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NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
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NRC INFORMATION NOTICE 2007-05: VERTICAL DEEP DRAFT PUMP SHAFT AND COUPLING FAILURES

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 Nuclear Regulatory Commission (NRC) is issuing this Information Notice (IN) to alert licensees to vertical deep draft pump shaft and coupling failures from intergranular stress corrosion cracking (IGSCC). 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 IN are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

Service Water Pump Shaft Failures at Columbia Generating Station

At Columbia Generating Station on June 14, 2005, following a start of the Service Water Pump 1A (SW-P-1A), control room operators noted that service water (SW) flow to the Division 1 residual heat removal heat exchanger was out of specification. Operators also noted the pump discharge head and pump motor current were lower than normal and declared the SW-P-1A inoperable. A subsequent surveillance test on SW-P-1A determined that pump performance had degraded and was operating at the intersection of the alert and action ranges of its performance curve. Energy Northwest, the licensee, proceeded to replace SW-P-1A with an available spare pump.

During disassembly of SW-P-1A, Energy Northwest determined that IGSCC failed the pump shaft end flanges on two of the shaft sections allowing the shaft sections to drop. This condition caused the pump impeller to contact the pump suction casing, which resulted in substantial wear of the pump impellers and degraded pump performance. The shaft drive keys remained captured between the shaft keyways and coupling sleeves such that the shaft segments and impeller continued to rotate. Energy Northwest conducted a metallurgical examination of the damaged pump shaft and also identified axial cracking on the impeller pump shaft segment and two diagonal cracks on the top column shaft.

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The metallurgical examination determined that the shaft material, TP410 martensitic stainless steel, was susceptible to tempering embrittlement (shaft material was tempered at 970 degrees Fahrenheit, which was conducive to tempering embrittlement). Tempering embrittlement reduced the corrosion resistance of the shaft material, thereby, increasing the material's susceptibility to IGSCC. Energy Northwest determined that SW-P-1B was also susceptible to the same failure mechanism as identified in SW-P-1A. However, it had not exhibited performance degradation based on past surveillance test results.

Interim corrective actions included additional monitoring of SW-P-1B to verify pump performance until a replacement pump could be procured and installed. A subsequent inspection of the as-found condition of SW-P-1B determined that the pump shaft had degraded in a manner similar to SW-P-1A, due to IGSCC, and that the pump impeller had degraded due to wearing on the suction casing.

A detailed evaluation of SW-P-1A historical computer data determined that although pump performance had met surveillance test acceptance criteria, pump performance had slowly degraded from as early as August 2000 and as late as December 2001. Similarly, a detailed evaluation of SW-P-1B historical computer data revealed that SW-P-1B had slowly degraded since August 2003.

Safety Function of SW-P-1A and SW-P-1B and Design Information

The standby SW system and ultimate heat sink function is to supply cooling water to remove heat from all nuclear plant equipment that are essential for safe and orderly shutdown of the reactor, to maintain it in a safe condition, and to remove decay heat from the reactor during shutdown conditions. During all normal operating conditions, including normal shutdown as well as emergency conditions, waste heat from the reactor auxiliary systems is transferred to the ultimate heat sink via the standby SW system.

SW-P-1A and SW-P-1B are deep draft vertical pumps manufactured by Byron Jackson, Model 28KXH3, and were originally designed to provide a minimum of 10,500 gpm rated flow at 500 ft of discharge head. Both pumps were installed in 1979 and, prior to the failure of SW-P-1A, had not been replaced, refurbished, removed, or opened for inspection since initial installation. Both pumps are exposed to the same environmental and physical conditions.

The standby SW pump design consists of five sections of shaft with four sets of shaft coupling components. Each set of shaft coupling components consists of two drive keys and a pair of split rings that are held by a shaft coupling (sleeve) which is located by two gib keys. At the point where two shaft sections join, the split rings are installed over mating shaft shoulder flanges. The shaft shoulder flanges were the failed components which allowed the pump shaft to drop, causing the pump impeller to rest on the casing bowl, thereby, resulting in milling and wear of the impeller into the bowl during operation.

Additional Service Water Pump Shaft and Shaft Coupling Failures

NRC review of Operating Experience records identified at least 23 essential SW pump shaft and coupling failures since 1983 involving more than six different pump manufacturers. Many of these failures involved IGSCC as a primary cause. Other causes of shaft and coupling failures included: misalignment, imbalance, installation errors, and deferred maintenance. Two incidents since 2001, involving IGSCC are:

(1) Perry experienced SW pump shaft coupling failures due to IGSCC in September 2003 and May 2004. These failures are described in NRC Inspection Reports 05000440/200401, dated July 2, 2004, and 050000440/2004008, dated August 4, 2004, (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML041900080 and ML042250254).

(2) VC Summer experienced SW pump shaft coupling failure during testing due to IGSCC in May 2001. This failure is described in NRC Inspection Report No. 50-395/02-06, dated April 1, 2002 (ADAMS Accession No. ML020920543).

BACKGROUND

Applicable Regulatory Documents

General Design Criterion (GDC) 1 (defined in Appendix A to Title10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities,") and Criterion XI (defined in Appendix B to 10 CFR Part 50) requires that all components (such as pumps and valves) that are necessary for safe operation be tested to demonstrate that they will perform satisfactorily in service. GDC 1 requires that components important to safety be tested to quality standards that are commensurate with the importance of the safety function(s) to be performed. Appendix B to 10 CFR Part 50 describes the requisite quality assurance program, which includes testing for safety-related components.

10 CFR 50.55a defines the requirements for applying industry codes and standards to boiling or pressurized water-cooled nuclear power facilities. This section requires that certain American Society of Mechanical Engineers Boiler and Vessel Pressure (ASME Code) Class 1, 2, and 3 pumps and valves be designed to enable inservice test (IST) and that testing be performed to assess operational readiness in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code). IST is intended to detect degradation affecting operation and assess whether adequate margins are maintained. The OM Code requires the licensee to show that the overall pump performance has not degraded from that required to meet its intended function. Establishing limits that are more conservative than the OM Code limits may be necessary to ensure that design limits are met. NRC IN 97-90 describes situations where ASME acceptance ranges were greater than those assumed in the accident analysis. OM Code acceptance criteria do not supersede the requirements delineated in a licensee's design or license basis.

Components within the scope of 10 CFR 50.55a are included in the scope of 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (the "Maintenance Rule"). The Maintenance Rule requires that licensees monitor the performance or condition of structures, systems, or components (SSCs) against licensee-established goals in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended function. Such goals are to be established, where practicable, commensurate with safety, and they are to take into account industry-wide operating experience. When the performance or condition of a component does not meet established goals, appropriate corrective actions are to be taken.

Operability limits of pumps must always meet, or be consistent with, licensing-basis assumptions in a plant's safety analysis. NRC Generic Letter 91-18 (replaced by Regulatory Issue Summary 2005-20) provides additional guidance on operability of components. NUREG-1482, Revision 1, “Guidelines for Inservice Testing at Nuclear Power

Plants" provides licensees guidelines and recommendations for developing and implementing programs for the IST of pumps and valves at commercial nuclear power plants.

Applicable NRC Information Notices

NRC IN 93-68, "Failure of Pump Shaft Coupling Caused by Temper Embrittlement During Manufacture," dated September 1, 1993, in part, described that type 410 stainless steel used in the manufacture of Byron Jackson pump shaft couplings may have low-impact strength due to inadequate heat treatment during manufacture, rendering the component susceptible to tempering embrittlement. Pump shafts containing temper-embrittled couplings could fail during operation if the pump has worn bearings, if the shaft is misaligned, or if shaft motion is impeded by silt or debris ingestion.

NRC IN 94-45, "Potential Common-Mode Failure Mechanism for Large Vertical Pumps," dated June 17, 1994, described a problem where differing coupling materials could experience galvanic corrosion resulting in a failure of the shaft coupling and subsequent failure of long shaft vertical pumps. NRC IN 94-45 also generally addressed a concern that current testing methodologies of vertical line shaft pump hydraulic and mechanical performance may not identify interference, before damage occurs, between the pump impellers and bowls caused by a change in shaft length.

NRC IN 97-90, "Use of Nonconservative Acceptance Criteria in Safety-Related Pump Surveillance Tests," dated December 30, 1997, describes examples that identify inadequacies in surveillance test procedure acceptance criteria that had the potential for, and in some cases did result in, pumps not meeting their accident analysis acceptance criteria.

Applicable NRC Inspection Procedures

NRC inspection guidance for SW pumps at operating nuclear plants is provided in: (1) Attachment 22, "Surveillance Testing," to NRC Inspection Manual IP 71111, "Reactor Safety: Initiating Events, Mitigating Systems, Barrier Integrity," available on the NRC's public Web site at: http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html; and in (2) Inspection Procedure 73756, "IST of Pumps and Valves," July 27, 1995, available on the NRC's public Web site at:
http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/ip73756.pdf

DISCUSSION

The SW pump shaft failure events at Columbia Generating Station and related industry operating experience demonstrate that IST alone might not be sufficient to ensure that pumps meet their accident analysis acceptance criteria. These failures might not be detected by commonly employed condition monitoring, during routine operations, or from surveillance test or IST results. Pump shaft and coupling failures can challenge operability even though performance degradation over time may appear consistent with normal wear. Operating experience also shows that pump shaft failures and coupling failures can result in sudden total loss of flow before standard performance monitoring techniques alert plant staff to the impending failure. Inspection and refurbishment of deep draft pumps on a periodic basis may reveal some of these failure mechanisms. Vibration analysis using more sophisticated tools (i.e., transducer on pump bowl, phase angle analysis) may be capable of identifying similar future failures. Enhanced condition-monitoring techniques or time-based inspections, may enhance early detection of degradation before failures in large vertical deep draft pumps occur.

CONTACTS

This IN requires no specific action or written response. Please direct any questions about this matter to the technical contacts.

/TQuay for MCase/

Michael J. Case, Director
Division of Policy & Rulemaking
Office of Nuclear Reactor Regulation

Technical Contacts: John McHale, NRR/DCI,
301-415-1261,
JJM1@nrc.gov

Tim Mitts, NRR/DIRS,
301-415-4067,
TMM5@nrc.gov

Note: NRC generic communications may be found on the NRC public web site, http://www.nrc.gov under Electronic Reading Room/Document Collections.