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

December 10, 2012

NRC INFORMATION NOTICE 2012-21: REACTOR VESSEL CLOSURE HEAD STUDS REMAIN DETENSIONED DURING PLANT STARTUP

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 or applicants for an early site permit, standard design certification, standard design approval, manufacturing license, or combined license under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees of an event involving detensioned reactor vessel closure head studs at a boiling-water reactor that resulted in leakage from the reactor vessel during startup operations and a manual scram. The NRC expects that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. Suggestions contained in this IN are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

<u>Brunswick Steam Electric Plant, Unit 2</u>

On November 16, 2011, the Brunswick Steam Electric Plant, Unit 2 (Brunswick 2, a General Electric boiling-water reactor) was in power ascension following a mid-cycle maintenance outage, which required reactor vessel disassembly. Following the outage, the reactor vessel was reassembled and operators commenced startup operations. With the reactor in Startup Mode (Mode 2) and at normal operating pressure, operators noted increasing drywell floor drain leakage. At 3:01 a.m. eastern standard time (EST), an Unusual Event was declared as a result of unidentified drywell leakage exceeding 10 gallons per minute (gpm). At 3:09 a.m. EST, a manual reactor scram was initiated from approximately 7 percent of rated thermal power due to the continued increase in unidentified drywell leakage. Following the scram, the reactor was depressurized and the unidentified leak rate decreased to less than 10 gpm within 1 hour. At 1:45 p.m. EST on November 17, 2011, with the reactor in Cold Shutdown (Mode 4), leak

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investigation activities determined that the reactor vessel head studs were not fully tensioned during startup operations; therefore, an unanalyzed condition existed at Brunswick 2. Subsequently, it was determined that none of the 64 reactor vessel head studs were adequately tensioned.

Reactor vessel head stud tensioning is accomplished by attaching a tensioning device to the stud's uppermost threads. Hydraulic pressure is applied to the tensioning device, which stretches the stud. With the stud elongated by the tensioning device, personnel rotate the stud nut until it makes firm contact with the washer on the head flange. When the hydraulic pressure is released, the nut maintains the tension and elongation in the stud, applying closure pressure to the flanges of the reactor vessel and head.

The licensee's investigation determined that this event was the result of errors made while operating the reactor vessel head stud tensioning equipment and during the validation process to ensure the head was properly tensioned. Following the event, the licensee assessed the stud tensioning process through equipment troubleshooting, review of the reactor vessel reassembly procedure (Procedure 0SMP-RPV502), and interviews with refuel floor personnel. The equipment was found to be fully functional. However, the licensee determined that personnel operating the stud tensioning equipment misinterpreted the digital display of the hydraulic pressure being applied to elongate the studs. Specifically, the licensee found that personnel incorrectly believed that the actual hydraulic pressure being applied to the tensioning device was a factor of ten greater than the pressure indicated on the device. As a result, none of the 64 studs were properly tensioned during the reactor vessel assembly process.

The Stud Elongation Measurement System (SEMS III) is used at Brunswick 2 to validate proper stud elongation. Based on interviews with personnel, the licensee determined that the refuel floor crew incorrectly concluded that the target stud elongation value of 0.045 inches was achieved when the elongation values indicated on the SEMS III device were only between ±0.004 inches. The licensee attributed this error to the crew incorrectly assuming that the elongation value of 0.045 inches was automatically deducted from the post-tensioned elongation indication on the SEMS III device. Furthermore, the elongation values of ±0.004 inches, as indicated on the SEMS III device, correspond to the stud elongation tolerance specified in Procedure 0SMP-RPV502. Accordingly, the crew compared the low reading on the SEMS III device to the stud elongation tolerance in the procedure and erroneously determined that acceptable stud elongation had been achieved. The quality control inspector concurred with the consensus opinion of the crew. As a result of these errors, the reactor vessel head studs were tensioned to only approximately 10 percent of the required amount. Therefore, Brunswick 2 reached Mode 2 with the head not properly tensioned. The increase in leakage and subsequent reactor scram were a direct result of this condition.

The licensee performed a post-event evaluation of the integrity of the reactor vessel closure components. The licensee concluded that no reactor coolant pressure boundary components were damaged or overstressed as result of the event. After completing the integrity evaluation, the reactor vessel was reassembled. Prior to plant restart, a hydrostatic test was completed to verify that proper head stud tensioning had been achieved.

The licensee attributed the root cause of this event to the failure to provide proper training and lack of procedure guidance to correctly interpret critical data used to validate that the reactor vessel head studs are properly tensioned. Specifically, the licensee concluded that the operator errors that occurred during the reactor vessel reassembly evolution were due to an inadequate understanding of the digital readings displayed on the hydraulic stud tensioning equipment and the SEMS III stud elongation measurement device. For both cases, the licensee determined that the crews relied on erroneous assumptions that led to incorrect conclusions.

Licensee Corrective Actions

The licensee revised the reactor vessel reassembly procedure (Procedure 0SMP-RPV502) to include detailed guidance on the proper use of the SEMS III stud elongation measurement equipment and the interpretation of hydraulic pressure indications on the stud tensioning device. The licensee also provided training to refuel floor crew personnel on the proper operation of the SEMS III and hydraulic stud tensioning equipment during reactor vessel reassembly. The licensee has revised its refuel floor training and qualification documents to include specific discussion on the correct operation of the SEMS III equipment and how to properly interpret hydraulic pressure indications on the stud tensioning device. The licensee has also revised Procedure 0SMP-RPV502 to require the necessary level of training regarding these activities, as provided in the revised training documents. In addition, the licensee has modified corporate-wide qualification programs for nuclear fleet refuel floor crew personnel to ensure that all refuel floor personnel at Brunswick maintain the necessary qualifications for performing their assigned activities and receive the necessary level of training on the SEMS III and stud tensioning equipment, as provided in the revised training documents.

The licensee noted that, prior to the event, a decision was made that a post maintenance reactor vessel pressure test was not necessary because there are no regulatory requirements to conduct this test following mid-cycle maintenance outages. Therefore, as a corrective action, the licensee revised plant procedures to require a pressure test of the reactor vessel following mid-cycle maintenance outages that require reactor vessel reassembly.

NRC Special Inspection Team Findings

An NRC special inspection team reviewed the circumstances surrounding this event. The inspection team reviewed the licensee’s actions prior to the event and identified examples of improper procedure adherence that contributed to the inadequate reactor vessel head stud tensioning. Specifically, the team determined that licensee personnel failed to properly pressurize the reactor vessel head stud tensioning equipment to the value specified in Procedure 0SMP-RPV502 because the tensioning equipment operators did not know how to correctly interpret the hydraulic pressure reading on the tensioning equipment display. The inspection team also determined that quality control personnel failed to verify proper reactor vessel stud elongation in accordance with stud elongation values specified in Procedure 0SMP-RPV502. Further, the inspection team determined that nine of the twelve refuel floor personnel performing reactor vessel reassembly did not have the necessary refuel floor support training, as required by Procedure TRN-NGCC-1000, “Conduct of Training.” Finally, based on its review of Procedure 0PLP-20, “Post Maintenance Testing Program,” which specifies “plant equipment shall be tested consistent with their safety functions following maintenance activities that may have impaired proper functioning of the components," the inspection team determined that the licensee failed to specify an adequate post maintenance test to verify the pressure retaining capability of the reactor vessel following a mid-cycle maintenance outage.

The Brunswick Steam Electric Plant, Unit 2, Licensee Event Report (LER) 50-324/2-2011-002, dated January 16, 2012, contains further discussion of this event. The LER is available on the NRC's public Web site under Agencywide Documents Access and Management System (ADAMS) Accession No. ML12031A167. Additional information is available in NRC Special Inspection Report 05000324/2011013, dated January 25, 2012, under ADAMS Accession No. ML120250556; and NRC Inspection Report 05000324/2012007, dated April 20, 2012, under ADAMS Accession No. ML12114A036.

DISCUSSION

Section 50.120, "Training and qualification of nuclear power plant personnel," of 10 CFR states, in part, that the training program must incorporate the instructional requirements necessary to provide qualified personnel to operate and maintain the facility in a safe manner in all modes of operation. Criterion V, "Instructions, Procedures, and Drawings," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 states, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

The root cause of this event was the failure to provide the necessary training and procedure guidance to correctly interpret critical indications on the stud tensioning and stud elongation measurement equipment for verifying that proper stud tensioning had been achieved. The failure to adequately tension the reactor vessel closure head studs during reactor vessel reassembly undermined the integrity of the reactor coolant pressure boundary, one of the primary barriers to fission product release, during startup operations.

In addition, a decision was made that a post maintenance reactor vessel pressure test was not necessary because there are no regulatory requirements to conduct this test following mid-cycle maintenance outages. However, the reactor vessel head was removed and reinstalled during this outage in the same fashion as during a refueling outage. Therefore, this event highlights the importance of conducting mid-cycle maintenance outage activities, particularly those that require reactor vessel disassembly and reassembly, with the same level of rigor as scheduled refueling outage activities.

This event also highlights the importance of human performance and oversight of maintenance activities. For example, operators of the stud tensioning equipment were not familiar with the pressure display, yet they proceeded with tensioning based on an incorrect interpretation of indicated tensioner pressure. In addition, a licensee lead mechanic and a quality control inspector signed a procedure checklist for stud elongation measurements using flawed data, based on incorrect explanations by other members of the maintenance crew. Other findings related to human performance can be found in the April 20, 2012, inspection report.

CONTACT

This IN requires no specific action or written response. Please direct any questions about this matter to the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) or Office of New Reactors project manager.

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