$	"A "
Core Support Shield	WCF-264	locking clip (for original UCB A-286 bolts) to core support shield bottom flange weld	MTIG or STIG with 308L	Fillet/Locking	$>1.0x10^{17}$, $<6.7x10^{20}$	"A "
Core Support Shield	WCF-265	locking clip (for original UCB A-286 bolts) to core support shield bottom flange weld	MTIG or STIG with 308L	Fillet/Locking	$>1.0x10^{17}$, $<6.7x10^{20}$	"A "

Assembly	Weld ID	Weld Description	Weld Process and Weld Metal	Weld Type	60-Yr Fluence, n/cm2, E>1.0 MeV	MRP 189 (Rev. 1)/ MRP-227 (Rev. 0) Category
CRGT	WC-40	washer to CRGT pipe and spacer bolt weld (WC-64 for top spacer)	MTIG or STIG with 308L	Fillet/Locking	most <$1.0x10^{17}$, some >$1.0x10^{17}$, <$6.7x10^{20}$	"A "
CRGT	WC-41	CRGT flange to CRGT pipe weld (all around except at the 4 scallops)	MTIG or STIG with 308L	Fillet	>$1.0x10^{17}$, <6.7E20	"A "
CRGT	WC-42	Type 304 dowel to CRGT flange weld	MTIG with 308L	Fillet/Locking	>$1.0x10^{17}$, <6.7E20	"A "
CRGT	WC-64	washer to CRGT pipe and spacer bolt weld (also see WC-40)	MTIG or STIG with 308L	Fillet/Locking	most <$1.0x10^{17}$, some >$1.0x10^{17}$, <$6.7x10^{20}$	"A "
CRGT	WC-67	CRGT flange cap screw (joining upper grid rib section) to CRGT flange weld	MTIG or STIG with 308L	Fillet/Locking	>$1.0x10^{17}$, <$6.7x10^{20}$	"A "
Flow Distributor	WC-57	flow distributor flange to flow distributor head circumferential seam weld	ASA with 308L	Double U-groove	>$1.0x10^{17}$, <$6.7x10^{20}$	"A "
Flow Distributor	WC-60	flow distributor (FD) bolt locking clip to flow distributor flange weld	MTIG or STIG with 308L	Fillet/Locking	>$1.0x10^{17}$, <$6.7x10^{20}$	"A "
Flow Distributor	WC-157	IMI guide support plate clamping ring vertical seam weld	MMA with 308	Double V-groove	>$1.0x10^{17}$, <$6.7x10^{20}$	"A "
Flow Distributor	WC-161	dowel (for clamping ring) to flow distributor flange weld	MTIG or STIG with 308L	Fillet/Locking	>$1.0x10^{17}$, <$6.7x10^{20}$	"A "
IMI Guide Tube	WC-160	spider to lower grid rib section weld	MTIG or STIG with 308L	Fillet	>$6.7x10^{20}$	Primary
IMI Guide Tube	WC-187	nut locking clip to in-core instrument guide tube nut weld	MTIG or STIG with 308L	Fillet/Locking	>$1.0x10^{17}$, <$6.7x10^{20}$	"A "
IMI Guide Tube	WC-188	nut locking clip to in-core instrument guide tube weld	MTIG or STIG with 308L	Fillet/Locking	>$1.0x10^{17}$, <$6.7x10^{20}$	"A "

Assembly	Weld ID	Weld Description	Weld Process and Weld Metal	Weld Type	60-Yr Fluence, n/cm2, E>1.0 MeV	MRP 189 (Rev. 1)/ MRP-227 (Rev. 0) Category
Lower Grid	WC-5	support post to lower grid forging weld	MTIG or STIG with 308L and 308	Fillet	$>6.7x10^{20}$	"A "
Lower Grid	WC-8	lower grid forging (or weldment) to lower grid shell forging weld	MTIG root pass with 308L, balance MMA with 308	J-groove	$>1.0x10^{17}$, $<6.7x10^{20}$	"A "
Lower Grid	WC-26	flow distributor plate to lower grid shell forging weld	STIG root pass with 308L, MMA mid-layer with 308, GMA surface layer with 308	J-groove	$>6.7x10^{20}$	"A "
Lower Grid	WC-27	support post to flow distributor plate weld	STIG with 308L	Fillet	$>6.7x10^{20}$	"A "
Lower Grid	WC-59	cap screw to lower fuel assembly pad weld	MTIG with 308L	Fillet/Locking	$>6.7x10^{20}$	Expansion
Lower Grid	WC-97	bolting plug to support post weld	MTIG root pass with 308L, balance MMA with 308.	J-groove	$>6.7x10^{20}$	"A "
Lower Grid	WC-99	locking pin (for lower grid rib section cap screw) to lower grid rib section weld	MTIG with 308L	Fillet/Locking	$>1.0x10^{17}$, $<6.7x10^{20}$	"A "
Lower Grid	WC-100	locking pin (for lower grid support post cap screw) to lower grid rib section weld	MTIG with 308L	Fillet/Locking	$>6.7x10^{20}$	"A "
Lower Grid	WC-101	shock pad bolt to shock pad fillet weld	MTIG or STIG with 308L	Fillet/Locking	$>1.0x10^{17}$, $<6.7x10^{20}$	"A "
Lower Grid	WC-102	guide block bolt to washer fillet weld	MTIG or STIG with 308L	Fillet/Locking	$>1.0x10^{17}$, $<6.7x10^{20}$	"A "
Lower Grid	WC-142	lower grid forging (or weldment) to lower grid shell forging weld	MTIG with 308L	Fillet	$>1.0x10^{17}$, $<6.7x10^{20}$	"A "

Assembly	Weld ID	Weld Description	Weld Process and Weld Metal	Weld Type	60-Yr Fluence, n/cm2, E>1.0 MeV	MRP 189 (Rev. 1)/ MRP-227 (Rev. 0) Category
Lower Grid	WC-142	support post to flow distributor plate weld	MTIG with 308L	Fillet	$>6.7\text{x}10^{20}$	"A "
Lower Grid	WC-142	flow distributor plate to lower grid shell forging weld	MTIG with 308L	Fillet	$>6.7\text{x}10^{20}$	"A "
Lower Grid	WC-173	fuel assembly support pad to lower grid rib section weld	MTIG or STIG with 308L	Fillet	$>6.7\text{x}10^{20}$	Expansion
Lower Grid	WC-240	orifice plugs to flow distributor plate weld	MTIG or STIG with 308L	Fillet	$>1.0\text{x}10^{17}$, $<6.7\text{x}10^{20}$	"A "
Lower Grid	WCF-103	guide block bolt washer to guide block-A and guide block-B weld	MTIG or STIG with 308L	Fillet/Locking	$>1.0\text{x}10^{17}$, $<6.7\text{x}10^{20}$	"A "
Plenum Cylinder	WC-34	plenum cylinder to bottom plenum cylinder flanges circumferential seam weld	ASA with 308L	Double U-groove	$>1.0\text{x}10^{17}$, $<6.7\text{x}10^{20}$	"A "
Plenum Cylinder	WC-34	plenum cylinder bottom flange to reinforcement weld	ASA with 308L	Double U-groove	$>1.0\text{x}10^{17}$, $<6.7\text{x}10^{20}$	"A "
Plenum Cylinder	WC-37	plenum cylinder vertical seam weld	ASA with 308L	Double U-groove	most $<1.0\text{x}10^{17}$, some $>1.0\text{x}10^{17}$, $<6.7\text{x}10^{20}$	"A "
Plenum Cylinder	WC-66	locking cup (for bolts joining plenum cylinder bottom flange to upper grid ring forging) to plenum cylinder bottom flange weld	MTIG or STIG with 308L	Fillet/Locking	$>1.0\text{x}10^{17}$, <6.7E20	"A "
Plenum Cylinder	WC-131	plenum cylinder reinforcement to plenum cylinder weld	MMA root pass with 308, Flux Core balance with 308	J-groove	most $<1.0\text{x}10^{17}$, some $>1.0\text{x}10^{17}$, $<6.7\text{x}10^{20}$	"A "
Plenum Cylinder	WC-132	plenum cylinder reinforcement to plenum cylinder weld (opposite WC-131 weld)	Flux Core with 308L	Fillet	most $<1.0\text{x}10^{17}$, some $>1.0\text{x}10^{17}$, $<6.7\text{x}10^{20}$	"A "

Assembly	Weld ID	Weld Description	Weld Process and Weld Metal	Weld Type	60-Yr Fluence, n/cm2, E>1.0 MeV	MRP 189 (Rev. 1)/ MRP-227 (Rev. 0) Category
Upper Grid	WC-23	cap screw to upper grid fuel assembly support pad weld	MTIG with 308L	Fillet/Locking	$>1.0x10^{17}$, $<6.7x10^{20}$	“A “
Upper Grid	WC-95	locking pin (for cap screw to joining upper grid rib section to upper grid ring forging) to upper grid rib section weld	MTIG or STIG with 308L	Fillet/Locking	$>1.0x10^{17}$, $<6.7x10^{20}$	“A “
Upper Grid	WC-173	fuel assembly support pad to upper grid rib section weld	MTIG or STIG with 308L	Fillet	$>1.0x10^{17}$, $<6.7x10^{20}$	“A “

Note:

1. The internal baffle-to-baffle bolt locking devices and locking welds are “Primary” and the external baffle-to-baffle bolt locking devices and locking welds are “Expansion’ per MRP-227 (Rev. 0).

Table 4-3
Combustion Engineering Components Containing Cast Stainless Steel and/or Structural Welds and Screened in for Potential Irradiation Embrittlement in MRP-191

Assembly/Component Name	Material Category	Material Type/Grade	Structural Weld	Neutron Fluence Region(1)	Embrittlement Screening	MRP-191/ MRP-227(2)
Lower Support Structure						
Core Support Plate	Austenitic SS	304 SS	Yes	Region 4	IE	Primary
Core support columns	Austenitic SS	304 SS	Yes	Region 4	IE	Expansion
Core support columns	CASS	CF8	Yes	Region 4	TE/IE	Expansion
Core support deep beams	Austenitic SS	304 SS	Yes	Region 4	IE	Primary
Core Support Barrel Assembly						
Lower cylinder	Austenitic SS	304 SS	Yes	Region 4	IE	Expansion
Core Shroud Assembly						
Shroud plates	Austenitic SS	304 SS	Yes	Region 5	IE	Primary
Former plates	Austenitic SS	304 SS	Yes	Region 5	IE	Primary
Ribs	Austenitic SS	304 SS	Yes	Region 5	IE	Expansion
Rings	Austenitic SS	304 SS	Yes	Region 5	IE	Expansion

Assembly/Component Name	Material Category	Material Type/Grade	Structural Weld	Neutron Fluence Region(1)	Embrittlement Screening	MRP-191/ MRP-227(2)
In-Core Instrumentation (ICI)						
ICI guide tubes	Austenitic SS	316 SS	Yes	Region 4	IE	Category A

Notes:

1. The neutron fluence regions are defined as follows:
   - Region 1: fluence < 1x1020 n/cm^2
   - Region 2: 1x1020 n/cm^2 ≤ fluence < 7x1020 n/cm^2
   - Region 3: 7x1020 n/cm^2 ≤ fluence < 1x1021 n/cm^2
   - Region 4: 1x1021 n/cm^2 ≤ fluence < 1x1022 n/cm^2
   - Region 5: 1x1022 n/cm^2 ≤ fluence < 5x1022 n/cm^2
   - Region 6: 5x1022 n/cm^2 ≤ fluence
2. The MRP-191 categorization in this table is Category A. The MRP-227 categorizations in this table are Primary, Expansion, Existing, and No Additional Measures.

Table 4-4
Combustion Engineering Components Containing Cast Austenitic Stainless Steel and/or Structural Welds that Fell Below the MRP-191 Irradiation Embrittlement Screening Criterion

Assembly/Component Name	Material Category	Material Type/Grade	Structural Weld	Embrittlement Screening	MRP-191/ MRP-227(1)	Peak 60 Year $>10^{17}$n/cm^2
Upper Internals Assembly						
Upper guide structure support plate	Austenitic SS	304 SS	Yes	None	Category A	No
Upper guide structure support flange – upper	Austenitic SS	304 SS	Yes	None	Category A	No
Upper guide structure support flange – lower	Austenitic SS	304 SS	Yes	None	Category A	No
Cylindrical skirt	Austenitic SS	304 SS	Yes	None	Category A	No
Grid plate	Austenitic SS	304 SS	Yes	None	Category A	No
Control rod shroud – grid ring	Austenitic SS	304 SS	Yes	None	Category A	No
Control rod shroud – grid beams	Austenitic SS	304 SS	Yes	None	Category A	No
Control rod shroud – cross braces	Austenitic SS	304 SS	Yes	None	Category A	No
Guide structure support system – guide structure plate	Austenitic SS	304 SS	Yes	None	Category A	No
Guide structure support system – support cylinder	Austenitic SS	304 SS	Yes	None	Category A	No
Reactor Vessel Level Monitoring System – support structure tubes	Austenitic SS	304 SS	Yes	None	Category A	Yes
Fuel Alignment Plate	Austenitic SS	304 SS	Yes	None	Primary	Yes

Assembly/Component Name	Material Category	Material Type/Grade	Structural Weld	Embrittlement Screening	MRP-191/ MRP-227(1)	Peak 60 Year $>10^{17} n/cm^2$
Lower Support Structure						
Core support beams	Austenitic SS	304 SS	Yes	None	Category A	Yes
Bottom plate	Austenitic SS	304 SS	Yes	None	Category A	No
ICI support columns	Austenitic SS	304 SS	Yes	None	Category A	Yes
Control Element Assembly (CEA)						
CEA shrouds	Austenitic SS	304 SS	Yes	None	Category A	Yes
CEA shrouds	CASS	CPF8, CF8	Yes	TE	Category A	Yes
CEA shroud bases	Austenitic SS	304 SS	Yes	None	Category A	Yes
CEA shroud bases	CASS	CF8	Yes	TE	Category A	Yes
CEA shroud extension shaft guides	Austenitic SS	304 SS	Yes	None	Category A	No
Modified CEA shroud extension shaft guides	CASS	CF8	Yes	TE	Category A	No
Instrument Tubes	Austenitic SS	304 SS	Yes	None	Primary	Yes
Internal/external spanner nuts	Austenitic SS	304 SS	Yes	None	Category A	No
CEA shroud tie rods	Austenitic SS	304 SS	Yes	None	Category A	No
Snubber blocks	Austenitic SS	304 SS	Yes	None	Category A	No
Core Support Barrel Assembly						
Upper cylinder	Austenitic SS	304 SS	Yes	None	Expansion	Yes
Upper core barrel flange	Austenitic SS	304 SS	Yes	None	Primary	No

Assembly/Component Name	Material Category	Material Type/Grade	Structural Weld	Embrittlement Screening	MRP-191/ MRP-227[1]	Peak 60 Year $>10^{17}$n/cm^2
Lower core barrel flange	Austenitic SS	304 SS	Yes	None	Primary, Expansion	No
Core barrel snubber lugs	Austenitic SS	304 SS, 321 SS, 348 SS	Yes	None	Category A	Yes
Core barrel outlet nozzles	Austenitic SS	304 SS	Yes	None	Category A	Yes
Thermal shield	Austenitic SS	304 SS	Yes	None	Category A	Yes
Core Shroud Assembly						
Guide lugs	Austenitic SS	304 SS, 348 SS	Yes	None	Category A	Yes
In-Core Instrumentation (ICI)						
ICI nozzle support plate	Austenitic SS	304 SS	Yes	None	Category A	No
ICI thimble support plate	Austenitic SS	304 SS	Yes	None	Category A	No
ICI thimble tubes – upper	Austenitic SS	304 SS	Yes	None	Category A	No

1. The MRP-191 categorization in this table is Category A. The MRP-227 categorizations in this table are Primary, Expansion, Existing, and No Additional Measures.

Table 4-5
Westinghouse Components Identified as Cast Austenitic Stainless Steel and/or Structural Welds in MRP-191 Identified for Potential Irradiation Embrittlement

Assembly/Component Name	Material Category	Material Type/Grade	Structural Weld	Neutron Fluence Region(1)	Embrittlement Screening	MRP-191/ MRP-227(2)
Upper Internals Assembly						
Control Rod Guide Tube Assemblies and Flow Downcomers						
Flanges-lower	Austenitic SS /CASS	304/CF8	Yes	Region 3	TE/IE	Primary
Mixing Devices						
Mixing devices	CASS	CF8	Yes	Region 3	TE/IE	Category A
Upper Plenum						
UHI flow column bases	CASS	CF8	No	Region 3	TE/IE	Category A
Upper Support Column Assemblies						
Column bases	CASS	CF8	No	Region 3	TE/IE	Category A
Lower Internals Assembly						
Bottom Mounted Instrumentation (BMI) Column Assemblies						
BMI column bodies	Austenitic SS	304 SS	Yes	Region 5	IE	Expansion
BMI column cruciforms	CASS	CF8	No	Region 5	TE/IE	No Additional Measures
BMI column extension tubes	Austenitic SS	304 SS	Yes	Region 5	IE	No Additional Measures

Assembly/Component Name	Material Category	Material Type/Grade	Structural Weld	Neutron Fluence Region[(1)]	Embrittlement Screening	MRP-191/ MRP-227[(2)]
Core Barrel						
Lower core barrel	Austenitic SS	304 SS	Yes	Region 4	IE	Expansion
Upper core barrel	Austenitic SS	304 SS	Yes	Region 4	IE	Expansion
Flux Thimbles (Tubes)						
Flux thimble tube plugs	Austenitic SS	304 SS	Yes	Region 5	IE	No Additional Measures
Flux thimbles (tubes)	Austenitic SS	316 SS	Yes	Region 5	IE	Existing
Lower Core Plate and Fuel Alignment Pins						
Lower core plate	Austenitic SS	304 SS	Yes	Region 5	IE	Existing
XL lower core plate	Austenitic SS	304 SS	Yes	Region 4	IE	Existing
Lower Support Column Assemblies						
Lower support column bodies	CASS	CF8	No	Region 5	TE/IE	Expansion

Notes:

1. The neutron fluence regions are defined as follows:

   Region 1: fluence < $1x10^{20}$ n/cm^2

   Region 2: $1x10^{20}$ n/cm^2 ≤ fluence < $7x10^{20}$ n/cm^2

   Region 3: $7x10^{20}$ n/cm^2 ≤ fluence < $1x10^{21}$ n/cm^2

   Region 4: $1x10^{21}$ n/cm^2 ≤ fluence < $1x10^{22}$ n/cm^2

   Region 5: $1x10^{22}$ n/cm^2 ≤ fluence < $5x10^{22}$ n/cm^2

   Region 6: $5x10^{22}$ n/cm^2 ≤ fluence

2. The MRP-191 categorization in this table is Category A. The MRP-227 categorizations in this table are Primary, Expansion, Existing, and No Additional Measures.

Table 4-6
Westinghouse Components Identified as Cast Austenitic Stainless Steel and/or Structural Welds in MRP-191 that Fell Below Irradiation Embrittlement Screening Criterion

Assembly/Component Name	Material Category	Material Type/Grade	Structural Weld	Embrittlement Screening	MRP-191/ MRP-227[(1)]	Peak 60 Year $>10^{17}$n/cm^2
Upper Internals Assembly						
Control Rod Guide Tube Assemblies and Flow Downcomers						
Enclosure pins	Austenitic SS	304 SS	Yes	None	Category A	No
Upper guide tube enclosures	Austenitic SS	304 SS	Yes	None	Category A	No
Flanges-intermediate	Austenitic SS	304 SS	Yes	None	Category A	No
Flanges-intermediate	CASS	CF8	Yes	TE	Category A	No
Guide plates/cards	Austenitic SS	304 SS	Yes	None	Primary	No
Upper Support Column Assemblies						
Extension tubes	Austenitic SS	304 SS	Yes	None	Category A	No
Upper Support Plate Assembly						
Deep beam ribs	Austenitic SS	304 SS	Yes	None	Category A	No
Deep beam stiffeners	Austenitic SS	304 SS	Yes	None	Category A	No
Inverted top hat (ITH) flange	Austenitic SS	304 SS	Yes	None	Category A	No
Inverted top hat (ITH) upper support plate	Austenitic SS	304 SS	Yes	None	Category A	No
Upper support ring or skirt	Austenitic SS	304 SS	Yes	None	Existing	No

Assembly/Component Name	Material Category	Material Type/Grade	Structural Weld	Embrittlement Screening	MRP-191/ MRP-227[1]	Peak 60 Year $>10^{17} n/cm^2$
Lower Internals Assembly						
Core Barrel						
Core barrel flange	Austenitic SS	304 SS	Yes	None	Primary	No
Core barrel outlet nozzles	Austenitic SS	304 SS	Yes	None	Expansion	Yes
Lower Support Casting or Forging						
Lower support casting	CASS	CF8	No	TE	Category A	Yes
Radial Support Keys						
Radial support keys	Austenitic SS	304 SS	Yes	None	Category A	No
Secondary Core Support (SCS) Assembly						
SCS base plate	Austenitic SS	304 SS	Yes	None	Category A	No
Interfacing Components						
Upper core plate alignment pins	Austenitic SS	304 SS	Yes	None	Existing	Yes

1. The MRP-191 categorization in this table is Category A. The MRP-227 categorizations in this table are Primary, Expansion, Existing, and No Additional Measures

5
GAP ANALYSES

This chapter provides an assessment of needed fracture toughness data requirements for development of evaluation acceptance criteria. In addition, this chapter identifies any additional test data or structural analyses that are required to fill potential gaps in existing data.

5.1 B&W Gap Analysis

5.1.1 CASS Items

As noted in Chapter 4, there are three CASS items in the B&W design PWR internals that are expected to exceed a fluence of $1x10^{17}$ n/cm^2, E > 1.0 MeV at the end of a 60-year lifetime. These three items are:

- IMI guide tube assembly spiders
- CRGT assembly spacer castings
- Plenum cylinder reinforcement castings (DB only)

The IMI guide tube assembly spiders are classified as a Primary item in MRP-227.[5-1] A visual (VT-3) examination of the IMI guide tube spiders is to be performed. The IMI guide tube spiders are being examined to detect spider arms that do not align with the lower fuel assembly support pad center bolt. The evaluation acceptance criteria recommended methodology in WCAP-17096[5-2] is to perform an analysis to show that one or more missing spider arms or a completely missing spider will not result in loss of function of the IMI guide tube. Thus, no fracture mechanics evaluations are needed. Since there are 52 IMI guide tubes in each B&W unit, a redundancy argument may also be adequate.

The CRGT assembly spacer castings are classified as an Expansion item in MRP-227. If necessary, a visual (VT-3) examination of the CRGT spacer castings is to be performed. The spacer castings have limited accessibility from the top or bottom of the CRGT through a center free-path (once the plenum assembly is removed from the vessel). Examination at the quarter points where the threaded connections are present is possible. These lanes are not blocked by the rod guide tubes. The examination would look for gross cracking of the spacer surface or evidence that the spacer is not approximately centered. The threaded fasteners are welded to the OD of the pipe column so it is possible that a degraded threaded location would not be detected. Since there are 69 CRDMs in each B&W unit, the evaluation acceptance criteria recommended methodology in WCAP-17096 is to perform a reactivity analysis to determine the number of CRDMs that are required for shut down of the reactor. Thus, no fracture mechanics evaluations are needed.

There are two plenum cylinder reinforcement castings (at DB only), which have not been included in any MRP or PWROG evaluations as of the preparation of this document. As noted in Table 4-1, only the lower edge of these castings are expected to exceed a fluence level of 1×10^{17} n/cm^2, E > 1.0 MeV at the end of a 60-year lifetime. Assuming the evaluations and conclusions would be the same as the wrought reinforcement plates at the other B&W units, this item would be classified as "No Additional Measures." It is also possible that this item could be dispositioned by reviewing the materials records and determining the ferrite content to be below the MRP-175[5-3] screening criteria, which would also classify it as "No Additional Measures."

Therefore, it is concluded that the CASS items in the B&W design PWR internals are redundant and/or potentially able to be analyzed for functionality in the anticipated degraded conditions. Replacement of the degraded item or component is also a potential option. Thus, no fracture toughness properties would be required for fracture mechanics analyses.

5.1.2 Austenitic Stainless Steel Welded Items

As noted in Chapter 4, there are 57 welded items in the B&W design PWR internals that are expected to exceed a fluence of 1×10^{17} n/cm^2, E > 1.0 MeV at the end of a 60-year lifetime. Of these 57 items, 36 will reach a 60-year lifetime fluence between 1×10^{17} and 6.7×10^{20} n/cm^2, E > 1.0 MeV. All of these 36 items have been classified as "No Additional Measures."

The majority of these 36 austenitic stainless steel welded items in the B&W design PWR internals are redundant, repairable, or potentially able to be analyzed for functionality in the anticipated degraded conditions. There are only a few locations where, if cracking were to occur and become identified, fracture toughness data may be required to perform a fracture mechanics analysis to justify continued operation in the degraded condition:

- Core barrel cylinder (top and bottom) flange circumferential welds
- Core barrel cylinder (top and bottom) vertical seam welds
- Core barrel top cylinder to core barrel bottom cylinder circumferential seam weld
- Thermal shield (top and bottom) vertical seam welds
- Thermal shield top cylinder to thermal shield bottom cylinder circumferential seam weld
- Core support shield cylinder to bottom flange circumferential seam weld
- Core support shield cylinder to outlet nozzle weld
- Core support shield cylinder to outlet nozzle fillet weld
- Core support shield cylinder vertical seam weld
- Flow deflector (U-baffle) to core support shield cylinder weld
- Flow distributor flange to flow distributor head circumferential seam weld
- Flow distributor plate to lower grid shell forging welds
- Lower grid forging (or weldment) to lower grid shell forging welds
- Plenum cylinder to bottom plenum cylinder flange circumferential seam weld

- Plenum cylinder bottom flange to reinforcement weld
- Plenum cylinder vertical seam weld
- Plenum cylinder reinforcement to plenum cylinder welds

There are limited data available on fracture toughness of austenitic stainless steel welds to perform analyses in these items. Thus, there is a gap in the information needed to determine both a defensible screening fluence and a lower bound fracture toughness value for long-term thermal embrittlement with low doses of irradiation in welded stainless steel components and items. However, as concluded in Chapter 3, there currently is no empirical evidence of a synergistic effect of irradiation and thermal aging for both CASS and austenitic stainless steel welds, but neither is there sufficient evidence to reject such a hypothesis.

In addition, as shown in the example calculations for various postulated flaw types provided in MRP-210 [5-4], using available lower bound fracture toughness properties for austenitic stainless steel materials including the most highly irradiated CASS, and austenitic stainless steel welds in the internals, adequate flaw tolerance remains. Some CASS and weld locations at lower dose between 1×10^{17} and 6.7×10^{20} n/cm^2, E > 1.0 MeV have more complex geometries and have not been evaluated by MRP-210; however these locations are not expected to require fracture mechanics analyses.

5.1.3 B&W Summary

Lowering the fluence screening values for the CASS and weld components to 1×10^{17} n/cm^2, would require additional items to be evaluated. Because most affected items have redundancy in the designs, they are potentially able to be analyzed for functionality in the degraded conditions. Replacement of the degraded item or component is also a potential option. Thus, no fracture toughness properties would be required for fracture mechanics analyses in most cases.

There are only a few B&W design austenitic stainless steel weld locations where, if cracking were to occur and become identified, fracture toughness data may be required to perform a fracture toughness evaluation to justify continued operation in the degraded condition. There remains a gap in the information needed to determine a lower bound fracture toughness value for long-term thermal embrittlement with low doses of irradiation in welded stainless steel components and items.

Assessment of the B&W design CASS components revealed that there are no gaps in required data to predict fracture toughness for potentially degraded items. To disposition the items, more detailed evaluations may be required, but fracture mechanics analyses are not needed. It is also possible that these items could be dispositioned by reviewing the materials fabrication records.

5.2 CE Gap Analysis

5.2.1 CASS Items

There are a number of gaps in the available data and models for the components fabricated from cast austenitic stainless steel or containing structural welds in the Combustion Engineering design PWR internals. These gaps are highlighted and addressed in this section.

MRP-191[5-5] indicated there are only four components fabricated from CASS in the CE design PWR internals:

- Core support columns
- CEA shrouds
- CEA shroud bases
- Modified CEA shroud extension shaft guides

The core support columns were dispositioned as an Expansion inspection component in MRP-227 [5-1] and do not require further analysis or measurement to address IE or TE. The other three components require further analysis.

The three remaining internals components (CEA shrouds, CEA shroud bases, and modified CEA shroud extension shaft guides) were evaluated to determine the likelihood of synergistic embrittlement.

The CEA shroud bases accumulate enough neutron fluence to be categorized as part of Region 2 per MRP-191. Components in fluence Region 2 are expected to receive between $1x10^{20}$ and $7x10^{20}$ n/cm^2, E > 1.0 MeV (0.15 to 1.1 dpa) by the end of the 60-year license period. Thus, the CEA shroud bases are expected to exceed the NRC screening level for synergistic embrittlement. However, CASS versions of the bases are present in only one plant. All other CE plants with this component have bases fabricated from wrought Type 304 stainless steel. Consideration of the shroud bases in a generic manner is not appropriate, since CASS shroud bases are present in only one plant.

The modified CEA shroud extension shaft guides are located at the top of the CEA shroud assembly approximately in line with the mating surface between the reactor vessel and the reactor vessel head. No specific fluence or stress analyses have been completed for this component, but in this region of the reactor, a very low accumulated fluence (probably lower than the $1x10^{17}$ n/cm^2, E > 1.0 MeV screening value) would be expected. Also, this component does not serve as a core support structure, and the loads are expected to be quite low. The knowledge gap for this component is the lack of analyses that address effects of fluence or stress. However, results of any analyses are expected to show low fluence and low stress. MRP-191 classified this component as a Category A component because of the low likelihood for failure.

The CEA shrouds constitute part of a core support structure in the CE design and are likely to receive enough fluence to exceed the $1x10^{17}$ n/cm^2, E > 1.0 MeV level. An initial survey of the CE plants indicated that a significant portion of the CE fleet has CEA shrouds fabricated from

CASS; the cast material. There are several reasons that further analysis or survey of plant materials records are likely to disposition this component. The first is that this component is likely to have less than the 20% delta ferrite content. According to MRP-175, centrifugal castings containing less than 20% delta ferrite can be screened out for consideration of thermal aging embrittlement. The one plant that does not contain centrifugally cast CEA shrouds will also have to consider the molybdenum content of the CASS material when taking this approach. The second reason is that the expert panel review summarized in MRP-191 determined that the shrouds had a low likelihood of failure.

5.2.2 Austenitic Stainless Steel Welded Items

The MRP-227 inspection recommendations include several components with structural welds potentially requiring fracture mechanics analysis. These welded components include the core support plate, core support deep beams, lower cylinder of the core support barrel assembly, core shroud plates, and core shroud former plates. These components are relatively large and experience a range of fluences. The toughness associated with any flaw in these components would depend on the length of service and location of the flaw. Toughness values in high fluence regions of the component may be based on bounding toughness curves for austenitic stainless steels. However, there is no data to demonstrate the validity of this toughness curve for weld metals beyond 15 dpa. A flaw in a welded section of a component well-removed from the core may occur at a low fluence location where it would be necessary to consider potential synergistic thermal and irradiation effects on the fracture toughness of the weld.

There is limited information available on fracture toughness of stainless steel welds to support analysis of the welds in these thick components at higher fluences. There is a gap in the information needed to determine a fracture toughness prediction model for long-term irradiations in welded stainless steel components. This data is required to support analysis of flaws in the components listed above.

Although several additional components included in the MRP-227 inspection plan have some apparent structural function (the core support columns, the core shroud ribs and rings, and the ICI guide tubes) there is significant redundancy built into these structural systems, and the structural evaluation would generally assume complete failure of any individual component with an observed flaw. In this case, determination of the fracture toughness would not be necessary. All of these (except the ICI guide tubes) are currently covered by either a Primary or Expansion inspection in MRP-227. The guide tubes were addressed in MRP-191 as a Category A item having both a low likelihood of failure and a low likelihood of causing damage even if they fail. Thus, there is no data gap associated with these components.

There are a total of 12 structural welds identified in MRP-191 that were not originally identified for irradiation embrittlement, but have peak neutron fluences in excess of the $1x10^{17}$ n/cm^2, E > 1.0 MeV screening limit suggested by the NRC for CASS materials. It should be noted that the ferrite content in reactor internals welds is generally well controlled. Applying the CASS assumptions to these welds is an extremely conservative assumption. However based on the

CASS screening fluence, the following welded components would be additionally identified for potential synergistic effects of irradiation and thermal embrittlement:

- Upper Internals Assembly
    - Reactor Vessel Level Monitoring System – support structure tubes
    - Fuel alignment plate
- Lower Support Structure
    - Core support beams
    - ICI support columns
- Control Element Assembly (CEA)
    - CEA shrouds
    - CEA shroud bases
    - Instrument tubes
- Core Support Barrel Assembly
    - Upper cylinder
    - Core barrel snubber lugs
    - Core barrel outlet nozzles
    - Thermal shield
- Core Shroud Assembly
    - Guide lugs

Of these twelve, the fuel alignment plate, instrument tubes and core barrel upper cylinder are already incorporated in the MRP-227 inspection program. The remaining nine components are not currently included in the MRP-227 program. Based on existing criteria there is no reason to expect that embrittlement should be a concern for these components. However, if data from ongoing programs indicates that synergistic effects of thermal aging and irradiation embrittlement can lead to degradation in low fluence, long time exposures, these assumptions might need to be re-evaluated. Large structures, such as the core barrel, thermal shield and fuel alignment plate are expected to be highly flaw tolerant. Other components in the lower support structure and CEA are part of redundant systems where failure of a single component is not expected to lead to failure of the structure. Data and models generated to support analysis of welds in the core barrel, core support plate, core shroud assembly, or core support deep beams could also be used to address this gap.

5.2.3 CE Summary

Although it is recognized that the effects of long term irradiation on the fracture toughness of CASS components is not fully understood, the MRP-227 inspection recommendations provide the basis for a comprehensive aging management program for CASS components the CE design.

Lowering the fluence screening values for the CASS components to $1x10^{17}$ n/cm^2, E > 1.0 MeV would have minimal impact on the program recommendations. The analysis of the CE CASS components revealed that using the more conservative screening fluence for irradiation embrittlement would identify potential concerns in the CEA shrouds and shroud bases. It should be noted that the cast shroud bases were used in only one plant. It is probable that either of these components could be dispositioned on a plant specific basis by reviewing the materials records for the fabrication of these components. Although the CEA shrouds are part of the core support structure, there are multiple shrouds and cracking in a single shroud would not imply failure of the structure. In this case, a flaw tolerance analysis of a single crack is highly unlikely.

There are a number of welded components already incorporated in the MRP-227 guidelines. Evaluation of a flaw detected in any of these welds would require an estimate of the fracture based on the local fluence. Current models, which provide lower bound estimates of irradiated fracture toughness are based primarily on testing of type 304 and type 316 stainless steels. Additional work is required to demonstrate that these models bound weld materials over the full range of exposure conditions.

Review of the Combustion Engineering components containing structural welds indicated that lowering the screening fluence to 10^{17} n/cm^2, E > 1.0 MeV would add nine welded components with potential embrittlement concerns. Most of these components are either highly flaw tolerant or are parts of redundant systems. In either case, detailed knowledge of the fracture toughness would not be required to resolve any issues.

The gaps identified in the review of CASS and welded components in the CE design are:

1. Need to confirm applicability of MRP-175 screening fluence to CASS and austenitic stainless steel welds.
2. Absence of a procedure to estimate fracture toughness in stainless steel welds of thick components like the core barrel, core support plate, core support deep beams, and core shroud assembly plates

5.3 Westinghouse Gap Analysis

5.3.1 CASS Items

There are relatively few CASS components in the Westinghouse design PWR internals. CASS was primarily used in non-structural or redundant components. Therefore, there are relatively few requirements for fracture toughness determinations. Six of the eight CASS components were already identified in the screening process for potential irradiation embrittlement concerns due to relatively high peak neutron fluences. However, there were no requirements for flaw tolerance analyses in the evaluation procedures for these components; therefore, there were no requirements for fracture toughness data. Thus, there are no data gaps identified for these six screened-in components.

There are two remaining CASS components that were not identified for irradiation embrittlement:

- Intermediate flange in the control rod guide tube assemblies

- Lower support castings

The intermediate flanges on the control rod guide tube assemblies are not expected to exceed the $1x10^{17}$ n/cm^2, E > 1.0 MeV screening limit because these flanges are well removed from the core. Cracking in the intermediate flanges of the control rod guide tube assembly is less probable than cracking in the lower flanges, where both fluences and bending stresses are expected to be higher. At most, the intermediate flange would be an Expansion item for the lower flange. Should a crack be observed in an intermediate flange in this component, a full flaw tolerance analysis would not likely be required, since the component in such a condition would probably be considered as failed. Therefore, there is no apparent need for fracture toughness data for the intermediate flange material.

A small fraction of Westinghouse plants have lower support castings rather than lower support forgings. Although the lower support casting is well-removed from the core, portions of this component may experience fluences greater than $1x10^{17}$ n/cm^2, E > 1.0 MeV. In the unlikely event that cracking is observed on the surface of the lower support casting, a flaw tolerance analysis might be undertaken. Since it is a large component, it may be possible to show by analysis that the stresses at the location of the crack are too low to drive crack propagation. If a flaw tolerance analysis is conducted, an estimate of the fracture toughness in the lower support casting would be required. It may be possible to assume a lower bound toughness equivalent to highly irradiated austenitic steel and demonstrate that significant margin against failure remains.

There is a potential gap for fracture toughness data to evaluate flaws in CASS lower support castings. However, there is a reasonable potential for demonstrating structural integrity with suitably conservative assumptions. There are no other data gaps for CASS components for the Westinghouse design.

5.3.2 Austenitic Stainless Steel Welded Items

MRP-227 places six of the eight weld containing components from Table 4-5 as either Primary, Expansion, or Existing Programs components. Components on this list potentially requiring fracture mechanics analysis include the upper core barrel, lower core barrel, lower core plate, and XL (Westinghouse extra long reactor design) lower core plate. These components are relatively large and experience a range of fluences. A flaw in a welded section of a component well-removed from the core may occur at a location where the fluence is below the normal irradiation embrittlement screening. In this case, it would be necessary to consider potential synergistic thermal and irradiation effects on the fracture toughness of the weld metal.

There are insufficient fracture toughness data for stainless steel weld metals to support analysis of welds in core barrels and lower core plates. There is a gap in the information needed to determine a fracture toughness prediction model for long-term irradiations in welded stainless steel components. These data are required to support analysis of flaws in both the core barrel structure and the lower core plate (both standard and XL designs).

Although several additional components in Table 4-5 have some apparent structural function (control rod guide tube assembly lower flange, upper head injection (UHI) flow column bases, upper support column assembly bases, and lower support column bodies) there is significant redundancy built into these structural systems and the structural evaluation would generally assume complete failure of any individual component with an observed flaw. In this case, determination of the fracture toughness would not be required. All components except the UHI flow column bases are already addressed by either Primary or Expansion inspections in MRP-227. MRP-191 classified the UHI flow column bases as having both a low likelihood of failure and a low likelihood of causing damage if they were to fail. On this basis, the UHI column bases would have been placed in the "No Additional Measures" group.

Only 2 of the structural welds identified in MRP-191 that were not originally identified for irradiation embrittlement, have peak neutron fluences in excess of the $1x10^{17}$ n/cm^2, E > 1.0 MeV screening limit suggested by the NRC for CASS materials. Both of these components are already incorporated in the MRP-227 inspection strategy.

5.3.3 Westinghouse Summary

There are no requirements for additional fracture toughness data to support analysis of CASS components that were identified for potential irradiation embrittlement.

The analysis of the Westinghouse CASS components revealed that decreasing the screening fluence to $1x10^{17}$ n/cm^2, E > 1.0 MeV would identify one additional component, the lower support casting. However, a relatively small fraction of Westinghouse plants have castings at this location. Furthermore, this component is deemed unlikely to experience stresses high enough to affect structural integrity because of its large size. Conservative analysis assumptions may be sufficient to demonstrate the structural integrity of the lower support castings.

There are a limited number of weld-containing components in the Westinghouse design that may require a flaw tolerance analysis to demonstrate acceptability of a flaw. Evaluation of a flaw detected in any of these welds would require an estimate of the fracture toughness based on the local fluence. Current models, which provide lower bound estimates of irradiated fracture toughness are based primarily on testing of type 304 and type 316 stainless steels. Additional work is required to demonstrate that these models bound weld materials over the full range of exposure conditions.

Review of the Westinghouse components containing structural welds revealed one gap:

- Absence of a procedure to estimate fracture toughness in stainless steel welds of thick components like the upper core barrel, lower core barrel, lower core plate, and XL (Westinghouse extra long reactor design) lower core plate.

5.4 References

[5-1] Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Rev. 0). EPRI, Palo Alto, CA: 2008. 1016596.

[5-2] WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," Westinghouse Electric Co. LLC Report, 2009.

[5-3] Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175) – EPRI, Palo Alto, CA: 2005. 1012081.

[5-4] Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internals Components (MRP-210). EPRI, Palo Alto, CA: 2006. 1016106.

[5-5] Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Designs (MRP-191). EPRI, Palo Alto, CA: 2006. 1013234.

6 SUMMARY, CONCLUSIONS, AND RECOMMENDATIONS

A review of the component items that could potentially be affected by a synergy between thermal aging and irradiation embrittlement was conducted for the PWR internals of the Babcock & Wilcox, Combustion Engineering, and Westinghouse designs. The component item lists created in MRP-189-Rev. 1, MRP-191, and recently identified items were examined to identify the items either fabricated from CASS or containing structural welds. These materials are potentially susceptible to thermal embrittlement, which could act in a synergistic fashion with the irradiation embrittlement that these items will also experience. The items were originally evaluated under the irradiation embrittlement limits set by MRP-175; however, the NRC has proposed a more stringent screening criterion in the current GALL report (lower fluence screening value for irradiation embrittlement), that may require the evaluation of additional items.

The GALL screening criterion has been evaluated with recently identified literature data on CASS and austenitic stainless steel materials. The reviewed literature included materials exposed to a variety of combinations of thermal aging and irradiation embrittlement. Thermal embrittlement is primarily controlled by the amount, size, and distribution of the ferrite phase in the duplex austenitic/ferrite structure. There are no test data or mechanisms to suggest that the original ferrite phase amount, size, and distribution or alpha-prime precipitation in the ferrite would be significantly altered by irradiation, especially at a neutron dose level of 1×10^{17} n/cm^2, E > 1.0 MeV. The available test data clearly indicate that this level of fluence does not change the fracture toughness of CASS or welds that have been completely thermally aged.

Fracture of thermally embrittled CASS and welds is by brittle cleavage fracture of the ferrite phase or separation of the ferrite/austenite phase boundaries. Inclusions also play a significant role in the fracture toughness of austenitic stainless steel welds. Fracture of highly irradiated CASS and welds on the other hand is by channel fracture, which exhibits localized plastic deformation. Therefore, prior to the formation of significant irradiation defects as evidenced by a detectable loss of fracture toughness due to irradiation embrittlement, there is no mechanism for the cleavage fracture of embrittled ferrite to be affected by irradiation defects. The minimum fluence level for a detectable fracture toughness loss due to irradiation embrittlement has been well established to be >0.5 dpa (3.3×10^{20} n/cm^2, E > 1.0 MeV) [3-10] in austenitic stainless steels. This level of irradiation is much higher than the screening threshold of 1×10^{17} n/cm^2, E > 1.0 MeV as suggested by the NRC to account for a potential synergistic effect.

The available test data have not shown any evidence to indicate a synergistic effect of thermal aging and irradiation. Given the wide scatter of fracture toughness values related to thermal and irradiation embrittlement, it has been the general practice to only use a minimum or lower bound fracture toughness in fracture mechanics based evaluations. As Chopra et al. recently concluded (see Section 3.1), the minimum fracture toughness below 0.3 dpa can be estimated by considering thermal embrittlement alone, and the minimum fracture toughness above 0.3 dpa can be estimated by using the lesser of the minimum predicted thermal embrittlement or the lower bound irradiated fracture toughness curve.

Nevertheless, there has not been sufficient testing to disprove any synergistic potential under all possible combinations of different thermal aging conditions and irradiation conditions. The most significant gap in the test data available is the lack of data from specimens exposed to low (less than ~1 dpa) radiation doses and long thermal aging times at PWR operating temperatures. This is represented graphically in Figure 3-9. The shaded areas in Figure 3-9 define regions where data exist, while the unshaded regions are regions where little or no data exist. As shown by the information presented earlier in this report, there currently is no empirical evidence of a synergistic effect of irradiation and thermal aging. Similarly, there is no empirical evidence on which to reject this hypothesis in the thermal exposure-irradiation exposure regimes represented by the unshaded regions of Figure 3-9. However, it appears to be clear from the available data presented in this report that the NRC recommended fluence screening criterion for a synergistic effect (i.e., $1x10^{17}$ n/cm^2, E > 1.0 MeV) is far too conservative and that the MRP-175 screening criterion (i.e., $6x10^{20}$ n/cm^2, E > 1.0 MeV) is sufficiently conservative.

There are a limited number of cast stainless steel components in all three reactor designs that did not exceed the original MRP screening limit of $6x10^{20}$ n/cm^2, E > 1.0 MeV, but are predicted to exceed the more conservative GALL limit of $1x10^{17}$ n/cm^2, E > 1.0 MeV. These components have been considered on a case-by-case basis. All of the CASS items could likely be dispositioned by examination of unit-specific material records or by more detailed analysis of the fluence or stress of the CASS items. In many cases, the loss of fracture toughness is not directly relevant to the suitability of the component for continued operation. While additional research to determine the appropriate screening fluence for irradiation embrittlement in CASS may have limited applicability to some components in the CE design, the expected impact on the aging management strategy is minimal. No other gaps related to embrittlement of CASS were identified.

The inspection requirements for CASS component items in MRP-227 provides augmented inspection to detect cracking in components that have potentially experienced a loss of fracture toughness. The MRP-227 inspection strategy considers the implications of loss of fracture toughness in the limited number of reactor internals component items that are potentially fabricated from CASS and provides relevant aging management recommendations. The inspection strategy is based on an item-by-item evaluation on a generic vendor design basis. This process defines an adequate Aging Management Program consistent with the intent of GALL AMP XI.M12 and X1.M13. Based on the findings of this study it is evident that implementation of the MRP-227 Guidelines provides appropriate aging management for irradiated CASS. The conclusions of this study support a recommendation to withdraw GALL AMP XI.M13 to allow establishment of requirements for aging management of CASS internals based on MRP-227 in GALL AMP XI.M16.

The austenitic stainless steel welds are a more significant gap because of the lack of test data. To address many of these items, data and analyses will be required to determine the screening fluence for irradiation embrittlement in stainless steel welds. A procedure that estimates the fracture toughness of a thick component with flaws will be required to permit analysis of any flaws observed.

It is impractical to consider producing data on long-term irradiation from test reactor irradiations. Therefore, the best way to generate data to fill this gap is to test components that have been removed from service in operating reactors. It is highly recommended that all plans for reactor decommissioning or component replacement be monitored to identify potential sources of CASS and austenitic stainless steel weld materials that would expand the current range of fracture toughness data.

Based on the conclusions and information gaps, the following recommendations are provided:

- Additional austenitic stainless steel weld data are needed
- Analytical efforts (including identification of stress, temperature, and fluence) for B&W, CE, and Westinghouse design CASS and austenitic stainless steel welded items should be completed

A
CASS AND AUSTENITIC STAINLESS STEEL WELDS IN PWRS INTERNALS, GALL REPORT (NUREG-1801, REV. 1)

A.1 GALL Tables for CASS and Stainless Steel Welds

GALL Volume 1, Table 1, ID 80 lists the following:

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Generic Item	Unique Item
80	PWR	Cast austenitic stainless steel reactor vessel internals (e.g., upper internals assembly, lower internals assembly, CEA shroud assemblies, control rod guide tube assembly, core support shield assembly, lower gird assembly)	Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement	Thermal Aging and Neutron Irradiation Embrittlement of CASS	No	R-111 R-140 R-153 R-171 R-183 R-191 R-206	IV.B2-37 IV.B2-21 IV.B3-1 IV.B3-18 IV.B4-4 IV.B4-21 IV.B4-28

GALL Volume 2, Table IV.B2 lists the following

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-21 (R-140)	IV.B2.5-m	Lower internal assembly Lower support casting Lower support plate columns	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/thermal aging and neutron irradiation embrittlement	Chapter XI.M13, “Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)”	No
IV.B2-37 (R-111)	IV.B2.1-g	Upper internals assembly Upper support column (only cast austenitic stainless steel portions)	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/thermal aging and neutron irradiation embrittlement	Chapter XI.M13, “Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)”	No

CASS and Austenitic Stainless Steel Welds in PWRs Internals, GALL Report (NUREG-1801, Rev. 1)

GALL Volume 2, Table IV.B3 lists the following

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3-1 (R-153)	IV.B3.2-e	CEA shroud assemblies	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/thermal aging and neutron irradiation embrittlement	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
IV.B3-18 (R-171)	IV.B3.5-f	Lower internal assembly Core support column	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/thermal aging and neutron irradiation embrittlement	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No

GALL Volume 2, Table IV.B4 lists the following

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4-4 (R-183)	IV.B4.3-d	Control rod guide tube (CRGT) assembly CRGT spacer casting	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/thermal aging and neutron irradiation embrittlement	Chapter XI.M13, “Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)”	No
IV.B4-21 (R-191)	IV.B4.4-g	Core support shield assembly Outlet and vent valve nozzles Vent valve body and retaining ring	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/thermal aging and neutron irradiation embrittlement	Chapter XI.M13, “Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)”	No
IV.B4-28 (R-206)	IV.B4.6-e	Lower grid assembly In-core guide tube spider castings	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/thermal aging and neutron irradiation embrittlement	Chapter XI.M13, “Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)”	No

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