From: BATCH, STAN <SBATC90@entergy.com>
Sent: Friday, January 12, 2007 10:14 AM
To: Wittich, Walter <wwittic@entergy.com>; Azevedo, Nelson F <nazeved@entergy.com>
Cc: FINNIN Ron <ron.finnin@AREVA.com>; FRONABARGER, DON <DFRONAB@entergy.com>
Subject: need to evaluate high cycle fatigue for IPEC baffle bolts?

See below. We just need to understand what Wolf Creek / W are stating so we can determine if it applies to IP or not. It would be nice to review this enough to determine that it only applies to newer plants like Wolf Creek- but I do not know enough to state that yet.

If you guys do not know about this W technical bulletin TB-03-02, maybe you can ask W about this.

Just FYI as an item to include in out baffle bolt review.

Thanks!
-----Original Message-----
From: BATCH, STAN
Sent: Thursday, December 28, 2006 1:58 PM
To: Ron Finnin (ron.finnin@AREVA.com)
Cc: FRONABARGER, DON; IVY, TED S
Subject: need to evaluate high cycle fatigue for IPEC baffle bolts?

The Wolf Creek LRA states an evaluation must be completed for **high cycle fatigue** of the RVI. The LRA specifically identifies an analysis of high cycle fatigue must be completed and refers to a **W technical bulletin TB-03-2 that states this must be addressed for license renewal**. Specifically, the LRA states:

Fatigue Analyses of Barrel-to-Former and Baffle-to-Former Bolts

Cracked baffle-to-former bolts were found in a few offshore reactors with designs and materials similar to Westinghouse units, multiple failures have occurred in Alloy A-286 internals bolting in B&W reactors; and stress corrosion cracking has occurred in tack-welded Alloy X-750 bolts in offshore Siemens and Kraftwerk Union internals. The failures have been attributed to a combination of fatigue, neutron embrittlement, and irradiation-assisted stress corrosion cracking (IASCC). With extended operation the stresses induced by differential void swelling between the bolts and bolted members, and between the bolted members, may also become significant, although the stresses will probably be somewhat mitigated by irradiation and thermal creep relaxation. All of these effects are time-dependent and all except fatigue and thermal creep depend on neutron fluence.

Fatigue in these bolts is the subject of an ASME code analysis, which is a TLAA. No other evaluations of these other effects have been introduced into the licensing basis at WCGS, and these bolted connection designs are therefore supported by no TLAAs addressing effects other than fatigue. This finding agrees with the conclusion of Westinghouse topical report WCAP-14577-A.

The Westinghouse topical report observed that the barrel-to-former and baffle-to-former bolts are in the category of components "...where the cyclic loadings are sufficiently uncertain to preclude the effective use of detailed fatigue design analysis, ..." and therefore

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for which "...alternatives for managing the effects of the age-related degradation are described...." The 40-year predicted usage factor in at least some of at least the baffle-to-former bolts is a significant fraction of the limit of 1.0. That is, the high predicted usage factor, the additional aging effects requiring mitigation, and the fact that some of these are synergistic (e.g., fatigue and the other cracking mechanisms) dictate that management of the fatigue usage factor in these bolts will be insufficient by itself, and that an aging management program must be constructed for the bolts which either adequately address all of these effects, or which will ensure their safety function despite these effects.

WCGS reviewed Westinghouse Technical Bulletin TB-03-2, "Reactor Internals Baffle-to-Former Bolt/Core Design Interface," and concluded that fatigue failures would not be expected during the original 40-year licensed operating period, but must be addressed for license renewal.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii); and Aging Management, 10 CFR 54.21(c)(1)(iii)

Confirmation of Negligible Effects of High-Cycle Fatigue

Since fatigue usage factor does not depend strongly on flow-induced vibration or other high-cycle effects that are time-dependent at steady-state conditions, but depends more strongly on effects of operational, upset, and emergency transient events, the increase in operating life to 60 years should not have a significant effect on fatigue usage factor so long as the number of design basis transient cycles remains within the number assumed by the original analysis. WCGS will obtain a design report amendment to either quantify the increase in high-cycle fatigue effects, or to confirm that the increase will be negligible. WCGS will complete this action before the end of the current licensed operating period. The analysis of these fatigue effects in reactor internals will thereby be revised for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

Cycle Count and Usage Factor Tracking for the Balance of Reactor Internals with Fatigue Analyses

Table 4.3-1 demonstrates that the specified set of primary coolant design basis transient events should not be exceeded during the 60-year period of extended operation. Therefore, once the negligible effect of high-cycle fatigue on critical components is confirmed, or is satisfactorily quantified, transient cycle counting under the WCGS fatigue management program will ensure that the design basis fatigue usage factor limit (1.0) will not be exceeded in any analyzed location in the reactor internals without being identified, and without an appropriate evaluation and any necessary mitigating actions. Fatigue in the reactor vessel internals (other than the barrel-former and baffle-former bolts, below) will therefore be adequately managed for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

The WCGS Metal Fatigue of Reactor Coolant Pressure Boundary program is described in Appendix B3.1.

Do you know anything about this?

IPEC00015124