UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

December 16, 2008

NRC REGULATORY ISSUE SUMMARY 2008-30
FATIGUE ANALYSIS OF NUCLEAR POWER PLANT COMPONENTS

ADDRESSEES

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to inform licensees of an analysis methodology used to demonstrate compliance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) fatigue acceptance criteria that could be nonconservative if not correctly applied.

BACKGROUND INFORMATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," requires that applicants for license renewal perform an evaluation of time-limited aging analyses relevant to structures, systems, and components within the scope of license renewal. The fatigue analysis of the reactor coolant pressure boundary components is an issue that involves time-limited assumptions. In addition, the staff has provided guidance in NUREG-1800, Rev. 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," issued September 2005. NUREG-1800, Rev. 1, specifies that the effects of the reactor water environment on fatigue life be evaluated for a sample of components to provide assurance that cracking because of fatigue will not occur during the period of extended operation. Since the reactor water environment has a significant impact on the fatigue life of components, many license renewal applicants have performed supplemental detailed analyses to demonstrate acceptable fatigue life for these components.

10 CFR 50.55a, "Codes and Standards," specifies the ASME Code requirements for operating reactors. Some operating facilities may have performed supplemental detailed analysis of components because of new loading conditions identified after the plant began operation.

SUMMARY OF ISSUE

The staff identified a concern regarding the methodology used by some license renewal applicants to demonstrate the ability of nuclear power plant components to withstand the cyclic loads associated with plant transient operations for the period of extended operation. This particular analysis methodology involves the use of the Green's (or influence) function to calculate the fatigue usage during plant transient operations such as startups and shutdowns.

The methodology involves performing a detailed stress analysis of a component to calculate its response to a step change in temperature. This detailed analysis is used to establish an influence function, which is subsequently used to calculate the stresses caused by the actual plant temperature transients. This methodology has been used to perform fatigue calculations and as input for on-line fatigue monitoring programs. The Green's (or influence) function methodology is not in question. The concern involves an input in which only one value of stress is used for the evaluation of the actual plant transients. The detailed stress analysis requires consideration of six stress components, as discussed in ASME Code, Section III, Subsection NB, Subarticle NB-3200. Simplification of the analysis to consider only one value of the stress may provide acceptable results for some applications; however, it also requires a great deal of judgment by the analyst to ensure that the simplification still provides a conservative result.

The staff has requested that recent license renewal applicants that have used this simplified methodology perform confirmatory analyses to demonstrate that the simplified analyses provide acceptable results. The confirmatory analyses retain all six stress components. To date, the confirmatory analysis of one component, a boiling-water reactor feedwater nozzle, indicated that the simplified input for the influence function did not produce conservative results in the nozzle bore area when compared to the detailed analysis. However, the confirmatory analysis still demonstrated that the nozzle had acceptable fatigue usage.

Licensees may have also used the simplified methodology in operating plant fatigue evaluations for the current license term. For plants with renewed licenses, the staff is considering additional regulatory actions if the simplified methodology was used.

BACKFIT DISCUSSION

This RIS informs addressees of a potential non-conservative calculation methodology and reminds them that the ASME Code fatigue analysis should be performed properly. For license renewal, metal fatigue is evaluated as a time-limited aging analysis per 10 CFR 54.21 (c). The associated staff review guidance is provided in Section 4.3, "Metal Fatigue Analysis," in NUREG-1800, Review 1, "Standard Rev. Plan for Review of License Renewal Applications for Nuclear Power Plants." For operating reactors, the ASME Code requirements are specified in 10 CFR 50.55a. This RIS does not impose a new or different regulatory staff position. It requires no action or written response and, therefore, is not a backfit under 10 CFR 50.109. Consequently, the NRC staff did not perform a backfit analysis.

FEDERAL REGISTER NOTIFICATION

A notice of opportunity for public comment on this RIS was published in the *Federal Register* FR 24094, on May 1, 2008. Comments were received from five industry commenters. The staff considered all comments. The staff's evaluation of the comments is publicly available through NRC's Agencywide Documents Access and Management System under Accession No. ML083450535.

CONGRESSIONAL REVIEW ACT

The NRC has determined that this RIS is not a rule as designated by the Congressional Review Act (5 U.S.C. §§801-808) and, therefore, is not subject to the Act.

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CONTACT

Please direct any questions about this matter to the technical contacts listed below.

/RA/

Timothy J. McGinty, Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Technical Contacts: Jerry Dozier, NRR
301-415-3925
E-mail: Jerry.Dozier@nrc.gov

John R. Fair, NRR
301-415-2759
E-mail: John.Fair@nrc.gov

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E-mail: Jerry.Dozier@nrc.gov

John R. Fair, NRR
301-415-2759
E-mail: John.Fair@nrc.gov

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ADAMS ACCESSION NO.: ML083450727

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