

PART 52, FUNCTIONAL DESIGN AND QUALIFICATION, AND PRESERVICE AND INSERVICE TESTING PROGRAMS FOR PUMPS, VALVES AND DYNAMIC RESTRAINTS

PROGRAM APPLICABILITY: IMC 2504 App B

73758-01 INSPECTION OBJECTIVES

01.01 Functional Design and Qualification

To evaluate the establishment, implementation, and results of the functional design and qualification of pumps, valves, and dynamic restraints (snubbers) during construction of nuclear power plants with a combined license (COL) in accordance with Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52.

01.02 Preservice and Inservice Testing Programs

To evaluate the establishment, implementation, and results of preservice testing (PST) and inservice testing (IST) programs for pumps, valves, and dynamic restraints during construction of nuclear power plants with a COL license in accordance with 10 CFR Part 52.

73758-02 INSPECTION REQUIREMENTS AND GUIDANCE

This inspection procedure (IP) provides inspection requirements and guidance for the functional design, qualification, and PST/IST programs for pumps, valves, and dynamic restraints at nuclear power plants under construction in accordance with 10 CFR Part 52.

The completion of this IP involves an initial program inspection, an implementation inspection of the functional design and qualification program, an implementation inspection of the PST/IST program, and a close-out inspection for the functional design, qualification, and PST/IST programs for pumps, valves, and dynamic restraints in preparation for plant startup. These inspection activities will be conducted at different times during the construction process. The close-out inspection for this IP should be completed 6 months before planned fuel loading in order to support an NRC staff finding on the completion of all operational programs consistent with the schedule for the finding that the inspections, tests, analyses, and acceptance criteria (ITAAC) have been met in accordance with 10 CFR 52.103(g).

To help clarify the distinct inspection activities, this IP has been prepared with four appendices as follows:

Appendix A, “Review of Functional Design, Qualification, and PST/IST Programs for Pumps, Valves, and Dynamic Restraints,”

Appendix B, “Implementation of Functional Design and Qualification Program for Pumps, Valves, and Dynamic Restraints,”

Appendix C, “Implementation of PST/IST Program for Pumps, Valves, and Dynamic Restraints,” and

Appendix D, “Close-Out Inspection for Functional Design, Qualification, and PST/IST Programs for Pumps, Valves, and Dynamic Restraints in Preparation for Plant Startup.”

The attachments to this IP provide more specific inspection requirements and guidance for functional design, qualification, and PST/IST programs for motor-operated valves (MOVs), air-operated valves (AOVs), and pyrotechnic-actuated valves (squib valves) to be used in nuclear power plants licensed under 10 CFR Part 52. Additional attachments for other components or associated activities may be included in the future.

The NRC regulations in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” specify requirements for structures, systems, and components (SSCs) important to safety that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. General Design Criterion (GDC) 1 in Appendix A to 10 CFR Part 50 states that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. GDC 1 also states that where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency, and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. GDC 1 also requires that a quality assurance (QA) program be established and implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50 specifies criteria for the QA program to provide adequate confidence that SSCs will perform their safety-related functions satisfactorily in service.

As of August 17, 2017, the NRC regulations in 10 CFR 50.55a incorporate by reference the American Society of Mechanical Engineers (ASME) *Operation and Maintenance of Nuclear Power Plants*, Division 1, OM Code: Section IST (commonly referred to as the OM Code) from the 1995 Edition through the 2012 Edition for implementation of PST and IST programs for pumps, valves, and dynamic restraints used in nuclear power plants. The ASME OM Code (1995 Edition through 2006 Addenda) specifies the performance of stroke-time testing of motor-operated valves (MOVs) on a quarterly frequency as part of the IST program. Beginning with the 2009 Edition, the ASME OM Code includes Mandatory Appendix III, “Preservice and Inservice Testing of Active Electric Motor Operated Valve Assemblies in Light-Water Reactor Power Plants,” which replaces quarterly stroke time testing of MOVs with periodic exercising at least every refueling outage, and periodic diagnostic testing based on capability margin up to a maximum interval of 10 years. Beginning with the 2011 Addenda, the ASME OM Code includes

Subsection ISTF, "Inservice Testing of Pumps in Light-Water Reactor Nuclear Power Plants – Post-2000 Plants," with PST and IST provisions for pumps in nuclear power plants that were (or will be) issued their construction permit, or COL for construction and operation, on or following January 1, 2000 (referred to herein as new reactors). Beginning with the 2012 Edition, the ASME OM Code includes PST and IST surveillance provisions for pyrotechnic-actuated (squib) valves in new reactors in Subsection ISTC, "Inservice Testing of Valves in Light-Water Reactor Nuclear Power Plants."

The NRC regulations in 10 CFR 50.55a include conditions for the implementation of the PST and IST provisions in specific editions and addenda of the ASME OM Code. As of August 17, 2017, the NRC regulations in 10 CFR 50.55a(b)(3) specify the following:

- (i) the acceptability of ASME Standard NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," where supplemented by 10 CFR Part 50, Appendix B, as necessary;
- (ii) periodic verification of MOV design-basis capability and acceptance of ASME OM Code, Mandatory Appendix III, with conditions;
- (iii) provisions for new reactors, including
 - (a) periodic verification of power-operated valve (POV) design-basis capability,
 - (b) check valve bi-directional testing,
 - (c) flow-induced vibration monitoring, and
 - (d) treatment of pumps, valves, and dynamic restraints with high safety significance;
- (iv) acceptance of Mandatory Appendix II, "Check Valve Condition Monitoring Program," with conditions;
- (v) acceptance of Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants," with conditions;
- (vi) a maximum 2-year test interval for manual valves;
- (vii) acceptance of 2012 Edition of Subsection ISTB, "Inservice Testing of Pumps in Light-Water Reactor Nuclear Power Plants – Pre-2000 Plants," but prohibition of the use of Subsection ISTB in the 2011 Addenda of the ASME OM Code;
- (viii) use of Subsection ISTE, "Risk-Informed Inservice Testing of Components in Light-Water Reactor Nuclear Power Plants," requires approval of an alternative in accordance with 10 CFR 50.55a(z);
- (ix) use of Subsection ISTF in the 2012 Edition of the ASME OM Code requires application of Mandatory Appendix V, "Pump Periodic Verification Test Program," and prohibition of the use of Subsection ISTF in the 2011 Addenda of the ASME OM Code;
- (x) acceptance of ASME OM Code Case OMN-20, "Inservice Test Frequency," for applicable editions and addenda of the ASME OM Code; and
- (xi) supplemental requirements for the implementation of paragraph ISTC-3700, "Position Verification Testing," in Subsection ISTC for valve position indication beginning with the 2012 Edition of the ASME OM Code.

In addition, the NRC regulations in 10 CFR 50.55a(f)(4) clarify the scope of the PST and IST program for pumps and valves to be consistent with the scope of the ASME OM Code. The regulations allow safety-related pumps and valves not classified as ASME Code Class 1, 2, or 3 to be addressed as part of an augmented IST program without requesting relief or an alternative. Otherwise, the basis for deviations from the ASME OM Code must demonstrate an acceptable level of quality and safety, or that implementing the Code provision would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety,

where documented and available for NRC review. See *Federal Register* Notice 82 FR 32934, dated July 18, 2017, for additional information.

More recently, ASME has published the 2015 and 2017 Editions of the ASME OM Code. Beginning with the 2017 Edition, the ASME OM Code includes Mandatory Appendix IV, "Preservice and Inservice Testing of Active Pneumatically Operated Valve Assemblies in Nuclear Reactor Power Plants," which requires quarterly stroke time testing and preservice performance assessment testing (PAT) for all AOVs, and periodic PAT for AOVs with high safety significance up to a maximum interval of 10 years. The NRC is considering proposed rulemaking to incorporate by reference the 2015 and 2017 Editions of the ASME OM Code with any appropriate conditions. For this proposed rulemaking, the NRC is also considering extending the current 12-month time period specified in 10 CFR 50.55a(f)(4) for the applicable ASME OM Code edition to be applied to the initial and periodic 10-year PST/IST programs.

In SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License without Inspections, Tests, Analyses, and Acceptance Criteria," the NRC staff discussed the level of programmatic information needed for approval of a COL without ITAAC for operational programs. In a **Staff Requirements Memorandum** dated May 14, 2004, the Commission stated that "fully described" for an operational program should be understood to mean that the program is clearly and sufficiently described in terms for scope and level of detail to allow a reasonable assurance finding of acceptability. The Commission noted that required operational programs should always be described at a functional level and an increasing level of detail where implementation choices could materially and negatively affect the program effectiveness and acceptability. The Commission also stated that the staff should continue the practice of inspecting relevant licensee procedures and programs in a similar manner as was done in the past and consistent with applicable inspection programs. The staff should also continue to ensure, consistent with the inspection and enforcement processes, that licensees address pertinent issues prior to fuel loading. To allow the staff to complete the necessary inspections, procedure-level information that has typically not been docketed for staff review should continue to be made available to NRC inspectors with sufficient time to allow the inspectors to complete the necessary inspections and resolve pertinent issues.

In SECY-05-0197, "Review of Operational Programs in a Combined License Application and General Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," the NRC staff defines operational programs for new nuclear power plants as programs that are required by regulation, are reviewed by NRC staff for acceptability with the results documented in the safety evaluation report (SER), and will be verified for implementation by NRC inspectors. SECY-05-0197 includes the PST and IST programs, and MOV Testing program, as operational programs. SECY-05-0197 discusses the information necessary for the staff to make a reasonable assurance finding on the acceptability of the operational program in the review of a COL application.

In their Final Safety Analysis Report (FSAR), COL applicants have incorporated by reference the functional design, qualification, and PST/IST program descriptions provided in the design control document (DCD) or FSAR submitted by the design certification applicant with supplemental information or departures. Therefore, the full description of the functional design, qualification, and PST/IST programs is provided by the combination of the design certification DCD/FSAR and the COL FSAR, together with the NRC SERs on the design certification application and the COL application. The NRC staff will conduct inspections of the functional design, qualification, and PST/IST programs during construction of a new nuclear power plant licensed under 10 CFR Part 52 to determine that the programs have been established, and are

being effectively implemented, in a manner that satisfies the NRC regulations and is consistent with the design certification DCD or FSAR provisions, COL FSAR provisions, NRC SER findings, and Commission license conditions.

The NRC inspector may request assistance from NRC headquarters technical staff in preparing for and conducting the inspection of the functional design, qualification, and PST/IST programs for pumps, valves, and dynamic restraints at nuclear power plants being constructed under 10 CFR Part 52. Training may be provided by headquarters staff for inspectors in preparation for performing inspections using this IP. Also, headquarters staff may be available to support the performance of inspections using this IP either directly on site or indirectly by telephone.

In planning the NRC inspection, the inspector should request the licensee to provide in advance materials that will be necessary for conducting the inspection. The inspector should request that the licensee provide the documentation of the functional design and qualification of pumps, valves, and dynamic restraints; PST/IST programs; surveillance and testing procedures; surveillance and test results to date; and component set-up information. Other materials that the inspector should review in preparing for the inspection include the design certification DCD/FSAR, COL FSAR (including COL Information Items), NRC SER, Commission license conditions, ITAAC for applicable plant components, plant technical specifications, Probabilistic Risk Assessment (PRA) reports, component or system design bases documentation, and procedures for the design change process. The inspector should also review the applicable edition of the ASME *Boiler & Pressure Vessel Code* (BPV Code) and OM Code as incorporated by reference in 10 CFR 50.55a. The inspector should review the licensee's planned use of ASME Code Cases as accepted in the applicable regulatory guides that have been incorporated in 10 CFR 50.55a.

The inspector should review the results of NRC audits of the implementation of the functional design, qualification, and IST programs conducted during review of COL applications. The inspector should review the results of ITAAC inspections and vendor inspections related to the functional design and qualification of pumps, valves, and dynamic restraints. The inspector should review the reports of any previous inspections performed using this IP for inspection findings and follow-up actions.

In SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," as accepted by the Commission in its Staff Requirements Memorandum dated June 28, 1995, the NRC staff specified the policy regarding the functional design, qualification, and inservice testing of RTNSS pumps and valves for new passive design nuclear power plants (such as the Vogtle Units 3 and 4 AP1000 reactors). In Supplement 2 to NUREG-1793 on the AP1000 design certification in Section 17.4.2, the staff states that activities of the reliability assurance program (RAP) after the design phase will be incorporated into existing plant programs, including the IST program. In NUREG-2124 on the Vogtle Units 3 and 4 COL application in Section 17.4.1, the staff indicates that operational phase reliability assurance activities (OPRAAs) are integrated into other plant programs. As of August 17, 2017, the NRC regulations in 10 CFR 50.55a specify that licensees shall assess the operational readiness of pumps, valves, and dynamic restraints within the scope of RTNSS for applicable reactor designs. As part of this inspection, the inspector should verify that the licensee has incorporated the activities to provide reasonable assurance that RTNSS pumps, valves, and dynamic restraints can perform their intended functions into plant programs.

02.01 Functional Design and Qualification

See Appendix A to this IP for inspection requirements and guidance for evaluating the functional design and qualification program for pumps, valves, and dynamic restraints. See Appendix B to this IP for inspection requirements and guidance for evaluating the implementation of the functional design and qualification program for pumps, valves, and dynamic restraints. See Appendix D to this IP for inspection requirements and guidance for evaluating the completion of the functional design and qualification process for pumps, valves, and dynamic restraints in preparation for plant startup.

02.02 Preservice and Inservice Testing Program

See Appendix A to this IP for inspection requirements and guidance for evaluating the PST/IST program for pumps, valves, and dynamic restraints. See Appendix C to this IP for inspection requirements and guidance for evaluating the implementation of the PST/IST program for pumps, valves, and dynamic restraints. See Appendix D to this IP for inspection requirements and guidance for evaluating the full implementation of the PST/IST program for pumps, valves, and dynamic restraints in preparation for plant startup.

73758-03 INSPECTION RESOURCE ESTIMATE

Completion of this IP for functional design, qualification, and PST/IST programs for pumps, valves, and dynamic restraints is expected to take 120 hours of direct inspection effort on average for each inspection specified in Appendices A to D if the programs are well developed and implemented. The inspection resource estimate might increase where the inspection scope is expanded based on the results of the inspection sample, or adverse operating experience is identified at other nuclear power plants, or issues are identified related to program development and implementation. As discussed in Appendix A to this IP, the inspection resource estimate might increase for the program review inspection where it is determined that a complete review of the design basis requirements and operating conditions for all safety-related pumps, valves, or dynamic restraints is appropriate for the prototype plant of a new nuclear power plant design.

73758-04 PROCEDURE COMPLETION

For construction activities, this IP is complete upon confirmation that the establishment, implementation, and results of the functional design, qualification, and PST/IST programs for pumps, valves, and dynamic restraints satisfy the NRC regulations specified in 10 CFR Part 52. To complete this IP, the staff should conduct at least four inspections using this IP as part of the Construction Inspection Program at each nuclear power plant site licensed under 10 CFR Part 52. The inspectors may use the results of ITAAC inspections in completing portions of this IP. Similarly, the inspectors may use the results of inspections using this IP to support completion of NRC review of specific component ITAAC.

The staff should perform an initial inspection using Appendix A to this IP early in the construction process prior to the installation of pumps, valves, and dynamic restraints to confirm the establishment of the functional design and qualification process and PST/IST programs for pumps, valves, and dynamic restraints, consistent with the NRC regulations and program descriptions in the design certification DCD/FSAR and COL FSAR as accepted in the applicable NRC SERs.

The staff should perform an inspection prior to or during initial installation of pumps, valves, and dynamic restraints to confirm the implementation of the functional design and qualification process using Appendix B to this IP.

The staff should perform an inspection following installation of pumps, valves, and dynamic restraints to confirm the implementation of the PST/IST programs using Appendix C to this IP.

Finally, the staff should perform an inspection using Appendix D to this IP 6 months before planned fuel loading to close out the NRC construction inspection activities for the functional design and qualification process, and the PST/IST programs that demonstrate the design-basis capability and operational readiness of pumps, valves, and dynamic restraints to perform their safety functions. The staff will use the results of the close-out inspection to support the NRC finding on the implementation of operational programs consistent with the schedule for reaching a finding on ITAAC completion in accordance with 10 CFR 52.103(g). This close-out inspection would identify any remaining follow-up actions to be addressed following fuel loading. In preparing the close-out inspection report, the inspector should document any follow-up actions for transition to the NRC operations inspection staff.

The inspectors are responsible for ensuring that each sample in the inspection procedure is completed and evaluated to a level that provides reasonable assurance that the licensee has met NRC regulatory requirements within the program area being inspected. The inspection sample was developed based on engineering judgment. Inspectors may expand the minimum number to aid in determining the extent of the condition, should compliance concerns arise. Completion of other recommended actions contained in this guidance should not be viewed as mandatory for the inspector to determine if an inspection sample has been adequately addressed. Should questions arise regarding sample size, inspectors should consult with the technical contacts at NRC headquarters for clarification.

73758-05 REFERENCES

American Society of Mechanical Engineers [ASME]: *Code for Operation and Maintenance of Nuclear Power Plants*, and Code cases.

ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants."

Code of Federal Regulations, Title 10, Part 52.

Federal Register, 64 FR 51370, "Industry Codes and Standard; Amended Requirements," September 22, 1999.

Federal Register, 82 FR 32934, "Incorporation by Reference of American Society of Mechanical Engineers Codes and Code Cases," July 18, 2017.

Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," June 28, 1989, and seven supplements (June 13, 1990; August 3, 1990; October 25, 1990; February 12, 1992; June 28, 1993; March 8, 1994; and January 24, 1996).

Generic Letter 90-09, "Alternative Requirements for Snubber Visual Inspection Interval and Corrective Action," December 11, 1990.

Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," August 17, 1995.

Generic Letter 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," September 18, 1996.

Information Notice 96-48, "Motor-Operated Valve Performance Issues," August 21, 1996, and Supplement 1, July 24, 1998.

Information Notice 97-16, "Preconditioning of Plant Structures, Systems, and Components Before ASME Code Inservice Testing or Technical Specification Surveillance Testing," April 4, 1997.

Information Notice 2003-15, "Importance of Followup Activities in Resolving Maintenance Issues," September 5, 2003.

Information Notice 2006-03, "Motor Starter Failures due to Mechanical-Interlock Binding," January 25, 2006.

Information Notice 2006-15, "Vibration-Induced Degradation and Failure of Safety-Related Valves," July 27, 2006.

Information Notice 2006-24, "Recent Operating Experience Associated with Pressurizer and Main Steam Safety/Relief Valve Lift Setpoints," November 14, 2006.

Information Notice 2006-29, "Potential Common Cause Failure of Motor-Operated Valves as a Result of Stem Nut Wear," December 14, 2006.

Information Notice 2008-09, "Turbine-Driven Auxiliary Feedwater Pump Bearing Issues," May 22, 2008.

Information Notice 2008-20, "Failures of Motor-Operated Valve Actuator Motors with Magnesium Alloy Rotors," December 8, 2008.

Information Notice 2010-03, "Failures of Motor-Operated Valves Due to Degraded Stem Lubricant," February 3, 2010.

Information Notice 2010-20, "Turbine-Driven Auxiliary Feedwater Pump Repetitive Failures," September 24, 2010.

Information Notice 2012-06, "Ineffective Use of Vendor Technical Recommendations," April 24, 2012.

Information Notice 2012-14, "Motor-Operated Valve Inoperable Due To Stem-Disc Separation," July 24, 2012.

Information Notice 2013-14, "Potential Design Deficiency in Motor-Operated Valve Control Circuitry," August 23, 2013.

Information Notice 2014-11, "Recent Issues Related to the Qualification and Commercial Grade Dedication of Safety-Related Components," September 19, 2014.

Information Notice 2015-09, "Mechanical Dynamic Restraint (Snubber) Lubricant Degradation Not Identified Due to Insufficient Service Life Monitoring," dated September 24, 2015.

Information Notice 2015-13, "Main Steam Isolation Valve Failure Events," December 10, 2015.

Information Notice 2016-09, "Recent Issues Identified When Using Reverse Engineering Techniques in the Procurement of Safety-Related Components," July 15, 2016.

Information Notice 2017-03, "Anchor/Darling Double Disc Gate Valve Wedge Pin and Stem-Disc Separation Failures," June 15, 2017.

Inspection Manual Part 9900, "Technical Guidance on Maintenance – Preconditioning."

Inspection Procedure (IP) 50090, "Pipe Support and Restraint Systems."

IP 61726, "Surveillance Observations."

IP 62707, "Maintenance Observations."

IP 62708, "Motor-Operated Valve Capability."

IP 62710, "Power-Operated Gate Valve Pressure Locking and Thermal Binding."

IP 70370, "Testing Pipe Support and Restraint Systems."

IP 71111.22, "Surveillance Testing."

Licensee Event Report 237/2011-005, "Standby Liquid Control Explosive Valve Failure," December 18, 2012, submitted by Exelon Generation (ADAMS Accession No. ML12363A044).

Licensee Event Report 424/2012-005-00, "Main Steam Isolation Valve Failure," November 29, 2012, submitted by Southern Nuclear Operating Company (ADAMS Accession No. ML12339A190).

NRC Inspection Procedure 95003 Supplemental Inspection Report 05000259/2011011, 05000260/2011011, and 05000296/2011011 (Part 1) for Browns Ferry Nuclear Plant, November 17, 2011 (ADAMS Accession No. ML113210602).

NRC Inspection Report No. 50-361 and 362/99-18, for San Onofre Nuclear Generating Station, Units 2 and 3, January 4, 2000 (ADAMS Accession Nos. ML003672750 and ML003672773).

NRC Inspection Report No. 50-237 and 249/2012008 for Dresden Nuclear Power Station, Units 2 and 3, April 23, 2012 (ADAMS Accession No. ML12114A356).

NRC Inspection Report No. 50-424 and 425/2017009 for Vogtle Electric Generating Plant, Units 1 and 2, dated January 23, 2018 (ADAMS Accession No. ML18024A566).

NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants."

NUREG-1793 (Supplement 2, September 2011), "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design."

NUREG-2124 (August 5, 2011), "Final Safety Evaluation Report Related to the Combined Licenses for Vogtle Electric Generating Plant, Units 3 and 4."

Regulatory Guide 1.82, Revision 4, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," March 2012.

Regulatory Guide 1.100, Revision 3, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," September 2009.

Regulatory Guide 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," August 1998.

Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code."

Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009.

Regulatory Issue Summary 2000-03, "Resolution of Generic Safety Issue 158, Performance of Safety-Related Power-Operated Valves under Design Basis Conditions," March 15, 2000.

Regulatory Issue Summary 2001-15, "Performance of DC-Powered Motor-Operated Valve Actuators," August 1, 2001.

Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," March 22, 2007.

Regulatory Issue Summary 2010-06, "Inservice Inspection and Testing Requirements of Dynamic Restraints (Snubbers)," June 1, 2010.

Regulatory Issue Summary 2011-13, "Followup to Generic Letter 96-05 for Evaluation of Class D Valves Under Joint Owners Group Motor-Operated Valve Periodic Verification Program," January 6, 2012.

Regulatory Issue Summary 2012-08, Revision 1, "Developing Inservice Testing and Inservice Inspection Programs under 10 CFR Part 52," July 17, 2013.

Special Inspection Report 05000400/2012008 (dated July 12, 2012) on the MSIV failure at Shearon Harris Nuclear Power Plant (ADAMS Accession No. ML12194A281).

Standard Review Plan Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," Revision 3, March 2007.

END

Appendices:

- A. Review of Functional Design, Qualification, and PST/IST Programs for Pumps, Valves, and Dynamic Restraints
- B. Implementation of Functional Design and Qualification Program for Pumps, Valves, and Dynamic Restraints
- C. Implementation of PST/IST Program for Pumps, Valves, and Dynamic Restraints
- D. Close-out Inspection for Functional Design, Qualification, and PST/IST Programs for Pumps, Valves, and Dynamic Restraints in Preparation for Plant Startup

Attachments:

- 1. Motor-Operated Valves
- 2. Air-Operated Valves
- 3. Pyrotechnic-Actuated Valves
- 4. Revision History for IP 73758

APPENDIX A - Review of Functional Design, Qualification, and PST/IST Programs for Pumps, Valves, and Dynamic Restraints

73758-APPA.01 INSPECTION OBJECTIVE

The objective of this inspection is to evaluate the development of the functional design, qualification and preservice testing (PST) and inservice testing (IST) programs for pumps, valves, and dynamic restraints during construction of a nuclear power plant licensed under 10 CFR Part 52. As part of this inspection, the inspector will confirm that the functional design and qualification process specified in the plant program and procedures for pumps, valves, and dynamic restraints is consistent with the provisions in the final safety analysis report (FSAR) submitted with the combined license (COL) application and its incorporation by reference of the design control document (DCD) or FSAR for the design certification application as accepted in the applicable U.S. Nuclear Regulatory Commission (NRC) safety evaluation reports (SERs). The NRC inspector will also determine whether the PST/IST programs satisfy the program description specified in the COL FSAR and its incorporation by reference of the design certification DCD/FSAR as accepted in the NRC SERs, and comply with the applicable regulatory requirements.

73758-APPA.02 INSPECTION REQUIREMENTS AND GUIDANCE

02.01 Functional Design and Qualification.

The NRC regulations in 10 CFR Parts 50 and 52 include requirements for the design and qualification of nuclear power plant components. As part of the review of the application for a COL under 10 CFR Part 52, the NRC staff evaluated the adequacy of the methodology specified in the applicable design certification DCD/FSAR and COL FSAR for the functional design and qualification of pumps, valves, and dynamic restraints at a nuclear power plant being licensed under 10 CFR Part 52. Specifically, the NRC staff followed the guidance in NRC Standard Review Plan (SRP) Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," in its review of the design certification and COL applications.

During plant construction, NRC inspectors will confirm the adequate implementation of the licensee's methodology for the functional design and qualification of pumps, valves, and dynamic restraints.

a. Responsibilities

(1) Inspection Requirements

Determine whether the licensee has appropriately assigned responsibilities and authority to persons and organizations for ensuring the functional design and qualification of pumps, valves, and dynamic restraints that perform a safety function at the nuclear power plant.

(2) Inspection Guidance

The NRC inspector should review the methodologies for the functional design and qualification of pumps, valves, and dynamic restraints with a safety function specified in the design certification DCD/FSAR and COL FSAR. The NRC review of those methodologies is described in the SERs on the applicable design certification and COL applications. Using this information, the inspector should determine whether the licensee has assigned responsibilities to persons and organizations for ensuring the functional design and qualification of pumps, valves, and dynamic restraints that perform a safety function at the nuclear power plant.

b. Program and procedures

(1) Inspection Requirements

Select three to five safety systems to review the functional design and qualification of a sample of pumps, valves, and dynamic restraints. The inspection scope may be expanded based on the results of the inspection sample or operating experience at other nuclear power plants. For the prototype plant of a new nuclear power plant design, a complete review of the design basis requirements and operating conditions for all safety-related pumps, valves, or dynamic restraints might be performed, if appropriate.

(2) Inspection Guidance

The NRC inspector will typically conduct a sampling inspection of the methodologies specified for the functional design and qualification of pumps, valves, and dynamic restraints. The inspector should select three to five safety systems to evaluate the methodologies for demonstrating the functional design and qualification of pumps, valves, and dynamic restraints. The inspection should focus on overall functional design and qualification activities for components in a representative sample of safety-significant systems.

The selection process should consider plant risk (associated with failures of pumps, valves and dynamic restraints), maintenance, and any identified programmatic weaknesses. The selected systems should contain a variety of pump and valve types to the extent possible. Where used in the nuclear power plant being inspected, pyrotechnic-actuated valves (squib valves) with new designs or high safety significance should be selected as part of the inspection sample. The selected dynamic restraints should be of at least three different load classifications and various degrees of accessibility (easy or difficult accessibility). For more details, see Inspection Procedure (IP) IP 50090, "Pipe Support and Restraint Systems," and IP 70370, "Testing Pipe Support and Restraint Systems." The sample of pumps, valves, and dynamic restraints may be expanded based on the inspection findings where concerns are raised regarding the capability of specific components to perform their design-basis safety functions. As discussed above, the inspection sample of safety-related pumps, valves, or dynamic restraints might be expanded for a prototype plant of a new nuclear power plant design.

The inspector should confirm that the functional design and qualification process specified in the plant program and procedures for pumps, valves, and dynamic restraints is consistent with the provisions in the design certification DCD/FSAR and COL FSAR as accepted in the applicable NRC SERs. The inspector should check the **inspections, tests, analyses, and acceptance criteria (ITAAC)** for the nuclear power plant to identify requirements for the functional design and qualification of specific pumps, valves, and dynamic restraints. The inspector may select pumps, valves, and dynamic restraints in the NRC staff list of targeted ITAAC.

The American Society of Mechanical Engineers (ASME) Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," includes provisions for the functional design and qualification of active mechanical equipment in nuclear power plants. In this standard, ASME incorporated lessons learned from valve operating experience and research programs for the design and qualification of components. The NRC staff has accepted the use of ASME QME-1-2007 in Revision 3 to Regulatory Guide (RG) 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," with specific conditions. The applicants for most nuclear power plants being licensed under 10 CFR Part 52 specified the implementation of ASME QME-1-2007 as accepted in Revision 3 to RG 1.100 in their licensing documentation. The inspector should review the licensee's program and procedures for the functional design and qualification of pumps, valves, and dynamic restraints in comparison to the methodologies described in the SERs on the design certification and COL applications.

In 2017, ASME published ASME Standard QME-1-2017, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," to provide updated qualification provisions for pumps, valves, and dynamic restraints. The NRC staff is preparing proposed Revision 4 to RG 1.100 to address the acceptance of ASME QME-1-2017 with any appropriate conditions. In that the qualification of motor-operated valves (MOVs) and power-operated valves (POVs) is specified as Tier 1 or Tier 2* provisions in several design certification rules, the inspector should discuss the application of ASME QME-1-2017 with the NRC headquarters technical staff where a licensee intends to implement ASME QME-1-2017.

The NRC staff is conducting Region and vendor inspections that provide information on the functional design and qualification of pumps, valves, and dynamic restraints. The inspector should contact the applicable headquarters staff to discuss Region and vendor inspections that should be reviewed in preparation for performing an inspection using this IP.

The inspector should evaluate the completion of the Qualification Report and Application Report in accordance with ASME Standard QME-1 as referenced in the applicable FSAR to support the functional design and qualification of pumps, valves, and dynamic restraints to confirm the provisions for the functional, dynamic, and environmental qualification have been implemented. In particular, the inspector should discuss with headquarters technical staff the specific aspects of functional, dynamic, or environmental qualification that should be confirmed as part of the review of the QME-1 Qualification Report and Application Report.

The inspector should verify that licensee programs are addressing the functional design and qualification of pumps, valves, and dynamic restraints within the scope of the regulatory treatment of non-safety systems (RTNSS) consistent with the NRC regulations and Commission policy in SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs." For example, SECY-95-132 states that the designer should establish graded requirements for SSCs based on the importance to safety of their functional reliability and availability missions. As of August 17, 2017, the NRC regulations in 10 CFR 50.55a specify that licensees shall assess the operational readiness of pumps, valves, and dynamic restraints within the scope of RTNSS for applicable reactor designs. Application of ASME QME-1-2007 as accepted in Revision 3 to RG 1.100 is one acceptable method for demonstrating the functional design and qualification of RTNSS pumps, valves, and dynamic restraints.

02.02 Preservice and Inservice Testing Program.

The NRC regulations in 10 CFR Parts 50 and 52 include requirements for the preservice testing and inservice testing of nuclear power plant components. As part of the review of the application for a COL under 10 CFR Part 52, the NRC staff evaluated the adequacy of the description provided in the applicable design certification DCD/FSAR and COL FSAR of the PST and IST programs for pumps, valves, and dynamic restraints at a nuclear power plant being licensed under 10 CFR Part 52. Specifically, the NRC staff followed the guidance in SRP Section 3.9.6 in its review of the design certification and COL applications.

The NRC regulations in 10 CFR 50.55a(f)(4)(i) require that the IST program for the initial 10-year IST program interval for a nuclear power plant licensed under 10 CFR Part 52 comply with the ASME OM Code edition and addenda incorporated by reference in the NRC regulations the specified time period (currently 12 months) before fuel load (or the optional Code Cases accepted by the NRC). In RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," the NRC staff discusses the acceptability with certain provisions of specific ASME OM Code Cases that provide alternatives to the IST provisions in the ASME OM Code. The NRC incorporates by reference specific revisions to RG 1.192 in 10 CFR 50.55a such that a licensee may implement an accepted ASME OM Code Case together with any RG 1.192 provisions where the ASME OM Code Case is applicable to the OM Code of record for that licensee's nuclear power plant without the need for the licensee to submit a request for NRC authorization to implement an alternative to the ASME OM Code that satisfies 10 CFR 50.55a(z).

During construction, the licensee will establish and begin implementation of the PST/IST programs for pumps, valves, and dynamic restraints that are described in its COL FSAR (as accepted by the NRC in the COL SER) in conformance with the NRC regulations and the ASME OM Code as incorporated by reference in the NRC regulations. PST/IST activities can include testing, inspection, surveillance and condition monitoring of components within the scope of the PST/IST programs. The licensee will conduct PST activities as part of the IST program to prepare pumps, valves, and dynamic restraints to perform their intended safety functions. The ITAAC for the nuclear power plant, as well as specific COL license conditions, will include PST activities for pumps, valves, and dynamic restraints. The NRC staff provides IST guidance in NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants."

The NRC staff reviewed the PST/IST program descriptions submitted by the COL applicant as part of the licensing process using SRP Section 3.9.6. The NRC inspector will determine whether the PST/IST programs satisfy the PST/IST program description specified in the COL FSAR as accepted in the NRC SER and comply with regulatory requirements, including the appropriate ASME Code edition used in developing the PST/IST programs.

a. Responsibilities

(1) Inspection Requirements

Determine whether the licensee has assigned responsibilities to persons and organizations for:

- (a) Preparation, review, and approval of the PST/IST program and procedures.
- (b) Scheduling of normal and increased frequency testing.
- (c) Performance of testing per approved procedures.
- (d) Performance of post maintenance testing.
- (e) Proper certification and calibration of test instruments.
- (f) Training for those personnel responsible for implementing the PST/IST program and procedures.

(2) Inspection Guidance

Using information from licensee documentation (including the design certification DCD/FSAR and COL FSAR) and applicable NRC SERs, the inspector should determine whether the licensee has assigned responsibilities to persons and organizations for the establishment and implementation of the PST/IST program for pumps, valves, and dynamic restraints that perform a safety function at the nuclear power plant.

b. Program and Procedures

(1) Inspection Requirements

Review the PST/IST program to determine the specification of the edition and addenda of the ASME OM Code and Code Cases to be applied in the PST/IST program consistent with the NRC regulations in 10 CFR 50.55a(f)(4)(i) that require the IST program to comply with the ASME Code edition and addenda (or acceptable ASME OM Code Cases) incorporated by reference in the NRC regulations the specified time period (currently 12 months) before fuel load. Determine whether the COL licensee has submitted, or is planning to submit, a request to apply an earlier edition of the ASME OM Code for the IST program to be implemented during the initial 10-year IST program interval as an alternative to the requirement in 10 CFR 50.55a(f)(4)(i). Determine whether the COL licensee has submitted, or is planning to submit, a request for relief from, or other alternatives to, the ASME OM Code in accordance with 10 CFR

50.55a. Regulatory Issue Summary (RIS) 2012-08, Revision 1, "Developing Inservice Testing and Inservice Inspection Programs under 10 CFR Part 52," dated July 17, 2013, provides information for use by licensees in requesting implementation of an edition or addenda of the ASME OM Code referenced in a design certification or COL application for the initial 10-year IST program.

Review the scope of the PST/IST programs and specified PST/IST program test parameters and test intervals in comparison to the IST program table provided in the design certification DCD/FSAR and COL FSAR as part of the licensing review. As of August 17, 2017, the NRC regulations in 10 CFR 50.55a(f)(4) clarify the scope of the PST and IST programs to be consistent with the scope of the ASME OM Code, and allow safety-related pumps and valves not classified as ASME Code Class 1, 2, or 3 to be addressed as part of an augmented IST program without requesting relief or an alternative where the basis for deviations from the ASME OM Code demonstrates an acceptable level of quality and safety, or implementing the Code provision would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety, where documented and available for NRC review.

Select three to five systems (with focus on risk-significant systems) for detailed review to assess the PST/IST program. These systems may differ from the functional design and qualification inspection performed under Section 02.01 of this IP appendix depending on the inspector's review of the licensee's PST/IST program.

As discussed in Section 02.01 of this IP appendix for the functional design and qualification of pumps, valves, and dynamic restraints, a complete review of the design basis requirements and operating conditions for all safety-related pumps, valves, or dynamic restraints might be determined to be appropriate for a prototype plant of a new nuclear power plant design. In such a situation, the scope of the review of the PST/IST program could be expanded to be consistent with the review of the functional design and qualification program for pumps, valves, and dynamic restraints.

Review the following aspects of the PST/IST program for applicable pumps, valves, and dynamic restraints in the selected systems:

- (a) Determine whether the pumps, valves, and dynamic restraints that perform safety-related functions in the selected systems are included in the PST/IST program.
- (b) Determine whether the PST/IST activities for the pumps, valves, and dynamic restraints of (a) above meet the PST/IST method and frequency requirements in the applicable ASME OM Code as incorporated by reference in the NRC regulations, except where the licensee implements ASME OM Code Cases as accepted in RG 1.192 or the NRC has granted relief or authorized alternatives.
- (c) Determine whether requests for relief or authorization for alternative testing have been submitted to the NRC. Verify that requested alternatives are not implemented in lieu of the Code requirements prior to NRC staff authorization.

- (d) Determine whether the PST/IST program includes applicable SER provisions for requests for relief and alternatives, and assess the adequacy of the implementation of the relief or alternative approach.
- (e) Determine whether the IST program includes applicable SER provisions for justification for deferring IST activities to cold shutdowns or refueling outages.
- (f) Review administrative controls for tracking PST/IST activities.
- (g) Review the PST/IST program, implementing procedures, and records to determine whether reference values and acceptance criteria are identified and are in accordance with Code limits as incorporated by reference in the NRC regulations.
- (h) Review program controls and PST/IST procedures for re-verifying or establishing reference values after component maintenance, replacement, or modification.
- (i) Review controls for post maintenance testing (PMT) to determine whether a component meets the PMT requirements prior to its return to service.
- (j) Review evaluation process for instruments found out of calibration to determine the effect on previous PST/IST results.
- (k) Review application of risk insights in IST program such as those allowed in ASME OM Code Cases as accepted with conditions in RG 1.192.
- (l) Determine whether applicable PST activities specified in the ITAAC and license conditions for the nuclear power plant are addressed for the sampled pumps, valves, and dynamic restraints.

(2) Inspection Guidance

The NRC inspector will review the ASME OM Code as incorporated by reference in 10 CFR 50.55a and Code Cases accepted in RG 1.192 that are applicable to the nuclear power plant being inspected as required in 10 CFR 50.55a(f)(4)(i). The inspector will determine whether the IST program complies with the ASME Code edition and addenda incorporated by reference in the NRC regulations the specified time period (currently 12 months) before fuel load (or acceptable ASME OM Code Cases). The inspector will review the requirements for the IST program based on any granted relief or authorized alternatives to requirements in 10 CFR 50.55a.

Consistent with the determination of the applicable PST/IST program requirements, the NRC inspector will review the plant-specific PST/IST program documentation for consistency with the PST/IST program description provided in the design certification DCD/FSAR and COL FSAR as accepted in the applicable NRC SERs. This will include review of the scope of the PST/IST program and the specified safety functions, surveillance tests, and test parameters and intervals.

The NRC inspector will typically conduct a sampling inspection of the PST/IST program established by the licensee for pumps, valves, and dynamic restraints. The inspector should select three to five safety systems as an initial sample. The selected systems should contain a variety of pump and valve types to the extent possible. The selection process should emphasize plant risk associated with potential failures of pumps, valves and dynamic restraints. A selection of 4 pumps, 10 to 20 valves, and 7 to 15 dynamic restraints is recommended. Where the nuclear power plant includes pyrotechnic-actuated valves (squib valves) with new designs or high safety significance, the inspector should select squib valves as part of the inspection sample. The selection could include pumps and valves in targeted ITAAC. The selected dynamic restraints should be of at least three different load classifications and various degrees of accessibility (easy or difficult accessibility). For more details, see IP 50090 and IP 70370.

The sample of pumps, valves, and dynamic restraints may be expanded based on the inspection findings where concerns are raised regarding the capability of specific components to perform their design-basis safety functions. As discussed above, the scope of the review of the PST/IST program could be expanded to be consistent with the review of the functional design and qualification program for pumps, valves, and dynamic restraints for a prototype plant of a new nuclear power plant design.

The inspector should review the design certification DCD/FSAR, COL FSAR, plant procedures, and other governing documents, as well as NRC SERs, to confirm that the sampled components are addressed within the PST/IST program. The overall programmatic aspects of the development, maintenance, and scheduling of the PST/IST program will be included in a number of administrative procedures. Procedures for implementing the testing will generally be operating procedures or surveillance procedures specific to a component or a group of components (e.g., one procedure may test only a pump, while another may test a pump and a group of valves, or may be limited to a group of valves or dynamic restraints).

The calibration of the instruments will generally be performed onsite, though some instruments (e.g., vibration monitoring) may be calibrated offsite.

The NRC staff is conducting vendor inspections that provide information on the PST and IST recommendations established by the vendors for pumps, valves, and dynamic restraints. The inspector should contact the applicable headquarters staff to discuss vendor inspections that should be reviewed in preparation for performing an inspection using this IP.

The inspector should verify that licensee programs are addressing inservice testing of RTNSS pumps, valves, and dynamic restraints consistent with 10 CFR 50.55a(b)(3)(iii)(D) and the Commission policy in SECY-95-132. For example, SECY-95-132 indicates that non-safety related piping systems with functions that have been identified as being important by the RTNSS process should be designed to accommodate testing of pumps and valves to assure that the components meet their intended functions. SECY-95-132 also states that specific positions on the inservice testing requirements for those components will be determined as part of the staff's review of plant-specific implementation of the RTNSS systems for passive reactor designs. SECY-95-132 states that to the extent practicable, the passive piping systems should be designed to accommodate the applicable Code requirements for quarterly testing of valves. SECY-95-132 specifies that passive system designs should incorporate provisions (1) to permit all critical check valves to be tested for performance, to the extent practicable, in both forward-

and reverse-flow directions, although the demonstration of a non-safety direction need not be as rigorous as the corresponding safety direction test, and (2) to verify movement of each check valve's obturator during inservice testing by observing a direct instrumentation indication of the valve position such as a position indicator or by using nonintrusive methods. SECY-95-132 also states that to the extent practicable, the design of non-safety related piping systems with functions under design-basis conditions that have been identified as being important by the RTNSS process should incorporate provisions to test power-operated valves in the system to assure that the valves meet their intended functions under design-basis conditions. SECY-95-132 indicates that the extent to which recovery from mispositioning will be applied to MOVs in important non-safety related systems will be determined when the staff reviews the implementation of RTNSS systems. The provisions in the ASME OM Code as incorporated by reference in 10 CFR 50.55a or accepted ASME OM Code Cases would be acceptable methods of performing inservice testing of RTNSS pumps, valves, and dynamic restraints. In addition, RIS 2000-03, "Resolution of Generic Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions," discusses attributes for POV periodic verification programs.

Using the NRC regulations, the applicable ASME OM Code and Code Cases, and the NRC guidance in RG 1.192 and NUREG-1482, the inspector should determine if the licensee's actions are adequate. Inspection guidance in specific areas is provided in the following paragraphs:

(a) PST/IST Program Review

(i) The NRC regulations in Appendix A to 10 CFR Part 50 require that nuclear power plant licensees test components important to safety to provide reasonable assurance that those components are capable of performing their safety functions. Appendix B to 10 CFR Part 50 requires that QA requirements be established for testing safety-related components at a nuclear power plant. The ASME OM Code states that the pumps, valves, and dynamic restraints within the scope of the PST/IST program are those that are required to perform a specific function in shutting down a reactor to the safe shutdown condition, in maintaining the safe shutdown condition, in mitigating the consequences of an accident, or in providing overpressure protection. As of August 17, 2017, the NRC regulations in 10 CFR 50.55a(f)(4) clarify that pumps and valves that are within the scope of the ASME OM Code must meet the IST requirements set forth in the ASME OM Code to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations allow safety-related pumps and valves not classified as ASME Code Class 1, 2, or 3 to be addressed as part of an augmented IST program without requesting relief or an alternative where the basis for deviations from the ASME OM Code demonstrates an acceptable level of quality and safety, or implementing the Code provision would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety, where documented and available for NRC review.

(ii) Where a licensee is implementing a Risk-Informed IST program, the NRC inspector should contact the headquarters engineering and PRA staff for guidance in evaluating the specific provisions for that program as accepted in the applicable NRC SER.

(iii) RG 1.192 specifies the acceptability of ASME OM Code Cases and applicable conditions for their use. The inspector should contact the headquarters engineering and PRA staff for assistance in the evaluation of the implementation of OM Code Cases accepted in specific revisions to RG 1.192 as incorporated in 10 CFR 50.55a.

(iv) The implementation of relief requests and requests for authorization of alternatives is discussed in NUREG-1482. Where the NRC staff has granted a relief request, the inspector should confirm the basis for the relief by determining that the design limitation and proposed IST method are as stated by the licensee and approved in the applicable SER. The inspector should determine whether the licensee is implementing the relief or alternative as stated in the request with any provisions imposed in the SER granting the relief or authorizing the request.

(v) The justification for deferral of testing to the cold shutdown or refueling outage conditions needs to adequately state the impracticalities of performing the testing during power operations, or cold shutdown conditions, as appropriate. Guidance is provided in NUREG-1482.

(vi) The administrative controls need to be adequate to ensure PST/IST activities meet the regulatory requirements. Guidance is provided in NUREG-1482.

(vii) Reference values need to be in the PST/IST procedures as they are necessary to perform testing. The acceptance criteria need to be readily available such that a determination of operational readiness can be made in an expeditious manner.

(viii) The procedures for accomplishing maintenance, replacement, or modification need to address PST/IST to establish new or reconfirm previous reference values.

(ix) Post maintenance testing will generally require performance of the IST procedures used for quarterly or cold shutdown or refueling outage testing.

(x) The inspector should discuss potential preconditioning of pumps, valves, and dynamic restraints prior to PST/IST activities with the licensee. Guidance on acceptable and unacceptable preconditioning is provided in several NRC documents, including NRC Inspection Manual Part 9900, Technical Guidance: Maintenance – Preconditioning of Structures, Systems, and Components Before Determining Operability; Information Notice (IN) 97-16, “Preconditioning of Plant Structures, Systems, and Component Before ASME Code Inservice Testing or Technical Surveillance Testing”; and NUREG-1482.

(b) Valve PST/IST

(i) Subsection ISTC, “Inservice Testing of Valves in Light-Water Reactor Nuclear Power Plants,” of the ASME OM Code specifies PST and IST requirements for valves within the scope of the ASME OM Code. The inspector should evaluate the program and procedures developed by the licensee to

implement ASME OM Code, Subsection ISTC, as incorporated by reference in 10 CFR 50.55a. The inspector should compare the valve PST/IST program and procedures with the valve PST/IST program description provided in the COL FSAR as accepted in the NRC SER on the COL application.

The inspector should also evaluate licensee procedures prepared for the use of OM Code Cases approved in RG 1.192, for relief granted by the NRC staff, or for alternatives authorized by the NRC staff, as applicable. The licensee needs to justify any departures from the valve PST/IST program description provided in the COL FSAR.

(ii) The licensee should have a procedure to determine changes in PST/IST parameters for power-operated valves by comparing measurements to either a reference value or the previous test measurement, as applicable to the Code of record. If measurements are compared to reference values, this method of comparison needs to be documented in the program. The inspector should evaluate a sample of the bases for assigning limiting values for acceptance criteria of power operated valves. The inspector should determine whether limiting values will be based on measurements when the valve is in good condition and operating properly. The limiting value needs to represent a reasonable deviation from the reference value and be within the Code allowable. The inspector should determine whether the limiting values exceed any design values. The inspector should determine whether limits are readily achievable during testing and that instructions are provided for actions to take if criteria are exceeded. Rapid acting valves can have a limiting stroke time of 2 seconds (see NUREG-1482 and ISTC-5114(c)).

(iii) The ASME OM Code contains requirements for the verification of valve position indication every 2 years in ISTC-3700, "Position Verification Testing." Beginning in 1985, the NRC staff in NUREG-1482 alerted licensees to ambiguous Code provisions regarding valve position indication. Appendix A to 10 CFR Part 50 states that where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. Operating experience has continued to reveal issues with valve position indication at nuclear power plants. See, for example, NRC IN 2012-14, "Motor-Operated Valve Inoperable Due to Stem-Disc Separation," dated July 24, 2012, and IN 2017-03, "Anchor/Darling Double Disc Gate Valve Wedge Pin and Stem-Disc Separation Failures," dated June 15, 2017. In light of continuing issues with valve position indication, the NRC regulations in 10 CFR 50.55a(b)(3)(xi) require that beginning with implementation of the 2012 Edition of the ASME OM Code, licensees shall supplement paragraph ISTC-3700 to verify that valve operation is accurately indicated by supplementing valve position indicating lights with other indications, such as flow meters or other suitable instrumentation, to provide assurance of obturator movement. The inspector should determine whether the licensee has provided adequate verification that valve position is accurately indicated such that valves are capable of performing their safety functions in accordance with the NRC regulations. See *Federal Register* Notice 82 FR 32934, dated July 18, 2017, for additional information.

(iv) Valves that have a specific leakage limit are designated as “Category A” valves in the OM Code. The leakage test may be of an individual valve or a group of valves. For pressure isolation valves, an individual leakage test is generally required. See NUREG-1482 for additional information.

(v) Guidance for inspection of the MOV, AOV, and squib valve programs is provided in the attachments to this inspection procedure.

(vi) Check valves are tested by ensuring the capability to full stroke to the position(s) required to fulfill the safety function(s) of the valve. The OM Code specifies testing of check valves in both directions. Guidance on testing check valves is provided in NUREG-1482.

(vii) Guidance on manual valves is provided in NUREG-1482.

(viii) Guidance on testing safety and relief valves is provided in NUREG-1482.

(ix) Valve replacement frequency needs to satisfy the applicable IST requirements.

(x) For Main Steam Isolation Valves (MSIVs), the inspector should evaluate the licensee’s procedures for periodic testing and surveillance activities for MSIVs to determine whether degradation in valve performance is occurring over time that might result in the MSIVs being incapable of performing their safety functions. For example, MSIV piston ring material might corrode and interfere with the operation of the valve. The inspector should contact the applicable headquarters staff to discuss operating experience with MSIV performance. See Special Inspection Report 05000400/2012008 (dated July 12, 2012) on the MSIV failure at Shearon Harris Nuclear Power Plant, and IN 2015-13, “Main Steam Isolation Valve Failure Events,” dated December 10, 2015.

(xi) Nozzle check valves represent a new check valve design with different features and performance characteristics than swing check valves. The inspector should determine whether the licensee establishes and implements PST/IST testing and surveillance consistent with the vendor recommendations. Where the ASME OM Code provisions are not sufficient for nozzle check valves, the licensee is responsible for establishing and implementing PST/IST activities that provide reasonable assurance of the operational readiness of nozzle check valves to perform their safety functions. The inspector should evaluate the licensee’s procedures for PST/IST activities that are intended to provide reasonable assurance of the operational readiness of nozzle check valves.

(xii) The inspector should determine whether the licensee has addressed the potential for pressure locking or thermal binding of gate valves within the scope of the program, such as by implementation of Generic Letter (GL) 95-07, “Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves.” Additional inspection guidance is provided in IP 62710, “Power Operated Gate Valve Pressure Locking and Thermal Binding.” The inspector should determine whether the licensee has considered operating experience at current nuclear power plants in avoiding pressure locking and thermal binding of valves. For example, in October 2012, the valve stem in two MSIVs at Vogtle

Unit 1 failed with the cause related to pressure locking or thermal binding. See Licensee Event Report (LER) 2012-005-00, "Main Steam Isolation Valve Failure," dated November 29, 2012, submitted by the Vogtle Unit 1 licensee.

(c) Pump PST/IST

(i) Subsection ISTB, "Inservice Testing of Pumps in Light-Water Reactor Nuclear Power Plants – Pre-2000 Plants," and Subsection ISTF, "Inservice Testing of Pumps in Light-Water Reactor Nuclear Power Plants – Post-2000 Plants," of the ASME OM Code specify PST and IST requirements for pumps within the scope of the ASME OM Code for applicable nuclear power plants. The inspector should evaluate the program and procedures developed by the licensee to implement ASME OM Code, Subsections ISTB and ISTF, as incorporated by reference in 10 CFR 50.55a. The inspector should compare the pump PST/IST program and procedures with the pump PST/IST program description provided in the COL FSAR as accepted in the NRC SER on the COL application. The inspector should also evaluate procedures prepared for the use of OM Code Cases approved in RG 1.192, for relief granted by the NRC staff, or for alternatives authorized by the NRC staff, as applicable. The licensee needs to justify any departures from the pump PST/IST program description provided in the COL FSAR.

(ii) PST/IST procedures need to include all steps necessary to comply with regulatory requirements and to ensure repeatable test conditions. Acceptance criteria need to be included in the procedure with instructions for actions to take if the criteria are exceeded.

(iii) The OM Code has established comprehensive pump testing provisions to be implemented as applicable to the Code of record. See NUREG-1482 for guidance on comprehensive pump testing.

(iv) Pump test instrument ranges and calibration accuracies must meet the PST/IST requirements. Guidance is provided in NUREG-1482.

(v) The PST/IST procedures need to specify that a pump be declared inoperable when the required action limits are exceeded. If a problem with instrumentation is suspected, instruments may be re-calibrated and the test re-run.

(vi) For systems with constantly changing demand, the licensee might establish multiple sets of reference values. Guidance on the use of pump curves is provided in NUREG-1482.

(d) Dynamic Restraint PST/IST

(i) Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants," of the ASME OM Code specifies PST and IST requirements for dynamic restraints within the scope of the ASME OM Code. The inspector should evaluate the program and procedures developed by the licensee to implement ASME OM Code, Subsection ISTD, as incorporated by reference in 10 CFR 50.55a. The inspector should compare the dynamic restraint PST/IST program and

procedures with the dynamic restraint PST/IST program description provided in the COL FSAR as accepted in the NRC SER on the COL application. The inspector should also evaluate procedures prepared for the use of OM Code Cases approved in RG 1.192, for relief granted by the NRC staff, or for alternatives authorized by the NRC staff, as applicable. The licensee needs to justify any departures from the dynamic restraint PST/IST program description provided in the COL FSAR.

(ii) PST/IST procedures need to include all steps necessary to comply with regulatory requirements of visual inspection of snubbers, to verify any physical damage, leakage or corrosion, or degradation that may interfere in proper function of the snubber. The COL FSAR (including COL Information Item responses) and NRC SER describe provisions for the snubber PST/IST program.

(iii) The licensee's procedures should specify snubber inservice testing that verifies the following:

Activation is within the specified range of velocity or acceleration in tension or compression.

Release rate, when applicable, is within the specified range in tension and in compression. For units specifically required not to displace under continuous load, ability of the snubber to withstand load without displacement.

For mechanical snubbers, drag force is within specified limits in tension and in compression.

For hydraulic snubbers, if required to verify proper assembly, drag force is within specified limits in tension and in compression.

(iv) Guidance for snubber PST/IST is provided in IP 50090 and IP 70370.

(v) RIS 2010-06, "Inservice Inspection and Testing Requirements for Dynamic Restraints," provides guidance regarding the requirements for inservice inspection and testing of snubbers under 10 CFR 50.55a(g) and 10 CFR 50.55a(b)(3)(v).

73758-APPA.03 RESOURCE ESTIMATE

Completion of Appendix A to this IP for the review of the functional design, qualification, and PST/IST programs for pumps, valves, and dynamic restraints is expected to take 120 hours of direct inspection effort. This resource estimate might increase if the inspection scope is expanded based on the results of the inspection sample or operating experience at other nuclear power plants. In addition, a complete review of the design basis requirements and operating conditions for all safety-related pumps, valves, or dynamic restraints might be appropriate for the prototype plant of a new nuclear power plant design. As a result, the inspection resource estimate might increase for the completion of Appendix A to this IP. The inspector should request that design basis information for pumps, valves, and dynamic restraints be provided by the licensee in advance of the inspection to allow review of the safety functions and operating conditions prior to the inspection.

73758-APPA.04 PROCEDURE COMPLETION

The initial inspection using Appendix A to this IP should be performed early in the construction process prior to the installation of pumps, valves, and dynamic restraints to confirm the establishment of the functional design and qualification process and PST/IST programs, consistent with the NRC regulations and program descriptions in the design certification DCD/FSAR and COL FSAR as accepted in the applicable NRC SERs.

APPENDIX B - Implementation of Functional Design and Qualification Program for Pumps, Valves, and Dynamic Restraints

73758-APPB.01 INSPECTION OBJECTIVE

The objective of this inspection is to evaluate the implementation of the functional design and qualification program for pumps, valves, and dynamic restraints during construction of a nuclear power plant licensed under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52.

73758-APPB.002 INSPECTION REQUIREMENTS AND GUIDANCE

02.01 Inspection Requirements.

Review the documentation supporting the functional design and qualification of sampled pumps, valves, and dynamic restraints in comparison to design-basis requirements for those components. Evaluate the activities to determine whether the licensee has verified the adequacy of the functional design and qualification of pumps, valves, and dynamic restraints to perform their intended safety functions.

02.02 Inspection Guidance.

The inspector should review the implementation of the licensee's functional design and qualification methodology for consistency with **American Society of Mechanical Engineers (ASME) QME-1-2007** as accepted in Regulatory Guide (RG) 1.100, "**Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,**" (Revision 3). For some new component designs, the licensee will need to adjust the specific provisions in ASME QME-1-2007 to demonstrate the functional design and qualification of critical parts and their performance for those new designs. For example, squib valves have design features beyond those specifically addressed in ASME QME-1-2007. Further, the licensee's functional design and qualification methodology for nozzle check valves may need to supplement the provisions in ASME QME-1-2007 because of the new design features of nozzle check valves compared to swing check valves.

ASME has published ASME QME-1-2017 to provide updated qualification provisions for pumps, valves, and dynamic restraints. The NRC staff is preparing proposed Revision 4 to RG 1.100 to address the acceptance of ASME QME-1-2017 with any appropriate conditions. In that the qualification of motor-operated valves (MOVs) and power-operated valves is specified as Tier 1 or Tier 2* provisions in several design certification rules, the inspector should discuss the application of ASME QME-1-2017 with the NRC headquarters technical staff where a licensee intends to implement ASME QME-1-2017.

The inspector should follow the general guidance for the selection of a sample of pumps, valves, and dynamic restraints discussed in Appendix A to this IP. The inspector should select some of the same components from the program review inspection to evaluate the follow-through of the functional design and qualification process for specific components. The inspector should also select different components to evaluate the broad implementation of the functional design and qualification process.

The inspector should determine whether the documentation supporting the functional design and qualification of sampled pumps, valves, and dynamic restraints is consistent with the design-basis requirements for those components. As part of the installation of pumps, valves, and dynamic restraints at a nuclear power plant, the licensee will conduct activities to verify their functional design and qualification. The inspector should confirm that the licensee verifies the functional design and qualification of pumps, valves, and dynamic restraints to perform their intended safety functions prior to being relied on to perform a safety function at the nuclear power plant.

When evaluating the functional design and qualification of valve assemblies, the inspector should consider the following items as applicable:

a. Qualification Plan

- (1) Has a Qualification Plan been prepared that satisfies ASME QME-1-2007 as accepted in Revision 3 to RG 1.100?
- (2) Does the plan specify the objectives of the valve qualification testing, including manufacturer, type, size, and rating, and its design-basis conditions, and the planned extrapolation of the qualification to other valve sizes and ratings, and range of performance conditions and allowable leakage rate?
- (3) Does the plan specify the objectives for qualification of the valve actuator, including manufacturer, type, size, and rating, and its design-basis conditions, and the planned extrapolation of the qualification to other actuator sizes and ratings, and range of performance conditions?
- (4) Does the plan implement ASME QME-1-2007 in accordance with RG 1.100 (Revision 3)?
- (5) Does the plan address applicable prerequisite qualification testing (such as seismic, temperature aging, and radiation aging) for metallic and nonmetallic parts?
- (6) Does the plan encompass the complete range of operating conditions and stroke time for the test valve assembly from normal operation to design-basis conditions?
- (7) Does the plan specify the requirements for extrapolation of the functional qualification to other valve assemblies or design-basis conditions that satisfy ASME QME-1-2007?
- (8) Does the plan specify the valve and actuator orientation for qualification?
- (9) Does the plan address the appropriate test fixture consistent with plant installation, or provide justification for differences between the test fixture and plant installation?
- (10) Does the plan address applicable valve packing requirements and assumptions?

- (11) Does the plan specify the acceptance criteria to be established for performance of the qualification testing?
 - (12) Does the plan specify the component initial setup, limits, maintenance or adjustments that are acceptable during performance of the qualification testing?
 - (13) Does the plan specify the application of a QA program that satisfies 10 CFR Part 50, Appendix B?
 - (14) Does the plan specify the required qualifications of test personnel?
 - (15) Does the plan specify the determination of activities to ensure that production valve assemblies will perform consistent with the qualified valve assembly?
 - (16) Does the plan specify the preparation of a Functional Qualification Report and Applicability Report in accordance with ASME QME-1-2007 sufficient to support close-out of applicable ITAAC for the qualified valve assemblies and their production valve assemblies?
- b. Observation of testing
- (1) Does the qualification testing satisfy ASME QME-1-2007 as accepted in RG 1.100 (Revision 3)?
 - (2) Does the test setup satisfy the requirements and assumptions in the Qualification Plan?
 - (3) Does the test setup provide for testing over the full range of operating conditions from normal operation to design-basis conditions with the specified stroke time and allowable leakage rate for the test valve assembly including justification documented for any differences?
 - (4) Does the valve and actuator orientation satisfy the plan requirements?
 - (5) Have the valve assembly internals been inspected with appropriate measurements for material surface condition, dimensions, edge radii, and clearances to satisfy qualification assumptions and to perform the qualification calculations and extrapolation methodology?
 - (6) Does the instrumentation appropriately provide sufficient information (such as system upstream and downstream pressure, differential pressure, flow rate, fluid temperature, stem thrust, stem torque, valve packing loads, valve leakage rate, stem friction coefficient, and actuator performance and output, as applicable) from multiple sources?
 - (7) Is the instrumentation calibrated and benchmarked to provide accurate measurements of pressure, flow rate, fluid temperature, stem thrust, stem torque, valve packing load, valve leakage rate, and actuator performance and output (as applicable) with specific uncertainty values?

- (8) Are the valve internal surfaces preconditioned immediately prior to qualification testing to establish a stable valve factor without exposure to air prior to testing?
 - (9) Are lubricant types and applications for the valve and actuator documented?
 - (10) Are post-testing inspections conducted of the valve assembly and its internals with appropriate measurements for surface condition, dimensions, edge radii, and clearances to evaluate qualification acceptance criteria for valve assembly performance and capability?
 - (11) Are deficiencies in the test instrumentation, valve assembly performance, or valve assembly parts appropriately documented and evaluated through a 10 CFR Part 50, Appendix B corrective action program?
- c. Evaluation of test data
- (1) Does the test data evaluation satisfy ASME QME-1-2007 as accepted in RG 1.100 (Revision 3)?
 - (2) Does the test data evaluation apply an industry-wide methodology (such as the Electric Power Research Institute (EPRI) MOV Application Guide or EPRI MOV Performance Prediction Methodology)?
 - (3) Are all applicable uncertainties included in the calculations (such as differential pressure and flow measurements, and torque and thrust measurements)?
 - (4) Are deficiencies in the test data or the results of the evaluation identified and addressed through a 10 CFR Part 50, Appendix B corrective action program?
 - (5) Are the results of the qualification testing specified for applicable valve assemblies, fluid conditions, differential pressure and flow rate, valve thrust and torque operating requirements, valve stroke time, valve packing load, valve leakage rate, valve factor or friction coefficient, stem friction coefficient, and actuator performance and output capability (as applicable) with applicable ranges and uncertainties?
 - (6) Are limitations of the qualification results regarding the scope of the qualified valve assemblies, their applications, fluid conditions, stroke time, packing load, or leakage rate adequately addressed and documented?
- d. Extrapolation of test data to other valve assemblies and applications
- (1) Does the extrapolation of the test data satisfy ASME QME-1-2007 as accepted in RG 1.100 (Revision 3)?
 - (2) Does the extrapolation apply an industry-wide methodology (such as the EPRI MOV Performance Prediction Methodology)?
 - (3) Are the specific valve assemblies, applications, fluid conditions, stroke time, and leakage rate within the scope of qualification extrapolation identified in the report?

- (4) Are the requirements for demonstrating the extrapolation of qualification to other valve assemblies and their applications consistent with ASME QME-1-2007 specified in the report?
- e. Post-qualification and post-installation requirements
- (1) Have applicable post-qualification requirements determined by the qualification testing (such as valve and actuator orientation, and internal valve assembly inspection and measurements) been identified and documented in the qualification report?
 - (2) Have post-installation testing requirements been established to demonstrate that production valve assemblies perform consistent with the qualified valve assembly in accordance with ASME QME-1-2007 as accepted in RG 1.100 (Revision 3)?
- f. Documentation
- (1) Does the Functional Qualification Report satisfy ASME QME-1-2007 as accepted in RG 1.100 (Revision 3)?
 - (2) Does the Functional Qualification Report describe the Qualification Plan, test performance, data evaluation, qualification scope, extrapolation, post-qualification and post-installation testing requirements, and corrective actions?
 - (3) Is the scope of the qualification (such as valve type, size and rating; valve orientation; blowdown flow; pump flow; fluid conditions; differential pressure; flow rate; temperature; valve thrust and torque operating requirements; stroke time; packing load; leakage rate; valve factor/friction coefficient, and actuator type, size, rating, and output, as applicable) specified in the report?
 - (4) Has the Functional Qualification Report been certified to be correct and complete, and to be in compliance with ASME QME-1-2007, by a registered professional engineer representing the organization responsible for the functional qualification?
 - (5) Does the Functional Qualification Report specify the preparation of an Applicability Report to demonstrate the suitability of any qualified valve assembly and associated production valve assemblies to meet the requirements of a specific application in accordance with ASME QME-1-2007?
 - (6) Are the Functional Qualification Report and the Applicability Report (when prepared) sufficient to support close-out of the ITAAC for the test valve assembly and valve assemblies qualified by extrapolation, and their production valve assemblies and applications?

The inspector should consider these items as applicable to the functional design and qualification of pumps and dynamic restraints sampled during the inspection. For example, the NRC staff accepted the functional qualification of pumps in ASME QME-1-2007 in Revision 3 to RG 1.100. In addition, the NRC staff provides guidance on pump qualification in RG 1.82 (Revision 4, March 2012), "Water Sources for Long-Term Recirculation Cooling Following a

Loss-of-Coolant Accident.” The inspector should also confirm the qualification of the pump seal material for its environmental conditions.

The inspector should review the licensee’s process for verification and validation of the engineering software used in safety-related applications for pumps, valves, and dynamic restraints. Where the licensee has not procured the engineering software as safety-related in accordance with Appendix B to 10 CFR Part 50, the inspector should confirm that the licensee’s procedures verify and validate the engineering software for its specific application. In particular, the licensee needs to establish adequate documented controls for the dedication of commercially procured engineering software in order to identify appropriate acceptance methods and critical characteristics for a commercially procured version of the engineering software. For example, the documented controls need to specify the manufacturer and version of the engineering software for verification in the commercial grade dedication instruction. Further, the documented controls need to specify acceptance criteria or methods to ensure an appropriate selection of an example problem with similar geometrical properties and boundary conditions to validate the use of the software for a specific safety-related application. Where applicable, the documented controls need to provide necessary guidance to ensure that finite element modeling results demonstrate convergence, such as through the use of more precise mesh size to arrive at a unique solution. The documented controls need to specify that the licensee collect and evaluate notices or errors issued by the software supplier and capture them in a nonconformance process to determine if they have an adverse effect on the versions of the software used in safety-related applications.

The licensee’s procedures should (1) include acceptance criteria for the comparison of the software results to the target results, (2) discuss the basis for the acceptability of the software in comparison to acceptance criteria, and (3) specify the bias and uncertainty values that would be included in the engineering calculations based on the validation of the software.

The NRC inspector should determine whether the licensee’s process, including oversight of its contractors, for verifying and validating the adequacy of computer software used in engineering applications conforms to the requirements in 10 CFR Part 50, Appendix B.

The NRC inspector should evaluate the process established by the licensee to control changes to the design for pumps, valves, and dynamic restraints. The licensee procedures should specify requirements for changes to the design of plant components, and address resolution of errors identified in licensing documents submitted to the NRC. The design change process needs to provide assurance that pumps, valves, and dynamic restraints receive adequate functional design and qualification where design changes are made to those components. The inspector should discuss the design change process with licensee engineers and review a sample of design change packages. The licensee’s design change process should involve personnel across multiple disciplines to address cross-cutting issues for design changes. The inspectors should determine whether the engineering design change process satisfies the requirements in 10 CFR Part 50, Appendix B.

The NRC inspector should verify that functional design and qualification of pumps, valves, and dynamic restraints within the scope of the regulatory treatment of non-safety systems (RTNSS) are being implemented consistent with 10 CFR 50.55a(b)(3)(iii)(D) and the Commission policy in SECY-95-132. For example, SECY-95-132 states that the designer should establish graded requirements for SSCs based on the importance to safety of their functional reliability and availability missions. Application of ASME QME-1-2007 as accepted in Revision 3 to RG 1.100

is one acceptable method for demonstrating the functional design and qualification of RTNSS pumps, valves, and dynamic restraints.

The inspector may obtain information on the acceptability of the functional design and qualification of pumps, valves, and dynamic restraints from Region inspections of equipment qualification and ITAAC completion, and NRC Vendor Branch inspections of equipment manufacturers. The inspector should discuss inspections related to pumps, valves, and snubbers with the applicable Region staff and NRC headquarters technical staff.

73758-APPB.03 RESOURCE ESTIMATE

Completion of Appendix B to this IP for the implementation of the functional design and qualification of pumps, valves, and dynamic restraints is expected to take 120 hours of direct inspection effort on average for each inspection at the site using the guidance in this IP. This resource estimate might increase if the inspection scope is expanded based on the results of the inspection sample or operating experience at other nuclear power plants.

73758-APPB.04 PROCEDURE COMPLETION

The inspection using Appendix B to this IP should be performed prior to or during initial installation of pumps, valves, and dynamic restraints to confirm the implementation of the functional design and qualification process before completion of installation activities.

APPENDIX C - Implementation of PST/IST Program for Pumps, Valves, and Dynamic Restraints

73758-APPC.01 INSPECTION OBJECTIVE

The objective of this inspection is to evaluate the implementation of preservice testing (PST) and inservice testing (IST) programs for pumps, valves, and dynamic restraints during construction of a nuclear power plant licensed under 10 CFR Part 52.

73758-APPC.02 INSPECTION REQUIREMENTS AND GUIDANCE

02.01 Inspection Requirements.

The following inspection requirements will be performed consistent with the status of the construction activities when the inspection is conducted. Where specific inspection requirements cannot be verified because of the status of plant activities, the inspector should verify that the licensee has established a process to implement those PST/IST activities when appropriate.

Review ongoing PST/IST activities for components in the selected systems as follows:

- a. Observe and evaluate PST/IST activities conducted for pumps, valves, and dynamic restraints in the PST/IST program, especially in the selected systems.
- b. Determine whether the instruments used for the PST/IST activities meet the specified range and calibration accuracies and that the calibration is current.
- c. Evaluate PST/IST results and corrective actions.

Evaluate PST/IST results for a sample of pumps, valves, and dynamic restraints in the selected systems as applicable during the pre-operational stage as follows:

- a. Review one year of test data, if available, for selected pumps, valves, and dynamic restraints.
- b. Determine whether applicable technical specification ACTION statements and applicable reporting requirements are satisfied when components are declared inoperable as a result of PST/IST activities.
- c. Review the method of test data comparison to previous test activities and actions taken on components indicating a degrading condition or a repetitive problem.
- d. Review the documented results of engineering evaluations performed over the previous 2 years, where possible, for components that did not satisfy the test acceptance criteria at any time during that interval, particularly root cause and apparent cause analysis of the problem and the bases for returning the components to an acceptable status.

- e. Review administrative controls for design modifications or replacement of pumps, valves, and dynamic restraints to determine whether PST/IST program requirements are reviewed for applicability when changes are made to plant procedures, systems, or components, where appropriate.
- f. Determine whether completed PST/IST documents require appropriate review and are maintained as quality controlled records.

Evaluate the following areas for PST/IST activities of a sample of at 10 to 20 valves in the selected systems:

- a. Evaluate the PST/IST method, acceptance criteria, and corrective action.
- b. Determine whether valves with remote position indication, including passive and manual valves, are subject to position indication verification in accordance with regulatory requirements.
- c. Evaluate leak rate testing of Category A valves of the OM Code.
- d. Evaluate the implementation of the programs for motor-operated valves (MOVs), air-operated valves (AOVs), and pyrotechnic-actuated (squib) valves described in the attachments to this IP.
- e. Evaluate the adequacy of check valve PST/IST activities, including the use of non-intrusive testing techniques, and disassembly and inspection, where applicable.
- f. Determine whether manual valves in the program are periodically exercised in accordance with regulatory requirements.
- g. Review the set pressure testing for safety and relief valves.
- h. Review the adequacy of non-reclosing pressure relief device (rupture disc) testing/replacement.

Evaluate the following areas for PST/IST activities of a sample of at least four pumps in the selected systems:

- a. Review pump PST/IST methods, acceptance criteria, and corrective action.
- b. Review pump PST/IST for the selected systems for compliance with the ASME OM Code provisions as incorporated by reference in the NRC regulations.
- c. Determine whether the range and calibration accuracies of PST/IST instruments meet Code requirements.
- d. Determine whether pumps are declared inoperable in completed test procedures when test results are in the "Required Action Range" or that the test frequency is doubled when the test results are in the "Alert Range."

- e. Determine whether PST/IST activities are performed at established reference values.

Evaluate the PST/IST activities of a sample of at least 7 to 15 dynamic restraints based on the ASME OM Code Subsection ISTD as incorporated by reference in 10 CFR 50.55a, and the guidance in IP 50090 and IP 70370.

02.02 Inspection Guidance.

The selection of a sample of pumps, valves, and dynamic restraints should emphasize plant risk (associated with potential failures of pumps, valves and dynamic restraints), maintenance, identified programmatic weaknesses, and PST/IST activities scheduled for performance during the inspection. Where the nuclear power plant includes squib valves with new designs or high safety significance, the inspector should select squib valves as part of the inspection sample. Where possible, the inspector should maximize the inspection effectiveness by evaluating areas involving multiple inspection requirements as follows:

System Risk. The results of an individual plant evaluation or a PRA report might reveal insights on important systems and components. The inspector should contact the NRC headquarters PRA staff during preparation for this inspection to obtain information on risk insights and issues related to new nuclear power plants in general or the plant to be inspected in particular. Plant PST/IST program correspondence may identify programmatic weaknesses for particular systems or components.

System Maintenance. Pumps or valves with a high incidence of corrective maintenance are good candidates for selection. The inspector might identify these components through discussions with the Resident Inspector, or plant maintenance or operations personnel; by a review of previous inspection reports; or through a search of licensee event reports (LERs) or the operating experience database.

The inspector should evaluate ongoing PST/IST activities and consider selecting systems with PST/IST activities scheduled during the inspection period. The attachments to this inspection procedure provide guidance for the inspection of PST/IST activities for MOVs, AOVs, and squib valves. PST/IST activities need to satisfy the applicable ASME OM Code provisions as incorporated by reference in the NRC regulations, OM Code Cases as accepted in RG 1.192, or authorized alternatives or granted relief requests. Instruments used for PST/IST activities need to meet the accuracy and range requirements and be within calibration.

The inspector should review results from completed PST/IST activities for pumps, valves, and dynamic restraints performed during the previous year. For example, a component will need to be declared inoperable in a timely manner and appropriate actions taken for test results where limiting values are not met. For components addressed by plant technical specifications that, if declared inoperable, would result in entering an ACTION statement, the inspector should determine whether appropriate information is provided in the test plans or test records, such that those responsible for the test can make a timely determination whether the data meet the acceptance criteria and the component is operable. The inspector should determine if the licensee is complying with applicable reporting requirements.

The inspector should evaluate the licensee's procedures for PST/IST data trending, any available trending activities, and actions taken for components found to be degraded or that require frequent corrective maintenance. For these components, the inspector should determine if an engineering evaluation was performed that adequately addressed the root cause. The inspector should assess the licensee's actions if the components represent a generic class of components at the plant or if the mode of degradation is likely to affect other components in the system. The inspector should review any engineering evaluations which were performed to return a component to operable status in lieu of other corrective actions.

The inspector should review at least one example (if available) of an instrument that was found to be out-of-calibration during PST or IST activities. The inspector should evaluate the acceptability of the licensee's corrective actions for this deficiency. The inspector should review the engineering evaluations that were performed to address the impact of the use of an out-of-calibration instrument on the operability of affected components.

The inspector should determine whether the licensee is implementing vendor recommendations regarding testing, preventive maintenance, and post-maintenance testing for pumps, valves, and dynamic restraints. Where the licensee is not implementing vendor recommendations, the licensee should provide justification for its approach. See NRC Information Notice (IN) 2012-06, "Ineffective Use of Vendor Technical Recommendations," April 24, 2012.

As part of the modification or replacement process during the pre-operational phase, the inspector should determine whether the licensee's controls consider the effect of the change on the required PST/IST provisions.

The inspector should determine whether the PST/IST documents are reviewed by the appropriate supervisor responsible for assessing operational readiness of components. The inspector should review the procedures and available database for trending of component performance.

The inspector should verify that licensee programs are incorporating the inservice testing of pumps, valves, and dynamic restraints within the scope of regulatory treatment of non-safety systems (RTNSS) consistent with 10 CFR 50.55a(3)(iii)(D) and the Commission policy in SECY-95-132. Appendix A to this IP lists Commission positions specified in SECY-95-132 for inservice testing of RTNSS equipment. The provisions in the ASME OM Code as incorporated by reference in 10 CFR 50.55a or accepted ASME OM Code Cases would be acceptable methods of performing inservice testing of RTNSS pumps, valves, and dynamic restraints. In addition, Regulatory Issue Summary 2000-03, "Resolution of Generic Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions," discusses attributes for successful periodic verification programs for power-operated valves.

73758-APPC.03 RESOURCE ESTIMATE

Completion of Appendix C to this IP for the implementation of the PST/IST programs for pumps, valves, and dynamic restraints is expected to take 120 hours of direct inspection effort. This resource estimate might increase if the inspection scope is expanded based on the results of the inspection sample or operating experience at other nuclear power plants.

73758-APPC.04 PROCEDURE COMPLETION

The inspection using Appendix C of this IP should be performed during implementation of the PST/IST programs during construction of the nuclear power plant.

APPENDIX D - Close-out Inspection for Functional Design, Qualification, and
PST/IST Programs for Pumps, Valves, and Dynamic Restraints in Preparation
for Plant Startup

73758-APPD.01 INSPECTION OBJECTIVE

The objective of this inspection is to close-out the NRC construction inspection activities for the functional design and qualification process and preservice testing (PST) and inservice testing (IST) programs for pumps, valves, and dynamic restraints in preparation for plant startup during construction of a nuclear power plant licensed under 10 CFR Part 52.

73758-APPD.02 INSPECTION REQUIREMENTS AND GUIDANCE

02.01 Functional Design and Qualification.

a. Inspection Requirements

Review the documentation supporting the functional design and qualification of safety-related pumps, valves, and dynamic restraints in comparison to their design-basis requirements. Evaluate the activities to determine whether the licensee has verified the functional design and qualification of pumps, valves, and dynamic restraints to perform their intended safety functions. Evaluate the activities to transition the construction programs related to functional design and qualification for pumps, valves, and dynamic restraints with safety functions to operational programs in preparation for fuel loading or being placed in service, as applicable.

b. Inspection Guidance

The inspector should determine whether the licensee has completed the functional design and qualification of pumps, valves, and dynamic restraints to perform their safety functions. Plant documentation should be available that demonstrates the completion of the functional design and qualification process for all safety-related pumps, valves, and dynamic restraints. The inspector should confirm that plant documentation supports the completion of ITAAC for the functional design and qualification of safety-related pumps, valves, and dynamic restraints by performing a detailed review of the documentation for a sample of components. As construction nears completion, the licensee will implement activities to transition the control of plant components to operational programs. The inspector should evaluate the activities to transition the construction programs for functional design and qualification for pumps, valves, and dynamic restraints with safety functions to operational programs in preparation for fuel loading or being placed in service, as applicable.

The inspector should discuss information on the acceptability of the functional design and qualification of pumps, valves, and dynamic restraints from Region inspections of equipment qualification and ITAAC completion, and NRC Vendor Branch inspections of equipment manufacturers with the applicable Region and headquarters technical staff.

02.02 PST/IST Program.

a. Inspection Requirements

Determine whether the licensee has established and implemented the PST/IST programs to support the startup of the nuclear power plant licensed under 10 CFR Part 52.

b. Inspection Guidance

The inspector should determine whether the PST/IST activities satisfy the applicable ASME OM Code provisions as incorporated by reference in the NRC regulations, ASME OM Code Cases as accepted in RG 1.192, and authorized alternatives or granted relief requests. The guidance under the PST/IST program implementation portion of this inspection procedure should be followed to sample the completion of the PST/IST activities.

The inspector should review the results of the completed PST/IST activities for pumps, valves, and dynamic restraints. All components within the scope of the PST/IST program need to have their operational readiness verified to perform their safety functions. The inspector should confirm that PST activities specified in the ITAAC for the sampled components have been satisfied. The inspector should confirm that the licensee is complying with applicable reporting requirements.

The inspector should evaluate the licensee's procedures for PST/IST data trending, any available trending activities, and actions taken for components found to be degraded or that require frequent corrective maintenance to determine whether safety-related components within the scope of the program have been addressed in accordance with NRC regulatory requirements.

The inspector should determine whether the licensee has incorporated appropriate PST/IST changes based on any modifications or replacements during the pre-operational phase.

The inspector should review any remaining issues regarding PST/IST activities that need to be addressed following fuel load. The inspector should contact NRC headquarters staff to discuss any remaining issues regarding PST/IST activities that have not been completed.

The PST/IST documents need to demonstrate operational readiness of the components within the scope of the PST/IST program in preparation for plant startup.

The inspector should verify that licensee programs have completed the functional design and qualification, and have implemented inservice testing activities for pumps, valves, and dynamic restraints within the scope of the regulatory treatment of non-safety systems (RTNSS) consistent with 10 CFR 50.55a(b)(3)(iii)(D) and the Commission policy in SECY-95-132.

As construction nears completion, the licensee will implement activities to transition the control of plant components to operational programs. The inspector should evaluate

the activities to transition the PST and IST programs for pumps, valves, and dynamic restraints to programs to be maintained during plant operation.

73758-APPD.03 RESOURCE ESTIMATE

Completion of Appendix D to this IP for the close-out review of the functional design, qualification, and PST/IST programs for pumps, valves, and dynamic restraints in preparation for plant startup is expected to take 120 hours of direct inspection effort. This resource estimate might increase if the inspection scope is expanded based on the results of the inspection sample or operating experience at other nuclear power plants.

73758-APPD.04 PROCEDURE COMPLETION

The inspection using Appendix D to this IP should be performed 6 months before planned fuel loading to confirm completion of the functional design and qualification process, and full implementation of the PST/IST programs, that demonstrate the design-basis capability and operational readiness of pumps, valves, and dynamic restraints to perform their safety functions. The results of the close-out inspection will be used to support the NRC finding on the implementation of operational programs consistent with the schedule for reaching a finding on ITAAC completion in accordance with 10 CFR 52.103(g). This close-out inspection would identify any remaining follow-up actions to be addressed following fuel load. In preparing the close-out inspection report, the inspector should document the follow-up actions for transition to the NRC operations inspection staff.

ATTACHMENT 1
MOTOR-OPERATED VALVES

73758-ATT1-01 INSPECTION OBJECTIVE

The objective of this attachment to Inspection Procedure (IP) 73758 is to provide guidance for the evaluation of the development and implementation of the program at a nuclear power plant under construction in accordance with Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," in Title 10 of the *Code of Federal Regulations* (10 CFR Part 52) to satisfy the regulatory requirements that motor-operated valves (MOVs) are capable of performing their safety functions over the full range of conditions from normal operation to design-basis accident conditions.

73758-ATT1-02 INSPECTION REQUIREMENTS

02.01 MOV Selection

Select a sample (5 to 10 MOVs) of risk-significant MOVs from 3 to 5 safety systems. The selection of MOVs should include consideration of various valve sizes, types, and manufacturers. The inspector should address a wide range of MOVs in the sample. The MOV sample may be expanded based on the inspection findings where concerns are raised regarding the capability of specific MOVs to perform their design-basis safety functions. For the prototype plant of a new nuclear power plant design, a complete review of the design basis requirements and operating conditions for all safety-related MOVs might be determined to be appropriate.

02.02 MOV Program Scope

Determine whether the scope of the MOV program is consistent with the NRC regulations. Review MOV program scope changes since the completion of any previous NRC review of the MOV program to determine that the appropriate safety-related MOVs are included in the program.

02.03 Design Calculations

Review design documents and calculations for MOV functional requirements under normal, abnormal, and accident conditions; motor and actuator sizing; methods for selecting, setting, and adjusting MOV switch settings; and modifications to the system or valves that could affect the MOV's capability in the as-modified configuration.

02.04 Design-Basis Verification, Preservice Testing and Inservice Testing

Review test documents for adequacy of functional design-basis, preservice testing (PST) and inservice testing (IST) test procedures, test equipment, training of test personnel, acceptance criteria, and test results. If the inspection schedule permits, observe actual testing of MOVs.

02.05 MOV Trending

Review available MOV trend reports, failure analyses, corrective actions, nonconformance reports, or other plant documents that may indicate that an MOV is not properly sized, has improper switch settings, or is not properly maintained.

02.06 Preventive Maintenance

Review MOV preventive maintenance to determine whether it is appropriate for the frequency of operation, working environment, and operational experience.

02.07 Corrective Actions

Determine whether the licensee is periodically reviewing data on MOV failures and the effectiveness of the corrective actions.

02.08 Post-Maintenance Testing

Review a sample of MOV maintenance packages and determine whether the post-maintenance tests and results demonstrate that the MOVs are capable of performing their design functions.

02.09 Operating Experience

Review the adequacy of licensee's processing and control of operating experience information and vendor notifications.

02.10 Periodic Verification

Evaluate development and implementation of the periodic verification program for MOV design-basis capability, including review of MOV periodic verification test results, both static and dynamic. Determine whether information from these tests is incorporated into the design and setup calculations for safety-related MOVs.

02.11 Program Changes

Review changes made in programs affecting safety-related MOVs since previous NRC reviews or inspections.

73758-ATT1-03 INSPECTION GUIDANCE

General Guidance

The NRC regulations in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," require that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Appendix A to 10 CFR Part 50 states that where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. Appendix A

to 10 CFR Part 50 also requires that a quality assurance (QA) program be established and implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 specifies criteria for the QA program to provide adequate confidence in the capability of safety-related SSCs to perform their design-basis functions.

In 10 CFR 50.55a, the NRC regulations incorporate by reference the American Society of Mechanical Engineers (ASME) *Operation and Maintenance of Nuclear Power Plants*, Division 1, OM Code: Section IST (commonly referred to as the OM Code) for implementation of an IST program for pumps, valves, and dynamic restraints used in nuclear power plants. The ASME OM Code (1995 Edition through 2006 Addenda) specified the performance of stroke-time testing of MOVs on a quarterly frequency as part of the IST program. Based on MOV operating experience and research results, the NRC determined that the ASME OM provision for quarterly stroke-time testing was inadequate to provide reasonable assurance of the operational readiness of MOVs to perform their safety functions. Therefore, the NRC regulations in 10 CFR 50.55a(b)(3)(ii) supplement the testing requirements for MOVs in the ASME OM Code by requiring that licensees implementing the ASME OM Code as part of the IST program at their nuclear power plants shall also establish a program to ensure that MOVs continue to be capable of performing their design-basis safety functions. As discussed later in this attachment, ASME has revised the PST and IST provisions for MOVs in the ASME OM Code beginning with the 2009 Edition.

In response to operating experience concerns regarding MOV performance, the NRC staff issued Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," on June 28, 1989, which requested that nuclear power plant licensees and construction permit holders ensure the capability of MOVs in safety-related systems to perform their intended functions by reviewing MOV design bases, verifying MOV switch settings initially and periodically, testing MOVs under design-basis conditions where practicable, improving evaluations of MOV failures and necessary corrective actions, and trending MOV problems. The NRC staff conducted inspections to review the development, implementation, and results of GL 89-10 programs. Licensees under 10 CFR Part 52 are expected to apply the lessons learned from GL 89-10 in developing and implementing their MOV programs.

The Electric Power Research Institute (EPRI) developed the MOV Performance Prediction Methodology (PPM) to determine dynamic thrust and torque operating requirements for gate, globe, and butterfly valves used in nuclear power plants. EPRI described the methodology in Topical Report TR-103237 (Revision 2, April 1997), "EPRI MOV Performance Prediction Program." On March 15, 1996, the NRC staff issued a safety evaluation report (SER) accepting the EPRI MOV PPM with certain conditions and limitations. On February 20, 1997, the staff issued a supplement to the SER on general issues and two unique gate valve designs. On April 20, 2001, the staff issued Supplement 2 to the SER on Addendum 1 to EPRI Topical Report TR-103237 addressing an update of the computer model.

On September 8, 1999, the Nuclear Energy Institute (NEI) submitted Addendum 2 to EPRI Topical Report TR-103237-R2, which described the development of the Thrust Uncertainty Method that takes into account conservatism in the EPRI MOV PPM to provide a more realistic (less bounding) estimate of the thrust required to operate gate valves than predicted by the PPM. In Supplement 3 (dated September 30, 2002) to the SER on the EPRI PPM, the NRC staff concluded that the Thrust Uncertainty Method developed by EPRI is acceptable for the prediction of minimum allowable thrust at control switch trip (or flow isolation) for applicable motor-operated gate valves under cold water applications within the scope of the Thrust Uncertainty Method, based on the NRC staff's review of Addendum 2 to the EPRI Topical Report as supplemented by NEI submittals dated January 5 and December 6, 2001, and June 10, 2002.

From 2004 to 2006, NEI submitted Addenda 3, 4, 5, 6, and 7 to the EPRI MOV PPM that the NRC staff reviewed with requests for additional information to NEI. In a letter dated February 24, 2009, the NRC staff forwarded to NEI Supplement 4 to the SER on the EPRI PPM. In the SER supplement, the NRC staff concluded that the PPM changes described in the PPM addenda improve the ability of licensees to predict the thrust and torque required to operate gate, globe, and butterfly valves, and that they are acceptable for reference by licensees.

On September 18, 1996, the NRC issued GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," requesting that each nuclear power plant licensee 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 bases of the facility.

Licensees under 10 CFR Part 52 are expected to apply the lessons learned from GL 96-05 in their programs for periodic verification of MOV design-basis capability.

In response to GL 96-05, nuclear power plant licensees developed an industry-wide Joint Owners Group (JOG) Program on MOV Periodic Verification. The NRC staff accepted the industry topical report on the JOG Program on MOV Periodic Verification in an SER dated September 25, 2006, and its supplement dated September 18, 2008. Nuclear power plant licensees committing to apply the JOG program in response to GL 96-05 are responsible for implementing the applicable conditions in the SER and its supplement. MPR-2524-A (November 2006), "Joint Owners Group (JOG) Motor Operated Valve Periodic Verification Program Summary," updates the topical report to reflect the NRC final SE, and includes the JOG response to NRC staff requests for additional information and the final SE as appendices to the report. The JOG program does not include actuator output capability as part of its long-term MOV program such that the licensee will need to address this aspect of MOV periodic verification on a plant-specific basis. In Regulatory Issue Summary (RIS) 2011-13 (January 6, 2012), "Followup to Generic Letter 96-05 for Evaluation of Class D Valves Under Joint Owners Group Motor-Operated Valve Periodic Verification Program," the NRC staff provided guidance for licensees in providing periodic verification of the design-basis capability of safety-related MOVs outside the scope of the JOG program.

On August 17, 1995, the NRC issued GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," to request that licensees perform, or confirm that they had previously performed, (1) evaluations of the operational configurations of safety-related, power-operated (including motor-, air-, and hydraulically operated) gate valves for susceptibility to pressure locking and thermal binding; and (2) further analyses, and any needed corrective actions, to ensure that safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding are capable of performing the safety functions within the current licensing basis of the facility. Licensees under 10 CFR Part 52 are expected to apply the lessons learned from GL 95-07.

Beginning with the 2009 Edition, the ASME OM Code replaces the quarterly MOV stroke-time testing requirements with periodic exercising and a performance-based diagnostic testing program described in Appendix III, "Preservice and Inservice Testing of Active Electric Motor Operated Valve Assemblies in Light-Water Reactor Power Plants," to periodically verify that MOVs are capable of performing their design-basis safety functions. As of August 17, 2017, the NRC updated 10 CFR 50.55a to incorporate by reference up through the 2012 Edition of the ASME OM Code, including Appendix III for MOVs. Through the specific MOV requirements in 10 CFR 50.55a or the updated ASME OM Code, licensees under 10 CFR Part 52 are required to establish a program that maintains the capability of their MOVs to perform the applicable design-basis safety functions.

Regulatory Guide (RG) 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," accepts with certain provisions the implementation of specific ASME OM Code Cases in lieu of the applicable provisions in the ASME OM Code as incorporated by reference in the NRC regulations. Licensees may implement the specific ASME OM Code Case accepted with certain provisions in specific revisions to RG 1.192 incorporated in 10 CFR 50.55a without submittal of a request for implementation of an alternative IST method to the NRC for review and authorization.

With respect to MOVs, RG 1.192 accepts with certain provisions ASME OM Code Cases OMN-1, "Alternative Rules for Preservice and Inservice Testing of Active Electric Motor-Operated Valve Assemblies in Light-Water Reactor Power Plants," and OMN-11, "Risk-Informed Testing for Motor-Operated Valves," that provide an alternative to quarterly MOV stroke-time testing through a program of exercising and diagnostic testing on a periodic frequency. ASME used the provisions of OM Code Cases OMN-1 and OMN-11 in developing the performance-based MOV diagnostic testing requirements in Appendix III to the 2009 Edition of the ASME OM Code. With the development of Appendix III in the ASME OM Code, ASME has limited the application of OM Code Cases OMN-1 and OMN-11 to earlier editions and addenda of the ASME OM Code.

The NRC regulations in 10 CFR 50.55a(f)(4)(i) require that the IST program for the initial 10-year IST program interval for a nuclear power plant licensed under 10 CFR Part 52 comply with the ASME Code edition and addenda incorporated by reference in the NRC regulations for the specified time period (currently 12 months) before fuel load (or acceptable ASME OM Code Cases). Therefore, the licensee will need to comply with the requirements in the ASME Code edition and addenda incorporated by reference in the NRC regulations for the specified time period (currently 12 months) before fuel load (or acceptable ASME OM Code Cases), or have been granted relief from, or authorized an alternative to, the ASME OM Code (such as authorization to implement an earlier Code edition or addenda for the initial 10-year IST program interval).

ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," includes provisions for the functional design and qualification of nuclear power plant active mechanical equipment (including MOVs). ASME prepared the 2007 Edition of the QME-1 standard to incorporate lessons learned from valve operating experience and research programs. The NRC staff has accepted the use of ASME QME-1-2007 in Revision 3 to RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," with specific conditions.

In 2017, ASME published ASME Standard QME-1-2017, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," to provide updated qualification provisions for pumps, valves, and dynamic restraints. The NRC staff is preparing proposed Revision 4 to RG 1.100 to address the acceptance of ASME QME-1-2017 with any appropriate conditions. In that the qualification of MOVs and power-operated valves is specified as Tier 1 or Tier 2* provisions in several design certification rules, the inspector should discuss the application of ASME QME-1-2017 with the NRC headquarters technical staff where a licensee intends to implement ASME QME-1-2017.

Design Certification and COL applicants have specified the implementation of ASME QME-1-2007 in their design certification DCD/FSAR or COL FSAR, as applicable. In addition, design certification and COL applicants have described their MOV program in the applicable design certification DCD/FSAR and COL FSAR. The NRC staff described its review of the MOV program in the SERs for the applicable design certification and COL applications.

Specific Guidance

03.01 MOV Selection

The NRC inspector should consider MOV risk insights and performance during selection of the sample of MOVs. For example, the inspector should review available MOV trend reports, nonconformance reports, licensee event reports, maintenance history or other plant documents to obtain insights into whether an MOV is properly sized or has proper switch settings. The inspector should focus on MOVs that are categorized as high risk and low capability margin. The selection of MOVs should include the consideration of various valve sizes, types, and manufacturers. To assist in the selection of an MOV sample, the inspector should request that the licensee provide a table of the safety-related MOVs including their function, safety significance, sizing and setting calculation assumptions, and operating margin.

03.02 MOV Program Scope

The NRC regulations establish the scope of the MOV program by requiring that licensees implementing the ASME OM Code establish a program to ensure that MOVs continue to be capable of performing their design-basis safety functions. Appendix B to 10 CFR Part 50 specifies criteria for the QA program to provide adequate confidence that SSCs will perform their safety-related functions satisfactorily in service.

The inspector should determine whether the licensee is applying the proper criteria when establishing the scope of MOV program. Where a licensee has modified the scope of its MOV program since the previous inspection, the inspector should determine whether the licensee has adequately justified the removal of any MOVs from its MOV program. The inspector should also review plant modifications and determine whether the new or modified MOVs were properly

incorporated into the program, as appropriate. Appendix A, "Considerations in Reviewing the Scope of Licensee MOV Program," to IP 62708, "Motor-Operated Valve Capability," provides additional guidance for the inspector regarding the scope of the licensee's MOV program.

03.03 Design Calculations

The inspector should review the methods used for determining the design-basis functional requirements for MOVs within the scope of the program under the applicable system and environmental parameters for normal, abnormal, and accident conditions that are used in selecting, setting, and adjusting switches (including torque, limit, bypass, and thermal overload). Motor sizing calculations must consider degraded voltage and elevated ambient temperature conditions. Use of appropriate actuator efficiency and the proper application factor must be justified. Adequate bases must exist for stem factors, valve factors, load sensitive behavior and other assumed parameters that are used in calculations used to size actuators. Licensees for nuclear power plants licensed under 10 CFR Part 52 have specified the implementation of the qualification provisions in ASME QME-1-2007 as accepted in Revision 3 to RG 1.100. As part of the functional design verification, the licensee may apply the EPRI MOV PPM where implemented in accordance with NRC acceptance. EPRI also provides guidance for design calculations in its MOV Application Guide.

The inspector should review the output capability calculations for the MOV actuators. For example, the NRC staff discussed ac-powered MOV actuator capability in Information Notice (IN) 96-48, Supplement 1 (July 24, 1998), "Motor-Operated Valve Performance Issues," which references Limatorque Technical Update 98-01 (updated by its Supplement 1). The NRC staff discussed dc-powered MOV actuator capability in RIS 2001-15 (August 1, 2001), "Performance of DC-Powered Motor-Operated Valve Actuators," which references Boiling Water Reactor Owners Group (BWROG) Topical Report NEDC-32958, "BWR Owners Group DC Motor Performance Methodology - Predicting Capability and Stroke Time in DC Motor-Operated Valves." As noted in RIS 2001-15, the NRC staff considers the BWROG methodology to be applicable to dc-powered MOVs in both BWR and pressurized water reactor nuclear power plants.

During GL 89-10 program inspections, the NRC staff provided four acceptable methods a licensee could use to demonstrate the design-basis capability of safety-related MOVs. The four methods for demonstrating MOV capability, in descending order of acceptability were:

1. Dynamic flow testing with diagnostics of each MOV where practicable. 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 should consider an appropriate increase in the valve factor during its design-basis evaluation and setup based on test data from similar valves.
2. Application of the EPRI MOV PPM. This method was initially developed for those valves that could not be dynamically tested. The PPM required internal measurements to provide assurance that the valve performance was predictable. The NRC staff later accepted the use of the PPM where dynamic testing for an MOV was practicable.

3. Where valve-specific dynamic testing was not performed and the PPM was not used, the NRC 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 least preferred approach (with the most margin required) was the use of valve test data from other plants or research programs because the licensee would have minimal information regarding the tested valve and its history. In such cases, the NRC inspector should perform an available capability evaluation of the MOV to provide confidence that the MOV had sufficient capability margin considering the uncertainties in the source of the data.

IP 62708 provides a list of issues to be addressed and assumptions to be justified as applicable in the MOV design calculations. These issues and assumptions include (a) valve factor (including seat area); (b) stem friction coefficient; (c) load sensitive behavior (rate of loading); (d) margins for stem lubrication degradation and springpack relaxation; (e) motor performance factor such as motor rating, efficiencies used in the open and close directions, application factor, power factor used in degraded voltage calculations, and ambient temperature; (f) basis for extrapolation method of partial differential pressure thrust measurements; (g) torque switch repeatability; (h) use of Limitorque, Kalsi, or other sources for increasing thrust and torque allowable limits; (i) equipment bias and uncertainties; (j) degradation assumptions; and (k) justification for grouping of MOVs for application of test data, performance characteristics, structural operating limits, and common-cause failure analysis.

The inspector should determine whether the licensee has addressed the potential for pressure locking or thermal binding of MOVs within the scope of the program, such as by implementation of GL 95-07 or other justified means. Additional inspection guidance is provided in IP 62710, "Power-Operated Gate Valve Pressure Locking and Thermal Binding."

Following the initial verification of MOV capability under design-basis conditions, the MOV switch settings will need to be re-verified if the MOV is replaced (which would constitute the need for a complete demonstration of design-basis capability), modified, or overhauled to the extent that the existing test results might not be representative of the MOV in its modified configuration. Because of the interrelationship of various operating parameters, the performance of the MOV can be affected by routine maintenance work, such as valve packing adjustments.

03.04 Design-Basis Verification, PST and IST Testing

The inspector should determine whether the licensee has conducted performance testing to demonstrate the functional design-basis capability of MOVs to perform their safety functions. The design-basis verification will provide the foundation for the PST and IST programs to demonstrate the operational readiness of MOVs prior to and during reliance on those MOVs to perform their safety functions.

The inspector should determine whether the licensee met the provisions in ASME QME-1-2007 as accepted in RG 1.100 (Revision 3) for demonstrating the functional design-basis capability of MOVs to perform their safety functions. If the licensee proposes to implement ASME QME-1-

2017, the inspector should determine whether the licensee is implementing the standard in an acceptable manner with assistance from the NRC headquarters technical staff.

The licensee will need to demonstrate that the applicable ITAAC are satisfied for the functional design and qualification, and PST activities, of MOVs to perform their safety functions. The inspector should confirm the completion of those activities for the sampled MOVs.

The inspector should determine whether the licensee meets the PST and IST testing provisions specified in the ASME OM Code, including Appendix III, as incorporated by reference in 10 CFR 50.55a (or acceptable ASME OM Code Cases) as described in the COL FSAR and accepted in the applicable NRC SER. The licensee will need to have obtained acceptance from the NRC for relief from or alternatives to the applicable ASME OM Code provisions. The inspector should determine the licensee's plans regarding the requirement in 10 CFR 50.55a(f)(4)(i) that the initial IST program comply with the ASME OM Code edition and addenda incorporated by reference in the NRC regulations for the specified time period (currently 12 months) before fuel load (or acceptable ASME OM Code Cases).

RG 1.192 as incorporated in 10 CFR 50.55a accepts the use of specific revisions to ASME OM Code Cases OMN-1 and OMN-11 with conditions as an alternative to the quarterly MOV stroke-time testing provisions in the ASME OM Code. RG 1.192 also accepts the use of ASME OM Code Case OMN-3 with conditions for the risk ranking of MOVs for use in implementing Code Cases OMN-1 and 11. Some nuclear power plants licensed under 10 CFR Part 52 submitted a request under 10 CFR 50.55a in their COL applications to implement a version of Code Case OMN-1 not accepted at that time in RG 1.192 as an alternative to the quarterly MOV stroke-time requirements in the ASME OM Code. The inspector should review the conditions placed on the use of the version of Code Case OMN-1 in the NRC SER for the COL application.

When the licensee is implementing ASME OM Code, Appendix III, or Code Case OMN-1, the inspector should review the licensee's consideration of the extension of the exercising of MOVs from a quarterly frequency to every refueling outage. As discussed in *Federal Register* Notice 64 FR 51370 (dated September 22, 1999) on page 51386 and *Federal Register* Notice 82 FR 32934 (dated July 18, 2017) on page 32946, the licensee should have sufficient information from the specific MOV, or similar MOVs, to demonstrate that exercising on a refueling outage frequency does not significantly affect component performance. This information may be obtained by grouping similar MOVs and staggering the exercising of the MOVs in the group equally over the refueling interval.

Where degradation in the performance of a high-risk MOV is identified when exercised or tested at an extended interval, the licensee needs to reapply the quarterly frequency for the exercise test interval for all high-risk MOVs and implement diagnostic testing of those MOVs at an interval that provides assurance of their design-basis capability throughout the test interval. The licensee should also evaluate the performance results for MOVs to determine whether the risk ranking of MOVs must be raised to a higher level based on those results.

ASME incorporated OM Code Cases OMN-1 and OMN-11 into the 2009 Edition of the ASME OM Code as Mandatory Appendix III to replace the requirements for quarterly stroke-time testing of MOVs. As of August 17, 2017, the NRC incorporated by reference into 10 CFR 50.55a the ASME OM Code up through the 2012 Edition with conditions on Appendix III similar to those imposed in RG 1.192 on OM Code Cases OMN-1 and OMN-11. In addition, the NRC imposed a condition requiring that when applying Paragraph III-3600, "MOV Exercising Requirements," of Appendix III to the ASME OM Code, licensees shall verify that the stroke

time of MOVs specified in plant technical specifications satisfies the assumptions in the plant's safety analyses. The inspector should determine the applicability of the most recent edition of the ASME OM Code to the nuclear power plant being inspected, and the implementation of the conditions specified directly in 10 CFR 50.55a or through RG 1.192 as incorporated by reference in 10 CFR 50.55a.

The inspector should review the table of MOV sizing and setting assumptions and margins provided by the licensee in identifying MOVs for more detailed review. The inspector should evaluate the MOV sizing and settings determined by the licensee for the sampled MOVs. For example, the licensee should use the best available MOV test data when sizing and setting its MOVs. The licensee should have justification for its assumption for each parameter in its MOV calculations. The licensee should assume a reasonable value based on industry test data for a parameter where it does not have plant-specific justification for the parameter. Where the licensee assumes realistic values based on test data for all parameters, the licensee should take action where the calculation predicts MOV capability problems. The licensee should undertake prompt evaluation of test results to determine capability under design-basis conditions prior to declaring the MOV operable and returning it to service. The licensee needs to have justification for the accuracy of its MOV diagnostic equipment. The licensee should monitor test data to affirm assumptions. The licensee should have justification for applying test data to valve groups.

The licensee needs to determine the capability margin for MOVs within the program, and validate its MOV program assumptions, including valve and stem friction coefficients, and load sensitive behavior, for gate and globe valves; and bearing friction coefficients for butterfly valves. Where a different approach is followed, the licensee needs to justify its approach.

When observing or evaluating MOV testing, the inspector should: (1) witness the testing if scheduling permits, (2) determine whether test equipment is setup and calibrated in accordance with vendor recommendations, (3) determine whether test personnel are properly qualified, (4) determine test equipment inaccuracies and test data accuracy, and (5) determine whether test results are adequately reviewed prior to declaring MOVs operable. The inspector should determine whether the licensee has justified the accuracy of MOV diagnostic equipment. The inspector should also determine whether the licensee has an adequate training program for personnel operating MOV diagnostic equipment and analyzing the information obtained. As part of that training, the licensee should ensure that plant personnel understand the inherent sensitivities and limitations of the diagnostic equipment.

The inspector should determine whether licensee activities prior to testing result in unacceptable preconditioning of the performance of the MOV. The NRC has prepared guidance in several documents, including Inspection Manual Part 9900, "Technical Guidance on Maintenance – Preconditioning," IP 61726, "Surveillance Observations," IP 62707, "Maintenance Observations," IP 71111.22, "Surveillance," and NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants." The NRC staff also alerted licensees to preconditioning issues in IN 97-16, "Preconditioning of Plant Structures, Systems, and Components Before ASME Code Inservice Testing or Technical Specification Surveillance Testing."

03.05 MOV Trending

The inspector should determine whether the licensee has procedures to periodically review MOV data on failures and corrective actions as part of a monitoring and feedback effort to establish trends of MOV performance. In addition to plant specific data, the monitoring and

feedback effort should include industry-wide MOV data. Examples of MOV parameters that may be trended include valve factor, stem factor (as-found and as-left), rate of loading/load sensitive behavior, actuator torque output, bearing coefficients, running load, motor current and voltage, torque switch settings, capability margin, and thrust and torque at control switch trip. The inspector should also evaluate the licensee's procedures for trending qualitative information on MOV performance.

03.06 Preventive Maintenance

The inspector should determine whether the licensee has implemented periodic MOV preventive maintenance based on MOV frequency of operation, working environment and operational experience. The inspector should evaluate these preventive maintenance activities during a walkdown of MOVs installed in the plant.

Examples of the licensee's preventive maintenance activities include the following items: Checking for indications of grease or oil leakage from the various sealed joints and shaft protrusions. Checking the mounting flange and valve yoke for cracks or damage. Checking fasteners for tightness. Lubrication of valve stem, main gear case, and limit switches. Checking valve stem and stem nut threads for damage by direct visual inspection or validated diagnostic methods. With regard to stem nut wear, operating experience has revealed that checking for bronze shavings below an MOV during a plant walkdown is not sufficient to identify significant thread wear of the stem nut prior to failure of the MOV to operate electrically or manually. Checking that the ball in the grease relief valve, if installed, is free to move. Sampling and analysis of the grease in main gear case. Checking spring pack for hardened grease. Checking that T-drains are installed, where appropriate, and are clear of paint and debris. Check limit switch compartment for cleanliness and general integrity of gears and wire terminals.

The inspector should also determine whether the licensee has an adequate training program for plant personnel performing MOV maintenance. The licensee should implement vendor recommendations for preventive maintenance or have justification for its alternate approach.

03.07 Corrective Actions

The inspector should determine whether the licensee's administrative procedures require that MOV failures, malfunctions, and deficiencies be promptly identified and corrected. The inspector should determine whether the licensee's procedures for analysis of MOV failures, justification of corrective actions, and trending of failures and corrective actions for the selected MOVs are adequate. The inspector should review any recent MOV failures and the resulting corrective actions. The licensee's failure analysis needs to include the results and history of each as-found deteriorated condition, malfunction, test, inspection, analysis, repair, or alteration. For example, a torque switch adjustment might overcome an increased actuator load, but does not identify and correct the cause of the increased actuator load. The application of a greater actuator torque might allow the MOV to be returned to service, but could lead to a repetitive or more serious failure. The inspector should determine whether the licensee performed the appropriate level of root cause analysis based on the significance of MOV failures, malfunctions, and deficiencies.

03.08 Post-Maintenance Testing

The inspector should determine whether the licensee's procedures require that MOVs be properly tested prior to return to service following maintenance. The licensee should follow the vendor recommendations for post-maintenance testing consistent with the NRC regulations or have justification for its alternate approach.

The inspector should review selected MOV maintenance packages and determine whether the post-maintenance tests demonstrate that the MOV is capable of performing its design function. For example, MOVs are set up to deliver thrust or torque values determined by calculations based on design-basis conditions. Stroking a valve following maintenance that could have adversely affected the capability of the MOV to provide the required thrust or torque does not demonstrate that the MOV is capable of operating during design-basis conditions. Since post-maintenance testing under design-basis conditions is not always feasible, the licensee might need to use other methods to ensure the maintenance performed has not rendered the MOV incapable of performing its intended function.

If the licensee chooses not to test an MOV following maintenance, the licensee needs to be able to justify that a test was not necessary to demonstrate the capability of the MOV to perform its safety function. For example, valve packing adjustment can affect MOV operation since the adjustment of packing could increase the torque required to open or close the MOV. In some instances, it might be difficult to test an MOV following the adjustment of packing during plant operation because plant conditions prohibit the cycling of the MOV. The inspector should determine whether the licensee has an adequate basis for not testing the MOV following the adjustment of the packing. For example, test data previously obtained could be used to demonstrate that the MOV's thrust or torque capability is not adversely affected at specific packing adjustment settings.

03.09 Operating Experience

The inspector should evaluate the consideration of experience in the MOV program at the nuclear power plant being inspected and from other nuclear power plants. Industry bulletins and NRC information notices alert licensees to operating experience issues with MOVs. For example, the NRC staff issued IN 2003-15, "Importance of Followup Activities in Resolving Maintenance Issues," in response to degradation of MOVs caused by the failure to incorporate adequate instructions for motor pinion key connections in maintenance procedures at an operating nuclear power plant. The NRC staff issued IN 2006-26, "Failure of Magnesium Rotors in Motor-Operated Valve Actuators," in response to degradation of magnesium rotors in MOV motors at several nuclear power plants. The NRC staff issued IN 2006-29, "Potential Common Cause Failure of Motor-Operated Valves as a Result of Stem Nut Wear," in response to significant degradation of stem nuts in numerous MOVs that went unidentified until MOV failures occurred at an operating nuclear power plant. The NRC staff issued IN 2008-20, "Failures of Motor-Operated Valve Actuator Motors with Magnesium Alloy Rotors," in response to degradation of MOV actuator motors with magnesium alloy rotors. The NRC staff issued IN 2010-03, "Failures of Motor-Operated Valves Due to Degraded Stem Lubricant," to alert licensees to potential adverse effects on MOV performance from degradation of lubricant used on valve stems. Limitorque (Flowserve Corporation) prepared a Safety Bulletin in June 2004 (following a tragic personnel accident at a fossil-fired power plant) to emphasize that the use of cheater bars or similar devices to operate MOV actuators is strictly prohibited. The NRC staff issued IN 2012-14, "Motor-Operated Valve Inoperable Due To Stem-Disc Separation," to inform

addressees of operating experience involving a motor-operated globe valve that failed at the connection between the valve stem and disc.

The inspector should discuss with NRR staff the resolution of the motor insulation qualification issue described in NRC Inspection Report 05000424 and 425/2017009 for Vogtle Electric Generating Plant, Units 1 and 2 (dated January 23, 2018).

The NRC staff issued IN 2017-03, "Anchor/Darling Double Disc Gate Valve Wedge Pin and Stem-Disc Separation Failures," to inform licensees of additional operating experience involving Anchor/Darling double-disc gate valves that failed at their stem-disc connection. In light of this operating experience, the inspector should verify that the licensee has justified the structural integrity of the stem-disc connections for its Anchor/Darling double-disc gate valves with threaded connections. The inspector should request assistance from the headquarters technical staff in evaluating the structural integrity of the stem-disc connections of Anchor/Darling double-disc gate valves.

The operating experience with valve stem-disc connections revealed a weakness in the ASME OM Code requirements for verifying valve position indication. As of August 17, 2017, the NRC revised 10 CFR 50.55a to specify a condition to supplement the ASME OM Code requirements for valve position indication beginning with the 2012 Edition of the ASME OM Code. The inspector should determine whether the licensee is complying with the NRC regulatory requirements to supplement the provisions in the ASME OM Code for valve position indication.

03.10 Periodic Verification

The inspector should evaluate the implementation of the program at the nuclear power plant to periodically verify MOV design-basis capability consistent with the NRC regulations in 10 CFR 50.55a(b)(3)(ii). The provisions in Appendix III of the ASME OM Code as incorporated by reference in 10 CFR 50.55a satisfy the regulatory requirements for periodic verification of MOV design-basis capability. The licensee may apply the lessons learned from the JOG program when implementing Appendix III of the ASME OM Code. The licensee described its MOV periodic verification program in the design certification DCD/FSAR and the COL FSAR. The NRC acceptance of the MOV periodic verification program is described in the applicable SER on the design certification DCD/FSAR and COL FSAR. The NRC staff accepted the JOG program for the valve operating requirements for applicable MOVs in an SER dated September 25, 2006, and its supplement dated September 18, 2008.

The inspector should determine whether the licensee's FSAR specifies that the JOG program will be implemented to satisfy the regulatory requirements to periodically verify the design-basis capability of MOVs. The inspector should review the specific attributes of the JOG program including, for example, proper classification of the valves, documentation of the valve material construction, service conditions, qualifying basis, and verification of proper valve factor being applied.

An example of the performance of an inspection of an MOV periodic verification program can be found in NRC IP 95003 Supplemental Inspection Report 05000259/2011011, 05000260/2011011, and 05000296/2011011 (Part 1) for the Browns Ferry Nuclear Plant, dated November 17, 2011 (ADAMS Accession No. ML113210602).

The inspector should determine whether the licensee is following the JOG program in risk ranking MOVs and classifying them based on valve type, construction, materials, service

conditions, manufacturer, and their susceptibility to degradation. The JOG process had four classification categories:

Class A: Valves are not susceptible to degradation based on test data.

Class B: Valves are not susceptible to degradation based on test data and engineering analysis.

Class C: Valves are susceptible to degradation as shown by test data.

Class D: Valves are not covered by the JOG program. Individual plants are responsible for justifying the periodic verification approach.

The inspector should determine whether the licensee has completed the MOV classification process and has documented the results. The inspector should determine whether the licensee is implementing the JOG program consistent with the JOG classification for the sampled MOVs. In particular, the MOVs in JOG Class A or Class B are determined to not be susceptible to degradation of valve operating requirements based on the JOG program. The inspector should determine whether MOVs in JOG Class A or Class B are periodically tested to demonstrate their output capability to satisfy the valve operating requirements. The inspector should determine whether MOVs in JOG Class C are periodically tested to demonstrate their design-basis capability.

For those MOVs in JOG Class D or where the licensee has not committed to implement the JOG program, the inspector should determine whether the licensee has established a plant-specific periodic verification program to ensure their continued design-basis capability. In RIS 2011-13, the NRC staff provides guidance for periodic verification of the design-basis capability of MOVs outside the scope of the JOG program. The licensee needs to have test data to support the periodic verification interval for those MOVs. NRC Inspection Report No. 50-361 and 362/99-18 (dated January 4, 2000) describes the NRC staff inspection of the GL 96-05 program at San Onofre Nuclear Generating Station, Units 2 and 3, which implemented a plant-specific MOV periodic verification program rather than the JOG program.

The inspector should review a sample of MOV periodic verification test results (both static and dynamic), and determine whether information from these tests is incorporated into the design and setup calculations for safety-related MOVs. In addition to valve operating requirements consistent with the JOG program, the inspector should determine whether the licensee addresses actuator output capability as part of its long-term MOV program. The inspector should review the documentation regarding the periodic verification of MOV design-basis capability and determine whether those commitments have been implemented.

The inspector should determine whether valve modifications have eliminated the original design basis capability of specific safety-related MOVs. The JOG program is intended to address valve degradation as it pertains to valve configuration, design, and system application. The JOG dynamic test program was not intended to provide data for the purpose of justifying valve design-basis capability. If a valve in service has a disallowing modification, the inspector should determine whether the licensee has obtained a new qualifying basis.

The inspector could contact the applicable NRC headquarters technical staff for assistance in evaluating the periodic verification of MOV design-basis capability for MOVs within the JOG program or outside its scope.

03.11 Program Changes

The licensee might have revised its MOV program since the previous NRC review or inspection. The inspector should discuss any MOV program changes with the licensee, and evaluate the justification of those changes consistent with the guidance in this IP.

ATTACHMENT 2 AIR-OPERATED VALVES

73758-ATT2-01 INSPECTION OBJECTIVE

The objective of this attachment to Inspection Procedure (IP) 73758 is to provide guidance for the evaluation of the development and implementation of the program at a nuclear power plant under construction in accordance with Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," in Title 10 of the *Code of Federal Regulations* (10 CFR Part 52) to satisfy the regulatory requirements that air-operated valves (AOVs) are capable of performing their safety functions over the full range of conditions from normal operation to design-basis accident conditions.

73758-ATT2-02 INSPECTION REQUIREMENTS

02.01 AOV Selection

Select a sample (5 to 10 AOVs) of risk-significant AOVs from three to five safety systems. The selection of AOVs should include consideration of various valve sizes, types, and manufacturers. The inspector should address a wide range of AOVs in the sample. The AOV sample may be expanded based on the inspection findings where concerns are raised regarding the capability of specific AOVs to perform their design-basis safety functions. For the prototype plant of a new nuclear power plant design, a complete review of the design basis requirements and operating conditions for all safety-related AOVs might be determined to be appropriate.

02.02 AOV Program Scope

Determine whether the scope of the AOV program is consistent with the NRC regulations. Review AOV program scope changes since the completion of any previous NRC review of the AOV program to determine that the appropriate safety-related AOVs are included in the program.

02.03 Design Calculations

Review design documents and calculations for AOV functional requirements under normal, abnormal, and accident conditions; actuator sizing; methods for selecting, setting, and adjusting AOV switch settings; and modifications to the system or valves that could affect the AOV's capability in the as-modified configuration.

02.04 Design-Basis Verification, Preservice Testing and Inservice Testing

Review functional design-basis, preservice testing (PST), and inservice testing (IST) documents for adequacy of test procedures, test equipment, training of test personnel, acceptance criteria, and test results. If the inspection schedule permits, observe actual testing of AOVs.

02.05 AOV Trending

Review available AOV trend reports, failure analyses, corrective actions, nonconformance reports, or other plant documents that may indicate that an AOV is not properly sized, has improper switch settings, or is not properly maintained.

02.06 Preventive Maintenance

Review AOV preventive maintenance to determine whether it is appropriate for the frequency of operation, working environment, and operational experience.

02.07 Corrective Actions

Determine whether the licensee is periodically reviewing data on AOV failures and the effectiveness of the corrective actions.

02.08 Post-Maintenance Testing

Review a sample of AOV maintenance packages and determine whether the post-maintenance tests and results demonstrate that the AOVs are capable of performing their design functions.

02.09 Operating Experience

Review the adequacy of licensee's processing and control of operating experience information and vendor notifications.

02.10 Periodic AOV Program Verification

Evaluate periodic verification of the implementation of the AOV program, including review of AOV periodic inservice test results. Determine whether information from these tests is incorporated into the design and setup calculations for safety-related AOVs.

02.11 Program Changes

Review changes made in programs affecting safety-related AOVs since previous NRC reviews or inspections.

73758-ATT2-03 INSPECTION GUIDANCE

General Guidance

The NRC regulations in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," require that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Appendix A to 10 CFR Part 50 states that where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. Appendix A to 10 CFR Part 50 also requires that a quality assurance (QA) program be established and

implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 specifies criteria for the QA program to provide adequate confidence in the capability of safety-related SSCs to perform their design-basis safety functions.

In 10 CFR 50.55a, the NRC regulations currently incorporate by reference the 1995 through 2012 Edition of the American Society of Mechanical Engineers (ASME) *Operation and Maintenance of Nuclear Power Plants*, Division 1, OM Code: Section IST (commonly referred to as the (OM Code) for implementation of an IST program for pumps, valves, and dynamic restraints used in nuclear power plants. The ASME OM Code (1995 Edition through 2015 Edition) specifies the performance of stroke-time testing of AOVs on a quarterly frequency as part of the IST program.

In the 2017 Edition, the ASME OM Code includes Mandatory Appendix IV, "Preservice and Inservice Testing of Active Pneumatically Operated Valve Assemblies in Nuclear Reactor Power Plants," which requires quarterly stroke time testing and preservice performance assessment testing (PAT) for all AOVs, and periodic PAT for AOVs with high safety significance up to a maximum interval of 10 years. The NRC is considering 10 CFR 50.55a rulemaking to incorporate by reference the 2017 Edition of the ASME OM Code with any appropriate conditions.

The NRC regulations in 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," require licensees to monitor the performance or condition of SSCs in a manner sufficient to provide reasonable assurance that these SSCs are capable of fulfilling their intended functions.

In response to generic concerns regarding motor-operated valve (MOV) performance, the NRC staff issued Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," on June 28, 1989, which requested that nuclear power plant licensees and construction permit holders ensure the capability of MOVs in safety-related systems to perform their intended functions by reviewing MOV design bases, verifying MOV switch settings initially and periodically, testing MOVs under design-basis conditions where practicable, improving evaluations of MOV failures and necessary corrective actions, and trending MOV problems. Licensees under 10 CFR Part 52 are expected to apply the MOV lessons learned from GL 89-10 in developing and implementing their AOV programs.

The Electric Power Research Institute (EPRI) developed the MOV Performance Prediction Methodology (PPM) to determine dynamic thrust and torque operating requirements for gate, globe, and butterfly valves used in nuclear power plants. The NRC staff described its review accepting the EPRI MOV PPM in a safety evaluation report (SER) and several supplements. In Information Notice (IN) 96-48 (August 21, 1996), "Motor-Operated Valve Performance Issues," and its Supplement 1 (July 24, 1998), the NRC staff indicated that lessons learned from the EPRI program were applicable to valves with other types of actuators.

On September 18, 1996, the NRC issued GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," requesting that each nuclear power plant licensee 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 bases of the facility. Licensees under 10 CFR Part 52 are

expected to apply the lessons learned from GL 96-05 in their programs for periodic verification of AOV design-basis capability.

In response to GL 96-05, nuclear power plant licensees developed an industry-wide Joint Owners Group (JOG) Program on MOV Periodic Verification. The NRC staff accepted the JOG Program for MOV Periodic Verification in SER dated September 25, 2006, and its supplement dated September 18, 2008. The JOG program evaluates degradation of the operating requirements for valves rather than the actuator output. Therefore, the lessons learned from the JOG program can be applied to valves powered by air actuators.

On August 17, 1995, the NRC issued GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," to request that licensees perform, or confirm that they had previously performed, (1) evaluations of the operational configurations of safety-related, power-operated (including motor-, air-, and hydraulically operated) gate valves for susceptibility to pressure locking and thermal binding; and (2) further analyses, and any needed corrective actions, to ensure that safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding are capable of performing the safety functions within the current licensing basis of the facility. Licensees under 10 CFR Part 52 are expected to apply the lessons learned from GL 95-07.

ASME OM Code Case OMN-12, "Alternative Requirements for Inservice Testing Using Risk Insights for Pneumatically and Hydraulically Operated Valve Assemblies in Light-Water Reactor Power Plants," allows a nuclear power plant licensee to implement a performance-based periodic testing program for AOVs in lieu of the quarterly stroke-time testing specified in the ASME OM Code. Regulatory Guide (RG) 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," incorporated in 10 CFR 50.55a accepts with certain provisions the implementation of ASME OM Code Case OMN-12.

Licensees may implement the ASME OM Code Cases accepted in RG 1.192 as incorporated in 10 CFR 50.55a without submittal of a request under 10 CFR 50.55a for implementation of an alternative IST method to the NRC for review and approval.

The NRC regulations in 10 CFR 50.55a(f)(4)(i) require that the IST program for the initial 10-year IST program interval for a nuclear power plant licensed under 10 CFR Part 52 comply with the ASME Code edition and addenda incorporated by reference in the NRC regulations the specified time period (currently 12 months) before fuel load (or acceptable ASME OM Code Cases). Therefore, the licensee will need to comply with the requirements in the ASME Code edition and addenda incorporated by reference in the NRC regulations for the specified time period (currently 12 months) before fuel load (or acceptable ASME OM Code Cases), or have been granted relief from, or authorized an alternative to, the ASME OM Code (such as authorization to implement an earlier Code edition or addenda for the initial 10-year IST program interval).

ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," includes provisions for the functional design and qualification of active mechanical equipment (including AOVs) in nuclear power plants. ASME prepared this revision to the QME-1 standard to incorporate lessons learned from valve operating experience and research programs. The functional qualification provisions for power-operated valves (POVs) in ASME QME-1-2007 apply to AOVs as well as other types of POVs. The NRC staff has accepted ASME QME-1-2007 in Revision 3 to RG 1.100, "Seismic Qualification of Electrical and

Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” with specific conditions.

In 2017, ASME published ASME Standard QME-1-2017, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” to provide updated qualification provisions for pumps valves, and dynamic restraints. The NRC staff is preparing proposed Revision 4 to RG 1.100 to address the acceptance of ASME QME-1-2017 with any appropriate conditions. In that the qualification of MOVs and POVs is specified as Tier 1 or Tier 2 provisions in several design certification rules, the inspector should discuss the application of ASME QME-1-2017 with the NRC headquarters technical staff where a licensee intends to implement ASME QME-1-2017.

The NRC issued NRC Regulatory Issue Summary (RIS) 2000-03, “Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions,” to discuss the application of lessons learned from MOV operating experience and research programs to POVs with other than motor actuators. For example, RIS 2000-03 includes a list of attributes for a successful POV design capability and long-term periodic verification program. RIS 2000-03 discusses the development of a JOG program on AOV periodic verification testing, and NRC comments on that program.

Licensees of nuclear power plants licensed under 10 CFR Part 52 have specified in their design certification DCD/FSAR or COL FSAR that additional testing, beyond that required by the ASME OM Code, will be performed as part of the AOV program. These licensee documents state that the AOV program will incorporate the attributes for a successful long-term periodic verification program for POVs as described in RIS 2000-03 by incorporating lessons learned from previous nuclear power plant operations and research programs as they apply to the periodic testing of AOVs and other POVs included in the IST program. The inspector should review the NRC SERs on the design certification DCD/FSAR and COL FSAR for the specific provisions to be addressed in the licensee’s AOV program.

Specific Guidance

03.01 AOV Selection

The NRC inspector should consider AOV risk insights and performance during selection of the sample of AOVs. For example, the inspector should review available AOV trend reports, nonconformance reports, licensee event reports, maintenance history or other plant documents to obtain insights into whether an AOV is properly sized or has proper switch settings. The inspector should focus on AOVs that are categorized as high risk and low capability margin.

The selection of AOVs should include the consideration of various valve sizes, types, and manufacturers. To assist in the selection of an AOV sample, the inspector should request that the licensee provide a table of the safety-related AOVs including their function, safety significance, sizing and setting calculation assumptions, and operating margin.

03.02 AOV Program Scope

Appendix B to 10 CFR Part 50 specifies criteria for the QA program to provide adequate confidence that valves will perform their safety-related functions satisfactorily in service. The inspector should determine whether the licensee is applying the proper criteria when establishing the scope of AOV program. Where a licensee has modified the scope of its AOV program since the previous inspection, the inspector should determine whether the licensee has

adequately justified the removal of any AOVs from its AOV program. The inspector should also review plant modifications and determine whether the new or modified AOVs were properly incorporated into the program, as appropriate.

03.03 Design Calculations

The inspector should review the methods used for determining the design-basis functional requirements for AOVs within the scope of the program under the applicable system and environmental parameters for normal, abnormal, and accident conditions that are used in selecting, setting, and adjusting switches. Licensees for nuclear power plants licensed under 10 CFR Part 52 have specified the implementation of the qualification provisions in ASME QME-1-2007 as accepted in Revision 3 to RG 1.100. As part of the functional design verification, the licensee may apply the EPRI MOV PPM where implemented in accordance with NRC acceptance, where applicable to AOVs. The inspector should review the design-basis capability for the sampled AOVs.

The inspector should determine whether the licensee has addressed the potential for pressure locking or thermal binding of AOVs within the scope of the program, such as by implementation of GL 95-07 or other justified means. Additional inspection guidance is provided in IP 62710, "Power-Operated Gate Valve Pressure Locking and Thermal Binding."

Following the initial verification of AOV capability under design-basis conditions, the AOV switch settings will need to be re-verified if the AOV is replaced (which would constitute the need for a complete demonstration of design-basis capability), modified, or overhauled to the extent that the licensee considers that the existing test results are not representative of the AOV in its modified configuration. Because of the interrelationship of various operating parameters, the performance of the AOV can be affected by routine maintenance work, such as valve packing adjustments.

03.04 Design-Basis Verification, PST and IST Testing

The inspector should determine whether the licensee has demonstrated the functional design-basis capability of AOVs to perform their safety functions. The design-basis verification will provide the foundation for the PST and IST programs to demonstrate the operational readiness of AOVs prior to and during reliance on those AOVs to perform their safety functions. The inspector should determine whether the licensee met the provisions in ASME QME-1-2007 as accepted in RG 1.100 (Revision 3) for demonstrating the functional design-basis capability of AOVs to perform their safety functions. Where appropriate, the inspector should evaluate the licensee's application of ASME QME-1-2017 with assistance from the NRC headquarters technical staff.

The inspector should determine whether the licensee has demonstrated that the applicable ITAAC are satisfied for the functional design and qualification, and PST activities, of the sampled AOVs to perform their safety functions.

The inspector should determine whether the licensee meets the PST and IST testing provisions specified in the ASME OM Code as incorporated by reference in 10 CFR 50.55a (or acceptable ASME OM Code Cases) as described in the COL FSAR and accepted in the applicable NRC SER. The licensee will need to have obtained acceptance from the NRC for relief from or alternatives to the applicable ASME OM Code provisions.

The NRC regulations in 10 CFR 50.55a(f)(4)(i) require that the IST program comply with the ASME Code edition and addenda incorporated by reference in the NRC regulations for the specified time period (currently 12 months) before fuel load (or acceptable ASME OM Code Cases). The inspector should determine the status of the incorporation by reference of the latest edition of the ASME OM Code into 10 CFR 50.55a for applicability to the nuclear power plant being inspected. For example, the NRC is preparing a proposed revision to 10 CFR 50.55a to incorporate by reference the 2017 Edition of the ASME OM Code, which includes Mandatory Appendix IV with updated PST and IST provisions for AOVs.

The inspector should review the design certification DCD/FSAR or COL FSAR for the nuclear power plant being inspected for additional testing, beyond that required by the ASME OM Code, that will be performed as part of the AOV program. These licensee documents state that the AOV program will incorporate the attributes for a successful long-term periodic verification program for POVs as described in RIS 2000-03 by incorporating lessons learned from previous nuclear power plant operations and research programs as they apply to the periodic testing of AOVs and other POVs included in the IST program. For example, the licensee documents state that AOV program will include the following elements:

Setpoints for AOVs will be defined based on current vendor information or valve qualification diagnostic testing, such that the valve is capable of performing its design-basis functions.

Periodic static testing will be performed to identify potential degradation, unless those valves are periodically cycled during normal plant operation under conditions that meet or exceed the worst case operating conditions within the licensing basis of the plant for the valve, which would provide adequate periodic demonstration of AOV capability. If necessary based on valve qualification or operating experience, periodic dynamic testing will be performed to re-verify the capability of the valve to perform its safety functions.

Sufficient diagnostics will be used to collect relevant data (e.g., valve stem thrust and torque, fluid pressure and temperature, stroke time, operating and/or control air pressure, etc.) to verify the valve meets the functional requirements of the qualification specification.

Test frequency will be specified and evaluated each refueling outage based on data trends as a result of testing. Frequency for periodic testing will be in accordance with the JOG AOV Program Document and the NRC staff comments on this program.

Post-maintenance procedures include appropriate instructions and criteria to ensure baseline testing is re-performed as necessary when maintenance on the valve, repair or replacement, have the potential to affect valve functional performance.

Guidance is included to address lessons learned from other valve programs specific to the AOV program.

Documentation from AOV testing, including maintenance records and records from the corrective action program are retained and periodically evaluated as a part of the AOV program.

The inspector should review the NRC SERs on the design certification DCD/FSAR and COL FSAR for the specific provisions to be addressed in the licensee's AOV program. RG 1.192 as

incorporated in 10 CFR 50.55a accepts the use of a specific revision to ASME OM Code Case OMN-12 with conditions as an alternative to the quarterly stroke-time testing provisions for AOVs in the ASME OM Code. The licensee will need to justify the implementation of the conditions specified in RG 1.192 where Code Case OMN-12 will be applied at the nuclear power plant.

The inspector should review the table of AOV sizing and setting assumptions and margins provided by the licensee in identifying AOVs for more detailed review. The licensee should use the best available AOV test data when sizing and setting its AOVs. The licensee should have justification for its assumption for each parameter in its AOV calculations. The licensee should assume a reasonable value based on industry test data for a parameter where it does not have plant-specific justification for the parameter.

Where the licensee assumes realistic values based on test data for all parameters, the licensee should take action where the calculation predicts AOV capability problems. The licensee should undertake prompt evaluation of test results to determine capability under design-basis conditions prior to declaring the AOV operable and returning it to service. The licensee should have justification for the accuracy of its AOV diagnostic equipment. The licensee should monitor test data to affirm assumptions.

The licensee should have justification for applying test data to valve groups.

The licensee needs to determine the capability margin for AOVs within the program, and validate its AOV program assumptions. Where a different approach is followed, the licensee needs to justify its approach.

When observing or evaluating AOV testing, the inspector should: (1) witness the testing if scheduling permits, (2) determine whether test equipment is setup and calibrated in accordance with vendor recommendations, (3) determine whether test personnel are properly qualified, (4) determine test equipment inaccuracies and test data accuracy, and (5) determine whether test results are adequately reviewed prior to declaring AOVs operable.

The inspector should determine whether the licensee has justified the accuracy of AOV diagnostic equipment. The inspector should also determine whether the licensee has an adequate training program for personnel operating AOV diagnostic equipment and analyzing the information obtained. As part of that training, the licensee should ensure that plant personnel understand the inherent sensitivities and limitations of the diagnostic equipment.

The inspector should determine whether licensee activities prior to testing result in unacceptable preconditioning of the performance of the AOV. The NRC has prepared guidance in several documents, including Inspection Manual Part 9900, "Technical Guidance on Maintenance – Preconditioning," IP 61726, "Surveillance Observations," IP 62707, "Maintenance Observations," IP 71111.22, "Surveillance," and NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants." The NRC staff also alerted licensees to preconditioning issues in IN 97-16, "Preconditioning of Plant Structures, Systems, and Components Before ASME Code Inservice Testing or Technical Specification Surveillance Testing."

03.05 AOV Trending

The inspector should determine whether the licensee has procedures to periodically review AOV data on failures and corrective actions as part of a monitoring and feedback effort to

establish trends of AOV performance. In addition to plant specific data, the monitoring and feedback effort should include industry-wide data. The inspector should also evaluate the licensee's procedures for trending qualitative information on AOV performance.

03.06 Preventive Maintenance

The inspector should determine whether the licensee has implemented periodic AOV preventive maintenance based on AOV frequency of operation, working environment and operational experience. The inspector should evaluate these preventive maintenance activities during a walkdown of AOVs installed in the plant. The inspector should also determine whether the licensee has an adequate training program for plant personnel performing AOV maintenance. The licensee should implement vendor recommendations for preventive maintenance or have justification for its alternate approach.

03.07 Corrective Actions

The inspector should determine whether the licensee's administrative procedures require that AOV failures, malfunctions, and deficiencies be promptly identified and corrected. The inspector should determine whether the licensee's analysis of any recent AOV failures, justification of corrective actions, and trending of failures and corrective actions for the selected AOVs is adequate. The inspector should review any recent AOV failures and the resulting corrective actions. The licensee's failure analysis needs to include the results and history of each as-found deteriorated condition, malfunction, test, inspection, analysis, repair, or alteration. The inspector should also determine whether the licensee performed the appropriate level of root cause analysis based on the significance of AOV failures, malfunctions, and deficiencies.

03.08 Post-Maintenance Testing

The inspector should determine whether the licensee's procedures require that AOVs be properly tested prior to return to service following maintenance. The licensee should follow the vendor recommendations for post-maintenance testing consistent with the NRC regulations or have justification for its alternate approach.

The inspector should review selected AOV maintenance packages and determine whether the post-maintenance tests demonstrate that the AOV is capable of performing its design function. Stroking a valve following maintenance that could have adversely affected the capability of the AOV to provide the required thrust or torque does not demonstrate that the AOV is capable of operating during design-basis conditions.

Since post-maintenance testing under design-basis conditions is not always feasible, the licensee must use other methods to ensure the maintenance performed has not rendered the AOV incapable of performing its intended function.

If the licensee chooses not to test an AOV following maintenance, the licensee needs to be able to justify that a test was not necessary to demonstrate the capability of the AOV to perform its safety function. For example, valve packing adjustment can affect AOV operation since the adjustment of packing could increase the force required to open or close the AOV.

In some instances, it may be difficult to test an AOV following the adjustment of packing during plant operation because plant conditions prohibit the cycling of the AOV. The inspector should

determine whether the licensee has an adequate basis for not testing the AOV following the adjustment of the packing.

03.09 Operating Experience

The inspector should evaluate the consideration of experience in the AOV program at the nuclear power plant being inspected and from other nuclear power plants. Industry bulletins and NRC information notices alert licensees to operating experience issues with AOVs.

03.10 Periodic AOV Program Verification

The inspector should evaluate the implementation of the program at the nuclear power plant to periodically verify AOV design-basis capability. The inspector should confirm that the licensee is implementing the periodic verification provisions of its AOV program described in the applicable design certification DCD/FSAR and COL FSAR.

Licensees of nuclear power plants licensed under 10 CFR Part 52 have specified in their design certification DCD/FSAR or COL FSAR that the attributes for a long-term periodic verification program for POVs described in RIS 2000-03 will be implemented. The inspector should review the documentation regarding the periodic verification of AOV design-basis capability and determine whether those commitments have been implemented.

03.11 Program Changes

The licensee might have revised its AOV program since the previous NRC review or inspection. The inspector should discuss any AOV program changes with the licensee, and evaluate the justification of those changes consistent with the guidance in this IP.

ATTACHMENT 3
PYROTECHNIC-ACTUATED VALVES

73758-ATT3-01 INSPECTION OBJECTIVE

The objective of this attachment to Inspection Procedure (IP) 73758 is to provide guidance for the evaluation of the development and implementation of the program at a nuclear power plant under construction in accordance with Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," in Title 10 of the *Code of Federal Regulations* (10 CFR Part 52) to satisfy the regulatory requirements that pyrotechnic-actuated valves (squib valves) are capable of performing their safety functions over the full range of conditions from normal operation to design-basis accident conditions.

73758-ATT3-02 INSPECTION REQUIREMENTS

02.01 Squib Valve Selection

Nuclear power plants licensed under 10 CFR Part 52 might use squib valves for safety-related functions. For example, new nuclear power plants with passive reactor designs use squib valves in safety-related applications with high safety significance. With the small number of squib valves in nuclear power plants licensed under 10 CFR Part 52, the inspector should select all squib valves for review as part of the initial inspection. The inspector could select a sample of the squib valves for review in subsequent inspections.

02.02 Squib Valve Program Scope

Determine whether all squib valves with safety functions are included in the squib valve program.

02.03 Design Calculations

Review design documents and calculations for squib valve functional requirements under normal, abnormal, and accident conditions.

02.04 Design-Basis Verification, Preservice Testing and Inservice Testing and Surveillance

Review plant documents for adequacy of design-basis verification, preservice testing (PST) and inservice testing (IST) test and surveillance procedures, equipment, personnel training, acceptance criteria, and surveillance results. If the inspection schedule permits, observe actual testing and surveillance of squib valves.

02.05 Squib Valve Trending

Review available squib valve trend reports, failure analyses, corrective actions, nonconformance reports, or other plant documents to determine whether the squib valves are properly sized and maintained.

02.06 Preventive Maintenance

Review squib valve preventive maintenance to determine whether it is appropriate for the frequency of operation, working environment, and operational experience.

02.07 Corrective Actions

Determine whether the licensee is periodically reviewing data on squib valve deficiencies and the effectiveness of the corrective actions.

02.08 Post-Maintenance Testing and Surveillance

Review a sample of squib valve maintenance packages and determine whether the post-maintenance testing and surveillance results demonstrate that the squib valves are capable of performing their design functions.

02.09 Operating Experience

Review the adequacy of licensee's processing and control of operating experience information and vendor notifications.

02.10 Periodic Testing and Surveillance Results

Evaluate development and implementation of the program for the review of squib valve periodic testing and surveillance results. Determine whether information from the surveillance is properly addressed by the licensee engineering staff.

02.11 Program Changes

Review changes made in programs affecting squib valves since previous NRC reviews or inspections.

73758-ATT3-03 INSPECTION GUIDANCE

General Guidance

The NRC regulations in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," require that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Appendix A to 10 CFR Part 50 states that where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. Appendix A to 10 CFR Part 50 also requires that a quality assurance (QA) program be established and implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 specifies criteria for the QA program to provide adequate confidence in the capability of safety-related SSCs to perform their design-basis safety functions.

In 10 CFR 50.55a, the NRC regulations incorporate by reference the American Society of Mechanical Engineers (ASME) *Operation and Maintenance of Nuclear Power Plants*, Division 1, OM Code: Section IST (commonly referred to as the OM Code) for implementation of an IST program for pumps, valves, and dynamic restraints used in nuclear power plants. ASME includes surveillance requirements for squib valves in Subsection ISTC of the ASME OM Code. In the 2012 Edition, the ASME OM Code specifies provisions for PST and IST surveillance of squib valves in new reactors (e.g., reactors receiving their combined license (COL) after January 1, 2000). The ASME OM Code squib valve provisions for new reactors are consistent with the provisions of the license condition for squib valves specified in COLs issued for AP1000 reactors. As of August 17, 2017, the NRC updated 10 CFR 50.55a to incorporate by reference up through the 2012 Edition of the ASME OM Code, including the PST and IST surveillance provisions for squib valves in new reactors.

The NRC regulations in 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," require licensees to monitor the performance or condition of SSCs in a manner sufficient to provide reasonable assurance that these SSCs are capable of fulfilling their intended functions.

ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," includes provisions for the functional design and qualification of nuclear power plant active mechanical equipment (including squib valves). ASME prepared the 2007 Edition of the QME-1 standard to incorporate lessons learned from motor-operated valve (MOV) programs implemented at operating nuclear power plants.

The NRC staff has accepted ASME QME-1-2007 in Revision 3 to RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," with specific conditions.

Nuclear power plants licensed under 10 CFR Part 52 specify the application of ASME QME-1-2007 as accepted in Revision 3 to RG 1.100 in their design certification DCD/FSAR or COL FSAR. Although ASME QME-1-2007 does not include a specific section on squib valve qualification, the inspector should review the licensee's application of the qualification methodology for power-operated valves in ASME QME-1-2007 to squib valves in the nuclear power plant under construction.

In 2017, ASME published ASME Standard QME-1-2017, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," to provide updated qualification provisions for pumps, valves, and dynamic restraints. ASME QME-1-2017 includes specific provisions for the qualification of squib valves based on lessons learned from the design and qualification of squib valves for the new reactors. The inspector should review the ASME QME-1-2017 for insights in evaluating the qualification process for squib valves in accordance with this inspection procedure. The NRC staff is preparing proposed Revision 4 to RG 1.100 to address the acceptance of ASME QME-1-2017 with any appropriate conditions. In that the qualification of MOVs and power-operated valves is specified as Tier 1 or Tier 2* provisions in several design certification rules, the inspector should discuss the application of ASME QME-1-2017 with the NRC headquarters technical staff where a licensee intends to implement ASME QME-1-2017. The NRC regulations in 10 CFR 50.55a(f)(4)(i) require that the IST program for the initial 10-year IST program interval for a nuclear power plant licensed under 10 CFR Part 52 comply with the ASME Code edition and addenda incorporated by reference in the NRC regulations for the specified time period (currently 12 months) before fuel load (or acceptable ASME OM Code

Cases). Therefore, the licensee will need to comply with the requirements in the ASME Code edition and addenda incorporated by reference in the NRC regulations for the specified time period (currently 12 months) before fuel load (or acceptable ASME OM Code Cases), or have been granted relief from, or authorized an alternative to, the ASME OM Code (such as authorization to implement an earlier Code edition or addenda for the initial 10-year IST program interval).

In light of the complexity and safety significance of some squib valves in new reactors, ASME prepared updated PST and IST testing and surveillance requirements for squib valves to be used in nuclear power plants licensed after January 1, 2000. These new requirements are published in the 2012 Edition of the ASME OM Code. The NRC has incorporated by reference the 2012 Edition of the ASME OM Code into 10 CFR 50.55a with applicable conditions. Therefore, nuclear power plants licensed under 10 CFR Part 52 will need to evaluate the applicability of the new squib valve surveillance requirements when implementing the ASME OM Code incorporated by reference in 10 CFR 50.55a for the specified time period (currently 12 months) before fuel loading.

Licensees of nuclear power plants licensed under 10 CFR Part 52 that use squib valves in safety applications include provisions for developing IST programs for squib valves in their COL FSAR. The COL FSARs typically specify that industry and regulatory guidance will be considered in the development of the IST program for squib valves. The FSARs also state that the IST program for squib valves will incorporate lessons learned from the design and qualification process for these valves such that surveillance activities provide reasonable assurance of the operational readiness of squib valves to perform their safety functions.

To supplement ASME OM Code provisions for squib valves prior to the 2012 Edition, the NRC specified license conditions for PST and IST surveillance of squib valves when issuing the COLs for nuclear power plants that use squib valves in safety applications. The license condition includes the following requirements:

Before initial fuel load, the licensee shall implement a surveillance program for specific explosively actuated valves (squib valves) that includes the following provisions in addition to the requirements specified in the ASME OM Code as incorporated by reference in 10 CFR 50.55a.

a. Preservice Testing

All explosively actuated valves shall be preservice tested by verifying the operational readiness of the actuation logic and associated electrical circuits for each explosively actuated valve with its pyrotechnic charge removed from the valve. This must include confirmation that sufficient electrical parameters (voltage, current, resistance) are available at the explosively actuated valve from each circuit that is relied upon to actuate the valve. In addition, a sample of at least 20 percent of the pyrotechnic charges in all explosively actuated valves shall be tested in the valve or a qualified test fixture to confirm the capability of each sampled pyrotechnic charge to provide the necessary motive force to operate the valve to perform its intended function without damage to the valve body or connected piping. The sampling must select at least one explosively actuated valve from each redundant safety train. Corrective action shall be taken to resolve any deficiencies identified in the operational readiness of the actuation logic or associated electrical circuits, or the capability of a pyrotechnic charge. If a charge fails to fire or its capability is not confirmed, all charges with the same batch

number shall be removed, discarded, and replaced with charges from a different batch number that has demonstrated successful 20 percent sampling of the charges.

b. Operational Surveillance

Explosively actuated valves shall be subject to the following surveillance activities after commencing plant operation:

- (1) At least once every 2 years, each explosively actuated valve shall undergo visual external examination and remote internal examination (including evaluation and removal of fluids or contaminants that may interfere with operation of the valve) to verify the operational readiness of the valve and its actuator. This examination shall also verify the appropriate position of the internal actuating mechanism and proper operation of remote position indicators. Corrective action shall be taken to resolve any deficiencies identified during the examination with post-maintenance testing conducted that satisfies the PST requirements.
- (2) At least once every 10 years, each explosively actuated valve shall be disassembled for internal examination of the valve and actuator to verify the operational readiness of the valve assembly and the integrity of individual components and to remove any foreign material, fluid, or corrosion. The examination schedule shall provide for each valve design used for explosively actuated valves at the facility to be included among the explosively actuated valves to be disassembled and examined every 2 years. Corrective action shall be taken to resolve any deficiencies identified during the examination with post-maintenance testing conducted that satisfies the PST requirements.
- (3) For explosively actuated valves selected for test sampling every 2 years in accordance with the ASME OM Code, the operational readiness of the actuation logic and associated electrical circuits shall be verified for each sampled explosively actuated valve following removal of its charge.

This must include confirmation that sufficient electrical parameters (voltage, current, resistance) are available for each valve actuation circuit. Corrective action shall be taken to resolve any deficiencies identified in the actuation logic or associated electrical circuits.

- (4) For explosively actuated valves selected for test sampling every 2 years in accordance with the ASME OM Code, the sampling must select at least one explosively actuated valve from each redundant safety train. Each sampled pyrotechnic charge shall be tested in the valve or a qualified test fixture to confirm the capability of the charge to provide the necessary motive force to operate the valve to perform its intended function without damage to the valve body or connected piping.

Corrective action shall be taken to resolve any deficiencies identified in the capability of a pyrotechnic charge in accordance with the PST requirements.

This license condition shall expire upon (1) incorporation of the above surveillance provisions for explosively actuated valves into the facility's inservice testing program, or (2) incorporation of inservice testing requirements for explosively actuated valves in

new reactors (i.e., plants receiving a construction permit, or combined license for construction and operation, after January 1, 2000) to be specified in a future edition of the ASME OM Code as incorporated by reference in 10 CFR 50.55a, including any conditions imposed by the NRC, into the facility's inservice testing program.

This license condition supplements the current requirements in the ASME OM Code for explosively actuated valves, and sets forth requirements for both pre-service testing and operational surveillance, as well as any necessary corrective action. The license condition will expire when either (1) the license condition is incorporated into the plant-specific IST program; or (2) the updated ASME OM Code requirements for squib valves in new reactors, as accepted by the NRC in 10 CFR 50.55a, are incorporated into the plant-specific IST program. For the purpose of satisfying the license condition, the licensee retains the option of including in its IST program either the requirements stated in this condition, or including updated ASME Code requirements.

The inspector should review the design certification DCD/FSAR, COL FSAR, COL license conditions, and ASME OM Code (and acceptable Code Cases) as incorporated by reference into 10 CFR 50.55a for the specified time period (currently 12 months) before fuel loading, for the PST and IST surveillance requirements for squib valves applicable to the nuclear power plant under construction. As indicated, the NRC has revised 10 CFR 50.55a to incorporate by reference up through the 2012 Edition of the ASME OM Code that includes PST and IST surveillance provisions for squib valves consistent with the COL license condition for squib valves in AP1000 reactors. Therefore, the implementation of the 2012 Edition of the ASME OM Code as incorporated by reference in 10 CFR 50.55a in the PST and IST programs for a new reactor will allow the COL license condition for squib valves to expire. As noted in the above COL license condition, the licensee retains the option of including in its IST program either the requirements stated in this condition, or the updated ASME OM Code requirements as incorporated by reference in 10 CFR 50.55a.

Specific Guidance

03.01 Squib Valve Selection

With the small population of squib valves in nuclear power plants, the inspector should select all squib valves with safety functions for the initial inspection. Subsequent inspections could include a sample of the squib valves.

03.02 Squib Valve Program Scope

Appendix B to 10 CFR Part 50 specifies criteria for the QA program to provide adequate confidence that valves will perform their safety-related functions satisfactorily in service. The inspector should determine whether the licensee has included squib valves with safety-related functions in the Appendix B program.

03.03 Design Calculations

The inspector should review the methods used for determining the design-basis functional requirements for squib valve within the scope of the program under the applicable system and environmental parameters. Licensees for nuclear power plants licensed under 10 CFR Part 52 have specified the implementation of the qualification provisions in ASME QME-1-2007 as accepted in Revision 3 to RG 1.100. The inspector should review the design-basis capability

calculations for the squib valves.

03.04 Design-Basis Verification, PST and IST Testing and Surveillance

The inspector should determine whether the licensee has conducted testing to verify the functional design-basis capability of squib valves to perform their safety functions.

The design-basis verification will provide the foundation for the PST and IST programs to demonstrate the operational readiness of squib valves prior to and during reliance on those valves to perform their safety functions. The inspector should determine whether the licensee met the provisions in ASME QME-1-2007 as accepted in RG 1.100 (Revision 3) for demonstrating the functional design-basis capability of squib valves to perform their safety functions. The inspector may use information in ASME QME-1-2017 as guidance in evaluating the qualification process for squib valves with assistance from the NRC headquarters technical staff.

The inspector should determine whether the licensee demonstrated that the applicable ITAAC are satisfied for the functional design and qualification, and PST activities, of squib valves to perform their safety functions. The inspector should confirm the completion of those activities for the selected squib valves.

The ASME OM Code specifies PST and IST for squib valves with conditions when incorporated by reference in 10 CFR 50.55a. In addition, some licensees have license conditions related to PST and IST programs for squib valves. The inspector should determine whether the licensee meets the PST and IST testing provisions specified as license conditions as well in the ASME OM Code as incorporated by reference in 10 CFR 50.55a as described in the COL FSAR and accepted in the applicable NRC SER. The inspector should determine whether the licensee has obtained acceptance from the NRC for relief from or alternatives to the applicable ASME OM Code provisions.

The surveillance requirements for squib valves in the COL license conditions include PST and IST activities. For example, the surveillance requirements include internal and external inspection of the squib valves, testing the squib valve actuation circuitry without firing the pyrotechnic charge, and firing a sample of pyrotechnic charges to verify their output capability. The inspector should review each of those surveillance requirements and evaluate the licensee activities to meet those requirements. For example, the inspector should determine whether the cabling and circuitry (including connectors) are designed and qualified to provide the proper amperage for the operation of the squib valves, including resistance over the full range for design temperatures, with establishment of appropriate post-installation and surveillance procedures.

The inspector should determine whether the licensee has conducted appropriate testing and surveillance of squib valves. For example, the licensee will need to undertake prompt evaluation of surveillance results to determine capability under design-basis conditions prior to declaring the squib valve operable. The licensee needs to have justification for the accuracy of its squib valve diagnostic equipment. The licensee should monitor surveillance data to affirm assumptions. The licensee needs to determine the capability margin for squib valves within the program and validate its program assumptions. Where a different approach is followed, the licensee needs to justify its approach.

When observing or evaluating squib valve surveillance, the inspector should: (1) witness the surveillance if scheduling permits, (2) determine whether equipment is setup and calibrated in accordance with vendor recommendations, (3) determine whether test personnel are properly qualified, (4) determine equipment inaccuracies and data accuracy, and (5) determine whether surveillance results are adequately reviewed prior to declaring the squib valves operable. The inspector should determine whether the licensee has justified the accuracy of diagnostic equipment. The inspector should also determine whether the licensee has an adequate training program for personnel operating diagnostic equipment and analyzing the information obtained. As part of that training, the licensee should ensure that plant personnel understand the limitations of the diagnostic equipment.

The inspector should determine whether licensee activities prior to testing result in unacceptable preconditioning of the squib valve performance. The NRC has prepared guidance in several documents, including Inspection Manual Part 9900, "Technical Guidance on Maintenance – Preconditioning," IP 61726, "Surveillance Observations," IP 62707, "Maintenance Observations," IP 71111.22, "Surveillance," and NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants." The NRC staff also alerted licensees to preconditioning issues in Information Notice (IN) 97-16, "Preconditioning of Plant Structures, Systems, and Components Before ASME Code Inservice Testing or Technical Specification Surveillance Testing."

03.05 Squib Valve Trending

The inspector should determine whether the licensee has procedures to periodically review data on squib valve deficiencies and corrective actions as part of a monitoring and feedback effort to establish trends of squib valve performance. In addition to plant specific data, the monitoring and feedback effort should include industry-wide squib valve data. The inspector should also evaluate the licensee's procedures for trending qualitative information on squib valve performance.

03.06 Preventive Maintenance

The inspector should determine whether the licensee has implemented periodic squib valve preventive maintenance. The inspector should evaluate these preventive maintenance activities during a walkdown of squib valves installed in the plant. The licensee should implement vendor recommendations for preventive maintenance or have justification for its alternate approach. The inspector should also determine whether the licensee has an adequate training program for plant personnel performing squib valve maintenance.

03.07 Corrective Actions

The inspector should determine whether the licensee's administrative procedures require that squib valve failures, malfunctions, and deficiencies be promptly identified and corrected. The inspector should determine whether the licensee's procedures for analysis of squib valve deficiencies, justification of corrective actions, and trending of failures and corrective actions for the selected squib valves are adequate. The inspector should review any recent squib valve deficiencies and the resulting corrective actions. The licensee's failure analysis needs to include the results and history of each as-found deteriorated condition, malfunction, test, inspection, analysis, repair, or alteration. The inspector should also determine whether the licensee performed the appropriate level of root cause analysis based on the significance of squib valve failures, malfunctions, and deficiencies.

03.08 Post-Maintenance Testing and Surveillance

The inspector should determine whether the licensee's procedures require that squib valves receive proper testing and surveillance prior to return to service following maintenance. The inspector should review selected squib valve maintenance packages and determine whether the post-maintenance testing and surveillance demonstrate that the squib valve is capable of performing its design function. The licensee should follow the vendor recommendations for post-maintenance testing consistent with the NRC regulations or have justification for its alternate approach. In that stroking a squib valve following maintenance is not feasible, the licensee must use other methods to ensure the maintenance performed has not rendered the squib valve incapable of performing its intended function. The inspector should determine whether the licensee has justified its post-maintenance testing and surveillance activities.

03.09 Operating Experience

The inspector should evaluate the consideration of experience in the squib valve program at the nuclear power plant being inspected and from other nuclear power plants. Industry bulletins and NRC information notices alert licensees to operating experience issues with squib valves. For example, at Dresden Unit 2 on October 28, 2011, during functional testing of the Standby Liquid Control (SLC) system, the "A" explosive valve failed to actuate during the test. The licensee replaced the failed explosive valve and tested the "B" SLC system to validate functionality. Further testing by the licensee revealed the failure of the "A" explosive valve was due to a faulty valve trigger assembly. Vendor analysis concluded that the failure was associated with thermal degradation of the primer's explosive material. The licensee determined that a heat trace modification accomplished in 2009 was installed too close to the 'A' squib valve trigger assembly. The prolonged exposure to heat greater than expected resulted in the accelerated thermal degradation of the explosive material and valve. The inspector should verify that the licensee has installed the sampled squib valves consistent with the assumptions for their environmental qualification. See Licensee Event Report (LER) 237-2011005R1 (ADAMS Accession No. ML12363A044) for more information. In addition, performance of squib valves in the SLC system at Vermont Yankee was discussed in Information Notice 86-13 (February 21, 1986), "Standby Liquid Control System Squib Valves Failure to Fire," and its Supplement 1 (August 6, 1986).

03.10 Periodic Testing and Surveillance Results

The inspector should evaluate the implementation of the program at the nuclear power plant to periodically verify squib valve design-basis capability. The inspector should review the documentation regarding the periodic verification of squib valve design-basis capability.

03.11 Program Changes

The licensee might have revised its squib valve program since the previous NRC review or inspection. The inspector should discuss any squib valve program changes with the licensee and evaluate the justification of those changes consistent with the guidance in this IP.

ATTACHMENT 4
Revision History for IP 73758

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Description of Training Required and Completion Date	Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information)
N/A	ML12314A205 04/19/13 CN 13-011	Initial issue to support inspections of construction programs described in IMC 2504, Construction Inspection Program: Inspection of Construction and Operational Programs. Completed 4 year search of historical CNs and found no commitments related to this Inspection Procedure.	N/A	ML13085A111
N/A	ML18222A281 09/06/18 CN 18-030	Proposed update to reflect lessons learned from nuclear power plant operating experience (including Anchor/Darling double-disc gate valve stem-disc connection integrity issues), vendor component qualification, NRC inspection results, recent ASME OM Code editions, and NRC rulemaking since the initial issuance of IP 73758.	N/A	ML18222A279
N/A	ML19364A004 02/06/20 CN 20-007	Revises the recommended sample size for valves and dynamic restraints in Appendices A and C to provide greater flexibility to inspectors and slightly lowers the inspection resource estimates. Also makes minor editorial changes.		n/a