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General Comment

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Attachments

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Comments on Draft NUREG-1800, Rev.2 and Draft NUREG-1801, Rev. 2

Comment 1:

In Table 4.1-3 of NUREG-1800, Rev. 2, ductility reduction of fracture toughness for the reactor vessel internals has been identified as a plant-specific time-limited aging analysis (TLAA). However, all operating reactors in the U.S. have been designed with no explicit embrittlement analysis based on the plant life for reactor core internals. Loss of fracture toughness due to radiation embrittlement was not considered in the design of light water reactor (LWR) core internal components constructed of austenitic stainless steels (SSs). These steels have used extensively in LWRs as structural alloys in the internal components of reactor pressure vessels because of their relatively high strength, ductility, and fracture toughness. Fracture of these steels occurs by stable tearing at stresses well above the yield stress, and tearing instabilities require extensive plastic deformation.

However, exposure to neutron irradiation for extended periods degrades their fracture properties. For example, at a neutron dose of 5-8 dpa the fracture toughness of SSs can decrease from a J_{Ic} value well above 200 kJ/m^2 to as low as 7.5 kJ/m^2 (or stress intensity factor K_{Jc} of 38 MPa $m^{1/2}$ or 34.5 ksi $in.^{1/2}$). Furthermore, while the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness. The decrease in fracture toughness of austenitic SSs with neutron exposure has been documented in several industry and NRC reports.

The current evaluation of the structural and functional integrity of the reactor vessel internals assumes that the irradiated materials will retain sufficient ductility to rule out the likelihood of unstable crack extension when the local applied stress intensity exceeds a K_{Jc} value of 38 MPa $m^{1/2}$ (34.5 ksi $in.^{1/2}$). The total neutron fluence for some vessel internal components that are highly stressed or have a high fatigue usage is likely to exceed 4 dpa within the 60-year life. As discussed in comment 2, the cumulative fatigue usage in air (i.e., without considering the effects of neutron irradiation and coolant environment on fatigue life) is typically above 0.5 for upper and lower core plate in PWRs, and for the core shroud in BWRs it is above 0.35. To prevent unstable crack extension if a crack is present in these components, a TLAA to evaluate the ductility reduction of fracture toughness of highly stressed vessel internals should be performed. In reactor vessel design, since the material fracture toughness is based on an explicitly assumed 40-year plant life, pursuant to 10 CFR 54.21(c)(1), the applicant should evaluate the ductility reduction of fracture toughness for the reactor vessel internals.

Comment 2:

In NUREG-1800, Rev. 2, Section 4.3.1, item #3, environmental fatigue calculations are recommended only for ASME Code Class 1 reactor coolant pressure boundary components. Similarly, in NUREG-1801, Rev. 2, Tables IV B1 through IV B4, fatigue of the reactor vessel internal components is considered a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. However, the effects of coolant environment are not included in the evaluation. It is well known that unlike carbon and low-alloy steels, which show little or no environmental effects on fatigue crack initiation in low corrosion potential environments such as hydrogen water chemistry BWR or PWR environment, the fatigue life of austenitic stainless steels (SSs) can be decreased by a factor of up to 10 in a these environments. Regulatory Guide (RG)-1.207 provides guidance for determining the acceptable fatigue life of ASME Class 1 components with consideration of the LWR environment. Fatigue evaluations for reactor vessel internal components performed by license renewal applicants, yield cumulative fatigue usage in air above 0.5 for upper and lower core plate in PWRs, and above 0.35 for BWR core shroud.

The design and construction of the core support structures of new reactors comply with the requirements of ASME Section III, Subsection NG, "Core Support Structures," and NUREG-0800, "Standard Review Plan" (SRP) Section 3.9.3. In addition, the design criteria, loading conditions, and analyses that provide the bases for the design of reactor internals other than the core support structures also meet the guidelines of Subsection NG-3000 and are constructed not to affect the integrity of the core support structures adversely (Subsection NG 1122). Subsection NG-3200 describes the requirements for the acceptability of a design by analysis, including an analysis for cyclic operation. The effect of service conditions such as neutron irradiation or the high-temperature coolant environment on fatigue life has to be included in the analysis. Subsection NG-2160 "Deterioration of Material in Service" states that it is the responsibility of the Owner to select material suitable for the conditions stated in the Design Specification (NCA-3250), with specific attention being given to the effects of service conditions upon the properties of the material.

The fatigue evaluations that are being performed for reactor core internal components of operating reactors do not include the possible effects of irradiation on fatigue life. Information regarding the effects of irradiation on fatigue crack initiation is, at best, very limited. The only data on the effect of neutron irradiation on fatigue crack initiation and fatigue crack growth rate were obtained under the fast breeder reactor program. Most of the data are on fatigue crack growth rates, and were obtained on SSs irradiated in fast reactors (primarily EBR-II) at 370-450°C and tested at 427-593°C. For Type 304 and 316 SS irradiated at 405-410°C to 1.2×10^{22} n/cm^2 (E>0.1 MeV), the crack growth rates at 427°C are up to a factor of 2 higher than those for nonirradiated material at low ΔK values (less than 40 ksi in.$^{1/2}$). A similar behavior is observed for Type 316 weld. Limited fatigue strain-vs.-life (S-N) data on irradiated SSs shows moderate decrease in life in the low-cycle regime and superior fatigue life in the high-cycle regime. There is no fatigue S-N data on materials irradiated under LWR conditions and tested at LWR operating temperatures. Several studies have shown significant differences in the microstructure and microchemistry of materials irradiated in LWRs and fast reactors. Specifically, the effect of cavities and He bubbles in austenitic SSs irradiated at temperatures above 320°C to high neutron dose levels in PWRs.

Since the possible effect of neutron irradiation on fatigue life is not considered in the current fatigue evaluations, if the known effect of LWR coolant environments on fatigue life is also ignored, the fatigue evaluations would be severely compromised. I would like to point out that SRP-LR (NUREG-1800, Rev.1) Section 4.3.1.1.1 states, "ASME Class 1 components, which include core support structures, are analyzed for metal fatigue. ASME Section III requires a fatigue analysis for Class 1 components that considers all transient loads based on the anticipated number of transients." Therefore, environmental effects should have been included in the fatigue evaluation for reactor core internal components, in support of license renewal. I request the staff to reconsider the regulatory position to consider environmental effects only for Class 1 pressure boundary components and not for the reactor core support structures.