UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
OFFICE OF NEW REACTORS
WASHINGTON, DC 20555-0001

April 24, 2015

NRC INFORMATION NOTICE 2015-04: FATIGUE IN BRANCH CONNECTION WELDS

ADDRESSEES

All holders of an operating license or construction permit for a nuclear power reactor under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

All holders of and applicants for a power reactor combined license, standard design certification, standard design approval, or manufacturing license under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." All applicants for a standard design certification, including such applicants after initial issuance of a design certification rule.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees about recent operating experience related to the structural integrity of recirculation system piping in boiling-water reactors (BWR) and to raise industry awareness regarding the possibility of emerging fatigue cracking in branch connections in all light-water reactors. The NRC expects that recipients will review the information contained in this IN for applicability to their facilities and consider actions, as appropriate, to avoid similar issues at their facilities. However, suggestions contained in this IN are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

The licensee event reports (LERs) detailed below represent three examples of fatigue failures in recirculation system piping, specifically in full-penetration groove welds in branch connections. Similar incidences occurred in recirculation system instrument lines in BWR units due to vibrations and were addressed in IN 95-16, "Vibration Caused by Increased Recirculation Flow in a Boiling Water Reactor," dated March 9, 1995[1].

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[1] Available in the Agencywide Documents Access and Management System (ADAMS) at Accession No. ML031060310.

ML15023A054

Hope Creek Nuclear Generating Station (2001)

On October 10, 2001, while performing a primary containment-walk down at the beginning of a refueling outage at Hope Creek Nuclear Generating Station, the licensee (PSEG Nuclear, LLC) observed a leak on the elbow tap of the “A” recirculation pump suction piping. The leak was producing a 3-4 foot long spray with the reactor vessel pressure at approximately 300-400 psig. Further investigation revealed that the leak was coming from the weld area of a 1-inch pipe-to-branch connection weld and was, therefore, a through-wall breach of the reactor coolant system pressure boundary. The licensee identified that a vibration-induced weld failure caused the leak. According to the LER, the vibration-induced failure was most likely caused by the second natural frequency of the piping becoming resonant with the 5X vane-passing frequency of the "A" recirculation pump. Additionally, the licensee found that accelerometers, which had been installed in 1991, added weight to the piping and altered its original frequency.

As a result of this event, the licensee: (1) removed the accelerometers, (2) performed walk-downs of all recirculation lines for indication of other failures and inspected equipment around the area of the leak to ensure no damage from leak impingement, (3) performed radiographic examinations on other lines and penetrant tests on all other susceptible welds (to determine extent of condition), and (4) removed the cracked weld and affected section of pipe.

Additional information is available in LER-50-354/2001-006-00, dated December 7, 2001 and can be found on the NRC’s public website in ADAMS under Accession No. ML020220237.

Hope Creek Nuclear Generating Station (2005)

On March 27, 2005, PSEG Nuclear, LLC identified a steam leak at the Hope Creek Nuclear Generating Station. The steam leak was coming from a 4-inch crack in the insulated decontamination port, which is connected to the “B” reactor recirculation loop between the suction of the pump and the suction isolation valve. The decontamination port is 4-inch SA-376 Type 304 stainless steel piping. The fatigue crack started near the outside diameter of the pipe-to-branch connection weld.

The through-wall crack in the decontamination port was caused by fatigue initiation and propagation. The licensee determined that an original subsurface weld defect propagated due to high cycle vibrations. Analysis revealed that, based on the geometry of the decontamination port, the natural frequencies of the “B” decontamination port coincided with the 5X vane-passing frequency of the reactor recirculation pump at normal operating speeds. The natural frequency of the decontamination port ranged from 122-135 Hertz (Hz), and the 5X vane-passing frequency of the reactor recirculation pump at normal operating speeds was 120-125 Hz.

Because of this event, the licensee modified the decontamination ports for both recirculation loops. The length of 4-inch decontamination port was reduced to increase its natural frequency in order to not coincide with the normal recirculation pump 5X vane-passing frequency. Additionally, the licensee’s review consisted of: finite element analysis, vibration analysis, modal analysis, isometric review, and review of previous in-service inspections (ISIs). Based on these reviews and as part of its extent of condition determination, the licensee inspected 23 welds using nondestructive examination techniques, and the inspections were satisfactory.

Additional information is available in LER-50-354/2005-002-00, dated May 26, 2005, and can be found on the NRC’s public website in ADAMS under Accession No. ML051540027.

Susquehanna Steam Electric Station, Unit 1 (2012)

On June 19, 2012, the licensee (PPL Susquehanna, LLC) for the Susquehanna Steam Electric Station Unit 1 identified the source of an increasing trend in drywell leakage to be an approximately 3-inch long crack. The crack was in the weld of the branch connection to the 4-inch diameter chemical decontamination pipe to the “1A" reactor recirculation pump suction line.

Metallurgical examinations revealed fatigue caused the crack. The root cause was that the calculated stresses were underestimated by use of an incorrect stress intensification factor (1.1 vs. 1.8) in the vibration analysis. The calculated vibration stress for the 4-inch decontamination line connection, considering the maximum extended power uprate vibration and stress data, exceeded the endurance limit by approximately 26 percent. Considering the primary frequency of vibration was 128.5 Hz, the fatigue life was approximately 4.9 years when exposed to the 5X recirculation pump vane-passing frequency, although the original vibration analysis yielded an infinite life. The normal operating speed range for the reactor recirculation pumps correlated to a 5X vane-passing frequency of 122-135 Hz. The configuration of the reactor recirculation pump suction line decontamination flange was such that its natural frequency was 135.8 Hz based on ambient temperature conditions. However, at operating temperature the natural frequency was approximately 129 Hz. Because the 5X vane-passing frequency of the reactor recirculation pumps was similar to the natural frequency of the decontamination flange, the vibrational accelerations were magnified. This led to relatively large bending moments and stresses at the branch connection (exceeding the endurance limit of the material) and resulted in fatigue failure. The crack was not caused by intergranular stress corrosion cracking (IGSCC); however, indications of IGSCC were found unrelated to the fatigue crack.

Because of this event, the licensee changed the natural frequency of the assembly by redesigning the 4-inch diameter chemical decontamination flange connection on both the "1A" and "1B" reactor recirculation pump suction lines. The length of the 4-inch diameter pipe was reduced from 6-inch to approximately 3.5-inch to produce a new configuration that was not susceptible to the cyclic fatigue caused by reactor recirculation pump 5X vane-passing frequency similarity to the natural frequency of the assembly. This modification also removed the sections with IGSCC, which were identified during metallurgical examinations. The modified decontamination flange connections were pressure tested (1035 psig) to ensure pressure boundary integrity. In addition, the licensee performed extent of condition inspections of similar reactor recirculation and reactor water cleanup system piping. Vibration related issues, additional fatigue flaws, or IGSCC were not identified. At the time of the event, the licensee determined that the Unit 2 recirculation system piping was capable of performing its required design function. The licensee planned for similar actions for Unit 2 during the next refueling outage in order to prevent recurrence.

Additional information is available in LER-50-387/2012-007-01, dated November 20, 2012, and can be found on the NRC’s public website in ADAMS under Accession No. ML123250703.

BACKGROUND

Related NRC Generic Communications

NRC IN 2005-08, “Monitoring Vibration to Detect Circumferential Cracking of Reactor Coolant Pump and Reactor Recirculation Pump Shafts,” dated April 5, 2005. The NRC issued this IN to alert addressees to the importance of timely detection of circumferential cracking of reactor coolant pump and reactor recirculation pump shafts to minimize the likelihood of consequential shaft failures.

NRC IN 2005-23, “Vibration-Induced Degradation of Butterfly Valves,” dated August 1, 2005. The NRC issued this IN to alert addressees to the vibration-induced degradation (loss of taper pins) of butterfly valves.

NRC IN 2006-15, “Vibration-Induced Degradation and Failure of Safety-Related Valves,” dated July 27, 2006. The NRC issued this IN to alert addressees of vibration-induced degradation and failure of valves.

NRC IN 2007-21, “Pipe Wear Due to Interaction of Flow-Induced Vibration and Reflective Metal Insulation,” dated June 11, 2007. The NRC issued this IN to alert addressees that a licensee identified significant wear marks on the outside wall of the chemical volume control system piping, which was subject to flow-induced vibration conditions.

DISCUSSION

The above operating experiences discuss that unexpected fatigue failures could occur in branch connection welds during normal operating conditions. These failures were caused by the affected piping becoming resonant with the reactor recirculation pump 5X vane-passing frequency. In addition, since fatigue failure is progressive, the affected plants’ ISI programs failed to identify the problems before actual failure took place.

Similar weld fatigue failures could be minimized by designing piping with natural frequencies that avoid pump vane-passing frequencies and by selecting welds that are vulnerable to fatigue failure for examination under the ISI program. In addition to NRC requirements, some licensees have chosen to implement programs for monitoring fatigue in branch connection welds.

CONTACTS

This information notice does not require any specific action or written response. Please direct any questions about this matter to the technical contacts listed below.

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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under NRC Library.

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