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

July 24, 2012

NRC INFORMATION NOTICE 2012-14: MOTOR-OPERATED VALVE INOPERABLE DUE TO STEM-DISC SEPARATION

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 early site permit, combined license, standard design certification, standard design approval, or manufacturing 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 recent operating experience involving a motor-operated valve (MOV) that failed at the connection between the valve stem and disc. 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

On October 23, 2010, at Browns Ferry Nuclear Plant Unit 1, reactor operators attempted to place residual heat removal (RHR) loop II in service to support refueling activities. At this time, the low-pressure coolant injection (LPCI) outboard injection valve in RHR loop II failed to open. Control room lights indicated that this motor-operated LPCI valve was open but there was no flow in the RHR loop II with the 1B RHR pump in service. Control room operators secured the 1B RHR pump and placed RHR loop I in service to provide shutdown cooling flow.

The LPCI valve that failed to open is a 24-inch Walworth angle globe valve. This model valve incorporates a three-part, stem-to-disc assembly design (Figure 1) which includes the valve stem, an upper disc skirt that slides over the stem, and a lower threaded disc that accepts the valve stem and is secured to the lower part of the upper disc skirt through a matching threaded area. Once threaded together, the upper disc skirt assembly is tack welded to the lower disc to prevent unthreading.

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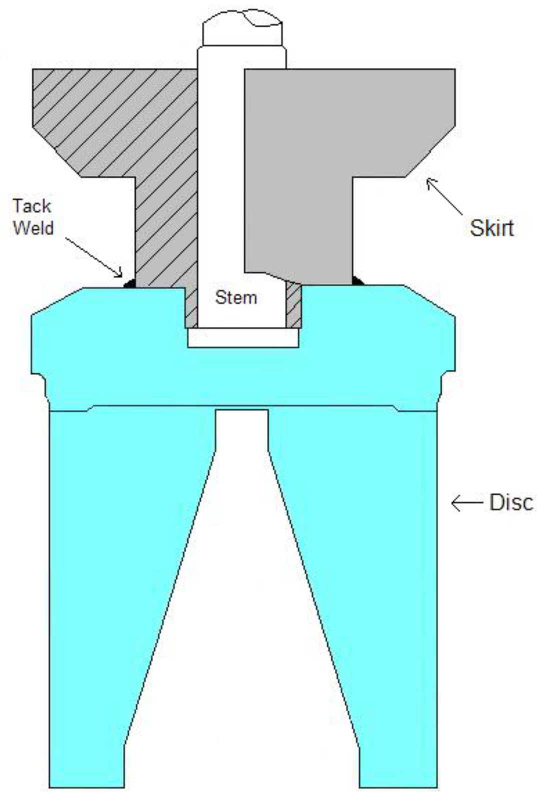

Figure 1
Disc/Skirt/Stem Assembly Cutaway View

Upon disassembly, the licensee discovered that the tack welds between the disc and skirt had failed and the lower disc of the LPCI valve had separated from the upper disc skirt and lodged in its seating area. The licensee determined that the threads on the upper disc skirt that interfaced with the lower disc threads were undersized. This contributed to the failure of the stem-to-disc connection during valve operations and the disc then becoming lodged in the seat. The licensee had last placed RHR loop II into service on March 12, 2009, with flow provided to the reactor vessel. The licensee indicated that the stem-to-disc connection for the LPCI valve failed in 2008 although the valve continued to function until it jammed in 2010. NRC inspectors reviewed this incident and issued a finding of high safety significance (red finding) because the RHR subsystem was inoperable for greater than the outage time allowed by the technical specification.

NRC inspectors also noted that the licensee failed to include the LPCI valve in the programs detailed in Generic Letter (GL) 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” and GL 96-05, “Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves.” The licensee explained that it made the omission because it considered the LPCI valve passive with no safety-related function to reposition. The NRC inspectors determined that the LPCI valve has an active safety-related function to close and thus should have been included in the scope of GL 89-10 program.

The failed LPCI valve is included in the scope of the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code) Inservice

Testing (IST) program. ASME designed OM Code IST activities to assess the operational readiness of components that are required to perform a specific function in (1) shutting down a reactor to the safe shutdown condition, (2) maintaining the safe shutdown condition, or (3) mitigating the consequences of an accident. The IST program at Browns Ferry, Unit 1 applies the 1995 Edition through the 1996 Addenda of the ASME OM Code. The IST activities implemented at Browns Ferry, Unit 1 did not reveal that the stem-to-disc connection failed in the LPCI valve or that the valve disc was lodged in the seat.

On June 8, 2011, the licensee for Browns Ferry, the Tennessee Valley Authority (TVA), appealed the NRC final significance determination for a red finding and notice of violation for Browns Ferry, Unit 1. In response, the NRC appointed an independent review panel to provide additional assurance that appropriate regulatory actions were being taken. The independent review panel did not alter the final finding results (see the review panel's letter dated August 16, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112280215)).

On September 23, 2011, the NRC inspectors completed Part 1 of a supplemental inspection following the guidance of Inspection Procedure 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Yellow Inputs or One Red Input," at Browns Ferry, Unit 1. This inspection focused on maintenance and testing programs related to the IST program, the MOV testing program, and the corrective action program (CAP). The NRC review of the IST program noted that the licensee had implemented augmented testing in the form of motor current signature analysis for the failed LPCI valve. The motor current signature data was collected at the motor control center (MCC) since 2006 at a 2-year refueling cycle interval. The NRC staff reviewed the augmented MCC testing and found that the licensee's procedures did not include appropriate quantitative or qualitative acceptance criteria for the captured motor current signature data as required by Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50. Post analysis of the failed LPCI valve data showed no unseating force in 2006 or 2008 but did show noticeable unseating force in 2010 after the valve had been rebuilt. The lack of unseating force in 2006 and 2008 data collection may have been the first indicator of a possible component problem.

The supplemental Inspection Procedure 95003 inspection also reviewed the MOV test program. The NRC review noted that Browns Ferry is a participant in the Joint Owners Group (JOG) and is committed to the JOG periodic verification program for addressing GL 96-05. Browns Ferry is implementing the final JOG program recommendation, which is required to be completed by September 25, 2012, per the September 25, 2006, NRC Safety Evaluation (ADAMS Accession No. ML061280315). The licensee estimates this implementation, which includes many valve modifications, will be completed by the September 2012 deadline. The NRC inspectors identified that several of the completed valve modifications nullified the original valve design basis capability established in GL 89-10 and that Browns Ferry did not apply appropriate methods for reestablishing the valve design basis for the modified valves and thus did not meet the requirements of 10 CFR 50.55a(b)(3)(ii).

Additional information concerning this issue appears in Browns Ferry Nuclear Plant–NRC Integrated Inspection Report 05000259/2010005, 05000260/2010005, 05000296/2010005, and Notice of Violation, dated February 9, 2011 (ADAMS Accession No. ML110400431), Browns Ferry Nuclear Plant–NRC Inspection Report 05000259/2011008, dated May 9, 2011 (ADAMS Accession No. ML111290482), and Browns Ferry Nuclear Plant–NRC Inspection Procedure 95003 Supplemental Inspection Report 05000259/2011011, 05000260/2011011, 05000296/2011011 (Part 1), dated November 17, 2011 (ADAMS Accession No. ML113210602).

BACKGROUND

The ASME OM Code (1995 Edition through 2006 Addenda) is incorporated by reference in 10 CFR 50.55a, "Codes and Standards," for implementation of the IST program for pumps, valves, and dynamic restraints used in nuclear power plants. The guidance of 10 CFR 50.55a(b)(3)(ii) supplements the testing requirements for MOVs in the ASME OM Code by requiring that licensees implementing the ASME OM Code as part of its IST program shall also establish a program to ensure that MOVs continue to be capable of performing their design-basis safety functions.

Criterion V of Appendix B to 10 CFR Part 50 requires that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that the important activities have been satisfactorily accomplished.

GL 89-10 requested that each nuclear power plant establish a program to demonstrate that safety-related MOVs are capable of performing their design-basis functions. The term "safety-related" refers to those systems and components that the nuclear power plant relies upon to remain functional during and following design-basis events to ensure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines of 10 CFR Part 100, "Reactor Site Criteria."

During the implementation of GL 89-10, NRC staff accepted four methods the licensee could use to demonstrate the design-basis capability of safety-related MOVs. In descending order of acceptability, the four methods for demonstrating capability are:

1. Dynamic testing at or near design-basis conditions with diagnostics of each MOV where practicable. Valves dynamically tested at less than design-basis conditions may be extrapolated with proper justification. Although the valve factor derived from the test data might be low because of minimal valve operating history or recent maintenance that exposed the stellite valve material to air, the dynamic testing provided assurance that the valve performance was predictable. The licensee would consider the need to increase the valve factor during its design-basis evaluation and setup based on test data from similar valves.

2. Electric Power Research Institute MOV performance prediction methodology (PPM). This method was developed for those valves that could not be dynamically tested. The PPM required internal measurements of the valve to provide assurance that the valve performance was predictable. NRC staff began accepting the use of the PPM even where dynamic testing for an MOV was practicable.

3. MOV valve grouping. Where valve-specific dynamic testing was not performed and the PPM was not used, the staff accepted grouping of MOVs that were dynamic tested at the plant to apply the plant-specific test information to an MOV in the group. Using plant-specific data allowed the licensee to know the valve performance and maintenance history and helped provide confidence that the valve performance was predictable.

4. The use of valve test data from other plants or research programs. The NRC ranks this as the least-preferred approach (with the most margin required) because the licensee would have minimal information regarding the tested valve and its history. In such cases, the NRC inspectors would perform an available capability evaluation of the MOV to provide confidence that the MOV had significant capability margin to close GL 89-10 for that MOV.

GL 96-05 superseded GL 89-10 and requested that each plant establish a program, or ensure the effectiveness of its current program, to verify on a periodic basis that safety-related MOVs continue to be capable of performing their safety functions within the current licensing basis of the facility. The program should ensure that the licensee can properly identify and account for changes in required performance resulting from degradation (such as those caused by age).

In response to GL 96-05, the nuclear industry joined together to form the JOG MOV periodic verification program. The JOG program consisted of three elements: (1) an "interim" MOV periodic verification program for licensees to use in response to GL 96-05 during development of a long term program, (2) a 5-year MOV dynamic diagnostic test program, and (3) a long-term MOV periodic diagnostic test program to be based on the information from the dynamic testing program. The nuclear industry designed the JOG program to answer the valve degradation question as it pertained to valve configuration, design, and system application.

ASME OM Code, 1995 Edition with the 1996 Addenda, Subsection ISTC 4.1, "Valve Position Verification," states that "Valves with remote position indicators to be observed locally at least once every 2 years to verify that valve operation is accurately indicated. Where practicable, this local observation should be supplemented by other indications such as use of flow meters or other suitable instrumentation to verify obturator position. These observations need not be concurrent. Where local observation is not possible, other indications shall be used for verification of valve operation."

ASME OM Code, 1995 Edition with the 1996 Addenda, Subsection ISTC 4.2.3, "Valve Obturator Movement," states that "The necessary valve obturator movement shall be determined by exercising the valve while observing an appropriate indicator, such as indicating lights that signal the required change of obturator position, or by observing other evidence, such as changes in system pressure, flow rate, level, or temperature, that reflects change of obturator position."

NUREG-1482, Revision 1, "Guidelines for Inservice Testing at Nuclear Power Plants" (ADAMS Accession No. ML050550290), Section 4.2.7, "Verification of Remote Position Indication for Valves by Methods Other Than Direct Observation," discusses the requirements of ASME OM Code, 1995 Edition with the 1996 Addenda, Subsection ISTC 4.1. The discussion emphasizes the importance of accurate position indication for safety-related valves under all plant conditions. The discussion states in part that "For certain types of valves that can be observed locally, but for which valve stem travel does not ensure that the stem is attached to the disk, the local observation should be supplemented by observing an operating parameter as required by Subsection ISTC-3700 and ISTC-3520 [4.1, 4.2, and 4.5]."

The ASME OM Code is a living document which allows it to be updated and improved. Since the issuance of the ASME OM Code 1995 Edition through the 1996 Addenda, there has been a major change to the format and numbering system. Currently, 10 CFR 50.55a recognizes the ASME OM Code 2004 Edition through the 2006 Addenda as the code of record. The requirement of ASME OM Code, 1995 Edition with the 1996 Addenda, Subsection ISTC 4.1, “Valve Position Verification,” is now under ASME OM Code, 2004 Edition with the 2006 Addenda, Subsection ISTC-3700, “Position Verification Testing.” The requirement of ASME OM Code, 1995 Edition with the 1996 Addenda, Subsection ISTC 4.2.3, “Valve Obturator Movement,” is now under ASME OM Code, 2004 Edition with the 2006 Addenda, Subsection ISTC-3530, “Valve Obturator Movement.”

DISCUSSION

This IN discusses operating experiences involving a failed safety-related valve in which the licensee failed to recognize the stem-to-disc separation for an extended period of time. Investigation of the valve failure identified weaknesses in the IST program, GL 89-10 program, and MOV testing procedures as well as improper implementation of the JOG program. This IN informs the industry of these issues so that other facilities can consider actions, as appropriate, to avoid similar problems.

ASME OM Code, 1995 Edition with the 1996 Addenda, ISTC 4.2.3 allows nuclear power plant licensees to monitor indicating lights in the control room when exercising a valve to meet quarterly stroke-time testing requirement. The NRC recognizes that indicating lights do not ensure that the valve obturator (stem/disc assembly) is moving properly between the appropriate open-to-close and close-to-open valve positions. For example, the internal mechanism of the valve and its operator (such as the position limit switches and stem-to-disc connection) must be intact and operating properly for the indicating lights to reflect actual valve position. Therefore, although not explicitly required, licensees should consider additional alternative parameters to verify that the indicating lights accurately reflect valve obturator position.

ASME OM Code, 1995 Edition with the 1996 Addenda, ISTC 4.1 requires confirmation on a 2-year frequency that the indicating lights reflect actual valve operation. ISTC 4.1 allows flexibility to nuclear power plant licensees in verifying that operation of valves with remote position indicators is accurately indicated. ISTC 2(b), “Owner’s Responsibility” states that the “Owner shall identify, categorize and list in the plant records each valve to be tested in accordance with the rules of this Subsection, including Owner-specified acceptance criteria. The Owner shall specify test conditions. The Owner shall ensure that the application, method, and capability of each nonintrusive technique is qualified.”

Licensee IST programs can help identify stem-to-disc separation as valves are tested. The NRC independent review panel assigned to investigate the Browns Ferry, Unit 1 MOV stem disc separation event reviewed IST requirements and concluded that the ASME OM Code is not clear with respect to the extent to which the Code requires certainty in the verification of obturator position during testing. Because of the ambiguity of the OM Code, it is possible for a testing program to meet the minimum requirements of the OM Code with respect to obturator position verification and valve operation being accurately indicated, but not fully meet the intent of verifying actual obturator position. Supplemental indicators such as flow measurement, system pressure changes, level, temperature, or adequate acceptance criteria for the augmented MCC testing can improve the likelihood of identifying valve failures.

Licensee IST programs that implement augmented testing (e.g., obtaining motor current signature data during valve stroke exercise), should contain qualitative or quantitative acceptance criteria for the data obtained. Criterion V of Appendix B to 10 CFR Part 50 requires safety-related components that are subjected to test activities be required to have appropriate instructions, procedures, or drawings and qualitative or quantitative acceptance criteria for determining that the activity has been successfully completed. Licensees are encouraged to review their safety-related component test procedures for compliance with Criterion V of Appendix B to 10 CFR Part 50.

A thorough evaluation of safety-related MOVs is important to ensure that valves are not inappropriately excluded from the MOV program. Consideration for all of a valve's functional requirements is important to ensure that a valve is appropriately included in the MOV program to meet all requirements. Licensees are encouraged to perform a periodic review of their MOV program scope for component applicability. Information on MOV program scoping may be found in GL 89-10 and GL 89-10, Supplement 1.

Additionally, in the implementation of the JOG MOV program for Browns Ferry to meet its commitment to GL 96-05, TVA included several valve modifications which disqualified the original design-basis capability verification that was obtained through GL 89-10. The final JOG document provides an approach for obtaining a qualifying design basis for valves that have had a disqualifying event. The JOG approach has a certain amount of dynamic testing for reaching a qualifying design basis to support the new valve configuration. As specified by the JOG final report, each plant is responsible for establishing a new design basis for those valves that have had a disqualifying event. Browns Ferry did not use the JOG approach for obtaining a qualifying design basis for the valve modifications. Instead, Browns Ferry took the least-preferred approach (method 4 stated above) for reestablishing the design basis of the modified valves. The licensee used similar valve test data obtained from the JOG final report in conjunction with an engineering analysis to justify the design-basis capability of the modified valves. Following NRC 95003 inspection activities, NRC inspectors determined that JOG test data was not intended to be used to establish initial design-basis capability of MOVs or modified MOVs. The NRC concluded that the methodology in the Browns Ferry program documentation for justifying the design-basis capability of the modified valves did not satisfy 10 CFR 50.55a(b)(3)(ii).

More information on this topic can be found in Browns Ferry Nuclear Plant–NRC Inspection Procedure 95003 Supplemental Inspection Report 05000259/2011011, 05000260/2011011, and 05000296/2011011 (Part 1) (ADAMS Accession No. ML113210602).

To determine if additional regulatory action is necessary, NRC staff plans to continue its evaluation of licensee implementation of the provisions in the ASME OM Code for valve position verification and obturator movement.

CONTACT

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

/RA by JLuehman for/

Laura A. Dudes, Director
Division of Construction Inspection
and Operational Programs
Office of New Reactors

/RA by SBahadur for/

Timothy J. McGinty, Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Technical Contact: Michael Farnan, NRR
301-415-1486
E-mail: Michael.Farnan@nrc.gov

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

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E-mail: Michael.Farnan@nrc.gov

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OFFICE	NRR/DE/EPTB*	Tech Editor*	NRR/DE/EPTB/BC*	NRR/DE/D*	NRR/DORL/D*	
NAME	MFarnan	CHsu	AMcMurtray (JHuang for)	PHiland	MEvans	
DATE	06/18/12	05/30/12	06/27/12	06/28/12	06/21/12	
OFFICE	NRR/DPR/PGCB/LA	NRR/DPR/PGCB/PM	NRR/DPR/PGCB/BC	NRR/DPR/PGCB/LA	NRO/DCIP/D	NRR/DPR/D
NAME	CHawes	ARussell	DPelton	CHawes	LDudes (JLuehman for)	TMcGinty (SBahadur for)
DATE	07/12/12	7/16/12	7/17/12	7/18/12	7/23/12	7/24/12

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