MRP Responses to U.S. Nuclear Regulatory Commission (NRC) Comments on MRP-175 "Materials Reliability Program: PWR Internals Aging Degradation Mechanism Screening and Threshold Values"

A. Thermal Embrittlement

1) Table 2-4 contains thermal aging embrittlement criteria for cast austenitic stainless steel (CASS). The NRC staff position on embrittlement of CASS is documented in a letter from Grimes (NRC) to Walters (NEI) dated May 19, 2000 (Agencywide Documents Access and Management System Accession No. ML003717179). The NRC staff position on determining the amount of ferrite is that the amount of ferrite should be calculated using Hull's equivalent factors or a method producing an equivalent level of accuracy (plus/minus 6 percent deviation between measured and calculated values). What is the impact of using Hull's equivalent factors on the Materials Reliability Program (MRP) analysis? It is recommended that this information should be included in any guidance to licensees.

   MRP Response
   As indicated in Section E.3 of MRP-175, the suggested CASS thermal aging embrittlement screening criteria for MRP efforts are based on the recognized industry efforts (references E-38 and E-39). Reference E-39 is the NRC staff position cited above and therefore has been included in the work.

2) In 2001, one Babcock and Wilcox (B&W) licensee experienced failures of two control rod drive mechanism 17-4 PH (precipitation hardened) leadscrew male couplings. The failures were attributed to thermal embrittlement of the 17-4 PH materials. It is recommended that the MRP should consider previous plant operating experience related to any failures of 17-4 PH materials due to thermal embrittlement in identifying susceptible 17-4 PH reactor vessel internal (RVI) components.

   MRP Response
   All martensitic precipitation-hardenable stainless steel alloys, which include Type 17-4 PH material, are screened as potentially susceptible to thermal aging embrittlement by the screening criteria in MRP-175 for additional evaluation. Operating experience is one of the considerations in determination of the categorization of each component item.

B. Void Swelling

For locations with temperatures greater than or equal to 320°C and neutron exposure greater than or equal to 20 displacement per atom (dpa), MRP-175, Appendix G establishes a 2.5 percent void swelling criteria. This criteria is not identified in Table 2-6, "Void Swelling Criteria for PWR Internals Materials." Was this criteria utilized in determining whether a component is susceptible to void swelling and whether additional functionality evaluation is necessary? Clarification regarding this issue could be beneficial.

MRP Response
The 2.5% void swelling criterion referred to in your question is not a screening criterion established by MRP-175. It is discussed in Appendix G as a suggested criterion for use in follow-on evaluations for component items once an item has been screened as potentially susceptible to void swelling. Therefore, no, the 2.5% void swelling value is not utilized in determining whether a component is susceptible to void swelling.

C. Irradiation-Enhanced Stress Relaxation

Table 2-7 provides thermal and irradiation-enhanced stress relaxation criteria for bolts and springs. Are all bolts with a preload screened in as susceptible to stress relaxation regardless of the neutron fluence? Section H.3 in the report indicates that a 60 percent loss of initial preload occurs due to irradiation-enhanced stress relaxation when the bolts or spring components are exposed to a threshold neutron fluence value of 1.3 X 10^{20} n/cm^2 (E > 1.0 MeV). How is the dose neutron fluence limit being used? Will the dose neutron fluence limit ensure that there is adequate preload on bolts to satisfy American Society of Mechanical Engineers (ASME) Code bolt stress criteria under design accident conditions considering horizontal loading (maximum coefficient of friction of 0.20) and plate/member bowing due to vertical loading? It is recommended that MRP document whether 60 percent loss of preload is taken into account in the current licensing basis bolt stress calculations.

MRP Response

Yes, all bolts and springs that require preload for functionality are initially screened in as potentially susceptible to thermal stress relaxation. Also, all bolted or spring locations exposed to a neutron fluence value of ≥ 1.3 X 10^{20} n/cm^2 (E > 1.0 MeV) are initially screened in as potentially susceptible to irradiation-enhanced stress relaxation and irradiation creep. All bolt or spring locations initially screened in by the screening criteria are then subjected to follow-on evaluations. Any existing current licensing basis bolt stress calculations are taken into account. It should be noted that the maximum dose on the component is used for screening purposes, whereas the follow-on evaluations take into account dose gradients that may result in relaxation less than 60%. Please note also that a neutron fluence value of 1.3 X 10^{20} n/cm^2 (E > 1.0 MeV) is considered a "screening value" for irradiation-enhanced stress relaxation and irradiation creep, not a "threshold value" that is indicative of the onset of irradiation-enhanced stress relaxation and irradiation creep.

D. Irradiation Assisted Stress Corrosion Cracking (IASCC)

1) Westinghouse's WCAP-14577 report, "License Renewal Evaluation: Aging Management for Reactor Internals," Revision 1-A establishes a threshold neutron fluence limit of 1 X 10^{21} n/cm^2 (E > 0. 1 MeV) for the initiation of IASCC in PWR RVI components. Section B.3 of the MRP-175 report, "Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values," specifies a threshold limit of 2 X 10^{21} n/cm^2 (E > 1.0 MeV) for the initiation of IASCC. The threshold value established by B&W's BAW-2248 report, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," stipulates a threshold limit of 1-2 X 10^{21} n/cm^2 (E > 1.0 MeV). Since the threshold limits specified in reports WCAP-14577, BAW-2248, and MRP-175 are different, it is recommended that MRP establish one threshold limit for the RVI components and document a basis for selecting a threshold fluence value of 2 X 10^{21} n/cm^2 (E > 1.0 MeV) in light of the information in the other reports.

MRP Response

One of the objectives of MRP-175 was to reach a consensus on a threshold value and a screening value (a.k.a., screening criteria) for each of the applicable PWR internals age-related degradation mechanisms (see Section 1.3). In most cases, a clear threshold value could not be ascertained. Due to the fact that numerous documents within the nuclear industry identify a variety of "threshold values" for IASCC, it was important for MRP-175 to reach a consensus on a threshold value and a screening value for IASCC, if possible. As noted in Appendix B, Section B.3, threshold values for IASCC in PWR internals materials are expected to vary considerably with material, heat, temperature, and neutron spectrum. A U.S. nuclear industry consensus was reached on a fluence threshold value of 2 X 10^{21} n/cm^2 (E > 1.0 MeV) and documented in MRP-175.

2) Section B.1.1 of the MRP-175 report states that at fluence levels greater than 6.7 X 10^{21} n/cm^2 (E > 1.0 MeV) there is little difference in IASCC behavior between cold worked and solution annealed stainless steel materials. It is recommended that MRP document information regarding the effect of cold work on IASCC for components that are exposed to fluence levels less than or equal to 6.7 X 10^{21} n/cm^2 (E > 1.0 MeV). If the onset of crack initiation due to IASCC is enhanced due to cold work on RVI components that are exposed to fluence levels less than or equal to 6.7 X 10^{21} n/cm^2 (E > 1.0 MeV), the threshold limits (as specified in Section B.3) for the cold worked RVI components and the welds should be reduced to a lower value. Does the threshold limit of 2 X 10^{21} n/cm^2 (E > 1.0 MeV) take into consideration the effect of cold work on RVI components or welds exposed to fluence levels less than or equal to 6.7 X 10^{21} n/cm^2 (E > 1.0 MeV)?

MRP Response

We believe that there is a misunderstanding here between the role of cold-work in IASCC and that recently cited as possibly contributing to IGSCC of severely cold-worked austenitic stainless steel components in PWRs. The latter is believed to be a factor in a few cases of IGSCC that have been observed in stainless steels in PWRs without any irradiation effects being involved. References B-6 and B-15 contain the currently published information regarding the effect of cold-work on IASCC behavior as a function of neutron dose. Cold-work up to the maximum level permitted (expressed as an upper yield strength limit of 95 ksi, which in most cases translates to between 10 and 20% cold-work) is a favorable factor that delays the onset of irradiation damage and irradiation hardening. This is due to the trapping of irradiation-induced point defects by dislocations, which delays the development of the irradiated microstructure responsible for irradiation hardening. Nevertheless, both solution-annealed and cold-worked stainless steels eventually attain the same degree of irradiation strengthening at fluence levels > 6.7 X 10^{21} n/cm^2 (E > 1.0 MeV). Additional evaluations by the MRP are currently underway and will soon be documented in another MRP report. As stated in the above response, a consensus was reached on a fluence threshold of 2 X 10^{21} n/cm^2 (E > 1.0 MeV) for IASCC based on the available data and material conditions.

3) Section B.3 states that IASCC is affected by bulk composition of the RVI components. The threshold limits as specified in Section B.3 do not take into consideration the effect of bulk composition on IASCC mainly due to lack of experimental data. It is recommended that the MRP document information regarding the effect of thermal neutron fluence on IASCC specifically taking into consideration the presence of boron in the austenitic stainless steel of the RVI component. The MRP should confirm that the threshold limit of 2 X 10^{21} n/cm^2 (E > 1.0 MeV) was developed taking into consideration the effect of variation in composition specifically elements like silicon, manganese, and boron, of the various RVI components.

MRP Response

Numerous studies have shown that no one parameter characterizing the irradiated microstructure and microchemistry can explain IASCC response in austenitic stainless steel materials, but rather a complex interaction of several variables appears to be involved (e.g., Reference B-44). At this time, due to the paucity of data, the threshold limit of 2 X 10^{21} n/cm^2 (E > 1.0 MeV) developed in MRP-175 is concluded to be conservative for IASCC functionality evaluations. This is specifically addressed for boron in response to the next point (4). As regards manganese, while it has been postulated that transformation of manganese into iron could liberate sulfur from MnS inclusions and thereby degrade IASCC resistance, no evidence has been obtained to support this hypothesis. By contrast, the segregation of Si to grain boundaries and other interfaces is widely believed to reduce the protective quality of the passive film since silica is highly soluble in high temperature water. However, there is no evidence of any quantifiable trend in IASCC susceptibility with silicon in the normal range of silicon contents in commercial stainless steel. At very high doses, precipitation of γ' (Ni_3Si) may cause additional irradiation-induced strengthening that could enhance IASCC susceptibility; but again, no conclusive evidence is available.

4) Section G.1 implies that at low neutron exposure (~ 10^{21} n/cm^2 thermal), 20 percent of natural boron will be converted to lithium, producing helium in the process. Since the original concentration of boron in austenitic stainless steel RVI components is not generally reported in the certified material test reports, it is difficult to assess the concentration of helium in the RVI components. The NRC staff recommends that the MRP address the synergistic effects of helium on IASCC in stainless steel RVI components. The potential effects of helium on IASCC should be taken into account in establishing neutron fluence threshold limits.

   MRP Response
   For highly irradiated austenitic stainless steels (e.g., neutron dose levels above 40 dpa), it is considered possible that very small helium bubbles could precipitate at grain boundaries and weaken them mechanically, but this hypothesis remains controversial. However, it is clear that at low neutron dose levels (around 2 X 10^{21} n/cm^2 (E > 1.0 MeV) when all the ^{10}B will have burned out long before), helium levels have not been observed to have any effect on the materials sensitivity to IASCC (c.f., Reference B-44). These data come from several different commercial heats of stainless steel, and there is no reason to believe that the boron concentrations are atypical of the range encountered in commercial products. Thereafter, helium in much higher concentrations comes primarily from nickel, via transformation of ^{58}Ni to ^{59}Ni, and therefore should not vary much from one stainless steel to another.

5) Section B.1.7 states that susceptibility to IASCC increased in Alloy X-750 when exposed to a neutron fluence greater than 1 X 10^{19} n/cm^2 (E > 1.0 MeV). No data is provided to substantiate this claim. Document how the threshold neutron fluence value of 2 X 10^{21} n/cm^2 (E > 1.0 MeV) specified in Table 2-2 of Section 2 of the MRP-175 report is applicable to Alloy 750 RVI components.

   MRP Response
   Reference B-46 contains the only available data (in simulated BWR conditions) indicating that susceptibility to IASCC of Alloy X-750 HTH Condition is increased at a fluence level of 1 X 10^{19} n/cm^2 (E > 1.0 MeV). As stated in Section B.1.7, the only available data regarding IASCC susceptibility in PWR primary conditions (References B-3 and B-4) indicate that Alloy X-750 materials behave approximately the same as Types 304 and 316 stainless steel materials. Therefore, it was concluded that the same threshold neutron fluence value would be applicable. It is noted that inspection of Alloy X-750 bolting for potential PWSCC is not governed by neutron dose levels.

6) It is widely understood that oxidizing ions are produced when the reactor coolant is exposed to neutron radiation. In RVI component crevices, these ions can play a major role in enhancing crevice corrosion. It is recommended that a discussion be provided regarding the effect of crevices in RVI components on the IASCC screening criteria listed in Table 2-2.

MRP Response

Crevice corrosion occurs when there are oxidizing species such as oxygen or H_2O_2 outside the crevice and none for all practical purposes within the crevice (because they are consumed on the crevice walls faster than they can diffuse into the crevice). In the case of exposure to ionizing radiations, the production of oxidizing species can occur everywhere and there is no differential concentration cell to drive crevice corrosion. In practice, in PWR primary water, dissolved hydrogen is also omnipresent and can diffuse rapidly along concentration gradients (if such were to exist) and eliminate oxidizing species produced radiolytically. Thus there is no reason to believe that the production of oxidizing species within crevices and their consumption by hydrogen will be any different within a crevice or outside a crevice. This has been proven experimentally and is cited in references B-3, B-7 and B-39.

7) In Table 3-3 of the MRP-175 report, the MRP states that the following components will not be inspected because they will not be exposed to neutron fluences greater than the screening criteria of 2×10^{21} n/cm^2 (E > 1.0 MeV) and, therefore, they are not susceptible to IASCC. The NRC staff recommends that the MRP document whether there are any localized areas in these components that can be exposed to neutron fluences greater than the screening criteria of 2×10^{21} n/cm^2 (E > 1.0 MeV). If so, MRP should consider that information when developing the aging management program that it intends to implement to ensure the integrity of these RVI components during the license renewal period:

- Lower Grid Assembly-to-Core Barrel Bolts
- Spiders
- Thermal Shield Cylinders
- Thermal Shield Restraint Hardfacing
- Retaining Ring

MRP Response

We believe that there is a misunderstanding in the interpretation of Table 3-3. The purpose of this table is to provide an example template for compiling PWR internals component items and the age-related degradation mechanisms that have been screened in as potentially applicable. As noted below the table (in note 1), the information in the table is fictitious and not directly applicable to any PWR design.

E Intergranular Stress Corrosion Cracking (IGSCC) / Stress Corrosion Cracking (SCC)

The recent failure of a stainless steel pressurizer heater sleeve at Braidwood, Unit 1 indicated that one of the root causes for the failure may have been related to the presence of a crevice-like geometry which led to IGSCC. It is recommended that a discussion be included regarding the effect of crevices in RVI components on the IGSCC screening criteria listed in Table 2-1.

MRP Response

The same basic response as to item D(6) applies here except in this case ionizing radiations are not a factor. In hydrogenated primary water, crevice corrosion is not plausible since differential concentration cells of oxidizing species cannot form. If crevice corrosion is shown by on-going investigations to have played a role at Braidwood Unit 1, then it could only have occurred if oxygen, not hydrogen, was present in the bulk pressurizer water at some point while it was hot (i.e., not at cold shut-down).

F. Category C Welds with Limited Accessibility

It is recommended that the MRP identify Category C welds that are not accessible for adequate inspection and discuss how current inspection technology is being used effectively in identifying the aging degradation in these welds. If the current technology cannot be effective in inspecting these welds, consider how the licensees can ensure that the intended function of these welds is adequately maintained during the license renewal period. Aging degradation of these uninspectable welds could generate loose parts. Consideration of the effect of loose parts, if any, on the structural integrity of the RVI components should be included in future inspection guidelines.

MRP Response

The MRP is currently in the process of evaluating the above recommendations. The MRP will discuss these recommendations in future meetings with the NRC staff.