UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
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NRC INFORMATION NOTICE 2007-02: FAILURE OF CONTROL ROD DRIVE MECHANISM LEAD SCREW MALE COUPLING AT A BABCOCK AND WILCOX-DESIGNED FACILITY

ADDRESSEES

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

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees about the discovery of a failed control rod drive mechanism (CRDM) lead screw male coupling during a maintenance activity at Oconee Nuclear Station (ONS), Unit 3. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

In a letter dated November 23, 2004, Framatome ANP (the vendor) informed NRC of the failure of two CRDM lead screw male couplings in a pressurized-water reactor, designed by Babcock and Wilcox (B&W), at ONS. During an outage at ONS in April 2001, Duke Power (the licensee) discovered that two tangs in two separate CRDM male couplings in ONS, Unit 3 were fractured.

CRDMs are mounted on the reactor vessel head. The lead screw and the roller nuts on the CRDMs convert rotary motion into linear motion. The lead screw connects the CRDM to the control rod assembly (CRA). Specifically, the male coupling at the end of the lead screw inserts into the female coupling on top of the CRA. The four equally spaced tangs on the male coupling must align with the four slots in the female coupling. The tangs and slots must align for the CRA and lead screw to be coupled properly. In addition, a stop pin inside the female coupling acts as a rotational hard stop that serves to confirm that the CRA is properly coupled to the lead screw.

The lead screw male coupling is made of 17-4 precipitation-hardened (PH) martensitic stainless steel (17-4 PH steel). Failure analysis of the lead screw male coupling indicated the material had lost ductility due to a phenomenon called thermal embrittlement. The licensee determined

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that the root cause of the male coupling failures was the combination of thermal embrittlement of the 17-4 PH steel and the excessive force that was used during refueling outages when coupling and decoupling the lead screw from the CRA. The licensee noted a low load on the male couplings during normal operation and a significant load only when the control rods are initially withdrawn from the core prior to start-up operation. The licensee concluded that a simultaneous failure of two or more male couplings is unlikely.

Licensee corrective actions included revising the procedure used during the refueling outage for coupling and decoupling the lead screw to minimize the force on the couplings. Additionally, the licensee performed control rod drop analyses for ONS, Units 1, 2, and 3 and concluded that the analyses were acceptable and bounding.

The vendor advised all B&W licensees to perform similar control rod drop analyses and to revise their procedure for coupling and decoupling the CRDM lead screw. Some B&W licensees have replaced the CRDM male couplings and scheduled an inspection of these couplings during future refueling outages. Other B&W licensees have revised their coupling and decoupling procedures to minimize the force applied to the CRDM male couplings.

DISCUSSION

When exposed to a light-water reactor temperature of 550° F, the 17-4 PH steel that has previously been subjected to aging (heat treatment) at 1100° F can experience thermal embrittlement and an increase in hardness (i.e., a reduction in Charpy “V” notch fracture toughness values)[1] . The operating experience at ONS shows that thermally embrittled 17-4 PH steel is susceptible to failure when exposed to unexpected loading conditions. These loading conditions may not be part of the original design basis analyses and may be introduced either during maintenance or normal operations. Therefore, synergistic effects of unexpected loading conditions and thermal embrittlement can cause failure of 17-4 PH steel components in the reactor coolant system (RCS).

Thermal embrittlement of 17-4 PH steel cannot be identified by typical in-service inspection activities. However, by performing visual or other inspections, licensees can identify cracks which could lead to failure of the embrittled component prior to component failure. Licensees, thus, can prevent the deleterious effects of thermal embrittlement in the 17-4 PH steel components by identifying aging degradation (i.e., cracks), implementing early corrective actions, and monitoring and trending age-related degradation.

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[1] “Fracture of Type 17-4 PH CRDM Lead Screw Male Coupling Tangs,” by H. Xu and S. Fyfitch, the 11th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, ANS: Stevenson, WA (2003).

CONTACTS

This information notice does not require any specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate project manager in the NRC's Office of Nuclear Reactor Regulation (NRR).

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