NUREG-1482, Revision 3

United States Nuclear Regulatory Commission

Protecting People and the Environment

Guidelines for Inservice Testing at Nuclear Power Plants

Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at the NRC's Library at www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and Title 10, "Energy," in the *Code of Federal Regulations* may also be purchased from one of these two sources:

1. **The Superintendent of Documents**
   U.S. Government Publishing Office
   Washington, DC 20402-0001
   Internet: www.bookstore.gpo.gov
   Telephone: (202) 512-1800
   Fax: (202) 512-2104

2. **The National Technical Information Service**
   5301 Shawnee Road
   Alexandria, VA 22312-0002
   Internet: www.ntis.gov
   1-800-553-6847 or, locally, (703) 605-6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: **U.S. Nuclear Regulatory Commission**
Office of Administration
Multimedia, Graphics, and Storage &
Distribution Branch
Washington, DC 20555-0001
E-mail: distribution.resource@nrc.gov
Facsimile: (301) 415-2289

Some publications in the NUREG series that are posted at the NRC's Web site address www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852-2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036-8002
Internet: www.ansi.org
(212) 642-4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX),(4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and the Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of the NRC's regulations (NUREG–0750).

NUREG-1482, Revision 3

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Guidelines for Inservice Testing at Nuclear Power Plants

Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants

Manuscript Completed: July 2020
Date Published: July 2020

Prepared by: Gurjendra S. Bedi

Office of Nuclear Reactor Regulation

ABSTRACT

The staff of the U.S. Nuclear Regulatory Commission (NRC) is issuing Revision 3 to the NUREG-1482, “Guidelines for Inservice Testing at Nuclear Power Plants,” to assist the nuclear industry in establishing a basic understanding of the regulatory basis for pump and valve inservice testing (IST) programs and dynamic restraint (snubbers) examination and testing programs. This NUREG also provides information regarding the NRC’s involvement in the development of the American Society of Mechanical Engineers (ASME) *Operation and Maintenance of Nuclear Power Plants*, Division 1, OM Code: Section IST (OM Code). In this NUREG, the NRC staff discusses OM Code inquiries, the inservice examination and testing of snubbers, pump and valve IST, the use of ASME code cases, conditions on the use of the OM Code, guidance for OM Code noncompliance, requests for alternatives to the OM Code at operating commercial nuclear power plants, and the development of IST programs for new reactors[1].

This NUREG report replaces Revision 0, Revision 1 and Revision 2 to NUREG-1482 and is applicable, unless stated otherwise, to editions and addenda (up to and including the 2017 Edition) to the OM Code, that are incorporated by reference in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a. In addition, the NRC staff discusses other IST program topics such as the NRC process for the review of the OM Code, conditions on the use of the OM Code, interpretations of the OM Code, and development of IST programs for new reactors. In this NUREG, the NRC staff provides guidance included in Revision 2 to NUREG-1482 that has been updated to reflect IST lessons learned and operating experience since the NUREG was previously issued.

All Subsections and paragraphs used in NUREG-1482, Revision 3, are from the 2017 Edition of the OM Code as incorporated by reference in 10 CFR 50.55a, unless otherwise noted. Appendix A contains guidance related to inservice examination and testing of snubbers. Appendix B contains guidance related to the treatment of pumps, valves, and dynamic restraints during implementation of 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants.”

Paperwork Reduction Act

[1] New Reactor (referred to as a post-2000 reactor in the ASME OM Code) is defined in NUREG-1482 as a nuclear plant that was issued (or will be issued) its construction permit, or combined license for construction and operation, by the applicable regulatory authority on or following January 1, 2000.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

EXECUTIVE SUMMARY

The information in NUREG-1482, Revision 0, issued April 1995, Revision 1, issued January 2005, and Revision 2, issued October 2013, described inservice testing (IST) programs in the past. Revision 3 to NUREG-1482 replaces Revision 0, Revision 1 and Revision 2 to NUREG-1482 and is applicable, unless stated otherwise, to editions and addenda (up to and including the 2017 Edition) to the American Society of Mechanical Engineers (ASME) *Operation and Maintenance of Nuclear Power Plants*, Division 1, OM Code: Section IST (OM Code), that are incorporated by reference in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a, “Codes and standards.” (*Federal Register*, Vol. 85, No. 86, pages 26540-26581, dated May 4, 2020).

The NRC regulations quoted in Revision 3 to NUREG-1482 are current as of June 3, 2020 (85 *Federal Register* 26540). Licensees should be aware that specific NRC regulations may have been modified since then. The latest NRC regulations are available at www.ecfr.gov.

This NUREG provides guidance for the inservice testing of pumps and valves, and inservice testing of dynamic restraints (snubbers) at nuclear power plants based on lessons learned since the issuance of previous revisions to NUREG-1482. Appendix A contains guidance related to inservice examination and testing of dynamic restraints (snubbers). Appendix B contains guidance related to the treatment of pumps, valves, and dynamic restraints during implementation of 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants.”

The guidelines and recommendations provided in this NUREG and its Appendices do not supersede the regulatory requirements specified in 10 CFR 50.55a. Further, this NUREG and its Appendices do not authorize the use of alternatives to, or grant relief from, the ASME OM Code requirements for inservice testing of pumps and valves, or inservice examination and testing of dynamic restraints (snubbers), incorporated by reference in 10 CFR 50.55a.

ABBREVIATIONS AND ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ADS	automatic depressurization system
AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
AOV	air-operated valve
ASME	American Society of Mechanical Engineers
ALWR	Advanced Light-Water Reactor
ATWS	anticipated transiant without scram
BEP	best efficiency point
BWR	boiling-water reactor
BWST	borated water storage tank
CFR	*Code of Federal Regulations*
CIV	containment isolation valve
COL	combined operating license
CPT	comprehensive pump test
CRD	control rod drive
CVCS	chemical and volume control system
DBD	design-basis document
DCD	design comtrol document
ECCS	emergency core cooling system
EPRI	Electric Power Research Institue
FIV	flow-indused vibration
FR	*Federal Register*
FSAR	final safety analysis report
FSER	final safety evaluation report
GDC	general design criterion
GE	General Electric Company
GL	generic letter
GPM	gallons per minute
GSI	generic safety issue

HCU	hydraulic control unit
HOV	hydraulic-operated valve
HPCI	high-pressure coolant injection
Hz	hertz
IEEE	Institute of Electrical and Electronics Engineers
IM	inspection manual
IN	information notice
INL	Idaho National Engineering Labpratory
IP	inspection procedure
ISI	inservice inspection
IST	inservice testing
ITAAC	inspections, tests, analyses, and acceptance criteria
JOG	Joint Owners Group
LAR	License Amendment Requrest
LCO	limiting condition for operation
LOCA	loss-of-coolant accident
LTOP	low temperature overpressure
LWR	light-water reactor
MOV	motor-operated valve
MSIV	main steam isolation valve
MSSV	main steam safety valve
NEI	Nuclear Energy Institute (formerly NUMARC)
NIC	Nuclear Industry Check Valve Group
NOED	Notice of Enforcement Discretion
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation (NRC)
NUMARC	Nuclear Management and Resources Council (now NEI)
OD	operabilty determination
OE	operating experience
OM Code	ASME *Operation and Maintenance of Nuclear Power Plants*, Division 1, OM Code: Section IST
PASS	post-accident sampling system
PdM	predictive maintenance
PIV	pressure isolation valve

PORV	power-operated relief valve
POV	power-operated valve
PPM	performance prediction methodology
PRA	probabilistic risk assessment
PTC	Performance Test Code
Psid	pounds per square inch differential
PST	preservice testing
PWR	pressurized-water reactor
P&ID	piping and instrument diagram
QA	Quality Assurance
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RFO	refueling outage
RG	regulatory guide
RHR	residual heat removal
RIS	regulatory issue summary
RTNSS	regulatory treatment of non-safety systems
RWST	refueling water storage tank
RWT	refueling water tank
SAR	safety analysis report
SBLC	standby liquid control
SBO	station blackout
SGCV	Code Committee Sub-Group on Check Valves (ASME)
SE	safety evaluation
SI	safety injection
SOV	solenoid-operated valve
SR	surveillance requirement
SRM	staff requests momorandum
SRP	Standard Review Plan
SSC	system, structure, and/or component
S/RV	safety/relief valve
STR	special treatment requirements
STS	Standard Technical Specification
TDI	Transamerica DeLaval
TRM	technical requirement manual
TSTF	technical specifications task force

TS	technical specification(s)
UFSAR	updated final safety analysis report
VEGP	Vogle Electric Generating Plant
WGCV	Working Group on Check Valves (ASME)

1 INTRODUCTION

1.1 Regulatory Basis

Section 55a, “Codes and standards,” of Part 50, “Domestic Licensing of Production and Utilization Facilities,” in Title 10, “Energy,” of the *Code of Federal Regulations* (10 CFR 50.55a) defines the requirements for applying industry Codes and standards to boiling- or pressurized-water-cooled nuclear power facilities. Each of these facilities is subject to the conditions in paragraphs (a), (b), (f), and (g) of 10 CFR 50.55a, as they relate to inservice inspection (ISI) and inservice testing (IST). By rulemaking effective September 8, 1992 (see 57 FR 34666; August 6, 1992), the U.S. Nuclear Regulatory Commission (NRC) established paragraph (f) of 10 CFR 50.55a to separate the IST requirements from the ISI requirements in paragraph (g).

The National Technology Transfer and Advancement Act of 1995 (P.L. 104-113) requires that if agencies establish technical standards, they must use technical standards that voluntary consensus standards bodies develop or adopt unless the use of such standards is inconsistent with applicable law or is otherwise impractical. P.L. 104-113 requires Federal agencies to use industry consensus standards to the extent practical; however, it does not require Federal agencies to endorse a standard in its entirety. The law does not prohibit an agency from generally adopting a voluntary consensus standard while taking exception to specific portions of the standard if those provisions are deemed to be “inconsistent with applicable law or otherwise impractical.” Furthermore, taking specific exceptions furthers the congressional intent of Federal reliance on voluntary consensus standards because it allows the adoption of substantial portions of consensus standards without the need to reject the standards in their entirety because of limited provisions that are not acceptable to the agency.

The American Society of Mechanical Engineers (ASME) *Operation and Maintenance of Nuclear Power Plants*, Division 1, OM Code: Section IST (OM Code) is a national, voluntary consensus standard. The NRC approves or mandates the use of editions and addenda to the OM Code in 10 CFR 50.55a through the rulemaking process of “incorporation by reference.” Once the OM Code Edition or Addenda is incorporated by reference into the NRC’s regulations, each provision of the OM Code that 10 CFR 50.55a incorporates by reference and mandates constitutes a legally binding NRC requirement imposed by rule. The staff notes that ASME no longer prepares addenda to the OM Code and that only new editions to the OM Code are published by ASME. Therefore, the staff has removed some references to addenda to the OM Code from NUREG-1482 where the discussion involves future updating of IST programs.

As of June 3, 2020, the NRC regulations in 10 CFR 50.55a(a)(1)(iv) incorporate by reference the 1995 Edition through the 2017 Edition of the OM Code promulgated by the ASME, in which Subsection ISTA provides general IST requirements and Subsections ISTB, ISTC, and ISTD provide the IST requirements for pumps (pre-2000 plants), valves, and dynamic restraints, respectively. Subsection ISTE provides requirements for risk-informing IST programs. Subsection ISTF provides IST requirements for pumps (post-2000 plants) in new reactors. Based on those requirements, each nuclear power plant licensee must establish IST programs, specify the components included in the program as well as the test methods and frequencies for those components, and implement the program in accordance with the OM Code as incorporated by reference in 10 CFR 50.55a.

Where a test requirement of the OM Code is determined to be impractical for a facility, the NRC’s regulations in 10 CFR 50.55a(f)(5)(iii) allow the licensee to submit a request for relief from that requirement, along with information to support the determination of impracticality, and the NRC staff to review that request under 10 CFR 50.55a(f)(6)(i). Relief requests generally detail the reasons for deviating from the OM Code requirements and propose alternative testing methods or frequencies. The Commission is authorized to evaluate licensees' relief requests and may grant the requested relief and/or impose alternative requirements, considering the burden that the licensee might incur if the OM Code requirements were enforced for the facility.

Pursuant to 10 CFR 50.55a(z)(1) and (2), the Commission may authorize the licensee to implement an alternative to the OM Code requirements, provided that the alternative ensures an acceptable level of quality and safety or the OM Code requirement presents a hardship without a compensating increase in the level of quality and safety.

Paragraph 50.55a(f)(4)(i) requires that inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during the initial 120-month interval must comply with the requirements in the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a(a) on the date 18 months before the date of issuance of the operating license under 10 CFR Part 50, or 18 months before the date scheduled for initial loading of fuel under a combined license (COL) under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," (or the optional ASME OM Code Cases listed in NRC Regulatory Guide (RG) 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," as incorporated by reference in 10 CFR 50.55a), subject to the conditions listed in 10 CFR 50.55a(b). During the initial 120-month IST program interval, this Code Edition is called the "Code of Record."

Paragraph 50.55a(f)(4)(ii) requires that an IST activity conducted during each 120-month interval following the initial interval must be conducted in compliance with the requirements of the latest edition and addenda of the OM Code incorporated by reference in the version of 10 CFR 50.55a(a) that is in effect 18 months before the start of the interval and subject to the conditions listed in 10 CFR 50.55a(b). During each successive 120-month IST program interval, this Code Edition is called the "Code of Record."

The regulations in 10 CFR 50.55a(f)(4)(iv) specify that inservice testing of pumps and valves may meet the requirements in editions and addenda of the OM Code that were published more recently than those that are incorporated by reference in 10 CFR 50.55a(a), subject to Commission approval and the conditions listed in 10 CFR 50.55a(b). Requests for approval to use later editions and addenda previously incorporated by reference in 10 CFR 50.55a may be made via letter to the NRC. The NRC will evaluate the request and provide a letter response. See NRC Regulatory Issue Summary (RIS) 2004-12, "Clarification on Use of Later Editions and Addenda to the ASME OM Code and Section XI," for additional clarification of this process.

The 10 CFR 50.55a regulations are applicable to pump and valve IST programs at operating reactors and are discussed throughout this NUREG. As a result of the unique wording in various paragraphs, note that the NRC **authorizes** licensee-proposed alternatives in accordance with 10 CFR 50.55a(z), **grants** relief and **imposes** alternative requirements in accordance with 10 CFR 50.55a(f)(6)(i) and 10 CFR 50.55a(g)(6)(i), or **approves** the use of later Code editions and addenda in accordance with 10 CFR 50.55a(f)(4)(iv) and 10 CFR 50.55a(g)(4)(iv).

The NRC regulations quoted in Revision 3 to NUREG-1482 are current as of June 3, 2020 (85 FR 26540). Licensees should be aware that specific NRC regulations may have been modified since then. The latest NRC regulations are available at www.ecfr.gov.

1.2 Regulatory History of NRC Staff IST Guidance

Over the years, the NRC has issued guidance for implementing IST requirements. After publishing the rule that established the IST requirements in 10 CFR 50.55a (see 41 FR 6256; February 12, 1976), the NRC sent letters to notify operating licensees of the new rule. In November 1976, after receiving inquiries from the licensees regarding acceptable methods for complying with the regulation, the NRC issued letters to licensees to transmit NRC staff guidance for complying with 10 CFR 50.55a(g), "Inservice Inspection Requirements." One of these transmitted letters to licensees during November 1976 containing NRC staff guidance is available in the NRC Agencywide Documents Access and Management System (ADAMS) Accession No. ML17215B409 for information.

To eliminate the backlog of IST program reviews for operating nuclear power plants, the NRC issued Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," dated April 3, 1989 (ADAMS Accession No. ML031150259). That generic letter included 11 technical positions that the NRC staff used in reviewing licensees' IST program relief requests and described alternatives to the Code requirements that the NRC staff considered acceptable. As a unique resolution of the backlogged IST program reviews, in GL 89-04, the NRC staff also approved six of the 11 technical positions (1, 2, 6, 7, 9, and 10), pursuant to 10 CFR 50.55a(g)(6)(i), with the provision that the licensee must perform the alternative testing delineated in the applicable position. The NRC staff approved these alternatives upon recognizing that it might be impractical to perform the required testing, and enforcing the requirements might pose an unnecessary burden on licensees.

GL 89-04 stated that licensees must document their use of Positions 1, 2, 6, 7, 9, and 10 in the IST program, but did not require the documentation to take the form of a relief request. The generic letter granted approval to follow the alternative testing delineated in Positions 1, 2, 6, 7, 9, and 10, pursuant to 10 CFR 50.55a(g) [now (f)] provided that the licensee followed the provisions of GL 89-04. For convenience, most licensees documented their use of these positions in relief requests; however, the NRC found other forms of program documentation to be acceptable as long as the provisions of the referenced positions were clearly documented and were discussed in adequate detail to ensure that they conformed with those provisions. Certain licensees may have submitted relief requests to ensure they adequately documented their conformance in their IST program, even though documentation in the IST program plan also would have been acceptable, as stated in GL 89-04.

The NRC staff held four public meetings to discuss GL 89-04 and stated that the generic letter was a first step toward resolving various problems associated with developing and implementing IST programs at nuclear power plants. The NRC staff had previously identified these problems through its reviews of licensees' IST programs, and by inspecting and auditing IST programs at plant sites, participating on the various ASME Code committees, and meeting with licensees and industry groups. The NRC staff then summarized the questions and answers from the four public meetings in a letter entitled "Minutes of the Public Meetings on Generic Letter 89-04," dated October 25, 1989 (ADAMS Accession No. ML19262F381). That letter contained useful information about how to apply the guidance in GL 89-04 and discussed issues of interest to licensees who attended the public meetings. In a subsequent letter, dated January 22, 1991, the NRC staff issued “Supplement to Minutes of the Public Meetings on Generic Letter 89-04” to address a question on stop-check valve testing.

Since the NRC issued GL 89-04, the NRC staff has improved its guidance regarding IST programs by revising 10 CFR 50.55a, and separating the IST and ISI programs in paragraphs (f) and (g), respectively, issuing additional guidance, and coordinating with ASME for periodic symposia on testing pumps and valves. The NRC intends to continue to improve its IST-related guidance through participation on OM Code committees and at meetings of technical organizations, as well as regular updates of the agency’s published guidance as future needs arise.

Over many years, the NRC has determined that some OM Code provisions were inadequate to assess the operational readiness of specific components to perform their safety functions. For example, the NRC supplemented the previous IST requirements in the OM Code for quarterly stroke-time testing of motor-operated valves (MOVs) in light of the inadequacy of such testing to assess the operational readiness of those power-operated valves (POVs). By its incorporation by reference in 10 CFR 50.55a, the OM Code is intended to assess the operational readiness of pumps, valves, and dynamic restraints (snubbers) in the IST program to perform their safety functions. Monitoring degradation might be one method of assessing operational readiness of those components if its initial design-basis capability has been demonstrated. In response to POV testing and operational experience, ASME developed appendices to the OM Code to provide IST provisions for diagnostic testing to assess the operational readiness of certain MOVs and air- or pneumatic-operated valves (AOVs).

1.3 NRC Recommendations and Guidance

The NRC staff guidance and recommendations in NUREG-1482, Revision 3, are based on the ASME OM Code, 1995 Edition through the 2017 Edition.

The recommendations herein replace the guidance and technical positions in GL 89-04. Note that specific relief is required to implement the guidance derived from GL 89-04. However, relief justification may refer to the positions in GL 89-04 with clarifying information to clearly show how it would apply to a licensee’s situation. To the extent practical, this document reflects the applicable section, subsection, or paragraph of the appropriate documents (subsections of 10 CFR Part 50; OM Code; regulatory guides; etc.).

The guidance presented herein is voluntary and may be used for requesting relief under 10 CFR 50.55a(f), or for authorization of an alternative under 10 CFR 50.55a(z). Licensees may also request relief or the use of an alternative which is not in conformance with this guidance. The NRC may grant relief or authorize the alternative if the licensee has addressed all of the aspects of the proposed relief or alternative in an acceptable manner.

1.3.1 NRC Review of the ASME OM Code

The first edition and addendum to the ASME OM Code that 10 CFR 50.55a incorporated by reference were the 1995 Edition and the 1996 Addendum. The NRC determines acceptability of new provisions in subsequent editions and addenda to the OM Code and the need for conditions on the use of the OM Code. Generally, the NRC staff participates with other ASME committee members in discussions and technical debates in the development of OM Code revisions. NRC committee representatives discuss the OM Code and technical justifications with other cognizant NRC staff to ensure an adequate technical review. Finally, NRC management reviews and approves the proposed agency position on the OM Code as part of the rulemaking to amend 10 CFR 50.55a to incorporate by reference new editions and addenda to the OM Code with any appropriate conditions. In addition, ASME also has a process for developing and publishing the OM Code. Therefore, there is confidence that ASME OM Code editions and addenda incorporated by reference in 10 CFR 50.55a (including necessary conditions) provide reasonable assurance of adequate protection of public health and safety with respect to inservice testing of pumps, valves, and dynamic restraints (snubbers) in nuclear power plants.

1.3.2 Exemptions

Under 10 CFR 50.12(a) and 10 CFR 52.7, the NRC may, either on its own initiative or upon application by any licensee, grant an exemption from the requirements of 10 CFR Part 50 that is authorized by law, does not present an undue risk to the public health and safety, is consistent with the common defense and security, and is appropriate because of special circumstances. If the NRC approves the application, the exemption relieves the licensee from compliance with the regulation(s) involved. Exemptions are normally not used for 10 CFR 50.55a(z) authorization of alternatives, or 10 CFR 50.55a(f) granting of relief.

1.4 Synopsis of Report

This discussion follows the format of a typical IST program plan, including Development and Implementation, General Guidance, Valves, Pumps, Technical Specifications, Code Noncompliance, and Risk-Informed IST.

Section 2, "Developing and Implementing an IST Program," describes existing IST requirements, discusses the scope of the IST program, and describes guidance for presenting information in IST programs, including cold shutdown justifications, refueling outage (RFO) justifications, and relief requests. Section 2 also includes a sample list of plant systems for boiling-water reactors (BWRs) and pressurized-water reactors (PWRs) that typically (but not necessarily) contain pumps or valves that perform a safety function and are subject to requirements of the OM Code.

Section 3, "General Guidance on Inservice Testing," describes the NRC's recommendations and its bases for several general aspects of IST programs. Sections 4 and 5 then become more specific, describing recommendations on valve-related and pump-related issues, respectively. Throughout Sections 3 through 5, this document discusses the IST requirements for which licensees have requested relief or proposed alternatives. This document also provides guidance concerning the types of information that licensees typically should (or in some cases must) include in their relief requests. Sections 3 through 5 also discuss related OM Code and regulatory issues and provide recommendations and guidance as needed. These discussions do not impose additional requirements beyond those imposed by the OM Code or the regulations and, as such, do not represent backfits.

These discussions are intended to clarify the existing requirements of the OM Code or the regulations and, as such, they may provide recommendations to ensure that licensees continue to meet the OM Code and other regulatory requirements.

Sections 6, 7, and 8 discuss the standard technical specifications, the process licensees should follow when they identify an OM Code nonconformance, and the development of a risk-informed IST program, respectively. Section 9 presents a list of related references.

This guidance is not equivalent to NRC staff positions in a regulatory guide, because licensees must request approval through the relief or alternative process described in 10 CFR 50.55a where the OM Code provisions as incorporated by reference in 10 CFR 50.55a will not be met.

2 DEVELOPING AND IMPLEMENTING AN INSERVICE TESTING PROGRAM

Licensees may use the following guidance for developing and implementing IST programs. This guidance supplements existing requirements and previously approved guidance on IST programs.

2.1 Compliance Considerations

The NRC regulations in 10 CFR 50.55a(f)(4) specify that pumps and valves that are within the scope of the ASME OM Code must meet the IST requirements set forth in the applicable editions and addenda of the OM Code that are incorporated by reference in 10 CFR 50.55a to the extent practical within the limitations of design, geometry, and materials of construction of the components. The IST program is intended to assess the operational readiness of the stated components. The regulations specify IST program requirements for the initial and successive 120-month intervals following plant startup.

Paragraph 50.55a(f)(4)(i) requires that inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during the initial 120-month interval must comply with the requirements in the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a(a) on the date 18 months before the date of issuance of the operating license under 10 CFR Part 50, or 18 months before the date scheduled for initial loading of fuel under a COL under 10 CFR Part 52 (or the optional ASME OM Code Cases listed in RG 1.192, as incorporated by reference in 10 CFR 50.55a), subject to the conditions listed in 10 CFR 50.55a(b).

Paragraph 50.55a(f)(4)(ii) requires that IST activities conducted during each 120-month interval following the initial interval must be conducted in compliance with the requirements of the latest edition and addenda of the OM Code incorporated by reference in the version of 10 CFR 50.55a(a) that is in effect 18 months before the start of the interval. As of June 3, 2020, the NRC regulations in 10 CFR 50.55a(a)(iv) incorporate by reference the 1995 Edition through the 2017 Edition of the OM Code subject to conditions.

The regulations incorporate by reference the OM Code with conditions to specify the IST requirements and, therefore, the OM Code IST provisions are legally-binding requirements. A plant's technical specifications (TS) which might include general and specific requirements for IST and other surveillance testing of pumps and valves, are part of the facility operating license and, therefore, are also legally-binding requirements. The plant's safety analysis includes information concerning the design limitations and functional requirements for the performance of pumps and valves for the given facility. The plant's IST program, including any relief requests and data analysis methods, describes the licensee's means for implementing the various requirements for the specific plant.

The implementing procedures include the lowest tier of IST elements. In addition, IST engineers often use other information (such as bases documents, vendor manuals, trend data, and graphs) in developing, maintaining, and implementing the plant's IST program.

Licensees must meet the regulations and TS if they identify a conflict between the regulations and their IST program or procedures. The regulations in 10 CFR 50.55a(f)(5)(ii) require that if an IST program for a facility conflicts with the TS, the licensee shall apply to the NRC for amendment of the TS to conform the TS to the revised program. The NRC staff provides guidance on cases where a licensee modifies its plant in a way that affects the basis for relief that the NRC has previously granted. Similarly, if a licensee has obtained the NRC's authorization of an alternative pursuant to 10 CFR 50.55a(z)(1) or (2), the licensee need not use that alternative if it subsequently determines that continued compliance with the OM Code requirements is warranted or necessary for particular circumstances that may preclude implementation of the authorized alternative. When a licensee revises an implementing procedure, the licensee typically ensures that the IST program continues to reflect the required testing. Similarly, when a system, subsystem, or component is modified, or an operating or test procedure or valve alignment is changed in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments," the licensee typically reviews the IST requirements to determine whether it must change the program for the affected components.

The NRC may grant relief from or authorize alternatives to IST requirements submitted as requests that include a discussion of the requirements, a description of the proposed alternative, and the justification for approval or use of the alternative. The following provisions in 10 CFR 50.55a address the process for authorizing alternatives or granting relief:

- Regulations in 10 CFR 50.55a(z)(1) allow the NRC to authorize an alternative to an ASME OM Code provision as incorporated by reference in 10 CFR 50.55a if the proposed alternative would provide an acceptable level of quality and safety. The NRC will normally authorize an alternative pursuant to this provision only if the licensee proposes a method of testing that is equivalent to, or an improvement of, the method specified by the OM Code as incorporated by reference in 10 CFR 50.55a, or if the alternative testing will comply or is consistent with later OM Code editions incorporated by reference in 10 CFR 50.55a.
- Regulations in 10 CFR 50.55a(z)(2) allow the NRC to authorize an alternative if compliance with the ASME OM Code provision as incorporated by reference in 10 CFR 50.55a would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The NRC may authorize an alternative pursuant to this provision if, although the proposed alternative testing does not comply with the OM Code, the increase in overall plant safety and quality attained by complying with the OM Code requirement is not justified in light of the difficulty associated with compliance.
- Regulations in 10 CFR 50.55a(f)(6)(i) include the following provision:

  > The Commission will evaluate determinations under paragraph (f)(5) of this section that code requirements are impractical. The Commission may grant relief and may impose such alternative requirements as it determines are authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

  Therefore, the NRC may grant relief pursuant to 10 CFR 50.55a(f)(6)(i) or may authorize alternatives if the licensee demonstrates that the design or access limitations make the OM Code requirement impractical. The NRC staff's evaluation considers the burden created by imposing the OM Code requirements on the licensee.

2.1.1 ASME OM Code Case Applicability

Code Cases prepared by ASME represent alternatives or additions to the OM Code. A Code Case is the official method of ASME for handling a reply to an inquiry or proposed alternative when study indicates that the OM Code wording needs clarification, or when the reply allows an alternative to the existing requirements of the OM Code, if ASME endorses the use of the alternative methods. ASME develops Code Cases through the OM Code committee consensus process to clarify the intent of existing OM Code requirements or to provide an alternative to a specific OM Code requirement. A Code Case may be issued for the purpose of providing alternative rules when justified, to permit early implementation of an approved revision when the need is urgent, or to specify IST methods for situations not covered by existing provisions of the OM Code.

The NRC reviews new or revised Code Cases to determine their acceptability for incorporation by reference in 10 CFR 50.55a through NRC regulatory guides. Accordingly, the NRC staff developed RG 1.192 as well as RG 1.193, "ASME Code Cases Not Approved for Use."

The regulations in 10 CFR 50.55a(a)(3)(iii) incorporate by reference RG 1.192. Licensees may implement the OM Code Cases listed in RG 1.192 without obtaining further NRC review or approval if the OM Code Cases are used in their entirety with any supplemental conditions specified in RG 1.192 and the licensee's IST Code of Record is applicable to the OM Code Case. RG 1.193 lists Code Cases not approved for general use. Licensees requesting the NRC's approval as part of a relief or alternative request to implement an OM Code Case listed in the RG 1.193 must show, at a minimum, that adequate protection to public health and safety is provided if the OM Code Case is applied.

If a licensee would like to use an OM Code Case with an edition or addendum of the OM Code to which it is not applicable, the licensee has the following options:

a. Request to use the OM Code Case as an alternative, beyond its stated applicability, and receive authorization by the NRC pursuant to 10 CFR 50.55a(z), or

b. If the OM Code Case is applicable to an edition or addendum of the OM Code later than the version of the OM Code being used by the licensee, the licensee could update its IST program to the later version of the OM Code pursuant to 10 CFR 50.55a(f)(4)(iv) or (g)(4)(iv) and then use the OM Code Case, provided the OM Code Case has been approved for use in the appropriate RG 1.192 and incorporated by reference into 10 CFR 50.55a. Note that (1) the later version of the OM Code must also have been incorporated by reference into 10 CFR 50.55a, (2) the licensee must update all related requirements of the respective edition or addenda, and (3) the update must be specifically approved by the Commission.

The NRC may authorize the use of an OM Code Case that it has not yet been approved for use in RG 1.192 if a licensee requests the use of the OM Code Case under 10 CFR 50.55a(z). The NRC may authorize the use of a Code Case that it has not yet been approved for use in RG 1.192 if a licensee requests the use of the Code Case under 10 CFR 50.55a(z). The authorization for a specific licensee to use a Code Case that is not listed in RG 1.192 does not authorize any other licensee to use the Code Case.

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," OMN-3, "Requirements for Safety Significance Categorization of Components Using Risk Insights for

Inservice Testing of LWR Power Plants," OMN-4, "Requirements for Risk Insights for Inservice Testing of Check Valves at LWR Power Plants," OMN-11, "Risk-Informed Testing for Motor-Operated Valves," and OMN-12, "Alternate Requirements for Inservice Testing Using Risk Insights for Pneumatically and Hydraulically Operated Valve Assemblies in Light-Water Reactor Power Plants (OM-Code 1998, Subsection ISTC)," as accepted in RG 1.192 include risk-informed provisions that licensees may apply in their IST programs. RG 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," describes an acceptable alternative approach for applying risk insights from a probabilistic risk assessment (PRA), in conjunction with established traditional engineering information, to make changes to a nuclear power plant's IST program. The approach described in RG 1.175 addresses the high-level safety principles specified in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and attempts to strike a balance between defining an acceptable process for developing risk-informed IST programs without being overly prescriptive. The alternative approach described in RG 1.175 must be authorized by the NRC pursuant to 10 CFR 50.55a(z)(1) on a plant-specific basis before being implemented by a given licensee. However, because 10 CFR 50.55a(z)(1) places no restrictions on the scope of alternatives that the NRC may authorize, licensees may propose risk-informed alternatives to their entire IST program or may propose alternatives that are more limited in scope (e.g., for a particular system or group of systems, or for a particular group of components). However, with the issuance of RG 1.192, licensees may use specific risk-informed IST methods without first obtaining NRC staff review and approval. Section 8 discusses risk-informed IST in greater detail.

2.1.2 Conditions to the ASME OM Code

The NRC regulations incorporate by reference specific editions and addenda (1995 through the 2017 Edition) to the ASME OM Code in 10 CFR 50.55a(a)(1)(iv), subject to conditions, such as those outlined below.

2.1.2.1 10 CFR 50.55a(b)(3)(i)—Quality Assurance

The OM Code references the use of either the 1994 Edition, 2008 Edition, and 2009-1a Addenda of the ANSI/ASME NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities," or the Owner's 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Program as part of its individual provisions for a quality assurance (QA) program. However, ANSI/ASME NQA-1 does not contain some of the QA provisions and administrative controls governing operational phase activities that would be required in order to use ANSI/ASME NQA-1 in lieu of an Owner's 10 CFR Part 50, Appendix B, QA program description. The NRC originally endorsed ANSI/ASME NQA-1 with the knowledge that it was not entirely adequate and that other commitments such as the ANSI/ASME standards must supplement it. Hence, ANSI/ASME NQA-1 is not acceptable for use without the other QA program provisions identified in TS and licensee QA programs.

2.1.2.2 10 CFR 50.55a(b)(3)(ii)(A), (B), (C), and (D) —Motor-Operated Valve Testing

This condition requires that licensees establish a program to ensure that MOVs continue to be capable of performing their design-basis safety functions. The condition in 10 CFR 50.55a(b)(3)(ii) supplements the quarterly MOV stroke-time testing requirement in Subsection ISTC of the OM Code prior to its 2009 Edition. In 1989, the NRC recognized that quarterly stroke-time testing is not sufficient to provide assurance of MOV capability under design-basis conditions. For example, in GL 89-10, "Safety-Related Motor-Operated Valve

Testing and Surveillance," dated June 28, 1989, the NRC stated that stroke-time testing alone is not sufficient to provide assurance of MOV operability under design-basis conditions. Therefore, in GL 89-10, the NRC staff requested licensees to verify the design-basis capability of their safety-related MOVs and to establish long-term MOV programs. The NRC subsequently issued GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," dated September 18, 1996, to provide updated guidance for establishing long-term MOV programs. The condition in 10 CFR 50.55a(b)(3)(ii) establishes a regulatory requirement for nuclear power plant licensees implementing the applicable editions and addenda to the OM Code to establish programs to periodically assess the design-basis capability of MOVs within the scope of the IST program at nuclear power plants.

In response to the MOV issues, ASME published Code Case OMN-1 for the OM Code to allow users to replace quarterly MOV stroke-time testing with a combination of MOV exercising at least every refueling outage and MOV diagnostic testing on a longer interval. In RG 1.192, the NRC addresses the acceptability of Code Case OMN-1 in lieu of the quarterly MOV stroke-time testing requirements in Subsection ISTC of the editions and addenda of the OM Code prior to the 2009 Edition. The implementation of OM Code Case OMN-1 as accepted in RG 1.192 may be used in satisfying the requirement in 10 CFR 50.55a(b)(3)(ii). ASME has incorporated Code Cases OMN-1 and OMN-11 into Mandatory Appendix III in the 2009 Edition through the 2017 Edition to the OM Code to replace quarterly MOV stroke-time testing with periodic exercising and diagnostic testing. Based on its review, the NRC specifies the following conditions for implementation of Appendix III to the OM Code in 10 CFR 50.55a(b)(3)(ii):

(A) *MOV diagnostic test interval.* Licensees shall evaluate the adequacy of the diagnostic test intervals established for MOVs within the scope of ASME OM Code, Appendix III, not later than 5 years or three refueling outages (whichever is longer) from initial implementation of ASME OM Code, Appendix III.

(B) *MOV testing impact on risk.* Licensees shall ensure that the potential increase in core damage frequency and large early release frequency associated with the extension is acceptably small when extending exercise test intervals for high risk MOVs beyond a quarterly frequency.

(C) *MOV risk categorization.* When applying Appendix III to the ASME OM Code, licensees shall categorize MOVs according to their safety significance using the methodology described in ASME OM Code Case OMN-3, "Requirements for Safety Significance Categorization of Components Using Risk Insights for Inservice Testing of LWR Power Plants," subject to the conditions applicable to OMN-3 which are set forth in RG 1.192, or using an MOV risk ranking methodology accepted by the NRC on a plant-specific or industry-wide basis in accordance with the conditions in the applicable safety evaluation.

(D) *MOV stroke time.* 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.

2.1.2.2.a. 10 CFR 50.55a(b)(3)(iii)(A), (B), (C), and (D)—New Reactors

In addition to complying the provisions in the OM Code with conditions specified in 10 CFR 50.55a(b)(3), the holders of operating licenses for nuclear power reactors that received construction permits under this part on or after the date 12 months after August 17, 2017, and holders of COLs issued under 10 CFR 50.52, whose initial loading occurs on or after the date 12 months after August 17, 2017, shall comply with the following conditions:

(A) *Power-operated valves.* Licensees shall periodically verify the capability of power-operated valves to perform their design-basis safety functions.

(B) *Check valves.* Licensees must perform bidirectional testing of check valves within the IST program where practicable.

(C) *Flow-induced vibration.* Licensees shall monitor flow-induced vibration from hydrodynamic loads and acoustic resonance during preservice testing [PST] or inservice testing to identify potential adverse flow effects on components within the scope of the IST program.

(D) *High risk non-safety systems.* Licensees shall assess the operational readiness of pumps, valves, and dynamic restraints within the scope of the Regulatory Treatment of Non-Safety Systems for applicable reactor designs.

2.1.2.3 10 CFR 50.55a(b)(3)(iv)(A), (B), (C), and (D)—Check Valves (Appendix II)

This condition supplements the provisions in Mandatory Appendix II, "Check Valve Condition Monitoring Program," to the ASME OM Code. Subsection ISTC of the OM Code permits the use of Appendix II as an alternative to other testing or examination provisions of Subsection ISTC. If a licensee elects to use Appendix II, the provisions of Appendix II become mandatory in accordance with the OM Code requirements. Based on its review, NRC specifies the following conditions for implementation of Appendix II to the OM Code:

(A) *Check valves: First Provision.* Valve opening and closing functions must be demonstrated when flow testing or examination methods (nonintrusive, or disassembly and inspection) are use.

(B) *Check valves: Second Provision.* The initial interval for tests and associated examinations may not exceed two fuel cycles or 3 years, whichever is longer; any extension of this interval may not exceed one fuel cycle per extension with the maximum interval not exceed 10 years. Trending and evaluation of existing data must be used to reduce or extend the time interval between tests.

(C) *Check valves: Third provision*. If Appendix II condition monitoring program discontinued, then the requirement of ISTC 4.5.1 through 4.5.4 must be implemented.

(D) *Check valves: Fourth Provision*. The applicable provisions of subsection ISTC must be implemented if the Appendix II condition monitoring program is discontinued.

Licensees applying Appendix II, 1995 Edition with the 1996 and 1997 Addenda, shall satisfy the requirements of 10 CFR 50.55a(b)(3)(iv)(A), (B), and (C). Licensee applying Appendix II, 1998 Edition through the 2002 Addenda, shall satisfy the requirements of 10 CFR 50.55a(b)(3)(iv)(A), (B), and (D). Appendix II of the OM Code, 2003 Addenda through the 2015 Edition, is acceptable for use with the following requirements. Trending and evaluation shall support the determination that the valve or group of valves is capable of performing its intended function(s) over the entire interval. At least one of the Appendix II conditions monitoring activities for a valve group shall be performed on each valve of the group at approximate equal intervals not to exceed the maximum interval as shown in Table II of 10 CFR 50.55a(b)(3)(iv). The NRC regulations in 10 CFR 50.55a(b)(3)(iv) do not specify any conditions for the use of Appendix II in the 2017 Edition of the ASME OM Code.

The condition in 10 CFR 50.55a(b)(3)(iv)(A) applies to the testing or examination of the check valve obturator movement to both the open and closed positions to assess its condition and confirm acceptable valve performance. ASME approved the bidirectional testing of check valves for inclusion in the 1996 Addendum to the OM Code. The NRC agrees with the need for a required demonstration of the bidirectional exercising movement of the check valve disk. The single direction flow testing of check valves will not always detect degradation of the valve. The classic example of this faulty testing strategy is that separation of the disk would not be detected during forward flow tests. The separated disk could be lying in the valve bottom or another part of the system and could move to block flow or disable another valve. Appendix II did not require bidirectional testing of check valves in the 1996 through 2002 Addenda to the OM Code. Hence, the condition in 10 CFR 50.55a(b)(3)(iv)(A) was included so that an Appendix II condition monitoring program includes bidirectional testing of check valves to assess their condition and confirm acceptable valve performance (as is required by the OM Code).

The condition in 10 CFR 50.55a(b)(3)(iv)(B) applies to the length of the check valve test interval. Appendix II would permit a licensee to extend check valve test intervals without limit. A policy of prudent and safe interval extension dictates that any interval extension must be based on sufficient experience to justify the additional time. Condition monitoring and current experience may qualify some valves for an initial extension, whereas the trending and evaluation of the data may dictate reduction in the testing interval for some valves. Extensions of IST intervals must consider plant safety and be supported by the trending and evaluation of both generic and plant-specific performance data to ensure that the component is capable of performing its intended function over the entire IST interval. Thus, the condition in 10 CFR 50.55a(b)(3)(iv)(B) limits the time between the initial test or examination and the second test or examination to two fuel cycles or 3 years (whichever is longer), with any extension of this interval not to exceed one fuel cycle per extension with the maximum interval not to exceed 10 years. An extension or reduction in the interval between tests or examinations would have to be supported by trending and evaluation of performance data.

The condition in 10 CFR 50.55a(b)(3)(iv)(C) applies to a licensee who discontinues a condition monitoring program when using the 1995 Edition of the OM Code with the 1996 and 1997 Addenda. A licensee who discontinues the use of Appendix II is required to implement the requirements of Subsections ISTC 4.5.1 through ISTC 4.5.4 of the 1995 Edition of the OM Code with 1996 and 1997 Addenda.

The condition in 10 CFR 50.55a(b)(3)(iv)(D) applies to a licensee who discontinues a condition monitoring program when using the 1998 Edition through the 2002 Addendum to the OM Code. A licensee who discontinues the use of Appendix II is required to implement the applicable provisions in Subsection ISTC of the 1998 Edition through the 2002 Addenda of the OM Code.

2.1.2.4 10 CFR 50.55a(b)(3)(vi) —Exercise Interval for Manual Valves

This condition requires that manual valves must be exercised on a 2-year interval rather than the 5-year interval specified in paragraph ISTC-3540 of the 1999 through 2005 Addenda to the

ASME OM Code, provided that adverse conditions do not require more frequent testing. The 1998 Edition and earlier versions of the OM Code specified an exercise interval of 3 months for manual valves. The 1999 Addendum to the OM Code revised paragraph ISTC-3540 to extend the exercise frequency for manual valves to 5 years; however, the NRC staff did not agree that there was sufficient justification to extend the exercise interval for manual valves to 5 years. See *Federal Register* Notice 67 FR 60520, 60531-32 (dated September 26, 2002). The condition in 10 CFR 50.55a(b)(3)(vi) does not apply to the 2006 Addendum through the 2017 Edition to the OM Code because ASME revised the exercise interval in paragraph ISTC-3540 of the 2006 Addendum to the OM Code to 2 years for manually-operated valves.

2.1.2.5 10 CFR 50.55a(b)(3)(vii) – Use of Subsection ISTB, 2011 Addenda of OM Code

This condition is required because in Subsection ISTB, "Inservice Testing of Pumps in Water-Cooled Reactor Nuclear Power Plants – Pre-2000 Plants," of the 2011 Addenda to the OM Code, the upper end of the "Acceptable Range" and "Required Action Range" for flow and differential or discharge pressure for comprehensive pump testing (CPT) was raised to higher values. The NRC staff on the ASME OM Code committees accepted the proposed increase of the upper end of the "Acceptable Range" and "Required Action Range" on the basis of a planned additional requirement for a pump periodic verification test program in the OM Code. However, the 2011 Addenda to the OM Code did not include a pump periodic verification test program. Since then, the 2012 Edition of the OM Code has incorporated Mandatory Appendix V, "Pump Periodic Verification Test Program," which supports the changes to the CPT in Subsection ISTB. Therefore, 10 CFR 50.55a(b)(3)(vii) prohibits the use of the Subsection ISTB to the 2011 Addenda of the OM Code. Licensees will be allowed to apply Subsection ISTB with the revised acceptable and required action ranges in the 2012 through 2017 Editions of the OM Code as incorporated by reference in 10 CFR 50.55a.

2.1.2.6 10 CFR 50.55a(b)(3)(viii) - Subsection ISTE

This condition requires that licensees who wish to implement Subsection ISTE, "Risk-Informed Inservice Testing of Components in Water-Cooled Reactor Nuclear Power Plants," of the OM Code, 2009 through 2017 Editions, must request and obtain NRC authorization in accordance with 10 CFR 50.55a(z) to apply Subsection ISTE on the plant-specific basis as a risk-informed alternative to the applicable IST requirements in the OM Code.

Nuclear power plant applicants for construction permits under 10 CFR Part 50, or COLs for construction and operation under 10 CFR Part 52, may describe their proposed implementation of the risk-informed IST approach specified in Subsection ISTE of the OM Code for NRC review in their application.

2.1.2.7 10 CFR 50.55a(b)(3)(ix) - Subsection ISTF

This condition is for applying Subsection ISTF, "Inservice Testing of Pumps in Water-Cooled Reactor Nuclear Power Plants - Post-2000 Plants," of the 2012 and 2015 Editions of the OM Code, and is similar to the previous condition in 10 CFR 50.55a(b)(3)(vii) for pre-2000 plants. While using Subsection ISTF of the OM Code, 2012 Edition, licensees must satisfy the requirements of Mandatory Appendix V of the OM Code, 2012 and 2015 Editions. However, the 2011 Addenda to the OM Code did not include a pump periodic verification test program. Since then, the 2012 Edition of the OM Code has incorporated Mandatory Appendix V, which supports the changes to the CPT in Subsection ISTF. Therefore, 10 CFR 50.55a(b)(3)(ix) prohibits the use of Subsection ISTF to the 2011 Addenda of the OM Code. Licensees will be allowed to apply Subsection ISTF with revised acceptable and required action ranges in the 2012 through 2015 Editions of the OM Code as incorporated by reference in 10 CFR 50.55a. The 2017 Edition of the OM Code incorporates Appendix V into Subsection ISTF such that the condition is not necessary when implementing the 2017 Edition of the OM Code.

2.1.2.8 10 CFR 50.55a(b)(3)(xi) - Valve Position Indication

This condition requires that licensees who are implementing paragraph ISTC-3700, "Position Verification Testing," in the ASME OM Code, 2012 Edition through the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a(a)(1)(iv), shall 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 proper obturator position for valves with remote position indication within the scope of Subsection ISTC including its mandatory appendices and their verification methods and frequencies.

2.1.3 Voluntary Use of Later Editions and Addenda to the ASME OM Code

The NRC regulations in 10 CFR 50.55a(f)(4) and (g)(4) establish the effective edition and addenda of the ASME OM Code or ASME *Boiler and Pressure Vessel Code* (BPV Code), as applicable, to be used by licensees in performing inservice testing of pumps and valves and inservice inspections of components (including supports).

10 CFR 50.55a(f)(4)(ii) and (g)(4)(ii) require the use of the latest edition and addenda of the applicable ASME Code that have been incorporated by reference 18 months prior to the beginning of each 120-month IST and ISI program interval. The ASME Code edition or addenda identified as applicable to the 120-month IST and ISI program interval is considered to be the "Code of Record" for the IST and ISI program interval.

10 CFR 50.55a(f)(4)(iv) and (g)(4)(iv) allow IST of pumps and valves and ISI of components (including supports) to meet the requirements set forth in subsequent editions and addenda to the "Code of Record" that are incorporated by reference in 10 CFR 50.55a(a), subject to conditions listed in 10 CFR 50.55a(b) and subject to NRC approval. The fact that these ASME Code editions and addenda have been incorporated by reference into the NRC regulations does not imply that NRC approval has already been granted for their use in lieu of the "Code of Record" for the specific licensee. If a licensee plans to apply a later edition or addenda of the ASME Code either at the outset or during its 120-month IST or IST interval, the licensee must request NRC approval to use the later ASME Code editions and addenda via a letter to the NRC. The licensee may not update the IST and ISI programs without NRC approval to use the later ASME Code editions and addenda. When proposing to use any or all of a later Code edition or addenda, the licensee must meet all related requirements of the respective editions or addenda, and any deviations are subject to NRC approval in accordance with 10 CFR 50.55a(f)(4)(iv) and (g)(4)(iv). The regulations do not specify when the licensee should submit the request to apply a later edition or addendum of the ASME Code. The regulations only specify that the licensee must receive NRC approval before use of a later ASME Code edition or addendum.

The amount of written documentation needed to support a request to use a later ASME Code edition and addendum that 10 CFR 50.55a(a) incorporates by reference is significantly less than that necessary for other types of requests for relief from or alternatives to the ASME Codes. For example, licensees are not required to provide specific justification for requests to use later

ASME Code editions and addenda that 10 CFR 50.55a(a) incorporates by reference. This is because the NRC has reviewed and accepted the provisions of those ASME Code editions or addenda, with any appropriate conditions, as part of the process for incorporation of the edition and addenda by reference in the regulations.

The purpose of the NRC review and approval for the use of later ASME Code editions and addenda is to provide and monitor consistency in the use of the appropriate ASME Code editions and addenda in IST and ISI activities for nuclear plant components.

When requesting to use editions and addenda of the ASME Code that have not been incorporated by reference into 10 CFR 50.55a, licensees must first request authorization to use these later editions and addenda as an alternative to the regulations pursuant to 10 CFR 50.55a(z). The NRC staff issued RIS 2004-12 and RIS 2004-16, “Use of Later Editions and addenda to the ASME Code Section XI for Repair/Replacement Activities,” dated October 19, 2004, to clarify this matter.

2.1.4 Identification of Code Noncompliance

NRC Inspection Manual Chapter (IMC) 0326 (September 30, 2019), “Operability Determinations,” provides the following guidance for Technical Specification (TS) operability determinations (ODs) versus ASME OM Code criteria:

> The TS normally applies to the overall performance of plant systems, but sometimes contains limiting values for the performance of certain components. The limiting values are specified to ensure that the operational limits established by the design basis and safety analysis are satisfied. The values (e.g., pump flow rate, valve closure time, valve leakage rate, safety/relief valve set point pressure) are criteria that can be used to verify operability. If at any time these values are not met, the system must be declared inoperable, the LCO [limiting condition for operation] must be declared not met, and the applicable conditions must be entered.
>
> The ASME OM Code establishes the requirements for preservice and inservice testing and the examination of certain components to assess their operational readiness. ASME OM Code acceptance criteria for IST activities include “required action ranges” or limiting values for certain component performance parameters. These required action ranges or limiting values, defined by the ASME OM Code as component performance parameters, may be more limiting than the TS values (which are accident analysis limits). Where IST requirements are incorporated into a facility’s surveillance requirements when performance data falls outside the required action range, regardless of whether the limit is equal to the TS limit or more restrictive, the surveillance requirement is not met (the word “inoperative” is used in the text of the ASME Code, i.e., the pump or valve is “inoperative”) and the LCO must be declared not met and the applicable conditions must be entered.
>
> When the required action range is more limiting than its corresponding TS, the corrective action need not be limited to replacement or repair; an analysis demonstrating the specific performance degradation does not impair operability would be acceptable. A new required action range may be established after such analysis, allowing a new OD.
>
> The NRC does not accept durations specified by the ASME OM Code for analyzing test results as a reason for postponing entry into a TS action statement. As soon as data are recognized as being within the required action range for pumps or as exceeding the

limiting-value full-stroke time for valves, the associated component must be declared inoperable, and if subject to TS, the completion time specified in the action statement must be started at the time the component was declared inoperable. For inoperable pumps and valves that are part of an ASME IST program but not subject to TS, only the actions required by the applicable sections of the ASME code are applicable.

Recalibrating test instruments and then repeating pump or valve tests are acceptable as an alternative to repair or replacement but cannot be done before declaring the pump or valve inoperable. However, if during a test it is obvious that a test instrument is malfunctioning, the test may be halted and the instruments promptly recalibrated or replaced. During a test, anomalous data with no clear indication of the cause must be attributed to the pump or valve under test. In that case, the licensee should evaluate to determine if this condition results in the loss of the presumption of operability and if so enter the OD process.

2.1.5 ASME OM Code Interpretations

The ASME issues “Interpretations” to clarify provisions of the OM Code. Users submit requests for interpretation and, after appropriate committee deliberations and balloting, ASME issues responses. Interpretations do not follow the same approval process as OM Code and Code Cases. The Code interpretations provide the meaning or the intent of the existing requirements in the OM Code. Licensees should exercise caution when applying interpretations as they are not specifically part of the OM Code’s incorporation by reference into 10 CFR 50.55a and have not received NRC approval. The NRC recognizes that the ASME is the official interpreter of the OM Code, but the NRC will not accept ASME interpretations that, in the NRC’s opinion, are contrary to NRC requirements or may adversely impact facility operations.

2.1.6 Impractical versus Impracticable

The terms “impractical” and “impracticable” are frequently used interchangeably in discussing IST program activities. Some of the definitions of these two terms are similar. Merriam-Webster defines impractical as not wise to put into practice, and impracticable as incapable of being performed. With respect to IST program activities, the NRC staff uses a reasonableness standard in applying these terms. Licensees should justify an activity as impractical or impracticable with respect to whether it is reasonable to expect the activity to be performed. For example, the construction of a structure requiring significant resources to enable an IST activity to be performed on the specific OM Code test interval might be considered impractical or impracticable if the activity could be performed readily on a slightly extended interval without significant degradation in the operational readiness of the pump, valve, or snubber to perform its safety function. The NRC regulations in 10 CFR 50.55a(f)(5) and (6) address relief requests where a licensee considers an ASME OM Code requirement to be impractical for its facility. The condition in 10 CFR 50.55a(b)(3)(iii)(B) for bidirectional testing of check valves in new reactors refers to practicable because it relates to the design capability to allow bidirectional testing of those valves.

2.2 Scope of Inservice Testing Programs

General Design Criterion (GDC) 1, “Quality Standards and Records,” in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, and Criterion XI, “Test Control,” in Appendix B to 10 CFR Part 50, require that all components (such as pumps and valves) that are important to safety be tested to demonstrate that they will perform satisfactorily in service. Among other things, GDC 1 requires that components that are important to safety must be tested to quality standards that are commensurate with the importance of the safety functions to be performed. Criterion XI requires, in part, that a test program shall be established to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures. For pre-GDC plants, the licensee should verify the scope of the IST program is consistent with the applicable licensing requirements.

Each licensee has the responsibility to demonstrate the continued operability or functionality of all components within the scope of its IST program. The NRC staff considers the ASME OM Code to apply to pumps, valves, and dynamic restraints installed in piping systems. For example, the NRC staff does not consider the OM Code to apply to rupture panels in building structures. An IST program, including implementing procedures, is subject to the requirements of 10 CFR Part 50, Appendix B, and OM Code, Subsection ISTA. Changes to the scope, test methods, or acceptance criteria should be reviewed for compliance with the requirements of 10 CFR 50.59, 10 CFR 50.55a, and 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants” (Maintenance Rule), as appropriate.

The TS for some plants may include IST requirements that are more restrictive than the NRC regulations.

2.2.1 Basis for Scope Requirements

The requirements for the scope of components to be included in an IST program are addressed in 10 CFR 50.55a(f)(4), and paragraph ISTA-1100, “Scope,” of the ASME OM Code.

The NRC regulations in 10 CFR 50.55a(f)(4), “Inservice testing standards requirement for operating plants,” state the following:

> Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves that are within the scope of the ASME OM Code must meet the inservice test requirements (except design and access provisions) set forth in the ASME OM Code and addenda that become effective subsequent to editions and addenda specified in paragraphs (f)(2) and (3) of this section and that are incorporated by reference in paragraph (a)(1)(iv) of this section, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The inservice test requirements for pumps and valves that are within the scope of the ASME OM Code but are not classified as ASME BPV Code Class 1, Class 2, or Class 3 may be satisfied as an augmented IST program in accordance with paragraph (f)(6)(ii) of this section without requesting relief under paragraph (f)(5) of this section or alternatives under paragraph (z) of this section. This use of an augmented IST program may be acceptable provided the basis for deviations from the ASME OM Code, as incorporated by reference in this section, demonstrates an acceptable level of quality and safety, or that implementing the Code provisions would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, where documented and available for NRC review.

The OM Code defines its scope by stating that IST programs shall include pumps and valves that are required to perform a specific function in (1) shutting down the reactor to a safe shutdown condition, (2) maintaining the safe shutdown condition, or (3) mitigating the consequences of an accident. The scope of the OM Code also covers pressure relief devices that are used to protect systems (or portions of systems) that perform a required safety-related function. Therefore, the scope of components to be included in an IST program must encompass the components covered in Subsection ISTA of the OM Code. The IST scope for dynamic restraints is discussed in Appendix A to this NUREG.

With respect to non-Code Class safety-related pumps and valves, the regulations in 10 CFR 50.55a(f)(4) specify that the IST requirements for pumps and valves that are within the scope of the ASME OM Code but are not classified as ASME BPV Code Class 1, Class 2, or Class 3 may be satisfied in an augmented IST program in accordance with 10 CFR 50.55a(f)(6)(ii) without requesting relief under 10 CFR 50.55a(f)(5) or alternatives under 10 CFR 50.55a(z). As an example, a licensee or applicant could specify the augmented IST program activities in the IST Program Plan with the basis for those activities indicated in the IST program tables similar to the justification for cold shutdown and refueling outage test frequencies.

Paragraph ISTA-1100 of the OM Code refers to components that are "needed to mitigate the consequences of an accident." The NRC regulations in 10 CFR 50.2, "Definitions," defines safety-related SSCs as follows:

> Safety-related structures, systems and components means those structures, systems and components [SSCs] that are relied upon to remain functional during and following design basis events to assure:
>
> (1) The integrity of the reactor coolant pressure boundary,
>
> (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
>
> (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

The language of ISTA-1100 and 10 CFR 50.2 is consistent with respect to mitigating the consequences of accidents with the exception that ISTA-1100 does not include specific references to exposure guidelines provided in the NRC regulations. Therefore, the NRC staff interprets paragraph ISTA-1100, in specifying pumps and valves that are required to perform a specific function in (1) shutting down the reactor to a safe shutdown condition, (2) maintaining the safe shutdown condition, or (3) mitigating the consequences of an accident, to be consistent with the definition of safety-related as specified in the NRC regulations. Licensees and applicants could use their identification of the safety-related pumps and valves with respect to design basis events in the Final Safety Analysis Report (FSAR) for their nuclear power plants to identify the pumps and valves within the scope of paragraph ISTA-1100. Similarly, other events such as anticipated transient without scram (ATWS), station blackout (SBO), and low temperature overpressure protection (LTOP) might have NRC requirements for components relied upon to mitigate their events that are safety-related and not within the scope of the OM Code.

The NRC staff notes that the regulations include requirements for a wide range of potential adverse events at nuclear power plants. For example, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to 10 CFR Part 50 provides specific requirements for mitigating fire events at nuclear power plants. Appendix R specifies requirements for plant SSCs that might be relied upon to mitigate fire events, including SSCs that are not classified as safety-related at a nuclear power plant. The NRC staff does not consider that a pump or valve relied on only to satisfy the Appendix R requirements must also be included in the IST program required by the OM Code as incorporated by reference in 10 CFR 50.55a.

Components within the scope of 10 CFR 50.55a are included in the scope of 10 CFR 50.65. Licensees may elect to consolidate testing for pumps and valves, designating any non-Code components as such in the IST program.

Table 2.1, "Typical Systems and Components in an Inservice Testing Program for a Pressurized-Water Reactor," and Table 2.2, "Typical Systems and Components in an Inservice Testing Program for a Boiling-Water Reactor," (which appear at the end of this chapter) provide examples of systems and components that licensees typically include in their IST programs. These tables are not intended to be all-inclusive, but they may form the basis for the initial review of a licensee's IST program scope.

Figure 2.1, "Flow Chart - Development of Inservice Testing Program for Pumps and Valves," (which appears at the end of this chapter) provides a quick reference to regulatory requirements for development of an IST program for pumps and valves. For complete details, see 10 CFR 50.55a.

2.2.2 Examples of Omitted Components

During IST program reviews and inspections, the NRC staff has noted that licensees do not always include the necessary equipment in the scope of their IST programs. Licensees should review their IST programs to ensure adequate scope. Components that are frequently erroneously omitted from IST programs include the following examples:

(a) BWR scram system valves

(b) control room chilled-water system pumps and valves

(c) accumulator vent valves or motor-operated isolation valves

(d) auxiliary pressurizer spray system valves

(e) boric acid transfer pumps

(f) valves in the emergency boration flow path

(g) control valves that have a required fail-safe position

(h) valves in mini-flow lines

(i) control rod drive (CRD) system check valves

(j) keep fill systems

(k) excess flow check valves

Licensees should review whether these components meet the scoping requirements, and the safety significance of these components, to ensure that the IST activities are adequate to demonstrate their continued availability. Licensees should also recognize that the pumps and valves listed above do not apply to every plant and do not satisfy the scope required by

Subsection ISTA for all plants. For example, items c, d, e, and f do not apply to BWRs. Each licensee should review the list and determine which items apply to its facility.

2.2.3 Testing of Additional Components

As discussed above, licensees are required to test safety-related components to demonstrate that they will perform satisfactorily in service in accordance with 10 CFR Part 50, Appendices A and B. The IST program and augmented IST program for pumps, valves, and dynamic restraints within the scope of the OM Code are addressed in 10 CFR 50.55a.

An IST program is also a reasonable vehicle to periodically demonstrate the operational readiness of pumps and valves that are not covered by the ASME OM Code. Thus, if a licensee voluntarily chooses to include non-Code components in its OM Code IST program (or some other licensee-developed testing program) and, as a result, is unable to meet certain OM Code provisions for these components, the regulations (10 CFR 50.55a) do not require the licensee to submit a relief or alternative request to the NRC.

Nonetheless, the licensee should maintain documentation that provides assurance of the continued operational readiness, or as required the continued functionality of the non-Code components through the performed tests.

2.2.4 Components in IST Programs

The licensee is responsible for determining whether a component is required to be included within its IST program in compliance with 10 CFR 50.55a(f) and paragraph ISTA-1100 of the ASME OM Code, or whether that classification is optional under paragraph ISTA-1320. Specifically, paragraph ISTA-1320 states that optional construction of a component within a system boundary to a classification higher than the minimum class established in the component design specification shall not affect the overall system classification by which applicable requirements of IST are determined. Thus, if a licensee changes the Code classification pursuant to 10 CFR 50.59, the pumps and valves may remain as “augmented components” (denoted as non-Code) in the IST program. (Note that NRC approval of a licensee amendment may be necessary, as determined by the evaluation conducted in accordance with 10 CFR 50.59.) RG 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” (ADAMS Accession No. ML17195A655) provides guidance for 10 CFR 50.59 implementation.

2.2.5 Components Added to IST Programs

The ASME OM Code establishes the requirements for preservice and inservice testing and examination of certain components to assess their operational readiness in nuclear power plants. Licensees have the responsibility to identify all components subject to test or examination within the scope of their IST program as required by ISTA-1500(a). During the life of an operating nuclear power plant, components may be added or deleted from the scope of the IST program due to plant modifications, changes in procedures, or changes in engineering safety analyses.

Each component that has been identified to be within the IST program scope is required by the NRC regulations to have a test plan prepared for the PST period, initial IST interval, and subsequent IST intervals. During the PST period, an initial set of reference values need to be determined under conditions as near as practical to those expected during subsequent inservice testing. Section ISTA-2000, "Definitions," in the OM Code defines the PST period as "the period of time following completion of construction activities related to the component and before first electrical generation by nuclear heat, in which component and system testing takes place, or in an operating plant prior to the component being initially placed in service." For components added to the scope of IST program, it is expected that a preservice test will be performed at the time when the component is added to the IST program. In addition, the component added to the IST program should undergo an engineering analysis review to determine if the component would require a verification test at its expected design parameters for performance of its safety function in accordance with 10 CFR 50.55a.

2.3 Systems Containing Safety-Related Pumps and Valves

The plant FSAR, TS, and other documents list the systems and components that are necessary to function to support the safe operation and shutdown of the plant. Tables 2.1 and 2.2 in this chapter list systems and components typically included in the IST programs for PWRs and BWRs. These tables are not intended to apply to all plants. The listed systems and components are not considered safety-related at every plant, and are not necessarily classified as ASME BPV Code Class 1, 2, or 3. For information on quality group and Code classifications, see RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition." The licensee's FSAR generally contains a section describing the Code classification of components. The IST program scope should be developed to be consistent with the FSAR.

2.4 IST Program Document

Within this discussion of the IST program document, Section 2.4.1 applies to pumps, and Section 2.4.2 applies to valves. These sections describe the information that licensees generally need in order to prepare the related sections of the IST program document.

The OM Code includes the following:

- Subsection ISTA includes general requirements for testing components.
- Subparagraph ISTA-3200(a) states that the Owner shall file IST plans with the regulatory authorities having jurisdiction at the plant site.
- Section ISTA-9000 addresses the records and reports that are required for these inspection and testing programs, and retention of these records.
- Paragraph ISTA-9210 states that the Owner shall prepare plans for preservice and inservice examinations and tests to meet the requirements of the OM Code.
- Paragraph ISTA-9220 states that licensees shall prepare test and examination records in accordance with the requirements of the respective articles of the OM Code.
- Sections ISTB-9000 and ISTC-9000 include additional guidance for the information that the IST program document must include for pumps and valves that perform safety functions.

- Nonmandatory Appendix A, "Preparation of Test Plans," and Supplement to Appendix A, "Preparation of Test Plans," to the OM Code gives voluntary guidance for licensees to use in preparing their inspection and test plans.

Licensees have found that pump and valve tables are a convenient format for the information. These tables typically include sufficient information to allow NRC inspectors to determine whether the testing complies with the OM Code requirements for test method and frequency. The tables also could note applicable NRC positions or recommendations for each pump or valve.

The NRC intends that the IST program should reflect design modifications and other activities performed under 10 CFR 50.59 that relate to pumps and valves within the scope of the program. Thus, the NRC staff recommends that the program plan submitted to the NRC should include documentation of the use of positions contained herein, and Code Cases.

In the 2020 Edition of the ASME OM Code, ASME deleted the requirement of ISTA-3200(a) to submittal of the IST Program Plan to regulatory authorities as more appropriate as a regulatory requirement rather than an ASME requirement. The NRC staff is considering the proposal of a regulatory requirement for submittal of the IST Program Plan by nuclear power plant licensees in a future 10 CFR 50.55a rulemaking.

2.4.1 Pumps

In preparing pump tables, licensees should consider the following information, which includes headings and a description of the text that licensees could include under each heading.

Title: List the applicable plant and unit.

Page number: Include the page number and total number of pages in the program document or the relevant section, such as "Page 15 of 135."

Program revision or revision date: List the program or page revision number and date (on each page). List the revision number for each program change submitted.

System, Code class, and group: List the plant system, Code class, and pump group, and briefly describe the service of the pump.

Pump identification: List a unique identifier for each pump. This identifier should be used consistently in all IST program documentation and design information such as system piping and instrument diagrams (P&IDs), test procedures, and relief requests.

P&ID number: List the applicable P&ID or figure that depicts the pump in the system.

Drawing coordinates: List the coordinates of the pump on the P&ID.

Test parameters: List each of the five parameters in Tables ISTB-5121-1, ISTB-5221-1, ISTB-5321-1, and ISTB-5321-2 (or similar ISTF tables) for each pump. A column or a footnote is typically used to list factors that affect testing. List a relief request or ASME OM Code Case number where the testing will not be performed in accordance with the OM Code. Notes can be used where OM Code testing would otherwise be required. A relief request is not required if the test requirement is exempted by the OM Code.

Relief request(s): List any applicable relief requests in the pump table. Table 2.3, "Example Data Table for Pumps," provides an example of a data table for pumps.

Notes: Notes at the end of the pump table can be used to clarify IST activities for specific pumps. Notes may also be used to describe augmented testing activities for non-Code pumps consistent with 10 CFR 50.55a(f)(4).

2.4.2 Valves

In preparing valve tables, licensees should consider the following information, which includes headings and a description of the text that licensees could include under each heading. Table 2.4, "Useful Abbreviations for Valve Data," lists common abbreviations used in valve data tables.

Title: List the applicable plant and unit.

Page number: Include the page number and total number of pages in the program document or the relevant section, such as "Page 15 of 135."

Program revision or revision date: List the program or page revision number and date (on each page). List the revision number for each program change submitted.

System, Code class, and group: List the plant system, Code class, and valve group, and briefly describe the service of the valve.

Valve identification: List a unique identifier for each valve. This identifier should be used consistently in all IST program documentation and design information such as P&IDs, test procedures, and relief requests. If valves such as excess flow check valves are grouped together in the table, the number of valves and the valve number must be clearly indicated.

P&ID number: List the applicable P&ID or figure that depicts the valve in the system.

Drawing coordinates: List the coordinates of the valve on the P&ID.

Valve type: List the valve type (e.g., gate, globe, check, or relief).

Valve size: Specify the valve size in inches, fractions of an inch, or metric units.

Actuator type: List the type of valve actuator (e.g., motor, solenoid, pneumatic, hydraulic, or self) with the type and function of each valve.

Code category: Specify the Code category (or categories), as defined in paragraph ISTC-1300. This determines the applicable subsections of the OM Code. For example, a motor-operated gate valve could be in OM Code Category A or B, while a self-actuated check valve could be in Category C or A/C.

Active/Passive: State whether a valve is active or passive, as defined in Section ISTA-2000 in the OM Code. Section ISTA-2000 in the OM Code defines active valves as "valves that are required to change obturator position to accomplish a specific function in shutting down a reactor to the safe shutdown condition, maintaining the safe shutdown condition, or mitigating the consequences of an accident." Section ISTA-2000 defines passive valves as "valves that maintain obturator position and are not required to change obturator position to accomplish the required function(s) in shutting down a reactor to the safe shutdown condition, maintaining the safe shutdown condition, or mitigating the consequences of an accident." Therefore, a valve is defined as either active or passive when initially included in the IST program based on its required safety function. A valve need not be classified as active if it is only temporarily removed from service or from its safety position, such as manually opening a sample valve for a short time to take a sample, while maintaining administrative control over the valve. A passive valve removed from service or in its non-safety position for days, weeks, or months is expected to be classified as an active valve and satisfy the applicable OM Code requirements as incorporated by reference in 10 CFR 50.55a. If the plant is in an operating mode that does not require a passive valve to be maintained in its "passive" (safety) position, the position of the valve may be changed without imposing IST requirements on the valve. By contrast, if a valve is routinely repositioned during power operations (or has an active safety function), it is an active valve. If a valve is repositioned to create a new alignment (e.g., as a corrective action for a condition of another valve in the line), an evaluation (considering the impact on the IST program) may be necessary to ensure its operational readiness before placing the valve in a new position, as determined on a case-by-case basis.

In preparing Revision 3 to NUREG-1482, the NRC staff was requested to provide additional guidance for the classification of a valve as active or passive. In response, the NRC staff expanded this section of NUREG-1482. In summary, the NRC staff notes that a valve is classified as active or passive by its safety function. A licensee may not change the classification of a valve simply by establishing administrative controls over its position. Containment isolation valves (except where always closed) are always considered to be active components based on their safety function to prevent release of radioactive material from containment. Check valves are always considered to be active valves in the IST program.

As defined in ASME OM Code, Subsection ISTA, paragraph ISTA-2000, passive valves maintain their obturator position and are not required to change obturator position to accomplish their required function. By this definition, passive valves that remain in their safety position upon loss of motive power and do not need to be fail-safe tested as required by the ASME OM Code.

<u>Safety position</u>: List the safety function position(s), and specify both positions for valves that perform a safety function in both the open and closed positions. Valves must be exercised to the position(s) required to fulfill their safety function(s). Check valve tests must include both open and close tests.

<u>Tests performed</u>: Specify which tests are to be performed on each valve.

<u>Test frequency</u>: List the actual frequency for each test to be performed. If it would be impractical (rather than a convenience consideration) or burdensome to perform the test at the frequency specified in the Code, reference cold shutdown or refueling outage justifications or relief requests for the alternative test frequency.

<u>Relief requests and cold shutdown/refueling outage justifications</u>: List any applicable relief request(s). In addition, when the testing is deferred to cold shutdowns or refueling outages, reference the technical justification for the test frequency. Notes at the end of the valve table may be used to provide the technical justification.

Remarks: Include any pertinent information that is not stated elsewhere in the table such as a brief functional description of the valve.

Notes: Notes at the end of the valve table can be used to clarify IST activities for specific valves. Notes may also be used to describe augmented testing activities for non-Code valves consistent with 10 CFR 50.55a(f)(4).

2.4.3 Piping and Instrument Diagrams and Drawings

The NRC staff recommends that licensees' program submittals should include P&IDs or system drawings to assist in locating the pumps and valves that are included in the program and such drawings should be the latest revision at the time the program is submitted to the NRC. This information will assist the NRC staff in reviewing relief requests or proposed alternatives. Inservice inspection boundary system drawings and isometrics, or reduced-size drawings, are suitable for inclusion in the program document. If the reduced-size drawings are not complete P&IDs, the NRC staff may request a set of full-size drawings for use in evaluating relief requests. A partial submittal of the program containing relief requests could include applicable drawings to support the relief requests or to supersede previous IST program drawings. Licensees need not update their program drawings regularly, but if drawings change because of modifications, or if the changes affect relief requests, the NRC staff recommends that licensees revise and resubmit the drawings in the next periodic submittal of revisions to the IST program document. The NRC staff also recommends that licensees include detailed drawings with relief request submittals and submittals to supplement the IST program. Such drawings are helpful because the NRC's technical staff who review relief requests do not maintain a set of FSARs for each plant. Drawings are helpful in reviewing relief requests, regardless of whether they are submitted as part of the program document or as an attachment to a relief request or proposed alternative.

2.4.4 Bases Document

The NRC staff recommends that each licensee create a bases document for the IST program. A paper discussing the creation and management of a bases document is included in Supplement 1 to NUREG/CP-0123, "Proceedings of the Second NRC/ASME Symposium on Pump and Valve Testing," dated November 1992. Bases documents have typically included a description of the methods used in preparing the IST program, with a list of each pump and valve in a system within the scope of the ASME OM Code, the basis for including (or excluding) the pump or valve, and the basis for the testing applied to each component. Although not required by the NRC, the bases document may help licensees ensure the continuity of their IST programs when the responsibilities of personnel or groups change. A bases document will also enable the plant staff to clearly understand the reasons that the components are either in the program or not, as well as the basis for testing (or not testing) certain functions. Although not a "licensing-basis document" (unless the licensee takes action to make the document part of the licensing basis for a plant), the bases document is a useful reference for licensee reviews performed under 10 CFR 50.59 when changes are being considered for a facility.

2.4.5 Deferring Valve Testing to Cold Shutdown or Refueling Outages

Exercising valves on a cold shutdown or RFO frequency does not constitute a deviation from the ASME OM Code. Paragraph ISTC-3520 provides guidance for testing valves during cold shutdown or RFOs if it is impractical to test during operation. The licensee should list the affected valves in the program document (or include in the table) and include cold shutdown or

RFO justifications for each affected valve or group of valves. The NRC staff recommends that licensees should include these cold shutdown and RFO justifications in their IST program submittals to the NRC.

Check valves that can be stroked quarterly, but must be monitored by a nonintrusive technique to verify full stroke, may be full-stroke tested during cold shutdown or RFOs if another method of verifying full-stroke exists during such plant conditions. The NRC would not require a licensee to invest in nonintrusive equipment for the purpose of testing check valves quarterly (instead of testing them during cold shutdown or RFOs), even though the use of nonintrusive techniques is recommended where practical.

A licensee may request relief from quarterly testing where such testing would impose a hardship (e.g., entering a limiting condition for operation of 3 to 4 hours in duration, repositioning a breaker from “off” to “on” and necessitating manual operator actions to restore the system if an accident were to occur while the test was in progress). For such situations, the risk associated with quarterly testing may outweigh the benefits that might otherwise be achieved. (Section 3.1.2 provides guidance on these types of situations.) Thus, it is appropriate for licensees to weigh the safety impact against the benefits of testing as a basis for deferring testing from a quarterly frequency to cold shutdown or RFOs. NUREG/CR-5775, “Quantitative Evaluation of Surveillance Test Intervals Including Test-Caused Risks,” dated February 1992 (ADAMS Accession No. ML19172A254) describes a method for making this comparison.

In the event of a planned or unplanned maintenance outage, a licensee may decide to test some or all valves in a cold shutdown mode, rather than waiting for the RFO. In making this decision, the licensee should consider the duration of the shutdown and the extent of other outage activities. The requirements of Subsection ISTC for testing valves in systems that are out of service may apply for extended outages that last for several months. Guidance on minimizing shutdown risk also may apply for extended outages.

Impractical conditions justifying test deferrals may include the following situations that could result in an unnecessary plant shutdown, cause unnecessary challenges to safety systems, place undue stress on components, cause unnecessary cycling of equipment, or unnecessarily reduce the life expectancy of the plant systems and components:

- inaccessibility
- testing that would require major plant or hardware modifications
- testing that has a high potential to cause a reactor trip
- testing that could cause system or component damage
- testing that could create excessive plant personnel hazards
- existing technology that will not give meaningful results

In the licensing process, the NRC staff weighs the possible safety consequences and benefits of performing a required test as part of TS surveillance, including circumstances in which one train is out of service. Nonetheless, any related guidance provided by the NRC staff does not supersede the TS requirements. For example, if testing is specified as part of the TS surveillance, the cycling of nonredundant valves in a remaining operable train may not be deferred to the next cold shutdown when one train is out of service, even though their failure would cause a loss of total system function. In this case, a TS change or enforcement discretion would be necessary to defer testing.

The NRC expects licensees to comply with required IST test frequencies. The OM Code does not require documentation for valves that are not tested during a cold shutdown outage other than as required for maintaining the IST schedule. The NRC does not have a position on the efforts a licensee expends in performing cold shutdown valve testing during a short outage. The NRC staff, however, expects licensees to expend a reasonable "good faith" effort.

This issue is further discussed in Sections 3 and 4, which provide guidance on deferring testing.

2.5 Relief Requests and Proposed Alternatives

Relief Requests

Licensees can request that the NRC grant relief from an ASME OM Code or ASME BPV Code, Section XI, requirement in accordance with 10 CFR 50.55a(f)(5)(iii) and (iv), and (g)(5)(iii) and (iv). Requests made under these paragraphs are called "relief."

The NRC regulations in 10 CFR 50.55a(f)(4) require licensees to test pumps and valves in the IST program to the "extent practical" within the limitations of the design, geometry, and materials of construction. The regulations in 10 CFR 5.55a(f)(5) and (g)(5) allow licensees to request relief from ASME OM Code requrements that are "impractical" for the facility. For example, OM Code, Subsection ISTC, requires that the limiting stroke time for POVs be specified by the licensee and measured within limits based on the full-stroke time of the valves. At some plants, the scram discharge volume vent and drain valves are not designed to be individually actuated. These valves are required by TS to close within a specified time (45 seconds for some plants) upon receipt of a scram signal. The valves are tested quarterly by cycling the valves to assess their operational readiness and performing a valve sequence response time test during each RFO. This testing is essentially a design basis test of the valve combination. Requiring these valves to be stroke timed individually is impractical and places a burden on the licensee because of the extensive modification that would be required to the system to individually stroke the valve. In addition, jumpering the control circuit during plant operation to test these valves individually would be impractical because of the potential for a reactor scram. Some licensees may have difficulty fully implementing these ISTC-5131 required tests, and, in certain cases, because of the impracticality of implementation, a request for relief under 10 CFR 50.55a(f)(5) would be appropriate. In addition to design, geometry, and materials issues, licensees might also identify testing requirements specified in the ASME OM Code that are impractical because their performance might be unwise or unreasonable for certain pumps or valves and their applications at a specific facility. Licensees may request relief from such requirements in accordance with 10 CFR 50.55a(f)(5) or (g)(5).

In accordance with the regulations, when updating a program to a later edition of the OM Code, licensees must implement the updated program at the beginning of a 120-month IST program interval. The regulations state that in cases in which a licensee determines that an OM Code-specified pump or valve test is impractical and is not included in the revised IST program, it must submit a relief request demonstrating the basis for its determination to the NRC no later than 12 months after the previous 120-month IST program interval ends, or 12 months after the current IST program interval starts. However, experience has shown that licensees also identify impractical test provisions throughout the interval. In such cases, licensees may request relief as soon as they identify the condition. Where the OM Code requirements are impractical, the licensee would test the applicable components using the method proposed in the relief request in the period of time from the beginning of the new interval or time of discovery (or from the time of identification) and using method granted by the NRC for the applicable IST program interval.

<u>Alternative Requests</u>

Licensees can request that the NRC authorize an alternative to an ASME OM Code requirement in accordance with 10 CFR 50.55a(z). Requests made under 10 CFR 50.55a(z) are called "alternatives."

The OM Code establishes the requirements for preservice testing and IST and the examination of certain components to assess their operational readiness in water-cooled reactor nuclear power plants. These requirements apply to pumps, valves, pressure relief devices, and snubbers within the scope of the OM Code. The requirements are constantly being reviewed and improved in order to meet the basic function of maintaining the safe and reliable operation and maintenance of nuclear power plants.

It is understood that not all plants are designed the same. It is also understood that the general requirements developed in the OM Code may not be applicable or that complying with these requirements may be difficult. Licensees may propose alternatives to the OM Code provided that (1) the alternative would provide an acceptable level of quality and safety under 10 CFR 50.55a(z)(1); or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety under 10 CFR 50.55a(z)(2). Hardships generally involve reductions in radiation exposure to as low as reasonably achievable (ALARA), challenges to operators or plant equipment, components that are somewhat unique in design such as jockey (waterleg) pumps, or systems where pump flow is fixed and cannot be adjusted.

Licensees shall not implement proposed alternatives to the OM Code requirements under 10 CFR 50.55a(z) until the NRC staff completes its evaluation and authorizes the alternative. For example, if a licensee proposes to implement a pump vibration program based on the use of spectral analysis rather than the OM Code-specified method, the licensee must continue to meet the OM Code requirements until the NRC staff completes its evaluation and authorizes the alternative.

2.5.1 Justifications for Relief or Alternatives

In determining whether to grant relief from the OM Code requirements or to authorize alternatives, the NRC staff considers the merits of the submitted technical information. In requesting relief or use of an alternative, the licensee would typically identify the specific OM Code requirement and associated paragraph for which relief or use of an alternative is requested, describe the proposed alternative(s), describe the basis for relief or authorization of the proposed alternative(s), and clarify the burden that would result if the NRC enforced the specified requirements. Situations that warrant granting relief or authorizing alternatives (as determined by the NRC staff in previous safety evaluations for plant-specific requests) may include the following examples:

1. In complying with the OM Code requirements, the licensee would not obtain information that would be more useful than the information that is currently available. For example, installing an analog gauge with a range of three times the reference value (or less) to

comply with OM Code requirements may not yield more accurate readings than those provided by the gauge that is presently installed (see Section 5.5.1).

2. Compliance with the OM Code is impractical because of design limitations. Imposition of the OM Code requirements would require significant system redesign and modifications. For example, a flow meter does not meet the accuracy requirements of ISTB-3510 and Table ISTB 3510-1 because the present system configuration does not have a straight section of pipe of sufficient length in which to measure flow accurately (see Section 5.5).

3. The required measurements or appropriate observations cannot be made because of physical constraints. Examples include a component located in an area that is inaccessible during power operation or a pump that is totally immersed in system fluid.

4. The need to keep personnel radiation exposure ALARA may present an adequate justification. The licensee should include information about the general area radiation field, local hot spots, plant radiation limits and stay times, the amount of exposure personnel would receive in performing the testing, and the safety significance of deferring testing or performing an alternative method. ALARA relates to controlling exposure during an activity, not specifically to eliminating activities; however, it may be a basis for relief or for deferring a test on the basis of hardship when exposure limits are prohibitive for performing testing (or possibly for accessing a valve for repair in the event that it could fail during a test). If the exposure limits are prohibitive, the licensee should defer testing to cold shutdown or refueling outages during which the exposure limits would no longer be prohibitive. ALARA is part of an overall program, including activities such as IST, as required by 10 CFR 20.1101, "Radiation Protection Programs." The NRC has not established "predetermined acceptable limits" for deferring an IST activity, based on maintaining occupational exposure ALARA.

5. Testing as required by the OM Code could cause significant equipment damage. For example, shutting off cooling flow to an operating pump by exercising a valve in the cooling flow path could damage the pump.

6. Failure of a component during testing could disable multiple trains of a reactor safety system and cause a loss of system function. For example, a motor-operated suction valve common to both trains of high-pressure safety injection could not be tested during power operation because a failure of the valve would result in both trains being out of service. Another example would be where a valve is undergoing maintenance in one train and testing of the valve in the other train would result in a loss of system function.

Inconvenience or administrative burden are not, in and of themselves, adequate justification for deviating from the OM Code requirements. Similarly, entering a TS LCO is not, in and of itself, adequate justification for deviating from the OM Code-specified frequency, except when entering the LCO would be prohibited because the total system function would be out of service.

2.5.2 Categories of Relief or Alternative Requests

The NRC staff categorizes relief or alternative requests as follows:

- General: A general relief or alternative request is appropriate when the requested relief or alternative applies to a broad range of similar components in the program, such as all pumps or all containment isolation valves.

- Specific: A relief or alternative request is specific when the requested relief or alternative applies only to a single component or a specified group of similar components in the program, such as service water pump discharge check valves.

2.5.3 Content and Format of Relief or Alternative Requests

As a minimum, the NRC staff recommends that each relief or alternative request should include the following information:

- Title and relief or alternative request number: Licensees should title each relief or alternative request and specify a unique identifier. The identifier should remain unique to avoid confusion with later revisions. Examples include (1) “Relief Request Number 1,” (2) “Safety Injection Pumps Relief Request,” or (3) “Check Valves in Series Relief Request.” The staff recommends that the licensee determine whether the request involves relief under 10 CFR 50.55a(f)(5) or (g)(5), or an alternative under 10 CFR 50.55a(z); and label the request as relief or an alternative as appropriate.
- Page number: List the page number and total number of pages in the program document or the relevant section, such as “Page 15 of 135.”
- Program revision or page revision date: List the program or page revision number and date (on each page). List the revision number for each program change submitted.
- Code of Record: List the applicable Code Editions for the plant’s IST interval.
- System and Code class: List the plant system and Code class of the system in which the component is located. Pump/valve category or group: List the ASME category or group for each pump or valve (i.e., A, A/B, A/C, B, C, or D).
- Component identification: List the identification number for each component in a specific relief or alternative request. Each individual component need not be listed in a general relief or alternative request, such as one for all pumps in the IST program. However, the NRC staff recommends that the list of program components (pump or valve table) should include the relief or alternative request number.
- Component function: Briefly describe the functions of the affected components and specify the function that is the subject of the relief or alternative request.
- OM Code test requirement(s): List and describe the OM Code requirement(s) from which relief or alternative is being requested.
- Basis for relief or alternative: Clearly state the legal basis under which relief or an alternative is requested, and then explain the reasons why complying with the OM Code requirements is impractical, poses a hardship, or otherwise should not be enforced. Include all information that the NRC staff might need to complete its review. For example, most relief requests for check valves list the test direction(s) for which relief is requested.
- Proposed alternative testing: Clearly and thoroughly discuss the proposed alternative in sufficient detail to clearly demonstrate why it is a reasonable alternative to the OM Code requirement, and provide a technical basis for its acceptability.
- IST Interval: Provide the start and end dates of the 10-year IST interval for which relief or alternative is requested.

- Drawings and/or diagrams: If the relief request or alternative testing is complex, or if drawings or diagrams are available for further clarification, include them in the relief or alternative request, or include them in the IST program document and reference them in the relief or alternative request.

- References: List references to FSAR sections, technical specifications, and other pertinent documents. Any document referenced in the relief or alternative request should be submitted to the NRC on the plant docket. If a document is not docketed but contains pertinent information, the relief request should explicitly include the information (if it is not readily available to the NRC staff and the public), rather than merely referencing the document.

To improve the effectiveness and efficiency of the request process, the Nuclear Energy Institute (NEI) developed a white paper entitled, "Standard Format for Requests from Commercial Reactor Licensees Pursuant to 10 CFR 50.55a, Revision 1," dated June 7, 2004 (ADAMS Accession No. ML070100400). This white paper provides useful guidance for determining the appropriate regulatory requirement under which a "relief request" is submitted for NRC approval, as well as the appropriate format and content to use in the request. The term "relief request" is used loosely in this instance to denote the various types of submittals allowed by 10 CFR 50.55a, including alternatives to the regulations [10 CFR 50.55a(z)], impractical relief requests [10 CFR 50.55a(f)(5)], and requests to use later Code Editions and Addenda [10 CFR 50.55a(f)(4)(iv)]. The NRC staff has reviewed the NEI White Paper and encourages licensees to use the specified format and content.

2.5.4 Revising NRC-Authorized Relief or Alternative

RG 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated May 2019, provides guidance related to use of 10 CFR 50.59 process. This 10 CFR 50.59 process does not allow the licensee to change an NRC granted or authorized relief request or alternative.

NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, dated November 2000, states that licensees' activities that are controlled by the regulations in 10 CFR 50.55a take precedence over 10 CFR 50.59. RG 1.187 endorses NEI 96-07, Revision 1.

2.5.5 NRC Temporary Verbal Authorization of an Alternative Request

On rare occasions, the NRC may grant verbal authorizations as an alternative under 10 CFR 50.55a(z) when, because of unforeseen circumstances, licensees need NRC authorization before the agency is able to issue its written safety evaluation as described in NRC Office Instruction LIC-102, Revision 3 "Review of Relief Requests, Proposed Alternatives, and Requests to Use Later Code Editions and Addenda,," (ADAMS Accession No. ML18351A218).

Temporary verbal authorization for an alternative under 10 CFR 50.55a(z) is subject to the following:

- The proposed alternative is in writing, and all the information that the NRC requires to complete the safety evaluation has been docketed.

- An identified need for the verbal authorization is recognized given the circumstances of the licensee's request.

- The NRC has completed its review and determined that the proposed alternative is technically justified, but the agency has not yet formally documented it in a safety evaluation.
- The technical branch and reactor licensing branch chiefs have agreed to the verbal authorization.

Verbal authorization is most likely conveyed in a telephone conversation with a summary of the NRC staff evaluation. As such, appropriate NRC personnel who are normally involved in authorizing the alternative must be present in the telephone conversation. The NRC project manager should promptly (i.e., in 1 or 2 days) generate a record of the conversation; this record will meet the definition of an Official Agency Record (OAR) and must be entered into ADAMS and made publicly available. The NRC should issue the final written authorization within 150 days after giving verbal authorization.

2.5.6 NRC Authorization of Proposed Alternative Similar to Prior NRC-Authorized Alternative and Reliefs

Licensees occasionally submit alternative and relief requests that are very similar to NRC-authorized alternative and relief requests for the previous 10-year IST program intervals when updating their IST program in accordance with 10 CFR 50.55a(f)(4)(ii). This practice is acceptable provided that the licensee compares the requirements between the old and new ASME OM Codes and evaluates whether changes to the alternative request are necessary. For example, the OM Code has provisions added for exercising check valves such as disassembly and condition monitoring programs. Addressing the check valve disassembly and condition monitoring programs in the alternative request may be appropriate if these provisions were not included in the OM Code upon which the original alternative request was based. Furthermore, the addition of disassembly and condition monitoring programs to the OM Code may eliminate the need for the alternative request.

Licensees also should review new OM Code Cases before submitting an alternative and relief request for updated IST programs. For example, OM Code Case OMN-9, “Use of Pump Curve for Testing,” provides an alternative method for testing centrifugal and vertical line shaft pumps when the licensee is unable to obtain a specific reference value in accordance with Subsection ISTB of the OM Code. The NRC conditionally approved Code Case OMN-9 in RG 1.192, Revision 3. Code Case OMN-16, “Use of Pump Curve for Testing,” incorporates all the conditions specified for approval of Code Case OMN-9. OMN-16 is approved for use in RG 1.192, Revision 3. The use of Code Case OMN-9 or OMN-16 may eliminate the need for an alternative request.

2.6 IST Program Documents

The applicable ASME OM Code, Subsection ISTA, “General Requirements,” provides documentation requirements such as the following:

- ISTA-3200(a) requires that IST plans shall be filed with the regulatory authorities having jurisdiction at the plant site. This requirement is currently in the OM Code, but the ASME is removing this requirement beginning with the 2020 Edition of the OM Code. The NRC staff is considering the proposal of a regulatory requirement for submittal of the IST Program Plan by nuclear power plant licensees in a future 10 CFR 50.55a rulemaking.

- ISTA-9000, “Records and Reports,” provides the requirements for preparation, submittal, and retention of records and reports.
- Nonmandatory Appendix A and the Supplement to Nonmandatory Appendix A in the ASME OM Code describe voluntary guidance for licensees for development of IST plans for pumps and valves.

As long as the IST program is consistent with the regulations, OM Code relief is not required. That is, deletions from or additions to the IST program do not necessarily require NRC approval. The burden is on each licensee to verify that its IST program is complete and includes all components that require IST activities, and that all such components are tested to the extent required by the OM Code. If a licensee deletes a particular component from its IST program, the NRC staff recommends that the licensee document the reason in an appropriate place.

The NRC staff expects each licensee to maintain its IST program up-to-date and ensure that it remains consistent with changes in plant configuration. If a particular relief request is no longer required because of changes in hardware, system design, or new technology, the licensee is expected to revise its program to withdraw the relief request. Conversely, if a system modification results in the addition of a component to the IST program, the licensee should ensure that it meets the OM Code requirements, or that a relief request is submitted for NRC review and approval, as appropriate.

Licensees not meeting ISTA-3200(a) must submit appropriate documents containing IST plans and submit a request for an alternative to the NRC pursuant to 10 CFR 50.55a(z). IST program documents submitted to the NRC are used to prepare for IST inspections and to address other licensing actions that may arise. Between a licensee’s 10-year IST program interval submittals, the NRC staff expects to receive up-to-date program documents when the licensee makes significant changes to the IST program to facilitate these regulatory activities.

2.7 Developing IST Programs for New Nuclear Power Plants

The nuclear industry has submitted applications for licenses to construct and operate new nuclear power plants. The NRC discusses policy and technical issues associated with new reactors, including the development of IST programs, in several Commission papers, such as the following:

- SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements” (ADAMS Accession No. ML003707849)
- SECY- 93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs” (ADAMS Accession No. ML003708021)
- SECY- 94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs” (ADAMS Accession No. ML003708068)
- SECY- 95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs” (Accession No. ML003708005)
- Applicable Staff Requirements Memoranda (SRMs)

In a public memorandum dated July 24, 1995, the NRC staff consolidated the discussion of the policy and technical issues associated with the regulatory treatment of non-safety systems (RTNSS) in new passive plant designs provided in SECY-94-084 and SECY-95-132, and their associated SRMs.

The NRC regulations in 10 CFR Part 52 include design certifications for specific new reactor designs, such as the Advanced Boiling Water Reactor (ABWR); AP600, AP1000, and APR1400 PWRs; ESBWR; and System 80+ reactor. In addition, suppliers of design certifications have submitted applications for, or updates to, certification of designs for several new reactors. The NRC regulations require new reactor suppliers to address the design of plant systems related to the performance of the IST program in their design certification application. While the design phase contains significant flexibility, new reactor vendors should design their plants to minimize the need for requests for relief from the IST provisions in the ASME OM Code. Under 10 CFR Part 52, the COL applicant has responsibility for the development of a plant-specific IST program for a COL to construct and operate a nuclear power plant.

The Commission's SRM, dated September 11, 2002, for Commission Paper SECY-02-0067, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Operational Programs (Programmatic ITAAC)," (ADAMS Accession No. ML020700641) stated that ITAAC for an operational program are unnecessary if the COL application fully describes the program and its implementation and the NRC finds them to be acceptable. The Commission also stated that the burden is on the COL applicant to provide the necessary and sufficient programmatic information for approval of the COL without ITAAC.

In its May 14, 2004, SRM for SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License Without Inspections, Tests, Analyses and Acceptance Criteria," (ADAMS Accession No. ML041350440) the Commission defined "fully described" as meaning that the program is clearly and sufficiently described in terms of the scope and level of detail to allow a reasonable assurance finding of acceptability. The Commission also noted that required programs should always be described at a functional level and at an increasing level of detail where implementation choices could materially and negatively affect the program effectiveness and acceptability. SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," (ADAMS Accession No. ML052770257) summarizes the NRC position regarding the full description of operational programs to be provided by COL applicants. The guidance in this NUREG may be used in developing and implementing the IST program for new nuclear power plants.

For a COL issued per 10 CFR Part 52, the NRC regulations in 10 CFR 50.55a(f)(4)(i) state that inservice tests conducted during the initial 120-month IST program interval to verify operational readiness of applicable plant components, whose function is required for safety, must comply with the requirements in the latest edition and addenda of the ASME OM Code, incorporated by reference in 10 CFR 50.55a(a) on the date 18 months before the date scheduled for initial fuel loading (or the optional Code cases listed in RG 1.192) subject to the conditions listed in 10 CFR 50.55a. As discussed in RIS 2012-08, Revision 1, "Developing Inservice Testing and Inservice Inspection Programs under 10 CFR Part 52," dated July 17, 2013 (ADAMS Accession No. ML13122A365), licensees may submit a request under 10 CFR 50.55a(z) to apply the edition and addenda of the OM Code specified in the COL application for the initial 120-month IST program interval as an alternative to the latest edition and addenda of the OM Code, where the differences between the IST provisions in these Code editions and addenda are addressed.

Several applications under 10 CFR Part 52 were submitted for COLs to construct and operate new nuclear power plants that reference certified reactor designs or designs under certification review. In addition to addressing design aspects related to the IST program, new reactor design certification applicants typically provide a description of generic aspects of the IST program to allow the COL applicants to incorporate by reference this design certification information in their COL application. The NRC staff reviews the description of the IST program in the COL application, with its incorporation by reference of IST provisions in the applicable design certification documentation, as part of the safety evaluation for the COL application. The NRC staff will conduct inspections of the development and implementation of the IST program following COL issuance.

ASME has a program underway to establish improved IST provisions in the OM Code for pumps, valves, and dynamic restraints to be used in new reactors. ASME has prepared a White Paper that discusses its plans to update the OM Code for new reactors. For example, the ASME White Paper identifies lessons learned from operating experience at current nuclear power plants, and from research sponsored by the nuclear industry and the NRC, that are applicable to IST programs for new reactors. The ASME White Paper also identifies new reactor issues that can affect IST programs to be developed for new reactors. As part of this effort, ASME has issued Subsection ISTF for IST activities for pumps in new reactors. ASME is working on updated IST provisions for valves in new reactors.

Lessons learned from nuclear power plant operating experience and research that should be considered in the development of IST programs for new reactors include, for example:

1. Design and qualification of pumps, valves, and snubbers to allow IST activities (including sufficient flow testing) to assess the operational readiness of those components, including ASME Standard QME-1-2017, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,” as accepted in Revision 4 to RG 1.100, “Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” to incorporate lessons learned in the qualification of mechanical equipment for nuclear power plants.

2. Performance and testing of MOVs that indicate the need for improved MOV activities, such as importance of adequate design and qualification, sufficient flow during testing to assess valve performance, consideration of MOV performance parameters (including valve disk and stem friction coefficients, reduced voltage, elevated temperature, and load sensitive behavior), use of adequate diagnostic instrumentation to allow proper evaluation and setup, improved maintenance and personnel training, monitoring of potential motor magnesium rotor degradation, and justification for motor control center testing.

3. Application of MOV lessons learned to other Power-Operated Valves (POVs).

4. Provisions for bidirectional testing of check valves.

5. Implementation of preservice testing and comprehensive pump testing (CPT) provisions without the need for relief from the OM Code provisions.

6. Consideration of potential adverse flow effects on plant components from flow-induced vibration resulting from hydrodynamic loads and acoustic resonance.

New reactor issues that should be considered in the development of IST programs for new nuclear power plants include, for example:

1. Description of IST programs by COL applicants in accordance with 10 CFR Part 52 with implementation of design certification provisions for design, qualification, and IST activities.
2. Coordination of PST and ITAAC so that testing is performed once for both purposes. For example, implementation of PST requirements and the 10 CFR Part 52 ITAAC closure and maintenance process both need to be accomplished. Under the 10 CFR Part 52 process, an applicant is required to meet OM Code requirements after a 10 CFR 52.103(g) finding is made, although it would be preferable to complete the PST requirements earlier.
3. Design, qualification, and IST and inspection activities for pyrotechnic-actuated (squib) valves that have high safety significance, and that represent more significant engineering challenges for new reactors than for current operating plants.
4. Design of plant systems and development of IST programs to minimize the need for relief from the OM Code provisions.
5. Design, qualification, PST, and IST activities for regulatory treatment for non-safety systems (RTNSS) components that perform safety significant functions.
6. Development and implementation of risk-informed IST programs, including programs under 10 CFR 50.69, “Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,” for new reactors.
7. Consideration of appropriate Code and standard modifications for design, qualification, PST and IST activities in response to application of software-based digital technology in mechanical components (e.g., pumps and valves).

Applicants for new nuclear power plants should consider the information in this NUREG and other sources in developing their IST programs.

The NRC regulations in 10 CFR 50.55a(b)(3)(iii) apply specific conditions for IST programs applicable to licensees of new nuclear power plants in addition to the provisions of the OM Code as incorporated by reference with conditions in 10 CFR 50.55a. Licensees of “new reactors” are identified in 10 CFR 50.55a as holders of operating licenses for nuclear power reactors that received construction permits under 10 CFR Part 50 on or after August 17, 2018, and COL holders issued under 10 CFR Part 52, whose initial fuel loading occurs on or after August 17, 2018.

Commission Papers SECY-90-016, SECY-93-087, SECY-94-084, and SECY-95-132, discuss IST programs for new reactors licensed under 10 CFR Part 52.

In recognition of new reactor designs and lessons learned from nuclear power plant operating experience, the ASME updated the OM Code to incorporate improved IST provisions for components used in nuclear power plants that were issued (or will be issued) construction permits, or COLs, on or following January 1, 2000 (defined in the OM Code as post-2000 plants). The first phase of the ASME effort incorporated IST provisions that specify full flow pump testing and other clarifications for post-2000 plants in the OM Code beginning with the 2011 Addenda. The second phase of the ASME effort incorporated preservice and inservice inspection and surveillance provisions for pyrotechnic-actuated (squib) valves in the 2012 Edition of the OM Code. The ASME is considering further modifications to the OM Code to address additional lessons learned from valve operating experience and new reactor issues. As described in the following paragraphs, 10 CFR 50.55a(b)(3)(iii) includes four specific conditions. Applicable licensees will need to satisfy 10 CFR 50.55a(b)(3)(iii)(A) for all AOVs. As discussed earlier, the NRC has prepared rulemaking to incorporate by reference the OM Code (2017 Edition), which includes Appendix IV for AOVs that will satisfy 10 CFR 50.55a(b)(3)(iii)(A) for AOVs.

The NRC regulations in 10 CFR 50.55a(b)(3)(iii)(A) require licensees within the scope of 10 CFR 50.55a(b)(3)(iii) to periodically verify the capability of POVs to perform their design-basis safety functions. While Appendix III to the OM Code addresses this requirement for MOVs with the conditions specified in 10 CFR 50.55a, applicable licensees (holder of OLs that received CPs on or after August 17, 2018, and holders of COLs whose initial loading occurs on or after August 17, 2018) will need to develop programs to periodically verify the design-basis capability of other POVs. RIS 2000-03, “Resolution of Generic Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions,” provides attributes for a successful long-term periodic verification program for POVs by incorporating lessons learned from MOV performance at operating nuclear power plants and research programs. Implementation of Appendix III to the OM Code as accepted in 10 CFR 50.55a(b)(3)(ii) satisfies 10 CFR 50.55a(b)(3)(iii)(A) for MOVs. Applicable licensees will need to satisfy 10 CFR 50.55a(b)(3)(iii)(A) for POVs.

Paragraph (A) of 10 CFR 50.55a(b)(3)(iii) is consistent with the Commission policy for new reactors summarized in an NRC Staff Memorandum, “Consolidation of SECY-94-084 and SECY-95-132,” dated July 24, 1995, that (a) the design capability of safety-related POVs should be demonstrated by a qualification test prior to installation; (b) prior to initial startup, POV capability under design-basis differential pressure and flow should be verified by a pre-operational test; and (c) during the operational phase, POV capability under design-basis differential pressure and flow should be verified periodically through a program similar to that developed for MOVs in GL 89-10. The condition in paragraph (A) specifies with the same level of detail as the condition in 10 CFR 50.55a(b)(3)(ii) that nuclear power plant licensees must establish a program to ensure the continued capability of MOVs in performing their design-basis safety functions. When establishing the MOV periodic verification condition, the NRC provided guidance in the final rule published September 22, 1999 (64 FR 51370) for licensees to develop acceptable programs that would satisfy the MOV periodic verification condition. Similarly, the NRC staff provided guidance in the *Federal Register* notice for the final rule dated July 18, 2017 (82 FR 32934), and summarized herein, for new reactor applicants and licensees to develop acceptable programs to periodically verify the capability of POVs to perform their design-basis safety functions.

In NUREG-2124, Volume 1 “Final Safety Evaluation Report [FSER] Related to the Combined Licenses for Vogtle Electric Generating Plant [VEGP], Units 3 and 4,” (ADAMS Accession No. ML12271A045) the NRC staff found the provisions established by the COL applicant for VEGP Units 3 and 4 in its FSAR, Revision 5, Section 3.9.6.2.2, “Valve Testing,” to periodically verify the capability of POVs, such as AOVs, solenoid-operated valves (SOVs), and hydraulic-operated valves (HOVs), to perform their design-basis safety functions to be acceptable. In particular, the VEGP Units 3 and 4 FSAR in Section 3.9.6, “Inservice Testing of Pumps and Valves,” specifies that:

> Power-operated valves other than active MOVs are exercised quarterly in accordance with OM ISTC, unless justification is provided in the inservice testing program for testing these valves at other than Code mandated frequencies.

Although the design basis capability of power-operated valves is verified as part of the design and qualification process, power-operated valves that perform an active safety function are tested again after installation in the plant, as required, to ensure valve setup is acceptable to perform their required functions, consistent with valve qualification. These tests, which are typically performed under static (no flow or pressure) conditions, also document the "baseline" performance of the valves to support maintenance and trending programs. During the testing, critical parameters needed to ensure proper valve setup are measured. Depending on the valve and actuator type, these parameters may include seat load, running torque or thrust, valve travel, actuator spring rate, bench set and regulator supply pressure. Uncertainties associated with performance of these tests and use of the test results (including those associated with measurement equipment and potential degradation mechanisms) are addressed appropriately. Uncertainties may be considered in the specification of acceptable valve setup parameters or in the interpretation of the test results (or a combination of both). Uncertainties affecting both valve function and structural limits are addressed. Additional testing is performed as part of the air-operated valve (AOV) program, which includes the key elements for an AOV Program as identified in the JOG AOV program document, Joint Owners Group Air Operated Valve Program Document, Revision 1, December 13, 2000 (References 203 and 204 from JOG Program Document) [JOG AOV Program Document, Revision 1, December 13, 2000 (ADAMS Accession No. ML010950310), and NRC comment letter dated October 8, 1999, to Nuclear Energy Institute (ADAMS Accession No. ML020360077)]. The AOV program incorporates the attributes for a successful power-operated valve long-term periodic verification program, as discussed in Regulatory Issue Summary 2000-03, Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions, by incorporating lessons learned from previous nuclear power plant operations and research programs as they apply to the periodic testing of air- and other power-operated valves included in the IST program.

For example, key lessons learned addressed in the AOV program include:

- Valves are categorized according to their safety significance and risk ranking.
- Setpoints for AOVs are defined based on current vendor information or valve qualification diagnostic testing, such that the valve is capable of performing its design-basis function(s).
- Periodic static testing is performed, at a minimum on high risk (high safety significance) valves, 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 required based on valve qualification or operating experience, periodic dynamic testing is performed to re-verify the capability of the valve to perform its required functions.
- Sufficient diagnostics are 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 is specified, and is evaluated each refueling outage based on data trends as a result of testing. Frequency for periodic testing is in accordance with References 203 and 204, with a minimum of 5 years (or 3 refueling cycles) of data collected and evaluated before extending test intervals.
- 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 attributes of the AOV testing program described above, to the extent that they apply to and can be implemented on other safety-related power-operated valves, such as electro-hydraulic operated valves, are applied to those other power-operated valves

Applicable applicants and licensees may follow the method described in the VEGP Units 3 and 4 FSAR in satisfying 10 CFR 50.55a(b)(3)(iii)(A), or may establish a different method, subject to evaluation by the NRC during the licensing process or inspections.

The NRC regulations in 10 CFR 50.55a(b)(3)(iii)(B) require that licensees within the scope of 10 CFR 50.55a(b)(3)(iii) perform bidirectional testing of check valves within the IST program where practicable. Nuclear power plant operating experience has revealed that testing check valves in only the flow direction can result in significant degradation, such as a missing valve disc, not being identified by the test. Nonmandatory Appendix M, “Design Guidance for Nuclear Power Plant Systems and Component Testing,” to the OM Code includes guidance for the design of new reactors to enable bidirectional testing of check valves. New reactor designs will provide the capability for licensees of new nuclear power plants to perform bidirectional testing of check valves within the IST program. Bidirectional testing of check valves in new reactors, as required by 10 CFR 50.55a(b)(3)(iii)(B), could be accomplished by valve-specific testing or condition monitoring activities in accordance with Appendix II to the OM Code as accepted in 10 CFR 50.55a. The NRC is specifying this provision for bidirectional testing of check valves for new reactors in 10 CFR 50.55a(b)(3)(iii)(B) to emphasize that new reactors should include the capability for bidirectional testing of check valves as part of their initial design.

The NRC regulations in 10 CFR 50.55a(b)(3)(iii)(C) require that licensees subject to 10 CFR 50.55a(b)(3)(iii) monitor flow-induced vibration (FIV) from hydrodynamic loads and acoustic resonance during preservice testing and inservice testing to identify potential adverse flow effects that might impact components within the scope of the IST program. Nuclear power plant operating experience has revealed the potential for adverse flow effects from vibration caused by hydrodynamic loads and acoustic resonance on components in the reactor coolant, steam, and feedwater systems. Therefore, the licensee will be required to address potential adverse flow effects on safety-related pumps, valves, and dynamic restraints within the IST program in the reactor coolant, steam, and feedwater systems from hydraulic loading and acoustic resonance during plant operation. In response to public comments, the NRC revised 10 CFR 50.55a(b)(3)(iii)(C) to clarify its intent that FIV monitoring of components may be conducted during preservice testing or inservice testing. This requirement will confirm that piping, components, restraints, and supports have been designed and installed to withstand the dynamic effects of steady-state FIV and anticipated operational transient conditions. As part of preservice testing activities, the initial test program may be used to verify that safety-related piping and components are properly installed and supported such that vibrations caused by steady-state or dynamic effects do not result in excessive stress or fatigue in safety-related plant systems.

In the VEGP Units 3 and 4 FSER, the NRC staff found the provisions established by the COL applicant for VEGP Units 3 and 4 in its FSAR, Revision 5, Section 3.9, “Mechanical Systems and Components,” Section 14.2.9, “Preoperational Test Descriptions,” and Section 14.2.10, “Startup Test Procedures,” with incorporation by reference of corresponding sections of the AP1000 Design Control Document (DCD), to monitor FIV from hydrodynamic loads and acoustic resonance during preservice testing or inservice testing to be acceptable. In particular, the NRC staff stated in the VEGP Units 3 and 4 FSER Section 3.9.6, “Inservice Testing of Pumps and Valves (Related to RG 1.206, Section C.III.1, Chapter 3, C.I.3.9.6, ‘Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints’)”:

> AP1000 DCD Tier 2, Section 3.9.2, “Dynamic Testing and Analysis,” describes tests to confirm that piping, components, restraints, and supports have been designed to withstand the dynamic effects of steady-state FIV and anticipated operational transient conditions. Section 14.2.9.1.7, “Expansion, Vibration and Dynamic Effects Testing,” in AP1000 DCD Tier 2, Chapter 14, “Initial Test Program,” states that the purpose of the expansion, vibration and dynamic effects testing is to verify that safety-related, high energy piping and components are properly installed and supported such that, in addition to other factors, vibrations caused by steady-state or dynamic effects do not result in excessive stress or fatigue to safety-related plant systems. Nuclear power plant operating experience has revealed the potential for adverse flow effects from vibration caused by hydrodynamic loads and acoustic resonance on reactor coolant, steam, and feedwater systems. … In its response, SNC [Vogtle Units 3 and 4 COL applicant] stated that it intended to use the overall Initial Test Program to demonstrate that the plant has been constructed as designed and the systems perform consistent with design requirements. SNC referenced the provisions in the AP1000 DCD for vibration monitoring and testing to be implemented at VEGP. For example, the applicant notes that AP1000 DCD Tier 2, Section 3.9.2.1, “Piping Vibration, Thermal Expansion and Dynamic Effects,” specifies that the preoperational test program for ASME BPV Code, Section III, Class 1, 2, and 3 piping systems simulates actual operating modes to demonstrate that components comprising these systems meet functional design requirements and that piping vibrations are within acceptable levels. SNC indicates that the planned vibration testing program described in AP1000 DCD Tier 2, Sections 14.2.9 and 14.2.10, with the preservice and IST programs described in AP1000 DCD Tier 2, Sections 3.9.3.4.4 and 3.9.6, will confirm component installation in accordance with design requirements, and address the effects of steady-state (flow-induced) and transient vibration to ensure the operability of valves and dynamic restraints in the IST Program. The NRC staff considers the response by SNC clarifies its application of the provisions in the AP1000 DCD to ensure that potential adverse flow effects will be addressed at VEGP. Therefore, the NRC staff considers Standard Content Open Item 3.9-5 to be resolved for the VEGP COL application.

As clarified in the final rule in response to public comments, a licensee may monitor components for adverse FIV effects during preservice testing or IST activities. Applicable applicants and licensees may either apply the methods described in the VEGP Units 3 and 4 FSAR in satisfying 10 CFR 50.55a(b)(3)(iii)(C) or develop their own plant-specific methods to satisfy 10 CFR 50.55a(b)(3)(iii)(C) for NRC review during the licensing process.

The NRC regulations in 10 CFR 50.55a(b)(3)(iii)(D) require that licensees within the scope of 10 CFR 50.55a(b)(3)(iii) establish a program to assess the operational readiness of pumps, valves, and dynamic restraints within the scope of the Regulatory Treatment of Non-Safety Systems (RTNSS) for applicable reactor designs. In SECY-94-084 and SECY-95-132, the Commission discusses RTNSS policy and technical issues associated with passive plant designs. Some new nuclear power plants have advanced light-water reactor (ALWR) designs that use passive safety systems that rely on natural forces, such as density differences, gravity, and stored energy to supply safety injection water and to provide reactor core and containment cooling. Active systems in passive ALWR designs are categorized as non-safety systems with limited exceptions. Active systems in passive ALWR designs provide the first line of defense to reduce challenges to the passive systems in the event of a transient at the nuclear power plant. Active systems that provide a defense-in-depth function in passive ALWR designs need not meet all of the acceptance criteria for safety-related systems. However, there should be a high level of confidence that these active systems will be available and reliable when needed. The combined activities to provide confidence in the capability of these active systems in passive ALWR designs to perform their functions important to safety are referred to as the RTNSS program. In the NRC Staff Memorandum, "Consolidation of SECY-94-084 and SECY-95-132," dated July 24, 1995, the NRC staff provided a consolidated list of the approved policy and technical positions associated with RTNSS equipment in passive plant designs discussed in SECY-94-084 and SECY-95-132. This new paragraph specifies the need for licensees to assess the operational readiness of RTNSS pumps, valves, and dynamic restraints.

The July 24, 1995, NRC staff memorandum summarizes the Commission policy positions related to inservice testing of RTNSS pumps and valves as follows:

> The NRC staff also concluded that additional inservice testing requirements may be necessary for certain pumps and valves in passive plant designs. The unique passive plant design relies significantly on passive safety systems, but also depends on non-safety systems (which are traditionally safety-related systems in current light-water reactors) to prevent challenges to passive systems. Therefore, the reliable performance of individual components is a very significant factor in enhancing the safety of passive plant design. The NRC staff recommends that the following provisions be applied to passive ALWR plants to ensure reliable component performance.
>
> 1. Important non-safety-related components are not required to meet criteria similar to safety-grade criteria. However, the 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. Specific positions on the inservice testing requirements for those components will be determined as a part of the NRC staff's review of plant-specific implementation of the regulatory treatment of non-safety systems for passive reactor designs.

2. The vendors for advanced passive reactors, for which the final designs are not complete, have sufficient time to include provisions in their piping system designs to allow testing at power. Quarterly testing is the base testing frequency in the Code and the original intent of the Code. Furthermore, the COL holder may need to test more frequently than during cold shutdowns or at every refueling outage to ensure that the reliable performance of components is commensurate with the importance of the safety functions to be performed and with system reliability goals. Therefore, to the extent practicable, the passive ALWR piping systems should be designed to accommodate the applicable Code requirements for the quarterly testing of valves. However, design configuration changes to accommodate Code-required quarterly testing should be done only if the benefits of the test outweigh the potential risk.
3. The 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 test need not be as rigorous as the corresponding safety direction test, and (2) to verify the 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 test methods.
4. Similarly, to the extent practicable, the design of non-safety-related piping systems with functions under design-basis condition that have been identified as being important by the RTNSS process should incorporate provisions to periodically test power-operated valves in the system during operations to assure that the valves meet their intended functions under design-basis conditions.
5. Mispositioning may occur through actions taken locally (manual or electrical), at a motor control center, or in the control room, and includes deliberate changes of valve position to perform surveillance testing. The NRC staff will determine if and the extent to which this concept should be applied to MOVs in important non-safety-related systems when the NRC staff reviews the implementation of the regulatory treatment of non-safety systems. (NRC Staff Memorandum, “Consolidation of SECY-94-084 and SECY-95-132,” July 24, 1995, pages 26–28).

Consistent with the Commission policy for RTNSS equipment, 10 CFR 50.55a(b)(3)(iii)(D) specifies that new reactor licensees shall assess the operational readiness of pumps, valves, and dynamic restraints within the RTNSS scope. This regulatory requirement will allow licensees flexibility in developing programs to assess operational readiness of RTNSS components that satisfy the Commission policy. Guidance on the implementation of the Commission policy for RTNSS equipment is set forth in NRC Inspection Procedure 73758, “Part 52, Functional Design and Qualification, and Preservice and Inservice Testing Programs for Pumps, Valves and Dynamic Restraints,” dated February 6, 2020 (ADAMS Accession No. ML19364A004).

Table 2.1 Typical Systems and Components in an Inservice Testing Program for a Pressurized-Water Reactor

Typical safety-related, Code-class system in pressurized-water reactors	Typical components in an IST program
Reactor coolant system and flow paths for establishing natural circulation	- Power-operated relief valves and associated block valves - Reactor high point and head vents - Primary system safety and relief valves (pressurizer Code safety valves) - Valves in any proposed flow path used for long-term core cooling or safe shutdown - Pressure boundary isolation valves - Valves in lines to pressurizer relief/quench tank
Main steam system	- Main steam isolation valves (MSIVs) - Main steam non-return valves (if applicable) - Secondary system safety and relief valves - Atmospheric dump valves - Auxiliary feedwater turbine steam supply valves - Steam generator blowdown isolation valves
High-pressure safety injection system	- High-pressure injection pumps and discharge check valves - Injection valves in injection flow path - Isolation valves - Valves for the refueling water storage tank (RWST) borated water storage tank (BWST), and refueling water tank (RWT), including vacuum breakers
Chemical and volume control or makeup system	- Charging or makeup pumps and suction/discharge check valves - Valves in charging/makeup flow path - Boric acid transfer pumps and suction/discharge check valves - Valves in emergency boration flow paths - Relief valves
Low-pressure safety injection system	- Injection pumps and suction/discharge check valves - Valves associated with safety injection accumulators and core flood tanks - Recirculation flow path valves, including containment sump isolation valves - Isolation valves (high-low pressure interface) - Relief valves
Shutdown cooling, residual heat removal, or decay heat removal systems	- Pumps and suction/discharge check valves - Valves in flow path - Isolation valves (high-low pressure interface) - Relief valves

Typical safety-related, Code-class system in pressurized-water reactors	Typical components in an IST program
Containment spray system	- Containment spray pumps and suction/discharge check valves - Valves in flow paths to spray header - Isolation valves - Valves in spray additive flow path - Spray additive tank valves, including vacuum breakers
Main feedwater system	- Main feedwater isolation valves
Auxiliary feedwater system	- Auxiliary feedwater pumps and suction/discharge check valves - Valves in flow path to steam generators - Valves in suction lines - Valves between normal and ultimate heat sink suction sources - Relief valves and isolation valves
Primary containment system	- Containment isolation valves (various systems) - Containment combustible gas venting valves - Containment atmosphere sampling valves (if within the scope of 10 CFR 50.55a)
Component cooling water system	- Component cooling water pumps and discharge check valves - Valves in letdown cooling water flow path - Valves in reactor coolant pump seal injection and cooling water flow path - Valves needed to isolate a rupture of the thermal barrier - Relief valves
Spent fuel pool/pit cooling system	- Spent fuel cooling pumps and suction/discharge check valves - Valves in flow path from ultimate heat sink source supply
Service water system	- Service water pumps and suction/discharge check valves - Valves in flow path to auxiliary feedwater system - Valves in flow paths to emergency room coolers - Valves in flow paths to containment emergency coolers - Valves in flow paths to emergency diesel generator heat exchangers - Isolation and cross-tie valves - Valves in ultimate heat sink source flow paths - Valves in standby or backup service water, if applicable
Emergency diesel generator system (within scope of 10 CFR 50.55a)	- Fuel oil storage and transfer pumps and valves - Diesel generator external cooling (service water) - Engine air start check valves - Air receiver relief valves

Typical safety-related, Code-class system in pressurized-water reactors	Typical components in an IST program
Ventilation systems	- Pumps and valves in control room emergency cooling water supply flow path
Instrument air system (if within the scope of 10 CFR 50.55a)	- Air supply to containment purge valves - Air supply to power-operated relief valves (PORVs) - Air supply to MSIVs

Table 2.2 Typical Systems and Components in an Inservice Testing Program for a Boiling-Water Reactor

Typical safety-related, Code-class system in boiling-water reactors	Typical components in an IST program
Nuclear boiler and reactor recirculation system	- Primary system isolation valves - Excess flow check valves - Safety Relief Valve/Automatic Depressurization System (SRV/ADS) valves
Main steam system	- MSIVs and actuator valves (pilot valves, accumulator check valves) - Main steam safety and relief valves - Main steam safety valve discharge rupture diaphragm valve - MSIV leakage valves
High-pressure core coolant injection (HPCI) system	- Pump and suction/discharge check valve - Valves in injection flow path - Isolation valves, including valves in test lines - Excess flow check valves - HPCI pump turbine valves, including turbine exhaust vacuum breakers (unless considered skid-mounted)
High-pressure core spray system	- Pumps and suction/discharge check valves - Valves in injection flow path - Isolation valves, including valves in test lines
Reactor core isolation cooling (RCIC) system	- Pump and suction/discharge check valve - RCIC pump turbine valves - Excess flow check valves - Isolation valves
Reactor water cleanup system	- Containment isolation valves
Residual heat removal (RHR) system	- RHR pumps and suction/discharge check valves - Isolation and cross-tie valves - Pump suction relief valves - RHR heat exchanger thermal relief valves - Valves in injection flow path - Flow control valves

Typical safety-related, Code-class system in boiling-water reactors	Typical components in an IST program
Spent fuel pool cooling system	- Fuel pool pumps and suction/discharge check valves - Ultimate heat sink supply valve
Feedwater coolant injection and isolation condenser system (if applicable)	- Reactor feedwater pumps and suction/discharge check valves - Condensate pumps and suction/discharge check valves - Condensate booster pumps and suction/discharge check valves - Emergency condensate transfer pump and suction/discharge check and isolation valves - Isolation and bypass valves - Vent valves - Makeup to condenser shell check valves
Standby liquid control (SBLC) system	- SBLC pumps and suction/discharge check valves - Relief valves - Injection line valves - Pyrotechnic-actuated (squib) valves
Main feedwater system	- Isolation valves
Primary containment system	- Containment isolation valves including excess flow check valves (various systems) - Containment atmosphere monitoring system valves - Containment atmosphere dilution system valves - Containment pressure suppression and vents
Closed cooling or component cooling water system	- Pumps and suction/discharge check valves - Valves in flow paths to safety-related coolers
Service water system	- Pumps and suction/discharge check valves - Isolation and cross-tie valves - Valves in flow paths to safety-related coolers - Valves in flow paths to diesel generator coolers - Valves in standby or backup service water - Valves in flow path from ultimate heat sink source - Valves in residual heat removal service water flow path
Control rod drive system (portions within the scope of 10 CFR 50.55a)	- Scram dump valves - Scram discharge volume vent valves - Scram discharge volume drain valves - Accumulator rupture disks - Hydraulic control unit control valves - Drive water backflow prevention valves
Emergency diesel generator systems (if within the scope of 10 CFR 50.55a)	- Fuel oil storage and transfer pumps and valves - Diesel generator external cooling (service water) - Engine air start check valves - Air receiver relief valves

Typical safety-related, Code-class system in boiling-water reactors	Typical components in an IST program
Ventilation systems	- Pumps and valves in control room emergency cooling water supply flow path
Instrument air system (if within the scope of 10 CFR 50.55a)	- MSIV accumulator check valves - MSIV pilot valves - ADS valve accumulator check valves - ADS pilot valves
Traversing incore probe system (if within the scope of 10 CFR 50.55a)	- Containment isolation valves

Table 2.3 Example Data Table for Pumps

PLANT NAME/UNIT

PUMP TESTING PLAN

Revision 3
Date: 1-15-03
Page: 1 of 3

Pump List					Parameters				
SYSTEM	PUMP ID	P&ID NO.	COORD.	PUMP GROUP	S	P	dp	Flow	V (PR-1)
Residual Heat Removal	RHR-01 RHR-02 RHR-03	M-402, Sh. 1 M-402, Sh. 2 M-402, Sh. 2	D-4 G-4 F-5	A	(1) (1) (1)	(2) (2) (2)	Q/2Y Q/2Y Q/2Y	Q/2Y Q/2Y Q/2Y	Q/2Y Q/2Y Q/2Y
Auxiliary Feedwater	AFW-01 AFW-02 AFW-03	M-408, Sh 1 M-408, Sh. 1 M-408, Sh. 1	B-5 B-8 B-11	B	(1) (1) Q/2Y	(2) (2) (2)	Q/2Y Q/2Y Q/2Y	Q/2Y Q/2Y Q/2Y	2Y 2Y 2Y
Service Water	SWS-01 SWS-02 SWS-03 SWs-04	M-335, Sh 1 M-335, Sh 2 M-335, Sh. 3 M-335, Sh. 4	F-9 D-4 E-8 C-4	A	(1) (1) (1) (1)	(2) (2) (2) (20	Q/2Y Q/2Y Q/2Y Q/2Y	PR-3 PR-3 PR-3 PR-3	PR-2 PR-2 PR-2 PR-2
Standby Liquid Control	SLC-01 SLC-02	M-367, Sh. 1 M-367, Sh. 1	D-9 D-4	B	(1) (1)	2Y 2Y	(3) (3)	Q/2Y Q/2Y	2Y 2Y

Note (1): Pump is directly coupled to a **content** speed synchronous or induction type driver
Note (2): Discharge pressure is a required parameter for positive displacement pumps only.
Note (3)" dP is not a required parameter for positive displacement pumps.

Legend:

S	Speed
P_i	Pressure
dP	Differential Pressure
PR	Pump Relief Request
Q	Quarterly
V	Vibration

Table 2.4 Useful Abbreviations for Valve Data

Parameter	Abbreviation	Description
Valve Type	GT	Gate valve
	GB	Globe valve
	CK	Check valve
	RV	Relief valve
	SC	Stop check
	BF	Butterfly valve
	DI	Diaphragm valve
	EX	Pyrotechnic-actuated valve
	BA	Ball valve
Actuator Type	MO	Motor-operated
	SO	Solenoid-operated
	AO	Air-operated
	HO	Hydraulic-operated
	SA	Self-actuated
	MA	Manual
	PA	Pilot-actuated
Safety Position(s)	O	Open
	C	Closed
	O/C	Both open and closed
	T	Throttled
Tests(s) Performed	FS	Full-stroke exercise valve to safety position(s)
	PS	Part-stroke exercise valve
	LT	Leak-rate test valve to Section XI requirements
	LJ	Leak-rate test valve to Appendix J requirements
	ST	Measure the full-stroke times of the valve
	FT	Observe the fail-safe operation of the valve
	PI	Verify the valve position indication
	RV	Safety and relief valve test
	EX	Pyrotechnic-actuated valve test
Test Frequency	Q	Test performed once every 92 days
	CS	Test performed during cold shutdowns, but not more frequently than once every 92 days
	RF	Test performed each reactor refueling outage
	2Y	Test performed once every 2 years
	RV	Test relief valve at OM schedule
	SD	Disassemble, inspect, and manually exercise one valve from specified group each reactor refueling outage

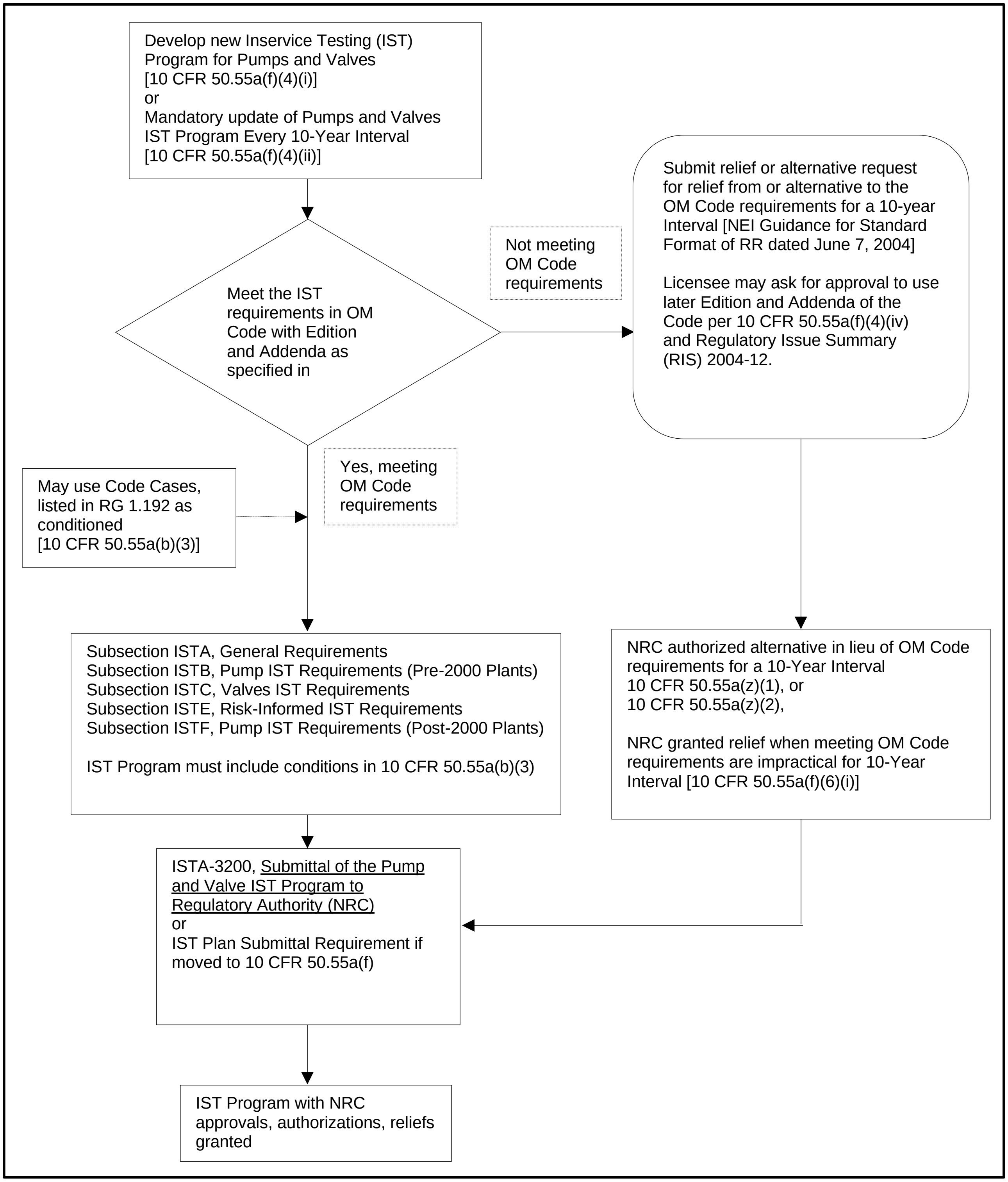

Figure 2.1 FLOW CHART – Development of Inservice Testing Program for Pumps & Valves*

* Note: Flow chart provided for guidance only. For complete details, see 10 CFR 50.55a

3 GENERAL GUIDANCE ON INSERVICE TESTING

3.1 Inservice Test Frequencies and Extensions for Valve Test

In the past, the ASME OM Code generally specified quarterly testing of valves. Now, the IST requirements and frequency are more dependent on the type of valve. For example, Mandatory Appendix II to the OM Code allows condition monitoring of check valves. Appendix III requires periodic exercising and diagnostic testing of MOVs. Mandatory Appendix IV requires quarterly stroke-time testing and performance assessment testing of AOVs based on their safety significance.

Where quarterly valve exercising is required, Subsection ISTC of the OM Code allows licensees to defer valve exercising to cold shutdown or refueling outages if it is not practical to exercise the valves during plant operation. Impractical conditions justifying test deferrals are those that could result in an unnecessary plant shutdown, cause unnecessary challenges to safety systems, place undue stress on components, cause unnecessary cycling of equipment, or unnecessarily reduce the life expectancy of the plant systems and components.

IST requirements and frequency for pumps depend on the pump group and the type of test. Table 3.1 (below) lists the tests and associated test frequencies required by the OM Code.

Table 3.1 ASME OM Code Tests and Test Frequencies for Pumps and Valves

Test	Frequency
Measure pump parameters	Subsection ISTB Once every 3 months (Group A, B Tests) Biennially (Comprehensive Test, Periodic Verification Test) Exceptions: - Pumps in systems that are out of service - Group B Pumps lacking required fluid inventory Subsection ISTF Inservice Test quarterly Periodic Verification Test biennially
Exercise Category A and B valves	Once every 3 months Exceptions: - Extension because of impracticality - Valves in regular use - Valves in systems out of service - MOVs covered by OM Code, Mandatory Appendix III

Test	Frequency
Measure stroke times of power-operated Category A and B valves	Once every 3 months Exceptions: - Extension because of impracticality - Valves in regular use - Valves in systems out of service - MOVs covered by OM Code, Mandatory Appendix III
Verify remote position indication	Once every 2 years, except where verified in accordance with OM Code appendices
Observe operation of fail-safe actuators for applicable valves	Once every 3 months, except for extension because of impracticality
Leak-test Category A and A/C valves	Once every 2 years
Test safety and relief valves, primary containment vacuum relief valves, and non-reclosing pressure relief devices	Test interval specified by ASME OM Code, Mandatory Appendix I
Exercise check valves	Once every 3 months Exceptions: - Extension because of impracticality - Valves in regular use - Valves in systems out of service - Check valves covered by OM Code, Mandatory Appendix II
Test pyrotechnic-actuated valves	20 percent tested once every 2 years. Charges shall not be older than 10 years. Additional testing and inspection required for pyrotechnic-actuated valves in post-2000 plants

3.1.1 Deferring Valve Testing to Each Cold Shutdown or Refueling Outage

For valves required to be exercised quarterly, the OM Code allows licensees to test valves during cold shutdowns if it is impractical to test the valves quarterly during plant operation. Paragraph ISTC-3500 provides guidance and alternatives. Therefore, exercising valves during cold shutdown outages does not constitute a deviation from the OM Code and does not require a relief request if the licensee determines that quarterly testing is impractical. Similarly, the OM Code allows licensees to test valves during each refueling outage if it is impractical to test the valves during cold shutdowns. In such instances, the licensee should identify the valves for which testing is deferred and the IST program document should specify the basis for determining that quarterly and/or cold shutdown testing is impractical.

In the past, the NRC staff has provided examples of valves that may be excluded from exercising (cycling) tests during plant operations. The excluded valves include the following examples:

1. All valves that would cause a loss of system function if they were to fail in a nonconservative position during the cycling test. Valves in this category would typically include all non-redundant valves in lines such as a single discharge line from the RWST or accumulator discharge lines in PWRs and the HPCI turbine steam supply and HPCI

pump discharge in BWRs. Other valves may fall into this category under certain system configurations or plant operating modes. For example, when one train of a redundant system [such as an emergency core cooling system (ECCS)] is inoperable, non-redundant valves in the remaining train should not be cycled because their failure would cause a loss of total system function.

2. All valves that would result in a loss of containment integrity if they failed to close during a cycling test. Valves in this category would typically include all valves in containment penetrations where the redundant valve is open and inoperable.
3. All valves that, when cycled, could subject a system to pressures in excess of their design pressures. For the purpose of a cycling test, it is assumed that one or more of the upstream check valves has failed unless positive methods are available for determining the pressure or lack thereof on the high-pressure side of the valve to be cycled. Valves in this category would typically include the isolation valves of the RHR or shutdown cooling system and, in some cases, certain ECCS valves.

The guidance in this NUREG and in the NRC's letters issued in 1976 to licensees do not supersede the TS requirements.

For valves required to be tested quarterly and such testing is practical, a licensee may request an alternative in accordance with 10 CFR 50.55a(z) to quarterly testing where such testing would impose a hardship (e.g., entering a limiting condition for operation of 3 to 4 hours in duration, repositioning a breaker from "off" to "on," and necessitating manual operator actions to restore the system if an accident were to occur while the test was in progress). For such situations, the risk associated with quarterly testing may outweigh the benefits that might otherwise be achieved. (Section 3.1.2 gives guidance on these types of situations.) Thus, it is appropriate for licensees to weigh the safety impact against the benefits of testing as a basis for deferring testing from a quarterly frequency to cold shutdown or refueling outages. NUREG/CR-5775, "Quantitative Evaluation of Surveillance Test Intervals Including Test-Caused Risks," dated February 1992, (ADAMS Accession No. ML19172A254) describes a method for making this comparison.

The following sections discuss issues related to deferring valve testing. These sections do not apply to testing that is required following maintenance or repair activities.

3.1.1.1 IST Cold Shutdown Testing

Although Subsection ISTC of the ASME OM Code does not include schedules for cold shutdown testing, an acceptable method is to ensure that the valves tested in the preceding cold shutdown are the last valves tested during the next cold shutdown, with the exception of valves that must be tested during each cold shutdown. The following is a sample schedule for 15 cold shutdown tests:

- First cold shutdown: Complete Tests 1, 2, 3, 4, 5, and 6
- Second cold shutdown: Complete Tests 7, 8, 9, and 10
- Third cold shutdown: Complete Tests 11, 12, 13, 14, 15, 1, 2, and 3
- Fourth cold shutdown: Complete Tests 4, 5, 6, and 7

Paragraph ISTC-3520, “Exercising Requirements,” discusses exercising valves during both plant operation and cold shutdown as circumstances and situations apply. While the discussion does not specifically address testing in hot standby or hot shutdown, valves should be exercised in the appropriate mode of operation. For a valve that cannot be tested in operation, testing might be practical during hot standby, hot shutdown, cold shutdown, or a refueling outage.

Valves that must be operable during cold shutdown may be tested during plant operation in accordance with paragraph ISTC-3520 or paragraph ISTC-3550, “Valves in Regular Use.” Paragraph ISTC-3550 applies if the component’s “normal use” is during cold shutdown.

3.1.1.2 Testing at a Refueling Outage Frequency for Valves Tested During Power Ascension

Paragraph ISTC-3520 specifies that valves that are tested on an RFO frequency should be tested before returning the plant to operation at power. Several licensees have indicated that they cannot test certain valves until power ascension begins. The NRC staff has included this section to provide guidance for such valves and to indicate that the operability TS would control the timing for testing such valves. It is intended that the IST program document will identify such valves as being tested on an RFO frequency, even though the plant may actually return to “operation” at power before the testing is completed. A similar intent applies to valves that are tested during power ascension from cold shutdowns (which are not RFOs); however, Subsection ISTC uses different language in discussing valves that are tested on a cold shutdown frequency.

Before beginning power ascension from an RFO, licensees normally complete the tests of those valves that are tested during each RFO. However, for valves that can only be tested during power ascension or at power, licensees may begin increasing the power level and changing modes in accordance with TS requirements and may test the applicable valves when plant conditions allow testing. This situation also could apply to valves that are tested during power ascension or at power following a cold shutdown outage.

NRC Recommendation

The ASME OM Code, paragraph ISTC-3520 requires that all valve testing scheduled for performance during an RFO shall be completed before returning the plant to operation. For valves which can only be tested during power ascension, TS requirements (for the valves or the associated system) determine when the valves are required to be operable. The testing for these valves may be scheduled for RFOs or during cold shutdown conditions, but completed during power ascension from the RFO.

The NRC staff has determined that testing of such valve during plant startup period but prior to reaching steady state full power operation is acceptable without a relief request, provided that the licensee meets all requirements of ISTC-3520.

Basis for Recommendation

The NRC staff has determined that the guidance in this section is consistent with OM Code, paragraph ISTC-3520, and the TS requirements and, therefore, is acceptable for meeting the OM Code provisions.

3.1.1.3 *De-Inerting Containment of Boiling-Water Reactors to Allow Cold Shutdown Testing*

According to 10 CFR 50.44, "Combustible Gas Control System for Nuclear Power Reactors," each BWR that is equipped with a Mark I or Mark II containment must have provisions for an inerted containment atmosphere during power operation to protect against a burn or explosion of hydrogen gas generated by the core metal-water reaction following a postulated loss-of-coolant accident (LOCA).

Licensees regularly monitor oxygen content in the containment atmosphere during normal power operation, and the plant's TS specify the maximum oxygen concentrations allowed. Since hydrogen generation is not a concern during cold shutdown or RFOs, the TS allow the containment atmosphere to be de-inerted. However, licensees do not routinely de-inert the containment during cold shutdown outages because of impracticality concerns associated with the time needed to de-inert and re-inert the containment, and because of the amount of nitrogen necessary for inerting.

For certain valves that are located in the inerted containment, paragraph ISTC-3500 allows licensees to perform testing during cold shutdown outages because it is not practical to test such valves during power operation. The NRC staff has determined that it is impractical to de-inert the containment during each cold shutdown outage solely to perform such routine testing or repair activities.

NRC Recommendation

The NRC staff considers it impractical to de-inert the containment merely to conduct regularly scheduled valve testing, and the OM Code allows licensees to defer such testing to an RFO when the containment is de-inerted for refueling or other reasons. The NRC staff also has determined that few outages require de-inerting, and it is unnecessary to maintain a separate schedule for valve testing. Consequently, testing is at the discretion of the licensee in the event of an extended cold shutdown that necessitates de-inerting the containment. Factors to be considered in the licensee's decision-making might include the length of the shutdown and the extent of other outage activities. In addition, for extended outages that last several months, the requirements of paragraph ISTC-3570 may apply for valves in systems that are out-of-service.

Basis for Recommendation

Subsection ISTC allows licensees to extend the test interval to defer valve testing to RFOs if such testing is impractical at quarterly intervals (during power operations) or during cold shutdown outages. Consequently, it is also acceptable for licensees to extend the test interval for those valves which cannot be tested unless the containment is de-inerted.

Unless the licensee has some other reason to enter the drywell during cold shutdown outages, the NRC staff regards it as impractical to de-inert the drywell during such outages merely to perform valve testing. The NRC staff's position is based on the time and effort needed to de-inert, re-inert, and replace lost nitrogen gas (which could delay the plant's return to power operation).

3.1.1.4 Stopping Reactor Coolant Pumps for Cold Shutdown Valve Testing

Licensees frequently defer the testing of certain valves in support systems that perform functions that are vital to the continued operability of the reactor coolant pumps, such as component cooling and the supply and return of seal water during cold shutdown. Exercising these valves while the reactor coolant pumps are operating could result in pump damage, and stopping the pumps could extend the cold shutdown period.

NRC Recommendation

The NRC staff recommends that licensees test the affected valves on an RFO schedule and during plant outages when the reactor coolant pumps are stopped for a sufficient period of time, but not more often than once every 92 days.

Basis for Recommendation

Subsection ISTC of the ASME OM Code allows licensees to extend the test interval to defer testing to RFOs when it is not practical to perform the tests during power operation or cold shutdown outages. The NRC staff believes that licensees need not schedule valve testing that requires stopping and restarting reactor coolant pumps during each cold shutdown solely to allow for the testing of such valves. This repetitive cycling would increase pump wear and stress, as well as the number of cycles of related plant equipment, and could extend the length of cold shutdown outages. Consequently, licensees may consider establishing a schedule to account for extended cold shutdown outages that would allow for valve testing when the reactor coolant pumps are stopped for a sufficient period of time. However, valves are to be tested at least during each RFO, as required by Subsection ISTC.

3.1.2 Entry into a Limiting Condition for Operation (LCO) To Perform Testing

The OM Code allows deferred testing until a cold shutdown or refueling outage, if testing is not practicable at power. The NRC staff believes that it is better to defer testing as allowed by the OM Code, rather than entering into an LCO ACTION to perform IST or requesting approval of a relief or exemption to perform such IST without entering into LCO ACTION. See Section 3.1.3 for guidance on scheduling of inservice testing.

3.1.3 Scheduling of Inservice Tests

The OM Code requires that testing is to be performed normally within a certain time period.

NRC Recommendation

To eliminate ambiguity concerning the IST intervals, the NRC staff recommends that licensees use the stated test frequency definitions in ISTA-3170, “Inservice Examination and Testing Frequency Grace,” of the OM Code (2017 Edition). See Table 3.2, below. For example, paragraph ISTC-3510 requires licensees to test Category A and B valves “nominally every 3 months.” For quarterly testing, the NRC staff recommends that licensees schedule the pump and valve tests such that a particular test is performed at approximately the same time within each quarter. For example, if a test procedure applies to many valves and, thus, requires 2 to 3 weeks or more to complete, the licensee would typically begin the procedure at approximately the same time in each quarter and include directions to perform tests in a specified order to ensure that specific valves are tested “at least once every 92 days.”

Table 3.2 ASME OM Code Terms for Inservice Testing Activities

Term	Required frequency for IST activities (at least once every):
Quarterly (or Every 3 months)	92 days
Semiannually (or Every 6 months)	184 days
Annually (or Every year)	366 days
x years	X calendar years where x is a whole number of years ≥ 2

The OM Code specifies performing the tests throughout extended shutdown periods for equipment that must be returned to operable status. Most equipment must be tested before being returned to service after being out-of-service for an extended period in accordance with TS requirements (if applicable). The OM Code provisions in paragraphs ISTB-3420 and ISTC-3570 specify that licensees need not follow the test schedule if the system in which the component was installed was declared inoperable or was not required to be operable. However, this applies only if the component was not out-of-service for repair or replacement. For repair or replacement, the component must be tested within 3 months of the system being returned to service.

Basis for Recommendation

To resolve the confusion regarding IST interval grace periods, ASME prepared OM Code Case OMN-20, “Inservice Test Frequencies,” with the 2012 Edition of the OM Code to address IST interval grace periods and the 25-percent margin. The OM Code Case OMN-20 is approved for use in RG 1.192, Revision 3, with conditions. ASME Code Case OMN-20 can be used with editions and addenda of the OM Code incorporated by reference in 10 CFR 50.55a. ASME Code Case OMN-20 has been incorporated in the OM Code (2017 Edition) in ISTA-3170. Prior to its acceptance in RG 1.192, ASME Code Case OMN-20 was approved for use in 10 CFR 50.55a.

The nuclear industry has developed TSTF-545 to remove the reference to the IST program from the standard technical specifications (STS). To avoid confusion with plant-specific technical specifications, licensees should follow the guidance in the OM Code (2017 Edition) or ASME Code Case OMN-20 as accepted in RG 1.192 for justifying grace periods for IST intervals.

The NRC staff considers that the grace periods for IST intervals allowed in the OM Code (2017 Edition) and ASME Code Case OMN-20 may be applied (except where prohibited as noted below) to IST intervals specified in requests for (a) relief from the OM Code that is granted by the NRC to a licensee in accordance with 10 CFR 50.55a(f)(6)(i), and (b) an alternative to the OM Code that is authorized by the NRC for use by a licensee in accordance with 10 CFR 50.55a(z). The submittal of a separate request by the licensee to the NRC for an IST interval grace period is not necessary, unless the staff specifically prohibits a grace period in granting the relief from, or authorizing the alternative to, the OM Code.

3.2 Initial 120-Month IST Program Interval

10 CFR 50.55a(f)(4)(i) requires that inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during the initial 120-month IST

program interval must comply with the requirements in the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a on the date 18 months before the date of issuance of the operating license under Part 50, or 18 months before the date scheduled for initial loading of fuel under a COL under Part 52 (or the optional ASME OM Code Cases listed in RG 1.192, as incorporated by reference in 10 CFR 50.55a(a)(3)(iii)), subject to the conditions listed in 10 CFR 50.55a(b).

3.3 120-Month Updates Required by 10 CFR 50.55a(f)(4)(ii)

10 CFR 50.55a(f)(4)(ii) requires licensees to revise their IST programs every 120 months to reflect the latest edition and addenda to the ASME OM Code incorporated by reference into 10 CFR 50.55a that is in effect 18 months before the start of the new 120-month IST interval.

After the initial 120-month interval, in accordance with 10 CFR 50.55a(f)(4)(ii), licensees must conduct inservice tests during successive 120-month intervals to verify the operational readiness of pumps and valves within the scope of the OM Code. In conducting these inservice tests, licensees must comply with the provisions of the latest edition and addenda of the OM Code incorporated by reference in 10 CFR 50.55a that is in effect 18 months before the start of the 120-month IST interval, subject to the conditions listed in paragraph (b). In addition, 10 CFR 50.55a(f)(5)(iv) specifies that where a pump or valve test requirement by the OM Code edition or addenda is determined to be impractical by the licensees and is not included in the new IST program interval, the basis for this determination must be submitted for NRC review and approval not later than 12 months after the expiration of the initial 120-month interval of operation from the start of the facility commercial operation and each subsequent 120-month interval of operation during which the test is determined to be impractical.

The ASME OM Code up to the 2017 Edition specifies in ISTA-3200, “Administrative Requirements,” that the Code user comply with the latest edition and addenda of the ASME OM Code adopted by the regulatory authority 12 months prior to the issuance of an Operating License or 12 months before initial fueling under a COL, and 12 months prior to the start of successful IST Program intervals. In that this NUREG cannot authorize alternatives to the ASME OM Code, licensees will need to submit an alternative request to follow the 18-month provisions in 10 CFR 50.55a(f)(4)(i) and (ii) rather than complying with the 12-month provisions in ISTA-3200, until the ASME OM Code is updated to reflect the 10 CFR 50.55a requirement.

3.3.1 Extension of IST Program Interval

The IST program interval may be extended in accordance with subparagraph ISTA-3120(c) in the ASME OM Code (2017 Edition):

> Each of the inservice examination and test intervals may be extended or decreased by as much as 1 yr. Adjustments shall not cause successive intervals to be altered by more than 1 yr. from the original pattern of intervals.

Subparagraph ISTA-3120(d) in the OM Code (2017 Edition) further states that, for units that are continuously out-of-service for 6 months or more, licensees may extend the IST interval during which the outage occurred for a period equivalent to the outage, and may extend the original pattern of intervals accordingly for successive intervals.

NRC Staff Recommendation

Licensees must establish the next updated IST program to the latest edition of the OM Code incorporated in the regulation 18 months before the new date. When extending its 120-month IST interval by as much as 1 year, as allowed by ISTA-3120, licensees may continue to apply the same OM Code edition and addenda from its current 120-month interval during this extended 1-year period. The NRC staff recommends that licensees document and inform the NRC of any change to extend before the start and end dates of the IST program interval allowed the OM Code. An extension beyond 1 year (other than for extended outages, as permitted by ISTA-3120) requires NRC authorization of an alternative to or exemption from the OM Code provisions of 10 CFR 50.55a, as applicable.

Basis for Recommendation

While it is not mandatory to maintain identical intervals for the ISI and IST programs, it is often desirable in order to maintain the same edition of the ASME BPV Code, Section XI, and ASME OM Code for all plant activities related to the ISI and IST programs. Even though 10 CFR 50.55a does not discuss extending the intervals, the Codes are incorporated by reference in the regulation and, therefore, have the same effect as the regulation. Although NRC approval is not required for 1-year extensions of the program interval, licensees would avoid any discrepancies in the program interval dates by informing the NRC of the extension and documenting it in the IST program document. Because the Codes do not allow extension beyond 1 year (other than for extended outages), such an extension would require NRC authorization of an alternative to or exemption from the Code provisions to comply with the regulatory requirements.

3.3.2 Concurrent Intervals

Several licensees have established concurrent intervals for all units at sites with multiple units, so that each unit is updated to a newer edition of the OM Code at the same starting date. Because the regulations do not specifically allow for adjustments to accommodate concurrent intervals among multiple units, when the interval start dates are to be concurrent, licensees may request a one-time alternative to or exemption from 10 CFR 50.55a or the OM Code, as applicable. If a licensee prefers not to request an alternative or exemption, the establishment of concurrent intervals would require that the licensee must perform program updates for a particular unit more often than once every 120 months. 10 CFR 50.55a(f)(4)(iv) permits IST of pumps and valves to meet the provisions in subsequent OM Code editions and addenda (or portions thereof) that are incorporated by reference in 10 CFR 50.55a, subject to the conditions listed, and subject to NRC approval. This regulation allows licensees to update their programs before the end of a 120-month program interval with NRC approval.

NRC Recommendation

If a licensee elects to use the same OM Code edition for multiple units, the licensee must request an alternative to the OM Code or exemption, to extend a unit’s interval by more than 1 year in order to place multiple units on a concurrent interval for the IST program. To establish concurrent intervals without an alternative or exemption, the licensee must update the referenced edition of the OM Code more frequently for the selected unit(s) to remain in compliance with 10 CFR 50.55a, except in the case where the interval dates are within 12 months, whereby the OM Code allowance for an extension would result in concurrent intervals. The NRC will likely authorize an alternative allowing the licensee to update to later editions of the OM Code, if the licensee uses the following guidelines:

1. Without obtaining an alternative or exemption, the licensee may perform the IST program for multiple units using the same edition of the OM Code at concurrent intervals if the initial interval for combining the programs is established such that no single unit is tested at an interval of more than 120 months (or no greater than the interval extension allowed by the OM Code). Thus, the licensee must use the interval for the first unit that was licensed for commercial operation to establish the interval dates and establish the correct OM Code edition according to the most recent required for either unit.
2. To exceed 120 months, other than as addressed in the OM Code for an extension, the licensee must first obtain authorization of an alternative to the OM Code or exemption from 10 CFR 50.55a unless the licensee intends to repeatedly update both units more often than the required 120 months. Under such an approach, the licensee would test each unit according to the most recent edition of the OM Code required for either unit.

The IST program document and the request for the alternative or exemption would typically describe the method for selecting the interval dates, specifying the dates at which the interval will begin and end, and comparing the effect of those dates with that of the dates that would otherwise be required.

Basis for Recommendation

The NRC staff believes that conducting IST programs for multiple unit sites using same OM Code edition could provide an improvement in program effectiveness.

3.3.3 Implementation of Updated Programs

Updating the IST program to a revised edition and addenda of the ASME OM Code is an extensive effort that involves changes to administrative and implementing procedures. Often, the revised requirements will necessitate establishing new reference values, such as by implementing a vibration program using velocity measurements rather than displacement measurements for pump testing. Implementing a new comprehensive pump test may be necessary for parameters that are not currently measured. New “reference values” for currently monitored parameters may not be necessary if previous reference values were acceptable. However, the OM Code does not specifically require licensees to establish new reference values simply because a later edition is used.

NRC Recommendation

The NRC staff recommends that, before beginning the first tests during the new interval, licensees revise the implementing procedures according to the appropriate requirements. When the testing requires baseline values to be reestablished to meet OM Code changes, this would typically involve establishing the new baseline (reference) values during the first quarterly or cold shutdown outage test performed in the new interval, if not before. Before performing tests during the first refueling outage, licensees would typically revise implementing procedures for the tests to be performed during that outage to incorporate any new requirements or components.

Before or during startup from the refueling outage, licensees must complete all tests that are required to be performed during the RFO, as required by the OM Code (ISTC-3500 and

Mandatory Appendix I, Section I-1300). If a licensee determines that timely implementation is not possible, the NRC staff recommends that the licensee submit a schedule to the NRC (1) before the beginning of the interval, or (2) before startup from the RFO if the interval begins while a plant is shut down for refueling.

For 120-month updated programs, the NRC staff recommends that licensees submit relief requests before the next inspection interval's start date to allow adequate time for NRC review and approval within 12 months after the interval start date (i.e., submit the updated program at least 3 to 6 months before the start date.)

In accordance with the regulations, when updating a program to a later edition of the OM Code, licensees must implement the updated program at the beginning of a 120-month interval. The regulations state that, where a pump or valve test specified by the OM Code is determined to be impractical and is not included in the revised IST program interval, the basis for this determination must be submitted for NRC review and approval not later than 12 months after the expiration of the initial 120-month interval of operation from the start of the facility commercial operation and for each subsequent 120-month interval of operation during which the test is determined to be impractical. However, experience has shown that licensees also identify impractical test provisions throughout the 120-month interval. In such cases, the NRC staff recommends that licensees request relief as soon as they identify the condition. Because the requirements are impractical, the licensee would test the applicable components using the method proposed in the relief request in the period of time from the beginning of the new interval (or from the time of identification) until the NRC staff completes its evaluation. This would include, for example, a situation in which a licensee identifies a solenoid valve that is within the scope of the IST program and is stroke-time tested but has no position indication, or if the licensee cannot meet the OM Code requirements because of design limitations and an alternative method may not comply with the OM Code requirements. Proposed alternatives to the OM Code requirements (rather than relief from "impractical" requirements) shall not be implemented until the NRC staff completes its evaluation (e.g., if a licensee proposes to implement a pump vibration program based on using spectral analysis, rather than the OM Code-specified method, the licensee must continue to meet the OM Code requirements until the NRC staff completes its evaluation).

Basis for Recommendation

When licensees update their IST programs to a revised edition of the OM Code, the NRC staff recognizes that changes might be completed over a period of time to allow for adequate review and approval; however, the NRC staff recommends completing the procedural revisions in a timely manner. The regulations do not allow a licensee to continue with a previous program while waiting for NRC review to grant relief requests and to authorize proposed alternatives for the next interval. The NRC staff recommendation that the request be submitted 3 to 6 months before the end of the inspection interval is based upon the expected time needed for the NRC staff to evaluate the request and advise the licensee.

3.3.4 General Comments on Inservice Testing Intervals

The NRC has received requests for IST programs and partial submittals that lack the dates of the intervals or the OM Code edition in use. Some licensees were not aware that the NRC may issue final rules amending 10 CFR 50.55a that are not reflected in the current printed version of the *Code of Federal Regulations*. Therefore, when those individuals revised their programs, they used the printed version of 10 CFR Part 50 to determine the OM Code edition cited in

10 CFR 50.55a at the specified time period before the interval start date. However, a more recent edition may have been incorporated by reference in 10 CFR 50.55a as noticed in the *Federal Register*, which may have resulted in the program being developed to an incorrect edition of the OM Code.

Additionally, several licensees have asked questions concerning phasing in the updated IST program. Generally, this is an acceptable approach for testing if the program does not involve any requests for relief from OM Code requirements.

NRC Recommendation

The NRC staff recommends that licensees include the interval dates and OM Code edition in each IST program submittal, regardless of whether it is for an entire program or only a partial submittal containing new or revised relief or alternative requests. The NRC staff must ensure that the interval dates are correct and that the evaluation is performed using the edition of the OM Code from which the licensee is requesting relief or an alternative. The NRC staff also recommends that licensees implement procedures to ensure that the individuals responsible for developing and maintaining the IST program are made aware of the regulatory changes made in 10 CFR 50.55a throughout the year.

For phasing in a new edition of the OM Code before the start of a new interval (or during an ongoing interval), the NRC staff recommends that licensees submit a plan and schedule to the NRC. If there are no issues that require NRC review, the testing can be phased into the appropriate edition of the OM Code (1) during the 18 months prior to the interval start date, or (2) during any time period identified by the licensee up to an interval start date, if the phasing-in begins in the middle of an interval and a licensee wants to use an edition of the OM Code that is more recent than that incorporated by reference in 10 CFR 50.55a.

Basis for Recommendation

The NRC staff has noted incorrect interval dates and OM Code editions cited in IST program submittals. The OM Code stipulates that licensees shall calculate the IST interval according to the number of calendar years that have passed since the power unit was placed into commercial service. Licensees may verify the licensing and commercial operation dates for their plants by reviewing the annual “NRC Information Digest” (NUREG-1350). For convenience, the licensees for several plants have established the initial interval as beginning on the date of their operating licenses or some other unspecified milestone. However, the NRC staff cautions that if the NRC revised 10 CFR 50.55a after the interval start date cited by the licensee and before the date of the operating license, and if the revision of 10 CFR 50.55a incorporated a later edition of the OM Code, the regulations may require use of a more recent edition than the licensee believes is required. Therefore, it is important that the IST program document state the OM Code edition and addenda used to develop the program, so that the NRC may verify the licensee’s correct use of the applicable OM Code edition and addenda.

3.4 Skid-Mounted Components and Component Subassemblies

The safety-related piping systems at a plant may include skid-mounted components or component subassemblies, such as valves in diesel air-start subassemblies, diesel skid-mounted fuel oil pumps and valves, steam admission and trip throttle valves for HPCI or auxiliary feedwater (AFW) pump turbine drivers, steam traps, and air supply system check valves and SOVs for main steam isolation valves. If the licensee’s FSAR identifies these components as safety-related, they are within the scope of the OM Code per ISTA-1100 as incorporated by reference in 10 CFR 50.55a. If the components are non-Code Class, the regulations in 10 CFR 50.55a(f)(4) allow the licensee to include those components in an augmented IST program. ISTC-1200, “Exclusions,” in paragraph (b) states that skid-mounted valves that are tested as part of the major component and are justified by the Owner to be adequately tested are excluded from the testing requirements of Subsection ISTC. ISTB-1200(c) and ISTF-1200(c) provide similar exclusions for skid-mounted pumps in pre-2000 and post-2000 reactors, respectively. If ISTB-1200(c), ISTC-1200(b), or ISTF-1200(c) apply to skid-mounted pumps and valves that are safety-related, these components continue to be subject to periodic testing in accordance with Appendix A and Appendix B to 10 CFR Part 50.

NRC Recommendation

The NRC staff considers that testing the major component is an acceptable means to verify the operational readiness of the skid-mounted components and component subassemblies if the licensee discusses this approach in the IST program document. Licensees should consider and document the specific measurements and attributes of major component testing which relate to the assessment of skid-mounted component condition. In addition, various continuous and periodic observations of the major components (such as System Monitoring Walkdowns or Operator Logs) may also support assurance of skid-mounted component readiness. This is acceptable for both Code components and non-Code components that are tested and tracked by an IST or augmented IST program.

Basis for Recommendation

Various pumps and valves that are procured as part of larger component subassemblies are often not designed to meet the requirements for components in ASME BPV Code Classes 1, 2, and 3. In RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” the NRC provides guidance on classifying components for Quality Groups A, B, C, and D. (For additional guidance, licensees should review NUREG-0800, Standard Review Plan.) When many of the components were procured, the IST requirements did not apply and, thus, the components might not have included features for IST performance. Licensees may, therefore, elect to use the IST program for testing these components and state in the IST program document that the surveillance tests of these components adequately test the skid-mounted components.

The OM Code addresses both components that are physically mounted on the skid, and those that are not mounted on the skid but function the same as skid-mounted components (e.g., check valves in the service water system that supply cooling water to a pump, or a system of valves that work together to perform a safety function), provided that testing the major component is adequate to test the function of the system component.

Testing of skid-mounted check valves are specifically discussed in Section 4.1.10.

3.5 Preconditioning of Pumps and Valves

3.5.1 Background

The NRC regulations in 10 CFR 50.55a require licensees to test pumps and valves at nuclear power plants to assess their operational readiness within the scope of the ASME OM Code. Criterion XI in Appendix B to 10 CFR Part 50 specifies that licensees must establish a test program to ensure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements and acceptance criteria contained in applicable design documents. Criterion XI further requires that test procedures must include provisions to ensure that (1) all prerequisites for the given test have been met, (2) adequate test instrumentation is available and used, and (3) the test is performed under suitable environmental conditions. Criterion XI then requires licensees to document and evaluate the test results to ensure that the test requirements have been satisfied. In order to effectively assess operational readiness, the performance of the tested pump or valve, and the conditions under which the pump or valve must be capable of performing its safety function, need to be fully understood. Any maintenance activities performed before actual inservice testing are called “preconditioning” or “grooming” and this will adversely affect the validity of the test results.

3.5.2 NRC Guidance

In Information Notice (IN) 97-16, “Preconditioning of Plant Structures, Systems, and Components Before ASME Code Inservice Testing or Technical Specification Surveillance Testing,” the NRC staff discussed the longstanding concern regarding unacceptable preconditioning of plant SSCs before testing. The NRC staff noted that experience has demonstrated that some testing cannot be performed without disturbing or altering the equipment. The NRC staff also stated that any such disturbance or alteration would be expected to be limited to the minimum necessary to perform the test and to prevent damage to the equipment. In addition, the NRC staff alerted licensees that, in certain cases, the safety benefit of some preconditioning activities might outweigh the benefits of testing in the as-found condition.

The NRC staff has provided guidance to the NRC’s regional offices and inspectors with respect to preconditioning of plant equipment prior to OM Code and TS testing. For example, guidance is found in the following documents (see Section 9, References, for the ADAMS Accession Numbers for these documents):

- NRC memorandum, dated July 2, 1996, from Frederick J. Hebdon, Director, Project Directorate II-3, Division of Reactor Projects I/II, Office of Nuclear Reactor Regulation, to Jon R. Johnson, Acting Director, Division of Reactor Projects, Region II, in response to Technical Assistance Request TIA 96-007: “Regulatory Acceptability of Lubricating Valves Prior to Surveillance Testing”
- NRC Inspection Manual Part 9900, “Technical Guidance: Maintenance – Preconditioning of Structures, Systems, and Components before Determining Operability,”
- Attachment 22, “Surveillance Testing,” to IP 71111, “Reactor Safety: Initiating Events, Mitigating Systems, Barrier Integrity”

The guidance provided in these documents is instructive for nuclear plant personnel in providing assurance that testing conducted as part of the IST program is capable of assessing the operational readiness of pumps and valves within the scope of the OM Code.

Preconditioning

NRC Manual Chapter 9900 (Technical Guidance: Maintenance) defines preconditioning as the “alteration, variation, manipulation, or adjustment of the physical condition of an SSC before

Technical Specification surveillance or ASME Code testing." Any maintenance activities performed before actual inservice testing are called "preconditioning" or "grooming" and this might adversely affect the validity of the test results.

Acceptable Preconditioning:

The licensee may consider several factors in determining whether an activity constitutes acceptable preconditioning of a pump or valve to be tested. For example, an activity would constitute acceptable preconditioning of a pump or valve if it is performed to protect personnel or equipment, or to meet the manufacturer's recommendations (or based on industry-wide operating experience). If a preventive maintenance activity (such as valve stem lubrication or pump venting) periodically occurs prior to testing, the licensee might justify the acceptability of this infrequent preconditioning of a pump or valve if the licensee evaluates the effect of the activity on the overall ability to assess the operational readiness of the pump or valve, and to trend degradation in its performance. As noted in the inspection guidance, the licensee should have evaluated and documented the activity as acceptable preconditioning before performing the testing.

Unacceptable Preconditioning

Some activities would constitute unacceptable preconditioning of a pump or valve to be tested under the IST program. NRC Manual Chapter 9900 (Technical Guidance: Maintenance) defines unacceptable preconditioning of an SSC as an activity before or during surveillance or testing that alters one or more of the SSC's operational parameters which results in acceptable test results. For example, a preventive maintenance activity might constitute unacceptable preconditioning of a pump or valve if the licensee routinely conducts the activity prior to testing. NRC Inspection Manual, IP 71111, Attachment 22, instructs NRC inspectors to evaluate the acceptability of any preconditioning of equipment in preparation for surveillance tests. Similarly, NRC inspectors also verify that licensees do not routinely schedule preventive maintenance activities prior to testing in order to help ensure that the test is passed satisfactorily. In addition to activities related to an individual pump or valve, maintenance or surveillance activities involving several SSCs, including the scheduling or timing of such activities, can inadvertently result in unacceptable preconditioning of a pump or valve.

NRC Manual Chapter 9900 (Technical Guidance: Maintenance) provides a series of questions that NRC inspectors should consider when evaluating the acceptability of an activity that appears to involve preconditioning of a plant SSC. With respect to pumps and valves, those questions can be interpreted as follows:

- Does the practice performed ensure that the pump or valve will meet its testing acceptance criteria?
- Would the pump or valve have failed the test without the preconditioning?
- Does the practice bypass or mask the as-found condition of the pump or valve?
- Is preventive maintenance routinely performed on the pump or valve just before testing?
- Is preventive maintenance on the pump or valve performed only for scheduling convenience?

According to the NRC technical guidance, an activity constitutes unacceptable preconditioning if an affirmative answer is determined in response to any of these questions, and the activity meets the definition of unacceptable preconditioning provided in the inspection guidance. Licensees are encouraged to consider such questions as part of their determination of whether an activity related to a pump or valve in their IST program constitutes unacceptable preconditioning.

The following are examples of acceptable preconditioning:

1. The running of prelube booster pumps prior to diesel starts is allowed by NUREG-1431, "Standard Technical Specifications – Westinghouse Plants," when documented and approved in technical specifications.
2. To help prevent damage due to hydro-locking, NUREG-1216, "Safety Evaluation Report Related to the Operability and Reliability of Emergency Diesel Generators Manufactured by Transamerica DeLaval (TDI), Inc. (August 1986)," recommends that TDI diesel generator engines be rolled or cranked with cylinder petcocks open, using the air start system to purge any water from the cylinders before performing preplanned startups and testing. This allowance is made for a limited number of starts per year. (Note: The licensee should evaluate the amount of water collected to determine if it is excessive and whether operability of the diesel is affected.) The NUREG still directs that a number of starts are conducted in the as-found condition. This has become an accepted industry practice for this particular diesel because experience has shown that TDI diesel cylinders were prone to collecting water in the cylinders when idle. However, rolling (or cranking) the diesel using the air start system may be considered unacceptable preconditioning for other types of diesels that are not prone to water accumulation.
3. Gas accumulation in the suction piping of pumps or condensate accumulation in turbine steam supply lines can result in an equipment performance degradation. Verifying pump suction piping is gas free and turbine steam supply lines are condensate free prior to planned equipment operation, including surveillance testing, is a good operating practice and may improve equipment reliability. Performing these evolutions may be acceptable provided that it does not remove a pre-existing adverse condition without proper identification and evaluation. However, routine uncontrolled pump venting or draining turbine steam supply lines directly preceding surveillance testing without proper controls is unacceptable preconditioning.

The following are examples where pump venting or draining condensate from turbine supply lines would be considered acceptable:

1. Periodic venting of pumps which is not routinely scheduled to be performed directly prior to surveillance testing, but may occasionally be performed before surveillance testing. For example, it would be acceptable if the licensee routinely vents a pump weekly and then randomly conducts pump surveillance tests at various times during the week.
2. Pump venting directly prior to surveillance testing is acceptable provided that the venting operation has proper controls. A technical evaluation is required to establish that the amount of gas vented would not adversely affect pump operation. If an unacceptable amount of gas is vented an operability evaluation of the as-found (pre-vented) condition is required.
3. Turbine steam supply lines can be drained prior to surveillance testing provided that evolution has proper controls. A technical evaluation is required to support that the condensate drained would not have an adverse effect on the turbine operation. If an

unacceptable amount of condensate is drained, an operability evaluation of the as-found condition is required.

If the licensee chooses to precondition SSCs, the effects on equipment performance or condition should be documented in an engineering evaluation. The engineering evaluation should be performed using procedures to ensure that design and licensing bases are satisfied.

The following examples of unacceptable preconditioning are taken from NRC inspection reports:

1. Electrical

   a. Inspectors noted that electrical loads were removed from a number of 480-volt circuit breakers before surveillance testing was performed. In addition, surveillance procedures instructed technicians to inspect, clean, and lubricate several breakers before performing as-found testing. Accordingly, a violation for failure to maintain appropriate test controls in accordance with 10 CFR Part 50, Appendix B, Criterion XI, was issued.

2. Mechanical

   a. Four air-operated containment spray flow control valves were unacceptably preconditioned before stroke time surveillance testing by having the valve stems lubricated. Accordingly, a violation was issued for failure to develop appropriate test controls in accordance with 10 CFR Part 50, Appendix B, Criterion XI. The licensee's administrative procedures failed to ensure that these stroke time tests were performed under suitably controlled conditions.

   b. During observation of surveillance testing, inspectors noted that certain heat exchangers had their air-operated inlet valve and outlet valves controlled by a single hand switch on the main control room panel. The safety function of these air-operated valves was to open or to remain open in response to an accident signal. A surveillance test was performed quarterly to ensure that the opening function was intact and was not degrading. During the test, with the valves closed, the hand switch was taken to the open position and the opening stroke time for the "A" valve was recorded. At the same time, the "B" valve cycled open. After both valves were closed, the hand switch was again taken to the open position, and the opening time for the "B" valve was recorded. The inspectors considered the manner in which this test was conducted to constitute inappropriate preconditioning of the "B" valves. Since the stroke time test of the "A" valve was performed first during each test, the stroke time of the "B" valve was always checked a short time after the "B" valve had been cycled with the "A" valve. Therefore, an as-found stroke time anomaly for one of the "B" valves may not be detected if the pretest stroke eliminated the anomaly. Accordingly, a violation for failure to maintain appropriate test control in accordance with Criterion XI of Appendix B to 10 CFR Part 50 was issued.

   c. Inspectors identified the practice of operating the turbine-driven AFW pumps immediately before performing surveillance tests and the practice of venting the RHR pumps immediately before performing surveillance tests as examples of unacceptable preconditioning. The failure to test safety-related equipment under suitably controlled conditions and in accordance with design and licensing bases was cited as a violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings."

d. Inspection findings indicated that plant operators had recognized that performing a series of different HPCI system and RCIC surveillance tests in sequence, without allowing sufficient time for the system to cool down between tests, would constitute unacceptable preconditioning. However, the licensee personnel did not identify or document the full extent of the preconditioning conditions, nor did they initiate appropriate corrective actions to ensure that preconditioning would not be repeated. The failure of the licensee personnel to fully recognize and initiate action to correct preconditioning concerns was identified as a violation of 10 CFR Part 50, Appendix B, Criterion V.

3. Instrumentation and Control

   a. Inspectors noted that the surveillance procedure for the containment and drywell hydrogen analyzer calibrations required technicians to check and adjust reagent gas flow before obtaining the as-found calibration data. However, adjusting the reagent gas flow could affect the as-found condition of the analyzer and invalidate the test results. The inspectors determined that the test procedure was inadequate. Accordingly, a violation was issued on the basis of TS 5.4.1.a, which requires that specific written procedures be established for surveillance tests.

3.5.3 ASME OM Code Guidance

The ASME OM Code relies on the licensee to determine whether an activity would constitute acceptable or unacceptable preconditioning of a pump or valve prior to testing under its IST program, except in a few limited instances. For example, OM Code Case OMN-1 specifies that MOV inservice tests shall be conducted in the as-found condition, and that maintenance activities shall not be conducted if they might invalidate the as-found condition for inservice testing. The 2009 Edition of the OM Code incorporates OM Code Case OMN-1 (including the as-found testing provision), and OMN-11 as Mandatory Appendix III to the OM Code, to replace quarterly MOV stroke-time testing with periodic exercising and diagnostic testing. Where the OM Code does not specify provisions for as-found testing of a pump or valve, the licensee is responsible for determining whether an activity constitutes acceptable or unacceptable preconditioning of a pump or valve to be tested under its IST program.

3.5.4 NRC Recommendation

The NRC staff has provided examples of acceptable and unacceptable preconditioning of plant components prior to testing in such documents as IN 97-16 and NRC Inspection Manual Part 9900 (Technical Guidance), “Maintenance – Preconditioning of Structures, Systems and Components before Determining Operability.” Where the OM Code does not provide specific provisions related to as-found testing of a pump or valve in the IST program, the NRC staff considers acceptable preconditioning to include such activities as (1) periodic venting of pumps, where not routinely scheduled directly prior to testing but occasionally performed before testing; (2) pump venting directly prior to testing, provided that the venting operation has proper controls with a technical evaluation to establish that the amount of gas vented would not adversely affect pump operation; (3) occasional lubrication of a valve stem prior to testing of the valve, where stem lubrication is not typically performed prior to testing; (4) unavoidable movement attributable to the setup and connection of test equipment; and (5) test instrument venting directly prior to equipment testing. In each instance of acceptable preconditioning, the NRC staff will expect the licensee to have available a documented evaluation of the preconditioning activity and a justification for continued confidence in the capability of the IST program to assess the operational readiness of the pump or valve. Generic evaluations may be acceptable as long as the evaluation bounds the conditions of specific activity performed on the SSC.

By contrast, the NRC staff considers unacceptable preconditioning of pumps and valves in the IST program to include such activities as (1) routine lubrication of a valve stem prior to testing the valve; (2) operation of a pump or valve shortly before a test, if such operation could be avoided through plant procedures with personnel and plant safety maintained; and (3) venting a pump immediately prior to testing without proper controls and scheduling. Licensees may evaluate applicable NRC staff documents to determine whether specific activities prior to testing constitute acceptable or unacceptable preconditioning of a pump or valve in the IST program. The NRC staff encourages licensees to contact their NRC resident inspector or NRR project manager if questions arise regarding potential preconditioning of a pump or valve to be tested under the IST program.

3.6 Testing in the As-Found Condition

The intent of the IST program is primarily to detect and monitor the degradation or rate of changes of a component after a period of operation or stand-by conditions. NRC staff review of IST programs during 1985 through 1991 with older editions of ASME BPV Code, Section XI, and the OM Code, had been interpreted to require that IST activities be performed in the “as-found” condition. However, later editions and addenda of the OM Code, with the exception of Mandatory Appendix I for safety/relief valves and Mandatory Appendix III for MOVs, do not specifically require licensees to test components in the “as-found” condition. If “as-found” tests were not performed, degradation mechanisms or rate of degradation resulting from previous period of operation or stand-by condition cannot be identified and obtained. Therefore, this section of NUREG-1482 is provided to ensure that the intent of the IST program is retained.

OM Code, Mandatory Appendix I, requires licensees to measure the initial lift of safety relief valves to determine whether additional valves are to be tested. OM Code, Mandatory Appendix III, paragraph III-3300(b), states that inservice tests shall be conducted in the as-found condition.

The “as-found” condition is generally considered to be the condition of a valve without pre-stroking or maintenance. The OM Code does not require stroke-time testing or check valve stroking prior to maintenance; however, degradation mechanisms may not be identified if the licensee does not perform any as-found testing. Therefore, the NRC staff encourages licensees to perform as-found testing, where practical. The NRC staff also cautions licensees to consider the timing of maintenance with regard to the required test intervals and the potential for preconditioning. Post-maintenance testing is required when the maintenance could have affected the valve’s performance. As-found testing may also apply to pumps in a similar fashion. Most inservice testing is performed in a manner that generally represents the condition of a standby component if it were actuated in the event of an accident (i.e., no preconditioning prior to actuation).

3.7 Testing at Power

In an effort to shorten refueling outages, many licensees are trying to perform as much maintenance, testing, and surveillance as possible with the nuclear power plant online. For example, several licensees have submitted alternative requests to obtain NRC authorization to conduct inservice testing once per refueling cycle, rather than during the RFO as prescribed by the OM Code. In preparing (and evaluating) such alternative requests, licensees (and the NRC staff) should consider several factors to ensure that the licensee's proposed alternative provides an acceptable level of quality and safety. If a licensee is testing a particular valve during RFOs, the licensee may have determined that it is impractical to test the valve quarterly during operation. The licensee's IST program document should, therefore, discuss the basis for deferring the testing from quarterly (and during cold shutdowns) to RFOs. Relief requests to perform testing once each refueling cycle with the nuclear power plant on line should be prepared in lieu of the RFO justification for each affected valve or group of valves. If necessary, the licensee should revise the RFO justification to be consistent with the relief request.

Licensees (and the NRC staff) should also consider whether the testing can be accomplished within the allowed outage time permitted by any applicable technical specification. In general, the time necessary to complete the testing should be significantly less than the allowed outage time. This is to preclude TS violations or the need to issue exigent TS amendments or notices of enforcement discretion (NOEDs). In addition, licensees should not conduct non-corrective maintenance/testing activities at power if the associated post-maintenance testing cannot reasonably be accomplished until the next outage.

Sometimes, there is a tradeoff between testing components at power (e.g., when they could be needed to mitigate the consequences of an accident) and testing them during outages (e.g., when there may be greater reliance on shutdown cooling or when other equipment is necessarily out-of-service). Licensees should quantitatively or qualitatively address the risks associated with testing components on line, rather than testing during the RFO. If the proposed testing could have a significant risk impact, or if its justification includes risk-related arguments, the relief request should be prepared and reviewed in accordance with RG 1.174, as applicable. Licensees should also identify any compensatory measures to be established as a means to reduce the impact (e.g., risk and operational worker safety) of testing with the nuclear power plant at power.

If relevant, licensees should also provide information on how testing at power (rather than testing during RFOs) will affect scheduled maintenance work windows for the applicable system (i.e., whether the testing can be completed within the work windows or whether it will extend either the shutdown or at-power work windows). In addition, licensees will need to develop a new estimate of the maintenance unavailabilities that reflects the increased maintenance activities at power, and will need to document the basis for the new estimate (e.g., use plant logs or maintenance data to include in the current estimate of the maintenance unavailabilities those activities that were being performed during shutdown that will now be performed at power)[2]

At times, testing (or the disassembly and inspection of components) during RFOs can be more advantageous than at-power operations from a worker safety perspective (for example, systems may be cold and depressurized). When requesting NRC authorization to perform testing with the nuclear power plant on line, licensees should consider worker safety and should discuss whether the applicable components can be adequately isolated and restored.

---

[2] It should be noted that the assessment of risk resulting from performance activities as required by 10 CFR 50.65(a)(4) of the Maintenance Rule is not sufficient justification for testing components at power. This assessment is required for maintenance activities performed during power operations or during shutdowns. The configuration risk management does not address the relative merits of testing at power verses testing during refueling outage.

In Section 11.2.3 of Nuclear Management and Resources Council (NUMARC) 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," the Nuclear Management and Resources Council (NUMARC), now NEI, provided additional guidance for conducting online maintenance and testing. It states, in part:

> Online maintenance [and testing] should be carefully managed to achieve a balance between the benefits and potential impacts on safety, reliability or availability. For example, the margin of safety could be adversely impacted if maintenance is performed on multiple equipment or systems simultaneously without proper consideration of risk, or if operators are not fully cognizant of the limitations placed on the plant due to out of service equipment. Online maintenance should be carefully evaluated, planned and executed to avoid undesirable conditions or transients, and to thereby ensure a conservative margin of core safety.

3.8 Potential Adverse Impact on Plant Components from Flow-Induced Vibration

Nuclear power plant operating experience has revealed that plant components can be adversely affected by flow-induced vibration caused by hydrodynamic and acoustic loading in nuclear power plant fluid systems. For example, BWRs have experienced damage to plant components in main steam systems as a result of vibration caused by acoustic resonance during power uprate operation. Flow-induced vibration has also resulted in damage to components in the reactor coolant system at a PWR plant. Licensees of new nuclear power plants, and licensees of currently operating plants planning to implement a power uprate, should ensure that the potential for adverse effects on plant components from flow-induced vibration is addressed during PST or IST activities to provide confidence in the established IST intervals. See OM Condition: Flow-Induced vibration, 10 CFR 50.55a(b)(3)(iii)(C).

4 SUPPLEMENTAL GUIDANCE ON INSERVICE TESTING OF VALVES

The NRC staff has developed the following recommendations for valves that may be a part of an IST program at nuclear power plants. The types of valves discussed herein are covered by the ASME OM Code. Specifically, these include check valves (Section 4.1), POVs (Section 4.2), safety/relief valves (S/RVs) (Section 4.3), and other miscellaneous valves (Section 4.4).

4.1 Check Valves

The NRC staff considers check valves, and other automatic valves designed to close without operator action following an accident, to be "active" valves that would be classified as such in the IST program (reference, for example, Section B 3.6.3 of the NUREG-1431, Revision 4, Westinghouse Standard Technical Specifications). Similar criteria could be applied to the opening function of a check valve. The flow through a check valve would be blocked by any condition that precludes flow through the system. For example, installing a flange or closing another valve (other than a check valve) in the line would block flow. A valve that is "positively held in place" would be one that has an operator or other auxiliary device that maintains the disk in an open or closed position, such as a stop check valve.

SECY-77-439, "Single Failure Criterion," dated August 17, 1977, which was referenced in several plants' licensing bases, discusses the failure of a check valve to move to its correct position as a passive failure; however, this does not correspond to the issue of "active" versus "passive" for the purpose of the IST program. SECY-94-084 specifies that check valves in passive cooling systems in new reactors might need to be addressed by the single failure criterion unless justified with a high reliability.

The OM Code defines valves that are self-actuating in response to some system characteristic, such as flow direction, for fulfillment of the required function(s) as Category C valves. The OM Code also defines valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required function(s) as Category A valves. Those check valves (Category C valves) that must also be leak-tight (Category A valves) would be designated as Category A/C in the IST program.

Whereas the OM Code only requires licensees to exercise Category C check valves on a periodic basis, Category A/C check valves must be leak tested in addition to being exercised. The NRC staff has found that, in many instances, licensees are not assigning check valves to Category A/C, despite the fact that the licensees take credit for the check valve providing an essentially leak-tight function.

When determining the proper categorization of a check valve, a licensee should consider all applicable aspects. For example, the licensee should determine (1) whether the flow requirements for connected systems can be achieved with the maximum possible leakage through the check valve, (2) the effect of any reduced system flows resulting from the leakage on the performance of other systems and components, (3) the consequences of the loss of water from the system, (4) the effect that backflow through the valve may have on piping and components, such as the effect of high temperature and thermal stresses, and (5) the radiological exposure to plant personnel and the public caused by the leak. If any of the above considerations indicate that Category C testing might not be adequate, licensees should assign the check valve to Category A/C and should comply with the associated leak testing requirements.

Licensees may refer to IN 91-56, "Potential Radioactive Leakage to Tank Vented to Atmosphere," dated September 19, 1991, for information on the categories assigned to valves that function to close. These valves may also function to prevent leakage above an assumed limit to prevent the plant from exceeding the limits in 10 CFR Part 100, "Reactor Site Criteria." Section 4.1.1 herein discusses backflow testing of check valves in series.

Paragraph ISTC-3550 of the OM Code discusses valves in regular use and states that valves that operate in the course of plant operation at a frequency that would satisfy the exercising requirements need not be additionally exercised, provided that the observations otherwise required for testing are made and analyzed during such operation, and recorded in the plant record at intervals no greater than specified in paragraph ISTC-3510. Even if licensees "exercise" check valves in accordance with paragraph ISTC-3550, they need to be included in the valve list in the IST program document, and the record (e.g., plant log, test procedure) needs to indicate that the test requirements are met.

For grouping valves in multiple nuclear power plants of like design and construction, if the plants are "identical" and the grouped valves have similar operational experience and otherwise meet the grouping criteria, it is acceptable for licensees to group valves from multiple plants. If a potentially generic problem is identified through disassembly and inspection during an RFO, the licensee must inspect all valves in the group in that plant during the RFO. If the other plant is not also in an RFO, inspection of the valves in the group that are installed in that plant may be deferred to the next RFO unless the licensee's evaluation of the problem indicates that it could impact the safety of continued operation. "Grouping" may also be applied to the use of nonintrusive techniques as discussed in Section 4.1.2 (below), although the focus is slightly different, in that all of the valves in the group are tested at a specifies maximum test interval as detailed in Appendix II of the OM Code. At least one of the nonintrusive techniques for a valve group need to be performed on each valve of the group at approximate equal intervals not to excess the maximum test interval; therefore, all valves in the group need to be in the same nuclear power plant.

The NRC issued the following Information Notices and Bulletins (BL) on IST for check valves:

- IN 82-08, "Check Valve Failures on Diesel Generator Engine Cooling System"
- IN 83-54, "Common Mode Failure of Main Steam Isolation Nonreturn Check Valves"
- IN 88-70, "Check Valve Inservice Testing Program Deficiencies"
- IN 2000-21, "Detached Check Valve Disk Not Detected by Use of Acoustic and Magnetic Nonintrusive Test Techniques"
- BL 83-03, "Check Valve Failures in Raw Water Cooling Systems of Diesel Generators"

4.1.1 Closure Verification for Series Check Valves without Intermediate Test Connections

Some plants have piping configurations that include two check valves in series with no provision for verifying that each valve can close. These valves might perform a safety function in the closed position. For example, the valves might be required to prevent the gross diversion of flow rather than to be leak-tight. The OM Code requires that each valve performing safety functions must be stroked to the position(s) required for the valve to perform those functions. The requirements for testing two check valves in a series configuration are addressed in paragraph ISTC-5223, “Series Valve in Pairs,” for pre-2000 plants as follows:

> If two check valves are in a series configuration without provisions to verify individual reverse flow closure (e.g., keepfill pressurization valves) and the plant safety analysis assumes closure of either valve (but not both), the valve pair may be operationally tested closed as a unit. If the plant safety analysis assumes that a specific valve or both valves of the pair close to perform the safety function(s), the required valve(s) shall be tested to demonstrate individual valve closure.

NRC Recommendation

Both check valves in a series pair should be tested to demonstrate individual valve closure if the plant safety analysis credits or otherwise requires both valves. For example, GDC 14, “Reactor Coolant Pressure Boundary,” requires licensees, in part, to test the valves in the reactor coolant pressure boundary to demonstrate extremely low probability of abnormal leakage. Pressure isolation valves are a special case of reactor coolant pressure boundary valves, which are generally required to be individually leakage tested at a frequency specified by the TS and the OM Code.

Systems containing series pair check valves might have provisions for verifying that at least one valve is capable of closing. These provisions enable the licensee to measure or observe operational parameters such as leakage, pressure, or flow during each quarter, each cold shutdown outage, or each refueling outage. However, testing the series pair as a unit provides no assurance that both valves close. The only indication of a problem would be the failure of both valves in the series. If the valve pair is operationally tested closed as a unit, as allowed by paragraph ISTC-5223, because the plant safety analysis assumes closure of either valve (but not both), and the tested unit closure capability cannot be demonstrated, both valves should be declared inoperable and corrective actions must be taken for both valves.

If it is not practical to flow test the pair of check valves in accordance with the OM Code, the licensee may demonstrate the closure safety function of each valve by other positive means, such as nonintrusive testing, or disassembly and inspection. However, licensees should not use these methods to verify leak tightness, which requires Category A valve testing.

Basis for Recommendation

The requirements of paragraph ISTC-5223 are contained in the OM Code, 2004 Edition with 2005 and 2006 Addenda through 2017 Edition, as incorporated by reference in 10 CFR 50.55a.

Subparagraphs ISTC-5221(a) and/or ISTC-5221(c) allow the use of nonintrusive examination, disassembly and inspection, or other positive means, for check valves that have no provision for testing individual valves or practical means to demonstrate the closure capability of each valve by flow, but not for verifying leak-tightness.

Keep-fill valves are a special case, in that they are redundant check valves in a system in which only one valve of a series is actually necessary to perform a system’s intended function. Licensees have proposed to exclude the upstream valve from the IST program. However, recognizing that neither valve can individually demonstrate a closure function, and that the OM

Code alternative allows the valve pair to be operationally tested closed as a unit, the NRC staff's position is that both valves need to be included in the IST program and should be operationally tested as a pair to prevent reverse flow. Upon observing leakage, the licensee should disassemble, inspect, and repair or replace both valves (as necessary) before return to service.

4.1.2 Exercising Check Valves with Flow and Nonintrusive Techniques

The OM Code requires both an open and close test for check valves and that licensees exercise check valves to the position(s) required to fulfill their safety function(s). To verify the disk position of check valves that do not have external disk position indication, the OM Code allows licensees to use indirect evidence (such as changes in system pressure, flow, temperature, or level) or other positive means. An acceptable test method must demonstrate by positive means that a check valve disk moves to the position necessary to fulfill its safety function. The demonstration might require that the valve be exercised "full-open" to the backstop. The full-stroke to the open position may be verified either by passing design flow or by other positive means such as nonintrusive techniques. The "other positive means" must be repeatable to meet the intent of the Code.

NRC Recommendation

The licensee may use nonintrusive techniques for IST of check valves. Relief is not required except as would be necessary for the testing frequency if the test interval extends beyond each RFO as allowed by the OM Code. The licensee may use nonintrusive techniques to verify the valve's capability to open, close, and fully stroke (as necessary) if it is qualified for the application in accordance with the plant's QA program requirements. A qualified nonintrusive technique is one that has been successfully and reliably demonstrated for the examination method and specific valve application. The licensee may qualify the technique and application on its own equipment, subcontract it to a vendor, or rely on the results of the Nuclear Industry Check Valve Group (NIC) evaluation of nonintrusive diagnostic techniques for check valves. Personnel training and qualification performed by a vendor in accordance with the licensee's QA program may be acceptable; however, the technique must also be qualified as described above. Records of techniques and qualification documentation shall be maintained in accordance with the licensee's QA program, and the NRC inspector may examine the licensee's records.

Basis for Recommendation

The NRC staff previously determined that nonintrusive testing methods appropriate for certain valve applications are acceptable to verify the capability of the valve to open, close, and fully stroke, provided that the licensee properly qualifies the testing methods used for the valve application in accordance with the plant's QA program requirements.
These techniques are considered "other positive means" in accordance with subparagraph ISTC-5221(a)(3). It is the licensee's responsibility to qualify and document the results in accordance with the plant's QA program requirements. Appendix B to 10 CFR Part 50 specifies criteria for the QA program, which includes, in part, requirements to ensure that nondestructive testing is controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable Codes, standards, specifications, criteria, and other special requirements. The NIC conducted an experimental research and testing program to evaluate available nonintrusive technologies to determine their acceptability and reliability for use in check valve testing applications. Information on qualification of nonintrusive testing is provided in the summary of the NRC's public workshops on the revision of NRC Inspection Procedure 73756. In response to a question about expectations for qualification of a nonintrusive test

method, the NRC indicated that a qualified nonintrusive test method is a technique that has been successfully and reliably demonstrated for the examination method and for the specific valve application. Other expectations and discussed in NRC IN 2000-21.

Qualification includes establishing a performance baseline when the check valve is known to be in good operating condition. A check valve's performance can be assessed against this baseline trace. One means to determine if a nonintrusive test method or technique will provide accurate, reliable, and repeatable results for a specific check valve is to qualify the method prior to its use. The qualification process may reveal that certain techniques or methods give inconclusive results for a particular application. Acoustic techniques and test methods are susceptible to other plant system noise being transmitted and masking or affecting the desired sound pattern and results. The NIC suggests the use of more than one technique to verify questionable results.

4.1.3 Full Flow Testing of Check Valves

The OM Code requires both an open and close test for check valves and that licensees exercise check valves to the position(s) in which they perform their safety function(s). A check valve's full-stroke to the open position may be verified by passing the necessary flow through the valve to perform its safety function. This is considered an acceptable full-stroke. Any flow rate less than this is considered a partial-stroke exercise. A valid full-stroke exercise by flow requires that the flow through the valve must be known. Knowledge of only the total flow through multiple parallel lines does not provide verification of flow rates through an individual valve and may not be a valid full-stroke exercise without further analysis.

4.1.3.1 Alternative to Direct Flow Measurement

Flow through a check valve must be known for a valid full-stroke exercise test, but an alternative to direct flow rate instrumentation may be acceptable. Any quantitative measure that has acceptance criteria that demonstrate the required flow through the check valve may be used to satisfy the full-stroke requirement. An indirect measure of flow may be acceptable. For example, a change in tank level over a specified period could be used. In another case, the acceptance criterion could be based on a change in flow rate of an instrumented line when flow is admitted from a non-instrumented line containing the check valve being tested. In any event, some form of quantitative criteria should be established to demonstrate full-stroke capability.

4.1.3.2 Flow through Parallel Lines

Knowledge of total flow through multiple parallel lines does not provide indication of flow through each individual path. The objective of inservice testing is to evaluate and investigate the possibility of degradation of individual components and to take corrective action before a component fails. Verification of total header flow rate might not identify a problem, developing or occurring, with an individual check valve in one of the parallel flow paths. With respect to the balancing of flow, TS requirements are based on the flow from one loop being lost through a postulated break. Consequently, that flow path is restricted or throttled to minimize significant diversion of flow. TS surveillance requirements were not intended to verify individual check valve operability. The licensee is expected to justify the use of a test method that does not verify full-stroke of individual check valves.

For example, in a Beaver Valley Power Station, Unit 1, Safety Evaluation Related to the IST Program and Relief Requests, dated January 24, 1992 (ADAMS Legacy Accession

No. 9201310094), the NRC informed the licensee of the results of an evaluation of flow through parallel lines and stated that a flow test through parallel lines without individual flow measurement may not be sufficient to indicate that the check valves in the lines are full-stroke exercised. Knowledge of only total flow through multiple parallel lines does not provide verification of flow rates through an individual valve and may not be a valid full-stroke exercise without further analysis.

4.1.3.3 Accident Condition Flow

The phrase, “necessary flow through the valve for safety function,” is the largest flow rate for which a licensee takes credit in a safety analysis for this check valve in any flow configuration. The safety analyses are those contained in the plant’s FSAR, or equivalent, but are not limited to the accident and transient analyses.

4.1.3.4 Check Valves Not Required to Be Fully Opened

For check valves that are never required to open fully (i.e., thermal expansion or siphon breakers), verification of design (safety) function is testing to confirm the capability of forward flow through the system. In addition to verifying its safety function performance, licensees should develop quantifiable acceptance criteria for the testing of these components. Verifying that a system is full is an acceptable means for verifying that the keep-fill check valves are capable of opening to provide flow when necessary.

4.1.3.5 Alternative to Full-Flow Testing

Full-flow testing of a check valve might not be achievable for certain valves. It may be possible to qualify other techniques to confirm that a valve is exercised to the position required to perform its safety function. To substantiate the acceptability of any alternative technique for meeting the OM Code requirements, licensees, as a minimum, need to address and document the following items in the IST program. Any alternative techniques for meeting OM Code requirements must be submitted to the NRC for authorization pursuant to 10 CFR 50.55a(z).

1. the difficulty of performing a full-flow test
2. a description of the alternative technique used and a summary of the procedures being followed
3. a description of the method and results of the program to qualify the alternative technique for meeting the OM Code
4. a description of the instrumentation used and the maintenance and calibration of the instrumentation
5. a description of the basis used to verify that the baseline data has been generated when the valve is known to be in good working order, such as recent inspection and maintenance of the valve internals
6. a description of the basis for the acceptance criteria for the alternative testing and a description of corrective actions to be taken if the acceptance criteria are not met.

The NRC staff’s position with respect to full-flow testing of check valves allows licensees flexibility in qualifying alternatives to full-flow testing. In general, licensees should demonstrate that the alternative test is quantifiable and repeatable. The alternative test must meet the intent of the OM Code. This qualification of the alternative test should be documented by the licensee and should be available for review by NRC inspectors. Any alternative techniques for meeting OM Code requirements must be submitted to the NRC for authorization pursuant to 10 CFR 50.55a(z).

The 1994 Edition of the OM Code and subsequent editions include the use of nonintrusive testing as other positive means for demonstrating check valve exercising. The criteria listed in the NRC staff's position for full-flow testing of check valves could be applied to the nonintrusive techniques.

4.1.4 Disassembly and Inspection Alternative to Flow Testing

Guidance regarding disassembly and inspection of certain check valves is included in subparagraph ISTC-5221(c) for use if the test methods of subparagraphs ISTC-5221(a) and ISTC-5221(b) are impractical or sufficient flow cannot be achieved.

NRC Recommendation

Testing of check valves by disassembly should be accomplished as follows:

- The sample disassembly and inspection program should involve grouping similar valves and testing a different valve in each group during each refueling outage. The sampling technique should require that each valve in the group be the same design (manufacturer, size, model number, and materials of construction) and have the same service conditions including valve orientation.
- At each refueling outage sampling, the licensee should verify that the disassembled valve is capable of full-stroking and that the internals of the valve are structurally sound (no loose or corroded parts). Also, if the disassembly is to verify the full-stroke capability of the valve, the disk should be manually exercised. While the valve is in a partially disassembled condition, the valve internals should be inspected and the condition of the moving parts evaluated. This inspection and evaluation should include verification that the valve disk is free to move. Following reassembly, a partial flow test should be performed, if practical.
- A different valve of each group should be disassembled, inspected, and manually full-stroke exercised at each successive refueling outage, until the entire group has been tested. If the disassembled valve is not capable of being full-stroke exercised or there is binding or failure of valve internals, then the remaining valves in that group should also be disassembled, inspected, and manually full-stroke exercised during the same outage. Once this is completed, the sequence of disassembly should be repeated unless the valve group is in the Condition Monitoring Program alternative and an extension of the interval can be justified.
- At least one valve from each group should be disassembled and examined at each refueling outage. All valves in each group should be disassembled and examined at least once every 8 years.

The disassembly/inspection must be qualified to evaluate the condition of the valve and to assess its continued operational readiness. The licensee is responsible for the development and implementation of a program to ensure that IST personnel are appropriately trained and qualified to perform the valve disassembly/inspections. Licensees should implement the provisions of ASME Nuclear Quality Assurance-1 (NQA-1), "Quality Assurance Program

Requirements for Nuclear Facilities." The NRC staff encourages licensees to review the ANSI standard for guidance in developing a program for the qualification of IST personnel.

Basis for Recommendation

OM Code, subparagraph ISTC-5221(c), allows the use of sample disassembly and inspection of check valves. Specifically, subparagraph ISTC-5221(c) provides requirements for sample disassembly and inspection of certain check valves, if the test methods of subparagraphs ISTC-5221(a) and ISTC-5221(b) are impractical, or if sufficient flow cannot be achieved or verified. ISTC-5221(c)(3) specifies requirements that every valve in each group must be examined at least once every 8 years. These OM Code requirements are incorporated by reference in 10 CFR 50.55a.

4.1.5 Reverse Flow Testing of Check Valves

The OM Code requires that Category C check valves (those that are self-actuated in response to a system characteristic such as pressure or flow direction) that perform a safety function in the closed position to prevent reverse flow must be tested in a manner that demonstrates that the disk travels to the seat on cessation or reversal of flow. In addition, for Category A/C check valves (those that have a specified leak rate limit and are self-actuated in response to a system characteristic), seat leakage must be limited to a specific maximum amount in the closed position for fulfillment of their function. Verification that a Category C valve is in the closed position can be achieved through visual observation, by an electrical signal initiated by a position-indicating device, by observation of appropriate pressure indication in the system, by leak testing, or by other OM Code-defined positive means (i.e., ISTC-5221).

4.1.5.1 Closure Capability of Check Valves that Do Not Have Defined Seat Leakage Limits

A plant's safety analysis may include a leakage limit for a particular valve, or may only require that the valve close to inhibit gross leakage. When a valve has a safety-related function to close to prevent diversion of flow between trains of a system, there may be a leakage limit based on the total system requirements. The OM Code does not specifically require these valves to be Category A. The basis for assigning valves to categories should be available.

Licensees may refer to IN 91-56, "Potential Radioactive Leakage to Tank Vented to Atmosphere," for information concerning the categories assigned to valves that function to close to prevent leakage above an assumed limit; thus preventing a plant from exceeding the limits of 10 CFR Part 100.

4.1.5.2 Permissible Leak Rates

Subparagraph ISTC-3630(e) requires that leakage rate measurement shall be compared with the permissible leakage rate specified by the licensee for a specific valve or valve combination. If the licensee does not specify leak rates, permissible leak rates are provided in subparagraph ISTC-3630(e). It should be noted that the OM Code does not provide either criteria or guidance concerning the methods that licensees should use to establish or specify the permissible leak rate of a particular valve. The OM Code recognizes that the leak behavior of a valve varies according to valve type and size, vendor, service conditions, safety-related functions, and other factors, and there is no simple leak rate rule that applies to all valves.

In general, licensees should set the leak rate limits within certain bounds. If the leak limits are too low, unnecessary valve repairs or adjustments can result. Leak limits that are too high could result in failure of the tests required by Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50, thereby leading to concerns regarding the leak-tight integrity of the containment. Appropriate permissible leak rates can only be developed and refined by analyzing and trending the leak rate data for specific valves or for similar valves at other plants. Therefore, the NRC staff has not provided specific guidance concerning leak rates. Licensees should document their methods for establishing the initial permissible leak rates and procedures for verifying compliance with the leak rate limits.

4.1.5.3 Closure Testing of Stop Check Valves

If a stop-check valve does not perform a safety-related function in the closed position, valve closure is only necessary to ensure a repeatable starting point for opening testing. Valves may be closed by using a handwheel or a hand switch. (Note: 1996 Addenda, and later Editions and Addenda, to the OM Code require bidirectional testing.)

If the use of a handwheel or hand switch to close a valve achieves the safety-related function of the system, exercising the valve by this method meets the OM Code requirements of paragraph ISTC-5221. By contrast, if closure of a stop check valve on cessation or reversal of flow is required to accomplish a safety-related function, its closure must be verified by reverse flow testing or other positive means, such as acoustic monitoring or radiography, to satisfy the OM Code.

When no other means of verification are possible, licensees may disassemble valves to verify valve closure. However, disassembly provides limited information on valve capability to seat on cessation or reversal of flow. Furthermore, if the method involves extensive disassembly, a post-reassembly test would be necessary in accordance with paragraph ISTC-3310, because disassembly and inspection can increase the probability of human error when the valve is reassembled. Licensees may evaluate the use of nonintrusive testing techniques and may implement such techniques if they are demonstrated to be effective in assessing closure capability, degradation, and incipient failure. Infrequent disassembly and inspection of the valves are appropriate to assess overall check valve condition, while reverse flow testing and nonintrusive testing provide an assessment of continued operational readiness.

4.1.5.4 Other Positive Means of Verification

Paragraph ISTC-5221 allows for other positive means of verification of obturator position. Examples from IST programs include verifying that a parallel centrifugal pump does not spin in reverse to verify closure of a pump discharge check valve, monitoring an upstream pressure indicator, monitoring a tank level, measuring the flow rate of a redundant train, or opening an upstream vent and drain valve.

4.1.6 Extension of Test Interval to Refueling Outage for Check Valves Verified Closed by Leak Testing

When it is impracticable for the licensee to verify check valve closure or opening during plant operation or cold shutdown, the OM Code allows the licensee to extend the check valve quarterly exercise test (both open and close) to the refueling outage. The closure verification may be performed in conjunction with the Type C leak rate test conducted in accordance with

Option A or Option B of Appendix J to 10 CFR Part 50. Licensees may also perform the open exercise test during the refueling outage or anytime during the fuel cycle interval.

NRC Recommendation

If no other practical means are available, it is acceptable for licensees to extend the quarterly closure or opening exercise test to a refueling frequency. In such instances, the licensee need to develop a refueling outage justification describing the impracticality of performing the quarterly closure test during plant operation or cold shutdown. The NRC staff has determined that the need to set up test equipment constitutes adequate justification to defer reverse flow testing of a check valve to a refueling outage. By referencing the refueling justification in the IST program document, the licensee may perform the closure exercise test during each reactor shutdown for refueling. A seat leak test is one method to verify that the obturator has traveled to the seat. All requirements of each individual valve category are applicable, although repetition of a common testing requirement is not required. Therefore, when the required performance of the Appendix J leak rate test coincides with a refueling outage exercise seat leak test, only the Appendix J test is required.

The OM Code states that open and close tests need only be performed at an interval "when it is practicable to perform both tests." The OM Code also states that licensees are not required to perform open and close tests at the same time if they are both performed during the same interval. For example, since the closure test is extended to the refueling outage by the refueling justification, the quarterly exercising test may also be extended to the refueling outage or may be performed anytime during the fuel cycle interval.

Basis for Recommendation

Subparagraph ISTC-5221(a) of the OM Code states that valve obturator movement observations shall be made by a direct indicator (e.g., a position-indicating device) or other positive means including seat leak testing. Therefore, a seat leak test is one method to verify that the obturator has traveled to the seat.

Subparagraph ISTC-3522(c) states that if exercising is not practical during plant operation and cold shutdowns, it shall be performed during refueling outages. A refueling outage justification shall document the extension of the exercise test to the refueling outage.

Subparagraph ISTC-3522(a) states that open and close exercise tests need only be performed at an interval when it is practical to perform both tests. This OM Code section also states that open and close tests are not required to be performed at the same time if they are both performed during the same interval.

4.1.7 Testing and Examination of Check Valves Using Manual Mechanical Exercisers

OM Code, subparagraph ISTC-5221(b), "Valve Obturator Movement," in part, requires that if a manual mechanical exerciser is used to test the check valve, the force(s) or torque(s) required to move the obturator to fulfill its safety function(s) shall meet the acceptance criteria specified by the Owner. This includes the following:

1. Exercise test(s) shall detect a missing obturator, sticking (closed or open), binding (throughout obturator movement), and the loss or movement of any weights.

2. Acceptance criteria shall consider the specific design, application, and historical performance.
3. If it is impractical to detect a missing obturator or loss or movement of any weight(s), other positive means may be used (e.g., seat leakage tests and visual observations to detect obturator loss and the loss or movement of external weight(s), respectively).

NRC Recommendation

In the past, the OM Code (through the OMa-1996 Addenda), subparagraph ISTC 4.5.4(b), required that, if a manual mechanical exerciser is used for IST movement of the obturator, the force or torque to initiate disk movement shall not vary by more than 50 percent from an established reference value. Licensees continuously experienced difficulty with this IST acceptance criterion. The manual mechanical exerciser assembly includes a packing gland to seal the hinge pin penetration of the valve body. The hinge pin seal packing introduces conditions that produce variations in friction forces over time. These variations make it difficult to establish a reference value that would be continually consistent and appropriate for use in the IST program.

The NRC received a number of requests for relief from the requirement that force or torque to initiate disk movement not vary by more than 50 percent of the reference value. Licensees also requested that the OM Code Working Group on Check Valves (WGCV) reexamine this requirement. The WGCV reexamination resulted in the change in the OM Code, OMa-2000, subparagraph ISTC-5221(b), which is discussed above. This change requires the Owner to specify the acceptance criteria within certain OM Code-defined expectations. In establishing an acceptance criterion for IST when using a manual mechanical exerciser, the Owner should consider the interactions, wear and effects of the valve parts on friction forces, and the valve preventive maintenance activities.

Basis for Recommendation

Mechanical exercisers are attached to a hinge pin that is fixed to the disk and penetrates the valve body. Many of these valves involve swing check valves that manufacturers supplied with a lever arm and counterweight modification. The counterweight is used to affect the opening or closing response of the disk to flow conditions, depending upon the lever arm’s location relative to the disk. The counterweight modification involves the use of a packing gland to seal the hinge pin penetration of the valve body. The seal packing introduces variations over time with regard to the required disk opening force and opening and closing responses of the disk, depending upon the type of packing material used, its condition, friction changes, leakage control adjustments, and the packing gland tightening procedure. Any wear of the hinge pin and bearing interfaces may exacerbate these variations. Disk opening and closing friction forces may also change as a result of valve preventive maintenance activities. Subparagraph ISTC-5221(b) of the OM Code, states that if a mechanical exerciser is used to exercise the check valve, the force(s) or torque(s) required to move the obturator to fulfill its safety function(s) shall meet the acceptance criteria specified by the Owner.

4.1.8 Check Valve Bidirectional Testing and Condition Monitoring Program

Bidirectional testing ensures that a check valve is adequately tested, regardless of its safety function. Such testing also improves the IST capability to detect valve degradation prior to valve failure. Two significant OM Code changes, in paragraphs ISTC 4.5.4(a) and ISTC 4.5.5, respectively, were introduced in the 1996 Addenda to the OM Code. Specifically, those changes included (1) a requirement for bidirectional exercise testing of the disk movement of check valves, and (2) a voluntary provision to use the condition monitoring program as an alternative to IST exercise testing for certain check valves. This integral two-part improvement to the OM Code provides interrelated requirements. The condition monitoring program allows licensees certain IST flexibility in establishing the types of test, examination, and preventive maintenance activities and their associated intervals, when justified based on the valve's performance and operating condition. These OM Code changes were developed so that licensees who elect not to implement condition monitoring in their IST program, would be required to bidirectionally test check valves as a default set of testing and examination requirements.

Subsection ISTC 4.5.4(a) of the 1996 Addenda to the OM Code, "Valve Obturator Movement," in part, requires that the necessary obturator movement during exercise testing shall be demonstrated by performing both an open and close test, and observations shall be made by a direct indicator (e.g., a positive-indicating device) or by other positive means (e.g., changes in system pressure, flow, level, temperature, seat leakage, testing, or nonintrusive testing and examination).

Subsection ISTC 4.5.5 of the 1996 Addenda to the OM Code, "Condition Monitoring Program," in part, provides an option for the Owner to establish a check valve condition monitoring program alternative to the testing and examination requirements of Subsections ISTC 4.5.1–4.5.4. The purpose of the program is to improve valve performance and optimize testing, examination, and preventive maintenance activities in order to maintain acceptable check valve performance. The program must be implemented in accordance with Appendix II, "Check Valve Condition Monitoring Program."

More recent editions of the OM Code continue to specify bi-directional testing of check valves and allow the implementation of check valve condition monitoring programs.

NRC Recommendation

The required testing or examination of the check valve obturator movements to both the open and closed positions, as required by paragraph ISTC-3522 in the OM Code, is necessary to assess the valve's operational condition, confirm the acceptability of its performance, and detect degradation prior to failure. Single-direction flow testing of check valves will not always detect functional degradation of the valves. In addition to check valves with both open and close safety functions, a check valve that has only a safety function in one direction (such as either open or close) needs to be tested in both directions to demonstrate that the disk is in its proper operating condition to provide reasonable assurance that the check valve is operationally ready to perform its open or close safety function.

The NRC staff considers the condition monitoring program approach of the OM Code, Appendix II, for check valve IST with the regulatory modifications in 10 CFR 50.55a(b)(3)(iv), to be an improvement over present OM Code requirements, and encourages licensees to implement the condition monitoring alternative.

Basis for Recommendation

The NRC incorporated the 1996 Addenda to the OM Code by reference in 10 CFR 50.55a on September 22, 1999 (64 FR 51370). Subsection ISTC 4.5.4(a) of the 1996 Addenda to the OM Code included the requirement that the necessary obturator movement during exercise testing must be demonstrated by performing both open and closed tests. More recent editions of the ASME OM Code continue this exercising requirement for check valves. The NRC agrees with the need for a required demonstration of bidirectional exercise testing of the movement of the check valve disk. Single-direction flow testing will not always detect degradation of the valve. The classic example of the flawed single-direction testing strategy is that the loss of the disk would not be detected during forward flow tests. The detached disk could be lying at the bottom of the valve body or another part of the system, and could move to block flow or disable another valve or component. One example of an undetected detached check valve disk lying at the bottom of the valve body is captured in the discussion of the event described in IN 2000-21. The NRC considers testing or examination of the check valve obturator movement to both the open and closed positions necessary to assess its condition and confirm acceptable valve performance.

The use of the OM Code, Appendix II, as incorporated by reference in 10 CFR 50.55a includes requirements that apply when extending check valve IST intervals, with regard to consideration of the plant safety significance, justification by trending of current and historical valve condition and performance data, maximum IST interval limits, stepwise IST interval limits, and bidirectional testing or examination. These requirements provide the licensee with knowledge of the valve's operating condition, monitor and verify valve performance over extended intervals, and provide a process to justify prudent IST interval extensions to reduce the burden of unnecessary IST.

The OM Code Committee, Working Group on Check Valves (WGCV) proposed changes to Subsection ISTC of the OM Code to address several regulatory modification issues. ASME issued these changes as part of OM Code, OMb-2003. All changes discussed above are incorporated in the 2004 through 2017 Editions of OM Code.

4.1.9 Instrumentation Requirements

Instruments used to verify that check valves fulfill their safety function(s) are not subject to the same range and accuracy requirements as instrumentation used for pump-related IST. However, the 1998 Edition of the OM Code added paragraph ISTC-3800, "Instrumentation," to provide specific requirements for instrumentation that is used in the testing and examination of valves. Specifically, the 1998 Edition of the OM Code, paragraph ISTC-3800, requires that instrumentation, including both measuring and test equipment and permanent plant instrumentation, used for valve testing and examination activities, shall (1) be properly controlled, calibrated, and adjusted in accordance with the Owner's QA program, and (2) have the accuracy, range, and repeatability characteristics necessary to verify compliance with the requirements of this subsection. In addition, instrumentation accuracy shall be considered when establishing valve test acceptance requirements.

NRC Recommendation

OM Code (2017 Edition), ISTC-3800, "Instrumentation," specifies requirements for instrumentation that is used in performing IST for check valve testing and examination activities. NRC staff expects that instrumentation that is used in IST must be controlled and calibrated, and must have the accuracy, range, and repeatability necessary to verify compliance with the requirements. Accuracy and repeatability of the instrumentation are important considerations in the IST of safety-related check valves. IST should be performed in a way that permits the results to be compared for indications of valve degradation trends. When instrumentation that is used in valve testing and examination is not properly controlled and calibrated, any valve degradation indications may be masked, thereby diminishing the usefulness of the valve test results.

Basis for Recommendation

The instrumentation requirements and QA activities specified in paragraph ISTC-3800 are needed to properly verify compliance with OM Code requirements and to detect and trend any operational degradation for assurance that the check valves will perform satisfactorily until the next IST. Appendix B to 10 CFR Part 50 provides for the QA program and includes requirements for IST of safety-related components, test control, and control of measuring and test equipment.

4.1.10 Skid-Mounted Valves and Component Subassemblies

The exercising or examination of each check valve that is contained in (or part of) a skid-mounted package or major component is not always practical, particularly if the valve is enveloped within the structure of the package or component. Although the valve's performance may support a safety function, the practicality of exercising or examining each valve separately, as required by the ASME OM Code, was not addressed prior to 1996 Addenda to the OM Code. Subparagraph ISTC-1200(c) of the 1996 Addenda to the OM Code through the 2017 Edition exempts skid-mounted valves from the individual testing requirement and allows the valves to be tested as part of the overall package or major component. Specifically, paragraph ISTC-1200, states, in part, "Skid-mounted valves and component subassemblies are excluded from this Subsection, provided they are tested as part of the major component and are determined by the Owner to be adequately tested."

NRC Recommendation

To satisfy the OM Code, testing major components that include check valves that are an integral portion of the major component and that support the major component's performance of its safety function requires that the licensee must determine that the check valves are adequately tested. The NRC staff expects that, as part of the licensee's testing and determination responsibility, the check valves will be identified in the IST plan, along with an explanation of how the testing of the major component adequately tests the valves. The licensee should review the safety significance of the identified valves to ensure that the IST program is adequate to demonstrate their continued operability. The documentation that provides assurance of the operational readiness of the valves through the performed tests should be available at the plant site.

Basis for Recommendation

GDC 1 in Appendix A to 10 CFR Part 50, in part, requires that components important to safety must be tested to quality standards commensurate with the importance of the safety functions performed.

4.1.11 Check Valves in New Reactors

The NRC regulations in 10 CFR 50.55a(b)(3)(iii)(B) require that licensees within the scope of 10 CFR 50.55a(b)(3)(iii) perform bidirectional testing of check valves within the IST program where practicable. Nuclear power plant operating experience has revealed that testing check valves in only the flow direction can result in significant degradation, such as a missing valve disc, not being identified by the test. Nonmandatory Appendix M, "Design Guidance for Nuclear Power Plant Systems and Component Testing," to the OM Code, 2011 Addenda and 2012 through 2017 Editions, includes guidance for the design of new reactors to enable bidirectional testing of check valves. New reactor designs will provide the capability for licensees of new nuclear power plants to perform bidirectional testing of check valves within the IST program. Bidirectional testing of check valves in new reactors, as required by 10 CFR 50.55a(b)(3)(iii)(B), could be accomplished by valve-specific testing or condition monitoring activities in accordance with Mandatory Appendix II to the OM Code as accepted in 10 CFR 50.55a. The NRC is specifying this provision for bidirectional testing of check valves for new reactors in 10 CFR 50.55a(b)(3)(iii)(B) to emphasize that new reactors should include the capability for bidirectional testing of check valves as part of their initial design.

As part of its review of design certification and COL applications, the NRC staff is evaluating whether new nuclear power plants will be designed to allow bidirectional testing of check valves.

Some new reactors include nozzle check valves in nuclear power plant systems to perform functions important to safety. Nozzle check valves are more complex than swing check valves used in most safety-related check valve applications in nuclear power plants. The provisions in the OM Code focus on exercising check valves to demonstrate their operational readiness. The internal mechanism of the nozzle check valve might require more precise IST provisions to provide assurance of their operational readiness for their specific applications. 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. Licensees of new reactors are responsible for satisfying 10 CFR Part 50, Appendix A, when developing IST provisions for nozzle check valves that provide reasonable assurance that they are capable of performing their safety functions.

The NRC staff will conduct inspections of the development and implementation of the IST program (including bidirectional testing of check valves and surveillance provisions for new check valve designs) during construction and operation of the new nuclear power plants.

4.2 Power-Operated Valves

POVs are equipped with actuators that use motive force to change the position of the valve obturator. The types of POVs may include, for example, MOVs, AOVs, HOVs, SOVs, and pyrotechnic-actuated (squib) valves. In addition, the OM Code, Subsection ISTC defines a power-operated relief valve (PORV) as a POV that can perform a pressure-relieving function and is remotely actuated by either a signal from a pressure-sensing device or controls switch and is not capacity certified under ASME BPV Code, Section III, overpressure protection requirements. Certain valves, such as main steam isolation valves (MSIVs) and valves that have a fail-safe function, may actuate open (or closed) on spring force. In the following sections, the NRC staff provides guidance concerning the implementation of specific OM Code provisions and associated regulatory requirements.

4.2.1 Stroke-Time Testing Reference Values for Power-Operated Valves

Paragraph ISTC-5113 of the ASME OM Code specifies that certain active POVs shall have their stroke times measured when exercised in accordance with the nominal 3-month schedule specified in paragraph ISTC-3500. Appendix III to the OM Code (2009 Edition and later) and Appendix IV to the OM Code (2017 Edition and later) specify separate IST requirements for MOVs and AOVs, respectively.

The OM Code includes provisions for establishing reference values and limiting values for POV stroke times. For example, reference values shall be determined from the results of preservice testing or inservice testing. Limiting values for stroke time for various types of POVs shall be specified by the Owner. Corrective actions shall be taken if acceptance criteria are not satisfied. If the limiting value of stroke time is exceeded, the POV shall be immediately declared inoperable.

The OM Code does not specify provisions for establishing the limiting value for stroke times, and it does not identify the relationship that should exist between those limits and the reference values for stroke time or any limits identified in the plant TS or safety analysis.

NRC Recommendation

The limiting value of full-stroke time should be based on the reference stroke time of a POV when it is known to be in good condition and operating properly. The limiting value should be a reasonable deviation from this reference stroke time, based on the size and type of the valve and power actuator. The deviation should not be so restrictive that it results in a POV being declared inoperable as a result of reasonable stroke time variations. However, the deviation used to establish the limiting value should be such that corrective action would be taken to provide assurance that the POV would remain capable of performing its safety function.

The limiting value for stroke time of a POV should be that point at which the licensee seriously questions continued operability. It is expected to be a value that is determined to be reasonable for the individual POV based on its characteristics and past performance, but not to exceed any safety analysis requirements. The value should not be based solely on the system requirements or values specified in safety analyses for system performance. When the identified limiting value is exceeded, the licensee shall declare the component inoperable and shall enter any applicable TS limiting condition for operation (LCO). After declaring the valve inoperable, the licensee should perform an analysis to identify the cause of the problem with the POV. If this analysis clearly demonstrates that the POV remains capable of performing its safety function, the analysis might constitute the corrective action required by the OM Code. The analysis must be documented.

Licensees should establish reference values that reflect the stroke time of the specific POV when in good condition and operating under applicable conditions. A licensee may establish additional sets of reference values as discussed in paragraph ISTC-3320, such as reference values that reflect test conditions of fluid pressure or flow in the system.

Licensees may use a quantitative multiplier on a reference time as a means of establishing a limiting value for stroke time. The licensee should document the justification for its selection of reference values for the stroke time of each POV, and should have this justification available at the plant site for review by NRC personnel.

Basis for Recommendation

The purpose of the limiting value of full-stroke time for a POV is to establish a value for taking corrective action on a degraded POV before it reaches the point where there is a high likelihood of failure to perform its safety function. While the TS provide assurance that important plant systems are capable of performing their safety functions in a timely manner during selected plant transient accidents and anticipated operational occurrences, the provisions of the OM Code are intended to ensure the continued operability of particular plant components. The distinct bases for these two documents (i.e., TS and OM Code) lead to criteria that may differ significantly. Nonetheless, the TS and OM Code are both needed to provide confidence that the nuclear power plant can be operated safely. Therefore, licensees must follow the more restrictive criteria of the two documents, even though this might result in a component or system being declared inoperable. For example, if the TS or safety analysis limit for a POV is less than the IST value established using the above guidelines, the TS or safety analysis limit should be used as the limiting value of full-stroke time. When the TS or safety analysis limit for a POV is greater than the IST value established using the above guidelines, the limiting value of full-stroke time should be based on the above guidelines instead of the TS or safety analysis limit. The TS and safety analysis limits are useful for analyzing data when a POV has indicated degraded performance and been declared inoperable. In accordance with the OM Code, the data may be analyzed to verify that the new stroke time represents acceptable POV operation.

4.2.2 Stroke-Time Measurements for Rapid-Acting Valves

The ASME OM Code allows licensees to establish the limiting stroke time of 2 seconds for POVs that stroke in less than 2 seconds. The OM Code also eliminates the acceptance criterion related to the reference value for stroke-time for those POVs. However, new technologies and new applications of existing technologies enable licensees to time the strokes of rapid-acting valves with accuracy measured in milliseconds. Using new technology, licensees could establish an appropriate limiting stroke time based on a multiple of a reference value to ensure that corrective actions are taken if degrading conditions are identified.

NRC Recommendation

The NRC staff recommends that licensees should determine whether continued reliance on the 2-second limiting stroke time criterion in the OM Code is appropriate when the actual stroke time can be measured in milliseconds.

Basis for Recommendation

The 2-second limiting stroke time for rapid-acting valves was based on measurement of stroke times using a stopwatch. Updated technology may improve the monitoring of the condition of these POVs or verify that a valve operates within a safety analysis limit that is less than 2 seconds.

4.2.3 Stroke Time for Solenoid-Operated Valves

The NRC is often asked to approve relief from the OM Code provisions to allow licensees not to measure the stroke times of enclosed SOVs that do not have position indication. If the licensee cannot time the stroke of an SOV by the conventional method using position indication, the licensee needs to propose a method to time the stroke of the valve or otherwise monitor the POV for degrading conditions to provide adequate assurance of its operational readiness. If the frequency provisions of the OM Code are met, the licensee does not need to request relief to use methods such as acoustics or diagnostic systems for stroke timing. If the licensee intends to apply a method to monitor for degradation other than by measuring stroke time, NRC authorization of the alternative is required pursuant to 10 CFR 50.55a(z). For example, an enhanced maintenance program or periodic replacement may be acceptable when testing methods cannot be used effectively.

NRC Recommendation

The NRC staff recommends that licensees should use advanced diagnostic techniques to obtain stroke-time measurements in accordance with the frequency provisions of the Code, and should also use those advanced techniques or maintenance programs to monitor the degradation of SOV performance. In addition, the NRC staff recommends that the technique should evaluate actual disk movement and not only movement of the pilot valve or valve stem.

Basis for Recommendation

In NUREG-1275, Vol. 6, “Operating Experience Feedback Report: Solenoid-Operated Valve Problems,” (ADAMS Accession No. ML063550260) the NRC described common-mode SOV problems that could significantly reduce plant safety. Several methods are available to measure stroke time or monitor the condition of SOVs using parameters such as the acoustic effects of disk movement, electric resistance, and the temperature of the coil. These advanced diagnostic techniques provide more precise means of monitoring SOV performance.

4.2.4 Supplement to the POV Stroke-Time Test Provisions of the ASME OM Code

Operational experience and valve testing programs have revealed weaknesses in the ability of stroke-time testing to assess the operational readiness of some POVs to perform their safety functions. In response to those weaknesses, ASME, the NRC, and various industry groups have taken action to provide improved confidence in the capability of POVs to perform their safety functions under design-basis conditions.

With respect to MOVs, the NRC’s regulations in 10 CFR 50.55a require that licensees who are implementing the OM Code (beginning with the 1995 Edition) must supplement the provisions for MOV stroke-time testing specified in the OM Code with a program to ensure that the MOVs continue to be capable of performing their design-basis safety functions. In a *Federal Register* notice (64 FR 51370) dated September 22, 1999, the NRC discussed the implementation of MOV programs in satisfying the requirement to supplement MOV stroke-time testing.

The NRC established Generic Safety Issue (GSI) 158, “Performance of Safety-Related Power-Operated Valves Under Design-Basis Conditions,” to evaluate whether additional regulatory actions were necessary to address performance issues for POVs. In RIS 2000-03, the NRC closed GSI-158 on the basis that current regulations provide adequate requirements to ensure verification of the design-basis capability of POVs, and no new regulatory requirements were needed. The NRC staff also stated that it would continue to monitor licensees’ activities to ensure that POVs are capable of performing their specified safety-related functions under design-basis conditions. In RIS 2000-03, the NRC staff provided attributes for an effective POV testing program that incorporates lessons learned from MOV research and testing programs.

In response to these valve issues, ASME updated the OM Code to replace the previous quarterly stroke-time testing with improved IST requirements for MOVs and AOVs in Appendix III and Appendix IV, respectively. For MOVs, Appendix III requires exercising on an RFO frequency and periodic diagnostic testing to demonstrate MOV capability. For AOVs, Appendix

IV requires an initial performance assessment test and quarterly stroke-time testing for all AOVs, and periodic performance assessment tests for high safety significant AOVs.

The NRC regulations in 10 CFR 50.55a incorporates by reference Appendix III and Appendix IV to the OM Code. The regulations specify conditions in 10 CFR 50.55a(b)(3)(ii)(A), (B), (C), and (D) for the implementation of Appendix III.

NRC Recommendation

Licensees are required to apply the OM Code as incorporated by reference in 10 CFR 50.55a, including Appendices III and IV, according to the applicable edition of their OM Code of Record. The NRC staff recommends that licensees apply lessons learned from their MOV programs established and implemented in response to GL 89-10 and GL 96-05 in implementing the provisions in the OM Code in accordance with the requirements in 10 CFR 50.55a. The NRC staff also recommends that licensees consider information provided in RIS 2000-03, as well as lessons learned from their own MOV programs, to improve confidence in the capability of other POVs to perform their safety functions.

Basis for Recommendation

In the 1980s and 1990s, operating experience at nuclear power plants revealed that weaknesses in the ability of stroke-time testing to assess the operational readiness of POVs allowed performance deficiencies to remain undetected for an extended period of time. In 10 CFR 50.55a, the NRC requires that licensees whose Code of Record is the 1995 Edition (or a later edition or addenda) of the OM Code must supplement their stroke-time testing of MOVs with programs to ensure that the MOVs are capable of performing their design-basis safety functions. As discussed in RIS 2000-03, the NRC staff determined that current requirements and guidance indicate the need for licensees to have confidence in the capability of all safety-related POVs to perform their design-basis functions. In RIS 2000-03, the NRC staff discusses industry activities to improve POV performance. The NRC staff also stated that it would continue to monitor licensees' activities to ensure that POVs are capable of performing their specified safety-related functions under design-basis conditions.

In GL 89-10, the NRC asked licensees to ensure that MOVs in safety-related systems have the capability to perform their intended functions by reviewing MOV design bases, verifying MOV switch settings initially and periodically, testing MOVs under design-basis conditions where practical, improving evaluations of MOV failures and necessary corrective actions, and trending MOV problems. The NRC subsequently issued GL 96-05 to request that licensees establish a program, or ensure the effectiveness of their current program, to verify, on a periodic basis, that safety-related MOVs continue to have the capability to perform their safety functions within the current licensing basis of the facility. The NRC staff reviewed licensees' activities in response to GL 89-10 and GL 96-05 through plant-specific inspections and reviews of submitted information.

Licensees have completed their GL 89-10 programs for the operational nuclear power plants. In response to GL 96-05, the owners' groups of nuclear power plant licensees established the Joint Owners' Group (JOG) Program on MOV Periodic Verification as an industry-wide effort to evaluate potential degradation of MOV operating requirements. Most licensees committed to implement the JOG MOV program as part of their response to GL 96-05. In a safety evaluation dated September 2006 and its supplement dated September 2008, the NRC staff accepted the application of the JOG Program on MOV Periodic Verification as described in the safety evaluation and its supplement. As discussed in the safety evaluation, the JOG program addresses the operating requirements of valves and, therefore, licensees are responsible for justifying the output capability of MOVs within the scope of the JOG program. Further, licensees are responsible for justifying the long-term periodic verification programs for MOVs or their applications that are outside the scope of the JOG program.

The NRC staff notes that the NRC regulations and documents contain requirements and guidance intended to provide assurance that safety-related POVs are capable of performing their safety-related functions. For example, the regulations in Appendices A and B to 10 CFR Part 50 require that licensees provide confidence that safety-related equipment (including POVs) is capable of performing its safety functions under design-basis conditions. Further, the regulations in 10 CFR 50.65 require that licensees monitor the performance of SSCs in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. With respect to air systems, the NRC staff issued GL 88-14, “Instrument Air Supply System Problems Affecting Safety-Related Equipment,” to ask licensees to verify (by test) that AOVs will perform as expected in accordance with all design-basis events. The NRC staff provided the results of studies of POV issues in NUREG-1275, “Operating Experience Feedback Report,” Volumes 2, 6, and 13; NUREG/CR-6644, “Generic Issue 158: Performance of Safety-Related Power-Operated Valves Under Operating Conditions,” and NUREG/CR-6654, “A Study of Air-Operated Valves in U.S. Nuclear Power Plants.” In RIS 2000-03, the NRC staff provided a list of attributes of a successful POV design capability and long-term periodic verification program based on lessons learned from NRC staff reviews of valve programs and plant visits. The NRC staff also prepared several information notices to alert licensees to IST issues related to POV performance. Specifically, these included IN 86-50, “Inadequate Testing to Detect Failures of Safety-Related Pneumatic Components or Systems;” IN 85-84, “Inadequate Inservice Testing of Main Steam Isolation Valves;” and IN 96-48, “Motor-Operated Valve Performance Issues,” which described lessons learned from MOV programs that are applicable to other POVs.

In response to POV operating and test experience, ASME initiated efforts to improve the OM Code provisions for assessing the operational readiness of POVs in IST programs at nuclear power plants. For example, ASME developed OM Code Cases OMN-1 and OMN-11. ASME also developed OM Code Case OMN-12 that provides guidance for an alternative to quarterly stroke-time testing for AOVs and HOVs. OMN-12 includes risk-informed provisions to allow licensees to obtain more precise performance data for use in assessing the operational readiness of POVs that are determined to have higher safety significance, while allowing licensees to obtain less precise data for POVs of lower safety significance.

In addition to the NRC and ASME, the nuclear industry took action to address POV performance issues. As discussed above, the industry developed the JOG Program on MOV Periodic Verification to share resources among licensees and to establish an improved response to the MOV issues to be addressed under GL 96-05. In addition, a Joint Owners’ Group on Air-Operated Valves (JOG AOV) established a voluntary program to improve confidence in the capability of safety-related AOVs to perform their design-basis functions. In RIS 2000-03, the NRC staff noted that it provided comments on the JOG AOV program in a letter to the NEI dated October 8, 1999. The NRC staff also stated that it would continue to monitor licensees’ activities to ensure that POVs are capable of performing their safety-related functions under design-basis conditions. If the industry does not adequately address POV functionality under design-basis conditions, the NRC staff indicated in RIS 2000-03 that additional regulatory action may be necessary.

When implementing Appendix III to the OM Code, licensees will need to address the conditions specified in 10 CFR 50.55a(b)(3)(ii). In particular, the regulations require licensees to establish a program to ensure that MOVs continue to be capable of performing their design-basis safety functions. Licensees may apply their lessons learned from the GL 89-10 and GL 96-05 programs in satisfying this condition. However, licensees might need to supplement their lessons learned for valve types (such as ball or plug valves) that might not have been dynamically tested as part of those programs. Licensees might obtain information from the valve vendor or operating and testing experience to satisfy this condition for specific valve types.

In 10 CFR 50.55a(b)(3)(ii)(A), the regulations require licensees to evaluate the adequacy of their diagnostic test intervals established for MOVs within the scope of Appendix III not later than 5 years or three RFOs (whichever is longer) from initial implementation of Appendix III. The NRC staff considers that licensees that implemented GL 89-10 programs and are currently implementing their GL 96-05 programs could apply lessons learned from those programs to justify the adequacy of their diagnostic test intervals in satisfying this condition.

In 10 CFR 50.55a(b)(3)(ii)(B), the regulations require licensees to ensure that the potential increase in core damage frequency and large early release frequency associated with the extension is acceptably small when extending exercise test intervals for high risk MOVs beyond a quarterly frequency. This condition was also applicable to the acceptance of OM Code Case OMN-1 as an alternative to quarterly stroke-time testing. The NRC staff considers that licensees may apply their plant PRA to justify the extension of the exercise interval for high risk MOVs in satisfying this condition.

In 10 CFR 50.55a(b)(3)(ii)(C), the regulations indicate that when applying Appendix III, licensees shall categorize MOVs according to their safety significance using the methodology described in OM Code Case OMN-3, “Requirements for Safety Significance Categorization of Components Using Risk Insights for Inservice Testing of LWR Power Plants,” subject to the conditions applicable to OMN-3 set forth in RG 1.192, or using an MOV risk ranking methodology accepted by the NRC on a plant-specific or industry-wide basis in accordance with the conditions in the applicable safety evaluation. This condition indicates acceptable alternatives to MOV risk ranking beyond the provision for the use of OM Code Case OMN-3 specified in Appendix III. The allowance for alternatives to OM Code Case OMN-3 relaxes the provisions in Appendix III.

In 10 CFR 50.55a(b)(3)(ii)(D), the regulations require that when applying Paragraph III-3600, “MOV Exercising Requirements,” of Appendix III, licensees shall verify that the stroke time of MOVs specified in plant TS satisfies the assumptions in the plant's safety analyses. Early editions and addenda of the OM Code specified that the licensee must perform quarterly MOV stroke-time measurements that could be used to verify that the MOV stroke time satisfies the assumptions in the safety analyses consistent with plant TS. Current reliance on plant TS to provide confidence in MOV stroke times has led to this condition. In satisfying this condition, the NRC staff notes that exercising of the MOVs specified in plant TS may verify that the stroke time does not exceed the safety analysis assumptions. Trending of MOV stroke time is not required by this specific condition.

When implementing Appendix IV, licensees must satisfy the IST requirements for AOVs with different testing provisions based on their risk significance. Licensees will need to establish the high or low safety significance of the AOVs within the OM Code scope to implement Appendix IV.

4.2.5 Alternatives to POV Stroke-Time Testing

Prior to the development of Appendices III and IV to the ASME OM Code, ASME developed alternatives to the OM Code provisions for stroke-time testing of POVs. As an alternative to MOV stroke-time testing, ASME developed OM Code Case OMN-1, which provides periodic exercising and diagnostic testing for use in assessing the operational readiness of MOVs. In OM Code Case OMN-11, ASME provided additional guidance for use with OM Code Case OMN-1 to emphasize the testing provisions for MOVs in the IST program that are determined to have high safety significance, while allowing less precise testing for MOVs that are determined to have lower safety significance. ASME has incorporated these code cases into Appendix III to the 2009 Edition of the OM Code to replace quarterly MOV stroke-time testing with periodic exercising and diagnostic testing.

With respect to AOVs and HOVs, ASME developed OM Code Case OMN-12 to provide an alternative to the OM Code stroke-time testing provisions that incorporates risk insights to focus on AOVs and HOVs in the IST program that are determined to have the highest safety significance, while allowing less emphasis on AOVs and HOVs that have lower safety significance. ASME has incorporated some provisions of this Code Case into Appendix IV to the OM Code in the 2017 Edition.

NRC Recommendation

The NRC regulations in 10 CFR 50.55a(b)(3)(ii) require nuclear power plant licensees implementing the OM Code incorporated by reference in 10 CFR 50.55a to supplement the MOV testing provisions in the OM Code with a program to periodically demonstrate the design-basis capability of safety-related MOVs. The NRC staff considers that licensees may apply the lessons learned from their MOV programs developed in response to GLs 89-10 and 96-05 in satisfying the regulatory requirement to supplement the MOV stroke-time provisions in the OM Code. The NRC staff also considers the provisions in OM Code Case OMN-1 as accepted in RG 1.192 to satisfy the regulatory requirement to supplement the quarterly MOV stroke-time provisions. ASME has replaced the quarterly MOV stroke-time testing provisions in Appendix III to the 2009 Edition through the 2017 Edition of the OM Code that incorporates the provisions in OM Code Cases OMN-1 and OMN-11. The NRC staff also considers the alternative approach to quarterly stroke-time testing of AOVs and HOVs in OM Code Case OMN-12 as accepted in RG 1.192 to provide an acceptable alternative to the OM Code stroke-time testing provisions for those POVs as discussed in RG 1.192. Therefore, the NRC staff recommends that licensees update their IST programs to implement Appendices III and IV to the OM Code as incorporated by reference in 10 CFR 50.55a. OM Code Cases OMN-1, OMN-11, and OMN-12, as accepted by the NRC (with certain conditions) in RG 1.192, are also acceptable as alternatives to the stroke-time testing provisions in the OM Code for applicable POVs. In the future, licensees will be responsible for modifying their MOV testing programs in accordance with the 10-year update requirements for IST programs.

Basis for Recommendation

ASME developed Appendices III and IV as well as OM Code Cases to address weaknesses in the ability of stroke-time testing to assess the operational readiness of POVs in IST programs at nuclear power plants. Appendices III and IV, and the Code Cases, incorporate risk insights to emphasize IST provisions for POVs that are determined to have the highest safety significance. The NRC has incorporated by reference Appendices III and IV of the OM Code in 10 CFR 50.55a with specific conditions. The NRC has accepted several Code Cases with certain conditions. For example, RG 1.192 allows licensees with an applicable Code of Record to implement OM Code Case OMN-1 (in accordance with the provisions in the regulatory guide) as an alternative to the OM Code provisions for MOV stroke-time testing, without submitting a request for relief from their Code of Record. In RG 1.192, the NRC staff also accepts (with certain conditions) the use of the risk-informed provisions in OM Code Case OMN-11 by applicable licensees, in conjunction with OM Code Case OMN-1. RG 1.192 also allows licensees with an applicable Code of Record to implement OM Code Case OMN-12 for AOVs and HOVs (with certain conditions) in lieu of the OM Code provisions for stroke-time testing, without the need to submit a relief request. Licensees with a Code of Record that is not applicable to the acceptance of these OM Code Cases may submit a request to apply those OM Code Cases, consistent with the indicated conditions, as an alternative to the OM Code that provides an acceptable level of quality and safety. ASME has incorporated OM Code Cases OMN-1 and 11 as Appendix III and OM Code Case OMN-12 as Appendix IV to the OM Code. Therefore, the NRC regulations will require licensees to modify their MOV and AOV testing programs to apply the provisions of Appendices III and IV to the OM Code in accordance with the 10-year update requirements for IST programs.

4.2.6 Main Steam Isolation Valves

In IN 85-84, "Inadequate Inservice Testing of Main Steam Isolation Valves (MSIVs)," the NRC staff described an inadequacy in the IST of MSIVs. Specifically, the NRC staff stated that two licensees were testing their MSIVs using the nonsafety-related instrument air to achieve closure. Fail-safe IST of MSIVs as required by paragraph ISTC-3560 necessitates the removal of the instrument air supply and electric power. Operating experience related to MSIVs is described in IN 94-08, "Potential for Surveillance Testing to Fail to Detect an Inoperable Main Steam Isolation Valve," and IN 94-44, "Main Steam Isolation Valve Failure to Close on Demand Because of Inadequate Maintenance and Testing."

NRC Recommendation

The NRC staff recommends that licensees review their inservice and fail-safe testing of MSIVs to ensure compliance with OM Code requirements.

Basis for Recommendation

The practice of performing IST of components that are relied on to mitigate the consequences of plant events using sources of power that were not considered in the safety analyses is inconsistent with the objective of periodic IST for fail-safe testing. In IN 85-84, the NRC staff alerted licensees that, with low or no steam flow, the MSIV might not close unless instrument air is available to power the actuator.

In its Service Information Letter 477, the General Electric Company (GE) described a related issue for BWRs in which excessive tightening of gland flanges in the MSIV can prevent the valve from closing in response to spring force alone. During a postulated design-basis event in which a recirculation line breaks with the MSIVs open, containment pressure may increase significantly, and may exert an opening force on the valve actuators inside containment. Under such circumstances, the MSIV springs alone will not close the MSIV unless the spring force can overcome the combination of the opening force caused by containment pressure and the resistive force caused by stem packing friction. GE recommended a review of packing chamber maintenance practices, "springs-only" full-stroke closing tests, a force balance in which containment pressure is considered, a leak-tightness test of the MSIV actuator and accumulator, and a modification of the applicable licensing-basis documents. GE noted that this would necessitate measurement of the actual valve stem travel because the final 10 percent of stem travel coincides with the weakest spring force. GE stated that, by monitoring position switches alone, a utility could not determine that the valve is fully closed because the switches monitor the valve only when it is 90-percent open or 90-percent closed. One BWR licensee identified that the MSIVs would not pass local leak rate testing after closing on spring force only.

4.2.7 Verification of Remote Position Indication for Valves by Methods Other Than Direct Observation

ASME OM Code (2017 Edition), paragraph ISTC-3700, "Position Verification Testing," states

> Valves with remote position indicator shall be observed locally at least once every 2 yr. to verify that valve operation is accurately indicated. Where practicable, this local observation should be supplemented by other indications such as the use of flow meters or other suitable instrumentation to verify obturator position. These observations need not be concurrent. Where local observation is not possible, other indications shall be used for verification of valve operation. Position verification for active MOVs shall be tested in accordance with Mandatory Appendix III of [the OM Code].

The NRC regulations in 10 CFR 50.55a(b)(3)(xi), "OM condition: Valve Position Indication," require that licensees who are implementing paragraph ISTC-3700 in the ASME OM Code, 2012 Edition through the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a, shall 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 proper obturator position for valves with remote position indication within the scope of Subsection ISTC including its mandatory appendices and their verification methods and frequencies.

For many years, the NRC staff has indicated that the provisions in ISTC-3700 to verify valve position indication were unclear. Since 1995, NUREG-1482 and its revisions have emphasized the importance of proper implementation of ISTC-3700 to provide reasonable assurance that the indicating lights for valves with remote position indication properly inform the plant operators whether the valve is open or closed. Recent operating experience, including 10 CFR Part 21 notices, where valve discs have become separated from the valve stems at nuclear power plants reinforced the need to supplement the ISTC-3700 provisions with a condition in 10 CFR 50.55a.

The requirement that valves with remote position indicators must be observed at least once every 2 years to verify that valve position is accurately indicated has been specified in the OM Code and the previous ASME/ANSI OM Part 10 for many years. The additional guidance for supplementing the local observation was not present in older versions of OM Part 10, such as in the 1983 version.

Many valves have no provision for verifying the position by direct observation. To verify the position by observation, licensees can disassemble the valve, which could introduce additional valve failure mechanisms. Other methods (such as nonintrusive techniques, causing the flow to begin or cease, leak testing, and pressure testing) can yield a positive indication of disk position.

NRC Recommendation

ISTC-3530, “Valve Obturator Movement,” allows obturator movement to be determined by indicating lights in the control room when exercising the valve to meet the quarterly stroke-time testing requirement of the OM Code. The valve position verification testing required by ISTC-3700 provides confirmation on a 2-year frequency that the indicating lights reflect actual valve operation. ISTC-3700 allows flexibility to licensees in verifying that operation of valves with remote position indicators is accurately indicated. Operating experience has revealed that indicating lights might not be sufficient to verify valve position. The extent of verification necessary for valve operation to satisfy ISTC-3700 will depend on the type of valve, the sophistication of the diagnostic equipment used in testing the valve, possible failure modes of the valve, and the operating history of the valve and similar valve types. After such consideration, the licensee will be responsible for developing and implementing a method to verify that valve operation is accurately indicated as required by ISTC-3700.

Basis for Recommendation

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. Nuclear power plant licensees are responsible for satisfying 10 CFR Part 50, Appendix A, where codes and standards are insufficient to provide reasonable assurance that components are capable of performing their safety functions.

In Section 4.2.5, “Verification of Remote Position Indication for Valves by Methods Other Than Direct Observation,” of the original issued version of NUREG-1482 dated April 1995, the NRC staff noted that the OM Code requires that valves with remote position indicators be observed at least once every 2 years to verify that valve position is accurately indicated. The NRC staff stated that if remote valve position cannot be verified by local observation at the valve, an acceptable approach is for the licensee to observe operational parameters such as leakage, pressure, and flow that give positive indication of the valve’s actual position. The NRC staff indicated its interpretation of the OM Code requirement by stating that for certain types of valves that can be observed locally, but for which valve stem travel does not assure the stem is attached to the disk, the local observation must be supplemented by observing an operating parameter as required in the OM Code. In the basis discussion, the NRC staff stated that accurate position indication for safety-related valves is important for reactor operation during all plant conditions. Therefore, the NRC staff noted that the OM Code requires verification of the accuracy of the remote position indication for all valves in the IST program with remote position indication. The NRC staff indicated that many positive ways are available to verify the indication that a valve is open or closed. For example, the NRC staff referenced leak-rate testing, inline flow rate instrumentation, and system and differential pressures for indication of valve position.

In Section 4.2.7 of Revision 1 to NUREG-1482, dated January 2005, the NRC staff discussed its interpretation of the OM Code for position indication verification. The NRC staff noted that ISTC-3700 requires verification of the accuracy of the remote position indication for all valves in the IST program with remote position indication. The NRC staff indicated that if licensees cannot verify remote valve position by local observation at the valve, an acceptable approach is for the licensee to observe operational parameters (such as leakage, pressure, and flow) that give a positive indication of the valve’s actual position. In Revision 1, the NRC staff modified the discussion on supplementing local observation of valves for which stem travel does not assure that the stem is attached to the disk with observation of operating parameters (i.e., changing “must” to “should”) because Revision 1 to the NUREG removed the preapproval of this approach and required Commission approval pursuant to 10 CFR 50.55a.

The OM Code requires licensees to verify the accuracy of the remote position indication for all valves in the IST program that have remote position indication. Paragraph ISTC-3700 states that where local observation is not possible, licensees shall use other indications to verify operation. Nuclear power plant operating experience has revealed that reliance on indicating lights and stem travel are not sufficient to satisfy the requirement in ISTC-3700 to verify that valve operation is accurately indicated for those valves where the integrity of the internal mechanism of the valve (such as stem-to-disk connection) cannot be assured. Criteria V of Appendix B to 10 CFR Part 50 requires safety-related components that are subjected to test activities be required to have appropriate instructions, procedures, or drawings and qualitative or quantitative acceptance criteria for determining that activities have been successfully completed. Therefore, licensees are responsible for developing a method to verify that valve operation is accurately indicated to satisfy ISTC-3700 requirements such that IST can help to identify the stem-to-disc separation as valves are tested.

The NRC staff discussed the OM Code provisions for valve position verification in IN 2012-14 (July 24, 2012), “Motor-Operated Valve Inoperable due to Stem-Disc Separation.”

In the modification to 10 CFR 50.55a issued in 2020, the NRC staff clarified 10 CFR 50.55a(b)(3)(xi) to require that licensees who are implementing ISTC-3700 in the ASME OM Code, 2012 Edition through the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a shall 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 proper obturator position for valves with remote position indication within the scope of Subsection ISTC including its mandatory appendices and their verification methods and frequencies. Although not specifically mentioned in 10 CFR 50.55a(b)(3)(xi), applicants and licensees may submit an alternative request in accordance with 10 CFR 50.55a(z) to propose that a combination of plant activities (such as periodic flow testing and Appendix J leakage testing) be used to demonstrate that the valve disk is in the open position during flow testing and is in the closed position during Appendix J leakage testing to satisfy the intent of 10 CFR 50.55a(b)(3)(xi) on a longer interval than the 2-year interval specified in ISTC-3700. The clarification of 10 CFR 50.55a(b)(3)(xi) reflects that valve position indication may be demonstrated according to the internal inspection interval for check valves in ASME OM Code, Appendix II; the diagnostic testing interval for MOVs in ASME OM Code, Appendix III; and the performance assessment testing interval for AOVs in ASME OM Code, Appendix IV, rather than the 2-year interval specified in ISTC-3700.

4.2.8 Requirements for Verifying Position Indication of Passive Valves

The OM Code does not restrict the verification of position indication to only active valves. Table ISTC 3500-1 of the OM Code indicates that the licensee must also locally verify the position indication for Category B passive valves. As discussed in Section 4.2.7 of this NUREG, the licensee is responsible for developing and implementing a method to verify that valve operation is accurately indicated as required by ISTC-3700. The extent of verification necessary for valve operation to satisfy ISTC-3700 will depend on the type of valve, the sophistication of the diagnostic equipment used in testing the valve, possible failure modes of the valve, and the operating history of the valve and similar valve types.

The OM Code does not require licensees to verify the indication at the remote panels. However, verification at remote panels is a good practice and provides confidence in the remote indication.

4.2.9 Control Valves with a Safety Function

In general, control valves that used only for system control would be exempt from IST as discussed in paragraph ISTC-1200. However, some control valves also perform safety or fail-safe functions (e.g., fail open, fail closed, or fail as-is), and such valves must be tested in accordance with the OM Code IST requirements. The NRC staff has received many requests for relief from stroke-time measurement requirements, based on the impracticality of performing the measurement by the conventional method using position indication lights. Typically, the control valves do not have position indication, and testing can only be performed by bypassing control signals. To allow stroke timing by bypassing the control signals of those control valves that have position indication lights, the licensee may have to drain systems, which might make it impractical to test at the OM Code-defined frequency.

NRC Recommendation

Control valves that perform a safety or fail-safe function must be tested in accordance with the OM Code provisions for IST to monitor the valves for degrading conditions. The NRC staff recommends that licensees apply OM Code Case OMN-8, “Alternative Rules for Preservice and Inservice Testing of Power-Operated Valves That Are Used for System Control and Have a Safety Function per OM-10, ISTC-1.1, or ISTA-1100,” as accepted in RG 1.192. OM Code Case OMN-8 states that stroke-time testing need not be performed for POVs when the only safety-related function of those valves is to fail safe. Any abnormality or erratic action experienced during valve exercising should be recorded in the test record and an evaluation should be performed. Code Case OMN-8 is endorsed in RG 1.192, Revision 3, with no conditions. The OM Code (2017 Edition) stated that Code Case OMN-8 is applicable to the 1995 Edition though the 2004 Edition. Therefore, a licensee must submit a request in accordance with 10 CFR 50.55a to use Code Case OMN-8 if the licensee’s Code of Record is later than the 2004 Edition, until ASME updates Code Case OMN-8 to more recent editions and RG 1.192 accepts the updated versions of Code Case OMN-8.

4.2.10 Pressurizer Power-Operated Relief Valve Inservice Testing

Power-operated relief valves (PORVs) were often not purchased as safety-related valves, and the function of these valves to provide pressure control for plant transients was not considered safety-related. The valves were not designed to serve as overpressure protection devices during power operations, as required by ASME BPV Code, Section III, but many have been used as low-temperature overpressure protection valves.

NRC Recommendation

Recognizing that the PORVs have shown a high likelihood of sticking open and are not needed for overpressure protection during power operation, the provisions in paragraphs ISTC-3500 and ISTC-5100 for exercising quarterly during power operation are not practical and, therefore, exercising may be performed during cold shutdown conditions. Paragraph ISTC-3310 requires licensees to perform testing after maintenance or repair. Test methods must confirm that the PORV has been reassembled correctly and is capable of performing its design function.

There have been instances (see IN 96-02, "Inoperability of PORVs Masked by Downstream Indications During Testing") where improper evaluation of testing failed to identify the incorrect reassembly of a PORV.

Previously approved NRC guidance included in GL 90-06 (see below) indicates that, because the PORVs function during reactor startup and shutdown to protect the reactor vessel and coolant system from low-temperature overpressurization conditions, they should be exercised before system conditions warrant vessel protection, and should also be exercised after the operational readiness of the block valves is ensured, by exercising and stroke-timing (as applicable) according to the following test schedule:

- Perform full-stroke exercising during each cold shutdown or, as a minimum, once each refueling cycle.
- Perform stroke timing during each cold shutdown, or as a minimum, once each refueling cycle.
- Perform fail-safe testing during each cold shutdown, or as a minimum, once each refueling cycle.
- Include the PORV block valves in the IST program, and test them quarterly to ensure protection against a small-break LOCA in the event that a PORV fails open.
- If the plant frequently enters cold shutdown mode, testing of the PORVs is not required more often than once every 3 months,

Basis for Recommendation

The NRC's guidance on the IST requirements for PORVs is included in GL 90-06, "Resolution of Generic Issue 70: Power-Operated Relief Valve and Block Valve Reliability, and Generic Safety Issue 94: Additional Low-Temperature Overpressure Protection for Light-Water Reactors, pursuant to 10 CFR 50.54(f)." In IN 89-32, "Surveillance Testing of Low-Temperature Overpressure-Protection Systems," the NRC discussed the stroke time assumptions made in plants' safety analyses for these PORVs, and the IST performed for these valves. Stroke times of the valves were unacceptable or were not measured in the direction required for low-temperature overpressure systems to prevent exceeding the limits in Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50. Implementation of the guidance in GL 90-06 has been coordinated between the licensees and the NRC project managers for each plant on a case-by-case basis.

4.2.11 Online Check Valve Sample Disassembly and Inspection

Licensees have proposed, as an alternative to ISTC-5221(c) and ISTC-5224, to perform sample disassembly and inspection of check valves in a group online. Paragraph ISTC-3510 of the OM Code, requires that check valves be exercised every 3 months. Subparagraph ISTC-3522(c) states that if exercising is not practicable during operation at power and cold shutdown, it shall be performed during refueling outages. ISTC-5221(c) allows disassembly of check valves every refueling outage as an alternative means to verify their operability. Instead of disassembly every refueling outage, ISTC-5221(c) provides the option of using a sample disassembly and inspection program for groups of identical valves in similar applications. Further, ISTC-5221(c)(3) states that at least one valve from each group shall be disassembled and examined at each refueling outage and all valves in each group shall be disassembled and examined at least once every 8 years. ISTC-5224 requires that check valves in a sample disassembly program that are not capable of being full-stroke exercised or have failed or have unacceptably degraded valve internals, shall have the cause of failure analyzed and the condition corrected.

ISTC-5224 also states that other check valves in the sample group that may also be affected by this failure mechanism be examined or tested during the same refueling outage to determine the condition of internal components and their ability to function. A licensee should fully describe how it plans to comply with the requirements in ISTC-5224 when submitting alternative requests for check valve group sample disassembly and inspection online. The plan description also should include information on management of examination and testing of all group valves should a scheduled valve inspection be declared inoperable. For example, licensees should explain how the disassembly and inspection of the other check valves in a group will be completed within the allowed system outage time.

4.2.12 POVs in New Reactors

The NRC regulations in 10 CFR 50.55a(b)(3)(iii) apply specific conditions for IST programs applicable to licensees of new nuclear power plants in addition to the provisions of the OM Code as incorporated by reference with conditions in 10 CFR 50.55a. Licensees of "new reactors" are, as identified in 10 CFR 50.55a, holders of operating licenses for nuclear power reactors that received construction permits under this part on or after the date 12 months after August 17, 2017, and COL holders issued under 10 CFR Part 52, whose initial fuel loading occurs on or after the date 12 months after August 17, 2017.

Commission Papers SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements;" SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs;" SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs;" and SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)," discuss IST programs for new reactors licensed under 10 CFR Part 52.

In recognition of new reactor designs and lessons learned from nuclear power plant operating experience, the ASME is updating the OM Code to incorporate improved IST provisions for components used in nuclear power plants that were issued (or will be issued) construction permits, or COLs, on or following January 1, 2000 (defined in the OM Code as post-2000 plants). The first phase of the ASME effort incorporated IST provisions that specify full flow pump testing and other clarifications for post-2000 plants in the OM Code beginning with the 2011 Addenda. The second phase of the ASME effort incorporated preservice and inservice inspection and surveillance provisions for pyrotechnic-actuated (squib) valves in the 2012 Edition of the OM Code. The ASME is considering further modifications to the OM Code to address additional lessons learned from valve operating experience and new reactor issues. As described in the following paragraphs, 10 CFR 50.55a(b)(3)(iii) includes four specific conditions.

The NRC regulations in 10 CFR 50.55a(b)(3)(iii)(A) require licensees within the scope of 10 CFR 50.55a(b)(3)(iii) to periodically verify the capability of POVs to perform their design-basis safety functions. While Appendix III to the OM Code addresses this requirement for MOVs with the conditions specified in 10 CFR 50.55a, applicable applicants and licensees will need to develop programs to periodically verify the design-basis capability of other POVs. RIS 2000-03 provides attributes for a successful long-term periodic verification program for POVs by incorporating lessons learned from MOV performance at operating nuclear power plants and research programs. Implementation of Appendix III to the OM Code as accepted in 10 CFR 50.55a(b)(3)(ii) satisfies § 50.55a(b)(3)(iii)(A) for MOVs.

Paragraph (A) of 10 CFR 50.55a(b)(3)(iii) is consistent with the Commission policy for new reactors summarized in an NRC Staff Memorandum, "Consolidation of SECY-94-084 and SECY-95-132," dated July 24, 1995, that (a) the design capability of safety-related POVs should be demonstrated by a qualification test prior to installation; (b) prior to initial startup, POV capability under design-basis differential pressure and flow should be verified by a pre-operational test; and (c) during the operational phase, POV capability under design-basis differential pressure and flow should be verified periodically through a program similar to that developed for MOVs in GL 89-10. The condition in paragraph (A) specifies with the same level of detail as the condition in 10 CFR 50.55a(b)(3)(ii) that nuclear power plant licensees must establish a program to ensure the continued capability of MOVs in performing their design-basis safety functions. When establishing the MOV periodic verification condition, the NRC provided guidance in the final rule published September 22, 1999 (64 FR 51370), for licensees to develop acceptable programs that would satisfy the MOV periodic verification condition. Similarly, the NRC staff provided guidance in the *Federal Register* notice for the final rule dated July 18, 2017 (82 FR 32934), and summarized herein, for new reactor applicants and licensees to develop acceptable programs to periodically verify the capability of POVs to perform their design-basis safety functions.

In NUREG-2124, "Final Safety Evaluation Report Related to the Combined Licenses for Vogtle Electric Generating Plant, Units 3 and 4," the NRC staff found the provisions established by the COL applicant for VEGP Units 3 and 4 in its FSAR, Revision 5, Section 3.9.6.2.2, "Valve Testing," to periodically verify the capability of POVs (such as AOVs, SOVs, and HOVs) to perform their design-basis safety functions to be acceptable. In particular, the VEGP Units 3 and 4 FSAR Section 3.9.6, "Inservice Testing of Pumps and Valves," specifies that:

> Power-operated valves other than active MOVs are exercised quarterly in accordance with OM ISTC, unless justification is provided in the inservice testing program for testing these valves at other than Code mandated frequencies. Although the design basis capability of power-operated valves is verified as part of the design and qualification process, power-operated valves that perform an active safety function are tested again after installation in the plant, as required, to ensure valve setup is acceptable to perform their required functions, consistent with valve qualification. These tests, which are typically performed under static (no flow or pressure) conditions, also document the "baseline" performance of the valves to support maintenance and trending programs. During the testing, critical parameters needed to ensure proper valve setup are measured. Depending on the valve and actuator type, these parameters may include seat load, running torque or thrust, valve travel, actuator spring rate, bench set and regulator supply pressure. Uncertainties associated with performance of these tests and use of the test results (including those associated with measurement equipment and potential degradation mechanisms) are addressed appropriately. Uncertainties may be considered in the specification of acceptable valve setup parameters or in the interpretation of the test results (or a combination of both). Uncertainties affecting both valve function and structural limits are addressed. Additional testing is performed as part of the air-operated valve (AOV) program, which includes the key elements for an AOV Program as identified in the JOG AOV program document, Joint Owners Group Air Operated Valve Program

Document, Revision 1, December 13, 2000 (References 203 and 204) [JOG AOV Program Document, Revision 1, December 13, 2000 (ADAMS Accession No. ML010950310), and NRC comment letter dated October 8, 1999, to Nuclear Energy Institute (ADAMS Accession No. ML020360077)]. The AOV program incorporates the attributes for a successful power-operated valve long-term periodic verification program, as discussed in Regulatory Issue Summary 2000-03, Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions, by incorporating lessons learned from previous nuclear power plant operations and research programs as they apply to the periodic testing of air- and other power-operated valves included in the IST program.

For example, key lessons learned addressed in the AOV program include:

- Valves are categorized according to their safety significance and risk ranking.
- Setpoints for AOVs are defined based on current vendor information or valve qualification diagnostic testing, such that the valve is capable of performing its design-basis function(s).
- Periodic static testing is performed, at a minimum on high risk (high safety significance) valves, 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 required based on valve qualification or operating experience, periodic dynamic testing is performed to re-verify the capability of the valve to perform its required functions.
- Sufficient diagnostics are 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 is specified, and is evaluated each refueling outage based on data trends as a result of testing. Frequency for periodic testing is in accordance with References 203 and 204 of the JOG Program, with a minimum of 5 years (or 3 refueling cycles) of data collected and evaluated before extending test intervals.
- 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 attributes of the AOV testing program described above, to the extent that they apply to and can be implemented on other safety-related power-operated valves, such as electro-hydraulic operated valves, are applied to those other power-operated valves

Applicable applicants and licensees may follow the method described in the VEGP Units 3 and 4 FSAR in satisfying 10 CFR 50.55a(b)(3)(iii)(A), or may establish a different method, subject to evaluation by the NRC during the licensing process or inspections.

4.2.13 Relationship between GL 89-10, GL 96-05, EPRI MOV Performance Prediction Methodology (PPM), Joint Owners Group (JOG), and Mandatory Appendix III for Motor-Operated Valves

On June 28, 1989, the NRC staff issued GL 89-10 in response to operating experience concerns regarding MOV performance at nuclear power plants. In GL 89-10, the NRC staff 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. In implementing MOV programs in response to GL 89-10, licensees and permit holders found many MOVs could not be practicably tested under design-basis conditions.

As part of the industry response to the MOV issue, the Electric Power Research Institute (EPRI) initiated a program to develop a methodology for bounding MOV performance to demonstrate the design-basis capability when valve specific test data is not available. The program included development of improved methods for prediction or evaluation of system flow parameters; gate, globe, and butterfly valve performance; and motor actuator rate of loading (load sensitive behavior) effects. The EPRI MOV Performance Prediction Methodology (PPM) includes computer models, software, and hand calculation models to implement individual valve and system models and methods. EPRI compared the predictions of the integrated methodology to valve test data from flow loop and in-situ tests. EPRI determined that the thrust and torque predictions from the methodology bounded the measured data for 173 of the 176 total analyzed valve strokes. EPRI stated that the use of the computer model assumes that the valve is in good condition and that model users will need to ensure that an adequate internal valve preventive maintenance program is established for the thrust and torque requirements predicted by the model to remain valid. The NRC staff and its contractor Idaho National Engineering Laboratory (now referred to as INL) evaluated the topical report for the EPRI MOV PPM. The NRC staff approved the use of the EPRI PPM, with limitations and conditions, specified in a safety evaluation dated March 15, 1996 (ADAMS Accession No. ML15142A761) and four supplements (ADAMS Accession No. ML090400621).

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

1. Dynamic testing at or near design-basis conditions with diagnostics of each MOV where practicable. Valves dynamically tested at less than design-basis conditions may be extrapolated with proper justification. Although the valve factor derived from the test data might be low because of minimal valve operating history or recent maintenance that exposed the Stellite valve material to air, the dynamic testing provided assurance that the

valve performance was predictable. The licensee would consider the need to increase the valve factor during its design-basis evaluation and setup based on test data from similar valves.

2. Implementation of the EPRI MOV PPM as accepted in the NRC safety evaluations. This method was developed for those valves that could not be dynamically tested. The PPM required internal measurements of the valve to provide assurance that the valve performance was predictable. The NRC staff accepted the use of the PPM even where dynamic testing for an MOV was practicable.
3. MOV valve grouping. Where valve-specific dynamic testing was not performed, and the PPM was not used, the staff accepted grouping of MOVs that were dynamically tested at the plant to apply the plant-specific test information to an MOV in the group. Using plant-specific data allowed the licensee to know the valve performance and maintenance history and helped provide confidence that the valve performance was predictable.
4. The use of valve test data from other plants or research programs. The NRC staff ranks this as the least preferred approach (with the most margin required) because the licensee would have minimal information regarding the tested valve and its history. In such cases, the NRC inspectors would perform an available capability evaluation of the MOV to provide confidence that the MOV had significant capability margin to close GL 89-10 for that MOV.

Based on a series of inspections, the NRC staff closed its review of the GL 89-10 program at each nuclear power plant. On September 18, 1996, the NRC issued GL 96-05 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. The program should ensure that the licensee can properly identify and account for changes in required performance resulting from degradation (such as those caused by age).

In response to GL 96-05, nuclear power plant licensees developed an industry-wide JOG Program on MOV Periodic Verification. The JOG program included the following three phases:

1. an interim plan of static diagnostic testing of safety-related MOVs at a frequency based on risk and margin
2. a JOG testing program to evaluate potential valve degradation whereby each participating licensee performed three static and dynamic diagnostic tests on two selected valves over a 5-year period with a minimum of 1 year between tests
3. a long-term plan of static diagnostic testing and, where necessary, dynamic diagnostic testing based on the JOG's evaluation of the valve performance data from the 5-year testing program

The JOG program tested over 150 different gate, globe, and butterfly valves in various applications, such as treated water, untreated water, hot water, cold water, and steam. The test results concluded that valves perform the same when they share the same attributes, such as valve type, fluid application, disk to seat material, disk to body material, and bearing material. The JOG program classified MOVs at nuclear power plants on the basis of their attributes and performance as follows:

- Class A: Class A valves are valves within the scope of the JOG program that have been determined not to be susceptible to degradation in their operating requirements based directly on testing performed in the JOG program or on other suitable bases (e.g., EPRI MOV PPM) as discussed in the JOG program.

- Class B: Class B valves are valves within the scope of the JOG program that have been determined not to be susceptible to degradation in their operating requirements based on testing performed in the JOG program, extended by analysis or engineering judgment to configurations and conditions beyond those tested.
- Class C: Class C valves are valves within the scope of the JOG program that have been determined to be susceptible to changes in the required thrust or torque based on the test results from the JOG program.
- Class D: Class D valves are valves that are determined to be outside the scope of the JOG program but within the scope of GL 96-05.

On February 27, 2004, the JOG submitted its topical report MPR-2524, "Joint Owners Group (JOG) Motor-Operated Valve Periodic Verification Program Summary" (ADAMS Accession No. ML040720092), to the NRC for review. The NRC staff accepted the industry topical report on the JOG Program on MOV Periodic Verification in an SER dated September 25, 2006 (ADAMS Accession No. ML061280315) and its supplement dated September 18, 2008 (ADAMS Accession No. ML082480638). MPR-2524-A (November 2006), "Joint Owners Group (JOG) Motor Operated Valve Periodic Verification Program Summary," (ADAMS Accession No. ML063490194) updates the topical report to reflect the NRC final SER and includes the JOG response to NRC staff requests for additional information and the final SER as appendices to the report. 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. In RIS 2011-13 (January 6, 2012), "Follow-up 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 conducting periodic verification of the design-basis capability of safety-related MOVs outside the scope of the JOG program. The JOG program evaluates degradation of the operating requirements for valves such that the lessons learned from the JOG program can be applied to valves with any power actuators. The JOG program does not include actuator output capability as part of its MOV program such that the licensee will need to address this aspect of MOV periodic verification on a plant-specific basis.

The OM Code establishes requirements for PST and IST activities to assess the operational readiness of MOVs within its scope used in water-cooled reactor nuclear power plants. The NRC regulations in 10 CFR 50.55a(b)(3)(ii) address the incorporation of the OM Code provisions for MOVs into 10 CFR 50.55a. In particular, 10 CFR 50.55a(b)(3)(ii) states in part that "Licensees must comply with the provisions for testing MOVs in OM Code, ISTC 4.2, 1995 Edition with the 1996 and 1997 Addenda, or ISTC–3500, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(iv) of this section, and must establish a program to ensure that MOVs continue to be capable of performing their design basis safety functions."

The 10 CFR 50.55a(b)(3)(ii) requirement for MOVs has two elements:

1. Licensee IST programs for MOVs at nuclear power plants must meet the OM Code requirements as incorporated by reference in 10 CFR 50.55a, including the conditions specified in 10 CFR 50.55a(b)(3)(ii)(A) to (D).

2. Licensees must implement an MOV program that provides continued assurance of the capability of MOVs to perform their design-basis safety functions.

The second element was a direct result of GL 89-10 that was further supported by GL 95-07 and GL 96-05. As noted above, GL 89-10 requested licensees to verify the design-basis capability of their safety-related MOVs by dynamic testing where practicable. GL 95-07 requested licensees to address concerns of potential pressure locking and thermal binding of power-operated gate valves. GL 96-05 requested licensees to develop programs to periodically verify MOV design-basis capability. The provisions in GL 96-05 superseded GL 89-10 for the periodic verification of MOV design-basis capability.

The OM Code committees combined the two elements of 10 CFR 50.55a(b)(3)(ii) by developing ASME OM Code Case OMN-1. The NRC staff approved Code Case OMN-1 with conditions in RG 1.192.

Code Case OMN-1 was incorporated into the OM Code beginning with the 2009 Edition as Mandatory Appendix III. The staff specified conditions for the implementation of Mandatory Appendix III in 10 CFR 50.55a(b)(3)(ii) when the applicable editions of the OM Code were incorporated by reference in 10 CFR 50.55a. Licensees may apply the lessons learned from their MOV programs, including implementation of the JOG MOV program, as accepted by the plant-specific SE in response to GL 96-05, in meeting the minimum requirements of Mandatory Appendix III. Licensees will need to ensure that the conditions in 10 CFR 50.55a(b)(3)(ii) are met when implementing Mandatory Appendix III to the OM Code. As noted above, the JOG program applied only to the valve operating requirements such that licensees must provide periodic verification of the output capability of the MOV actuators. Licensees need to follow their commitment change process when updating their MOV program from the JOG final MOV test program to the new requirements of Mandatory Appendix III.

Mandatory Appendix III requires a one-time design-basis verification test of the capability of each MOV to meet its safety-related design-basis requirements. Mandatory Appendix III indicates that an engineering evaluation may be used to satisfy this requirement where justified. The staff considers the implementation of the EPRI MOV PPM in accordance with the NRC safety evaluations to be an acceptable engineering evaluation. For MOVs within the scope of the GL 89-10 and GL 96-05 programs, the testing and/or engineering analysis performed to close out those generic letters may be used to meet the design-basis verification requirement in Mandatory Appendix III. For MOVs that were outside of the scope of GL 89-10 and GL 96-05 but within the IST program, a one-time test and/or engineering analysis is required by Mandatory Appendix III. In both cases, it is expected that licensees will have the design-basis verification test and/or engineering evaluation analysis formally documented for each MOV in the IST program.

Licensees that have evaluated MOVs using the EPRI MOV PPM to meet the requirements of a one-time design-basis verification test in Mandatory Appendix III will need to meet the EPRI condition of maintaining the valve in good condition for the EPRI MOV PPM to remain valid for that valve. A licensee should address the following aspects to provide assurance of acceptable valve condition:

1. Valve susceptibility to internal component degradation due to system operation and/or conditions.
2. Diagnostic testing that monitors and trends valve operating parameters (thrust, torque, running loads, motor current, motor power, etc.).

3. Periodic valve internal inspection/rework.
4. Valves that relied on vendor drawings to capture the internal dimensions to perform the EPRI MOV PPM calculation should be verified if a periodic valve internal inspection/rework is deemed necessary.

Following the initial demonstration of MOV design-basis capability, the IST program as required by Mandatory Appendix III of the OM Code as incorporated by reference in 10 CFR 50.55a with conditions will maintain the design-basis capability of MOVs by periodic exercising and diagnostic testing. The JOG program may be applied as part of the IST requirements specified in Mandatory Appendix III. The staff considers that the implementation of Mandatory Appendix III for MOVs within the scope of the OM Code will meet the requirements of 10 CFR 50.55a(b)(3)(ii) provided the specified regulatory conditions are satisfied.

4.3 <u>Safety and Relief Valves</u>

4.3.1 Scope

Paragraph ISTA-1100 of the ASME OM Code, Subsection ISTA, defines the scope of the valves subject to IST provisions to include pressure-relief devices that protect systems (or portions of systems) that perform a required function in shutting down the reactor to the safe shutdown condition, maintaining the safe shutdown condition, or mitigating the consequences of an accident that results from overpressure. Pressure-relief valves, which are installed in systems to protect against overpressure, might not, of themselves, appear to perform a specific function to shut down the reactor, maintain it in a safe shutdown condition, or mitigate the consequences of an accident. (Automatic depressurization valves in BWRs are an example of relief valves that perform both an overpressure protection function and a function to depressurize the primary system when opened on an automatic signal or by an operator.) However, they might be required to be included in the IST program and tested according to the schedules stipulated in Subsection ISTC and Appendix I of the OM Code. Specifically, Subsection ISTC of the OM Code clarifies that its requirements apply only to pressure-relief devices required for overpressure protection. The testing of these devices is to be included in 120-month updated IST programs.

Testing of “thermal relief valves” has been the subject of much discussion over the past several years. Contributing to some confusion, in many original system designs, so-called thermal relief valves were installed to protect isolated segments of piping that could be pressurized as a result of heating from some source, but were widely viewed as having no safety-related function in mitigating the consequences of accidents or ensuring any other system safety function. GL 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,” emphasized the importance of protecting certain isolated segments of piping from excessive thermally-induced pressurization, especially where containment integrity could be affected. In recent years, ASME has made changes to Appendix I to the OM Code to include specific requirements to periodically test or replace thermal relief valves.

The requirement to test safety and relief valves (S/RVs) that provide overpressure protection is based on the requirements of Section III of the ASME BPV Code, as well as the *USA Standard Code for Pressure Piping* (USAS B31.1) and the *USA Standard Code for Nuclear Power Piping*

(USAS B31.7). If the results of an overpressure protection “re-analysis” for a particular system indicate that a relief valve is not necessary, it may be removed from the scope of the IST program.

As required by ASME BPV Code, Section III, Article NX-7200, it is the Owner’s responsibility to prepare, certify, and file an Overpressure Protection Report for the facility. The Overpressure Protection Report defines the protected systems and the integrated overpressure protection provided. Article NX-7200 also contains requirements regarding verification that pressure relief devices are not required and reconciliation of the Overpressure Report following modifications.

4.3.2 Method of Testing Safety and Relief Valves

The NRC staff expects licensees to develop methods to test the entire assembly of safety and relief valves to demonstrate their operational readiness to perform the applicable safety functions. For example, the NRC staff does not consider testing of only a pilot assembly would be adequate to demonstrate the performance of the main valve disc for those types of safety and relief valves with pilot assemblies. The operational history of safety and relief valves does not support partial testing of valve assemblies.

4.3.2.1 BWR Safety/Relief Valve Stroke Testing

In recent years, the NRC staff has received numerous requests for relief or TS changes or both related to the stroke testing requirements for BWR dual-function main steam S/RVs. The 2003 Addendum and earlier editions and addenda to Mandatory Appendix I to the OM Code require the stroke testing of S/RVs after they are reinstalled following maintenance activities. A number of licensees have determined that in situ testing of the S/RVs can contribute to undesirable seat leakage of the valves during subsequent plant operation and have received approval to perform stroke testing at a laboratory facility coupled with in situ tests and other verifications of actuation systems as an alternative to the testing required by the OM Code. The revised subparagraph I-3410(d) in Mandatory Appendix I to the 2004 through 2017 Edition of the OM Code does not require licensees to stroke test S/RVs at reduced or normal system pressure following maintenance. Subparagraph I-3410(d) in the OM Code requires that each S/RV that has been removed for maintenance or testing, and reinstalled shall have the electrical and pneumatic connections verified either through mechanical/electrical inspection or testing before the resumption of electric power generation. Several licensees have requested and obtained NRC approval in accordance with 10 CFR 50.55a(f)(4)(iv) to use subparagraph I-3410(d) of the 2004 through 2017 Edition of the OM Code in place of subparagraph I-3410(d) of the 2001 Edition through the 2003 Addenda to the OM Code.

4.3.2.2 PWR Main Steam Safety Valve Set Pressure Testing

To reduce the need to remove valves from their installed position and the time required to perform set pressure testing, many PWR licensees perform testing of main steam safety valves (MSSVs) using in situ testing equipment with operating steam pressure. One advantage of this method is that actual environmental and fluid temperature conditions are used, in lieu of duplicating them in a test laboratory. However, this method has introduced inaccuracies because the set pressure is determined by a combination of the measured system operating pressure and the applied assisting force provided by the testing device. This assisting force is applied by pneumatic pressure on a piston or diaphragm and is converted to an equivalent additional amount of system pressure by dividing the force by the valve disk area against which the system pressure acts. Inaccuracies in the value of the disk area have caused some inaccuracies in the set pressure determination, as discussed in IN 94-56, "Inaccuracy of Safety Valve Set Pressure Determination Using Assist Devices."

4.3.3 Jack-and-Lap Process

In IN 91-74, "Changes in Pressurizer Safety Valve Setpoints Before Installation," the NRC stated that the setpoint changes in Dresser pressurizer safety valves could result, in part, from changes made during a jack-and-lap procedure that is performed after setpoint testing and before installation to reduce seat leakage. This procedure may have lacked adequate controls.

Many licensees avoid performing setpoint testing after jack-and-lap maintenance because this testing could lead to leakage. Subsection ISTC and Appendix I of the OM Code require that after repairing a valve or performing maintenance that could affect the valve's performance, the licensee must demonstrate that the performance parameters are acceptable by testing the valve before returning it to service. The licensee must test pressure relief devices as required by Subsection ISTC and Appendix I following replacement, repair, and maintenance.

The NRC staff recommends that, if a licensee chooses to use the jack-and-lap process and not re-verify the set pressure of the valves, the licensee must determine whether the maintenance activity is of an extent that a setpoint test is required after the valve is reassembled and reinstalled. If the jack-and-lap process is controlled so that the setpoint will not be affected, the licensee may not need to perform a test. Action in accordance with this recommendation necessitates determination of the effect of this activity and evaluation within the quality controls and quality assurance for the process. Controls include limits on the amount of material that is removed, the controls to ensure that the settings and adjustments of the valve parts that affect the setpoint are not changed, and the requirements in the maintenance procedure to address any unusual conditions that occur during the maintenance activity. The licensee may also consider industry experience to determine whether changes in the methods of performing this activity are necessary as plants accumulate more data. Because the NRC staff cannot make this determination by evaluating a relief request, relief is neither appropriate nor available for this activity.

4.3.4 Maintenance and Inspection of Safety and Relief Valves in Addition to ASME OM Code Requirements

Licensees should note that not all maintenance and inspection that may be needed to ensure continued functional capability of safety, relief, and pilot valves are necessarily performed as a result of inservice testing required by the OM Code. In a recent case involving some BWR S/RVs, additional periodic maintenance and inspection of certain internal parts were necessary to check for excessive wear and eventual binding of the main disks. This was discovered on valves that had successfully passed required inservice tests, and is discussed further in IN 2003-01, "Failure of a Boiling-Water Reactor Main Steam Safety/Relief Valve."

4.3.5 Scheduling of Safety and Relief Valve Testing

Appendix I to the OM Code requires that licensees must test a minimum size sample of valves within a valve group within a specified period. A penalty is also applied, in that additional valves must be tested when any of the samples fail to meet the necessary acceptance criteria. In determining the minimum acceptable sample size, fractions of valve numbers resulting from calculating the number of valves to be tested are to be rounded to the next higher whole number.

4.3.6 Use of ASME OM Code Case OMN-17

Many licensees have requested and obtained NRC authorization in accordance with 10 CFR 50.55a(z)(1) to use the provisions in OM Code Case OMN-17, "Alternative Rules for Testing ASME Class 1 Pressure Relief/Safety Valves," as an alternative to the 5-year test interval specified in the OM Code. Code Case OMN-17 allows an extension of the test frequency for S/RVs from 60 months to 72 months plus a 6-month grace period. The Code Case imposes a special maintenance requirement to disassemble and inspect each valve to verify that parts are free from defects resulting from the time-related degradation or maintenance-induced wear before the start of the extended test frequency. Although the OM Code does not require that S/RVs be routinely refurbished, refurbishment provides reasonable assurance that the S/RVs are operationally ready during the extended test interval. ASME published Code Case OMN-17 in the 2009 Edition through 2017 Edition of the OM Code. The Code Case is endorsed in RG 1.192, Revision 3, with no conditions. It is stated in the Code Case that it is applicable to the 1995 Edition though the 2006 Addenda. Therefore, a relief request is required to use Code Case OMN-17 if the licensee's Code of Record is later than the 2006 Addenda, until the limited applicability statement in Code Case OMN-17 is resolved.

4.4 <u>Miscellaneous Valves</u>

The following issues and NRC recommendations apply to miscellaneous types of valves.

4.4.1 Post-Accident Sampling System Valves

NUREG-0737, Clarification of TMI Action Plan Requirements, Section II.B.3, details the requirements and capabilities of post-accident sampling systems (PASSs) for sampling both the reactor coolant and the containment atmosphere. The PASS consists of pumps and valves that perform these and other functions. The PASS also includes valves that perform a containment isolation function.

NRC Recommendation

The IST program applies to any PASS valves within the scope of 10 CFR 50.55a and the OM Code. Such valves in the PASS that perform a containment isolation function must be included in the IST program as Category A or A/C, and must be tested to OM Code requirements except where relief has been granted or an alternative authorized.

The remaining valves in the PASS would typically be tested as required by the TS or other documents and need not be included in the IST program. However, the NRC staff recommends that if the licensee elects to include these valves in the IST or augmented IST program, a note should be included that the testing is beyond the scope of 10 CFR 50.55a.

In many cases, a licensee's TS have been amended to eliminate the requirements to have and maintain a PASS. If a PASS valve is eliminated from the TS but still performs a function within the scope of 10 CFR 50.55a and the OM Code, the valve should remain in the IST program.

4.4.2 Post-Maintenance Testing After Stem Packing Adjustments and Backseating of Valves to Prevent Packing Leakage

Paragraph ISTC-3310, "Effects of Valve Repair, Replacement, or Maintenance on Reference Values," in the ASME OM Code, Subsection ISTC, requires that, when a valve or its control system has been replaced, repaired, or has undergone maintenance that could affect the valve's performance, a new reference value shall be determined or the previous values reconfirmed by an inservice test before the time it is returned to service or immediately if not removed from service.

Examples of maintenance are provided in a footnote of paragraph ISTC-3310 and include: adjustment of stem packing, limit switches, or control system valves, removal of the bonnet, stem assembly, actuator, obturator, or control system components.

Backseating a valve may also affect its performance (e.g., cause damage to the valve or bind it into its backseat).

It may be necessary to adjust the stem packing during power operations in order to stop stem packing leaks on valves that must remain in position for operations to continue. Examples include MSIVs and main feedwater isolation valves. If the leakage does not pose a personnel safety hazard, licensees may adjust the packing without removing the valves from service. Alternatively, backseating a valve may stop packing leakage without the need to take the valve out of service. Licensees should exercise caution when performing such maintenance, as improper backseating or adjustment of valve stem packing could adversely affect the valve's functional capability. Licensees will need to have justification that backseating or packing adjustments do not adverse impact the operational readiness of the valve.

NRC Recommendation

The NRC staff has determined that it is acceptable for licensees to perform an engineering evaluation of the impact of adjusting valve stem packing or backseating a valve to demonstrate that the performance parameters are within acceptable limits if a stroke test cannot be performed under current plant conditions. If it is necessary to adjust the stem packing or backseat a valve to stop packing leakage and if a required stroke test or leak rate test is not practical in the current plant mode, the licensee must, at a minimum, justify by analysis that (1) the packing adjustment is within manufacturer-specified torque limits for the existing packing configuration, (2) the backseating does not deform the valve stem, and (3) the performance parameters of the valve are not adversely affected (including the stroke time of the valve). When intending to use backseating to stop a packing leak, the licensee will be expected to stroke the valve stem away from the backseat after the initial backseating operation to demonstrate that the valve stem will not become bound in the backseat by this temporary leakage mitigation method. In addition, the licensee must perform a confirmatory test at the first available opportunity when plant conditions allow testing. Packing adjustments beyond the manufacturer's limits may not be performed without (1) an engineering analysis showing that the performance parameters of the valve are not adversely affected, and (2) input from the manufacturer, unless tests can be performed after adjustments.

Examples of such valves are MSIVs and main feedwater isolation valves, which must remain open to continue power operations. The licensee must evaluate any data available from previous testing with the packing torqued to the specified limit, and must verify that the valve was leak tight and previously stroked within acceptable limits with the packing adjusted to the higher value, or from previous instances of backseating a valve.

Granting of relief under 10 CFR 50.55a(f) is not necessary because this action is in accordance with the OM Code requirements if the licensee can demonstrate that the performance parameters will not be adversely affected.

To properly implement this guidance, licensees must perform a partial-stroke test, if practical, to obtain further assurance that the valve stem is free to move. At the first opportunity when the plant enters an operating mode in which testing is practical, the licensee must test all valves that have had packing adjustments or been backseated without post-maintenance testing. The maintenance procedure used to adjust the packing must include the limits, and any changes to the torque limits are subject to a 10 CFR 50.59 review. Licensees should avoid adjusting redundant valves without performing post-maintenance testing. Backseating procedures should include precautions to prevent stem deformation.

To properly implement this guidance, a licensee must evaluate valves individually, unless it has established a valve packing program in which designated limits, justified by test data, allow adjustments that do not affect performance parameters. Specific or general relief is not appropriate for this activity. If the licensee cannot demonstrate that the packing adjustment does not adversely affect performance parameters, the OM Code requirements must be met for post-maintenance testing. Therefore, the licensee must consider this issue for each valve individually.

Basis for Recommendation

The NRC staff would not require a licensee to shut down a plant to perform IST unless the licensee has no alternative to ensure that the operational readiness of components is maintained or a safety issue exists. The IST requirements do not prohibit or discourage a licensee from making limited adjustments to valve packing to stop a leak that may be adversely affecting the valve or surrounding components. Therefore, the licensee can perform an analysis of the packing adjustment and, upon demonstrating that the adjustment does not adversely affect the stroke time (or leakage rate) such that it would not exceed its limiting value, can make the adjustment without a post-maintenance stroke time measurement (or leakage test). Confirmatory testing must be performed at the first available opportunity when plant conditions allow testing. This guidance applies only to valves that need adjustment during power operation and cannot be fully stroked in the plant operating mode. The guidance does not apply merely as a convenience to the licensee and does not supersede any related guidance associated with GL 89-10. IN 87-40, “Back Seating Valves Routinely to Prevent Packing Leakage,” provides information related to backseating valves. Both Westinghouse and General Electric issued guidance on performing backseating to minimize deformation to valve stems. Backseating is not listed as an example of a maintenance activity in OM Code, paragraph ISTC-3310. The licensee would have to assess the effect of backseating on valve operation and determine whether post-maintenance testing is required. Test results for MOV programs to address GL 89-10 and GL 96-05 may indicate whether backseating of a particular valve affects its stroke time. Any information would need to be included and documented in an evaluation, and the assessments would have to be valve-specific.

4.4.3 Manual Valves

The OM Code includes IST requirements for manual valves that meet its scope requirements. To comply with the OM Code, manual valves must be included in the IST program and tested in accordance with applicable OM Code requirements if they are required to perform a specific function in shutting down a reactor to the safe shutdown condition, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident. Applicable tests could include exercising, leak testing, and/or position indication verification, at the frequency specified in the OM Code. For some valves, no tests are specified depending on their active/passive classification and their performance and design attributes. For example, a passive manual valve with no position indication has no tests specified. However, it must be considered and listed in the IST program.

4.4.3.1 Manual Valve Exercise Interval

The 1998 Edition and earlier versions of the OM Code specified an exercise interval of 3 months for manual valves within the scope of the OM Code. In the 1999 Addenda to the OM Code, ASME revised paragraph ISTC- 3540 to extend the exercise frequency for manual valves to 5 years. The NRC staff did not agree that there was sufficient justification to extend the exercise interval for manual valves to 5 years. At that time, the NRC staff reviewed licensees' IST programs to determine the performance of manual valves that could only be exercised at cold shutdown or refueling outages. Based on its review, the NRC staff determined that a 2-year interval is justified for exercising manual valves. On September 26, 2002 in *Federal Register* (67 FR 60520), the NRC revised 10 CFR 50.55a(b)(3)(vi) to specify that manual valves shall be exercised on a 2-year interval. Nonetheless, licensees are not prohibited from exercising manual valves more frequently than every 2 years. In the 2006 Addenda to the OM Code, ASME revised ISTC-3540 to change the exercise frequency for manual valves to 2 years. The 2-year interval requirement for exercising manual valves continues up to and including the 2017 Edition of the OM Code.

4.4.3.2 Power-Operated Valves with Manual Valve Capability

Some power-operated valves (POVs) have manual valve capability. For those POVs, the valve can be manipulated to its desired position manually in the event of a loss of power or operating force (such as caused by motor failure as applicable). POVs that are relied upon to change position to complete their safety function in the absence of power or operating force as part of their response to design-basis events are expected to be treated and tested to the requirements of a manual valve in addition to the POV requirements in the OM Code. Although IST program requirements might not apply, POVs that are relied upon for manual capability to perform functions during beyond design-basis events are expected to satisfy the applicable requirements for those events.

4.4.4 Pressure Isolation Valves

Pressure isolation valves (PIVs) are defined as two normally closed valves in series that isolate the reactor coolant system (RCS) from an attached low-pressure system. PIVs are located at all RCS/low-pressure system interfaces. As such, PIVs are located within the reactor coolant pressure boundary (RCPB), which is defined in 10 CFR 50.2.

"Event V" PIVs are defined as two check valves in series at an RCS/low-pressure system interface, which may result in a LOCA that bypasses containment if they fail. The "Event V" PIVs comprise a subset of PIVs. "Event V" refers to the scenario described for this event in the "Reactor Safety Study" (WASH-1400).

On April 20, 1981, the NRC issued Orders to 32 PWRs and 2 BWRs, which required the specified licensees to conduct leak rate testing of their PIVs, based on plant-specific NRC-supplied lists of PIVs, and required the licensees to modify their TS accordingly. These Orders are known as the "Event V Orders," and the valves listed therein are the "Event V" PIVs.

Currently, the majority of operating plants operate using NUREG-1430 to 1434, Standard Technical Specifications (STS), that do not contain a listing of PIVs or Event V PIVs. Therefore, the NRC staff recommends that licensees should include a listing of PIVs (including Event V PIVs) in their 10-year IST programs to document IST testing requirements for each PIV. Licensees should also review their testing procedures to ensure that the PIVs are individually leak rate tested. (This position supersedes Position 4 of GL 89-04, because the improved STS no longer contain PIV listings.)

4.4.4.1 PIV Discussion in Generic Letter 87-06

GL 87-06 supersedes Position 4 of GL 89-04, because the STS do not contain PIV listings. The NRC staff used licensees' responses to GL 87-06 as input for the resolution of Generic Issue 105, "Interfacing Systems LOCAs at Light-Water Reactors," which was evaluated by the NRC's Office of Nuclear Regulatory Research. The results of studies of interfacing system LOCAs are provided in NUREG/CR-5124, "Interfacing Systems LOCA: Boiling-Water Reactors," and NUREG/CR-5102, "Interfacing Systems LOCA: Pressurized-Water Reactors." Generic Issue 105, which included the issue discussed in GL 87-06, was closed by memorandum from E. Beckjord to J. Taylor, "Technical Resolution of Generic Issue 105 (GI-105), 'Interfacing Systems Loss-of-Coolant Accident (ISLOCA) in LWRs,'" dated June 3, 1993. PIVs need to be included in and tested by the IST programs if they are not included as part of a licensee's TS.

4.4.4.2 Leak Rate Testing of PIVs

The leak rate testing specified in a plant's TS must meet the intent of paragraph ISTC-3600 of the OM Code, Subsection ISTC. A licensee must ensure that each PIV is individually leak tested (or that the measured leakage is adjusted) in accordance with the differential pressure requirements of the OM Code. If the TS are not sufficiently detailed to ensure individual valve leak testing, the licensee is responsible to ensure that the test procedures are themselves adequate for valves and valve combination leak testing.

NRC Recommendation

A licensee may consider the leakage testing performed to meet TS requirements to also meet IST requirements if the intent of the OM Code is met (e.g., leakage limits are established, corrective actions are taken as required, and valves are individually leak tested). However, a licensee must ensure that the test differential pressure specified in the TS, if applicable, is equivalent to the function maximum pressure differential, or that the measured leakage is adjusted to the function maximum pressure differential in accordance with the formula in paragraph ISTC-3630 of the OM Code.

Basis for Recommendation

Increasing pressure usually improves the seating of a valve. The OM Code allows that when leak testing those types of valves in which the service pressure will tend to diminish the overall leakage channel opening, as by pressing the disk into or onto the seat with greater force, the test differential pressure may be lower than the function maximum differential pressure.

The resulting leakage is to be adjusted according to the following formula from the subparagraph ISTC-3630(b)(4) of the OM Code:

$$\frac{L\,(maximum)}{L(test)} = \sqrt[2]{\frac{dP(maximum)}{dP(test)}}$$

where:

L = leakage

dP = differential pressure

While the NRC staff has accepted other aspects of the TS, the licensee must ensure that any testing requirements that are not specifically detailed in the TS are, nonetheless, imposed on the PIVs to comply with the OM Code leakage testing requirements. Generally, the same test will be used to meet both TS and IST requirements. The major difference between TS and IST requirements are related to the acceptance criteria specified in some TS between a nominal leakage limit and the upper leakage limit. (If allowed by TS, the upper leakage limit is considered acceptable as the acceptance criteria for IST.)

If the list of PIVs is removed from the TS, the leakage testing must be described in detail in the SAR or must be identified as in accordance with the requirements of the OM Code.

4.4.5 Containment Isolation Valves That Have Other Leak-Tight Safety Functions

Valves that function as containment isolation valves (CIVs) may have additional safety functions (i.e., other than isolation), such as pressure isolation, train separation, or preventing diversion of flow. The leakage testing for 10 CFR Part 50, Appendix J, might not adequately test these additional functions based on the pressure or fluid medium. For such valves, the requirements of both Appendix J and OM Code, Subsection ISTC, paragraph ISTC-3600 apply.

4.4.6 Testing Individual Scram Valves for Control Rods in Boiling-Water Reactors

BWRs are equipped with bottom-entry hydraulically driven control rod drive mechanisms with high-pressure water providing the hydraulic power. Each control rod is operated by a hydraulic control unit (HCU), which consists of valves and an accumulator. The HCU is supplied charging and cooling water from the control rod drive pumps, and the control rod operating cylinder exhausts to the scram discharge volume. Various valves in the control rod drive system perform an active function in scramming the control rods to rapidly shut down the reactor.

The NRC staff believes that those valves that must change position to provide the scram function should be included in the IST program and should be tested in accordance with the requirements of the OM Code, Subsection ISTC, except where relief has been granted in a safety evaluation report. Bidirectional exercise testing of check valves is required by the 1996 Addenda to the OM Code (and later editions and addenda).

The control rod drive system valves that perform an active safety function in scramming the reactor are the scram discharge volume vent and drain valves, scram inlet and outlet valves, scram discharge header check valves, charging water header check valves, and cooling water header check valves. With the exception of the scram discharge volume vent and drain valves, exercising the other valves quarterly during power operations could result in rapid insertion of one or more control rods. If practical, licensees should test control rod drive system valves at the OM Code-specified frequency. However, for those control rod drive system valves for which testing could result in rapid insertion of one or more control rods, the rod scram test frequency identified in the facility's TS may be used as the valve testing frequency to minimize rapid reactivity transients and wear of the control rod drive mechanisms. This alternative test frequency which is a deviation from the OM Code requirement should be clearly stated and documented in the IST program document, and this alternative or relief requires NRC approval in accordance with 10 CFR 50.55a.

Industry experience has shown that normal control rod motion may verify the cooling water header check valve moving to its safety function position. This can be demonstrated because rod motion may not occur if this check valve were to fail in the open position. If this test method is used at the OM Code-required frequency, the licensee should clearly explain in the IST program document that this is how these valves are being verified to close quarterly.

Closure verification of the charging water header check valves requires that the control rod drive pumps must be stopped to depressurize the charging water header. This test should not be performed during power operation because stopping the pumps results in a loss of cooling water to all control rod drive mechanisms, and seal damage could result. Additionally, this test cannot be performed during each cold shutdown because the control rod drive pumps supply seal water to the reactor recirculation pumps, and one of the recirculation pumps is usually kept running. Therefore, the HCU accumulator pressure decay test, as identified in the facility's TS may be used as the charging water header check valve alternative testing frequency for the reasons stated above. If this test is not addressed in the licensee's TS, this closure verification should be performed at least during each refueling outage, and this alternative test frequency which is a deviation from the OM Code requirement should be specifically addressed in the IST program document, and this alternative or relief requires NRC approval in accordance with 10 CFR 50.55a.

The scram inlet and outlet valves are power-operated valves that full-stroke in milliseconds and are not equipped with indications for both positions; therefore, it may be impractical to measure their full-stroke time as required by the OM Code. Verifying that the associated control rod meets the scram insertion time limits defined in the plant's TS can be an acceptable alternative method of detecting degradation of these valves. Also, it may be impractical and unnecessary to trend the stroke times of these valves because they are indirectly stroke timed, and no meaningful correlation may be drawn between the scram time and valve stroke time. Furthermore, conservative limits are placed on the control rod scram insertion times. If the above test is used to verify the operability of scram inlet and outlet valves, it should be specifically documented in the IST program document, and this alternative or relief requires NRC approval in accordance with 10 CFR 50.55a, as discussed above.

4.4.7 Use of Appendix J, Option B, in Conjunction with ISTC Exercising Tests for Check Valves

The OM Code, Subsection ISTC, paragraph ISTC-3522, requires licensees to exercise Category C valves every 3 months. Paragraph ISTC-3620, "Containment Isolation Valves," also requires licensees to seat leak test those Category A valves in accordance with Appendix J to 10 CFR Part 50. Specifically, Option B of Appendix J allows a variable seat leak testing frequency, based on component performance, and allows test intervals for valves with acceptable performance to be extended to once every three refueling outages. Therefore, for Category A/C valves, the OM Code requires two independent tests, including an exercising test and a seat leakage rate test.

The OM Code recognizes that when more than one distinguishing category characteristic applies, all requirements for each of the individual categories apply, although duplication or repetition of common testing requirements is not necessary. Therefore, a seat leak rate test is one acceptable method to verify the closure portion of an exercise test.

OM Code, Appendix II, "Check Valve Condition Monitoring Program," allows an alternative to the exercising testing requirements in the OM Code. The 1996 Addenda to the OM Code included two significant changes to IST of check valves to (1) correct certain anomalies in the way check valves were being exercised, and (2) codify a process for monitoring the valve's operating condition and performance. This integral two-part improvement to the OM Code provides interrelated requirements. Beginning with the 1996 Addenda, ASME modified Subsection ISTC 4.5.2, "Exercising Requirements," and Subsection ISTC 4.5.4, "Valve Obturator Movement," to require bidirectional testing to improve the detection of valve degradation and failure. The related change to Subsection ISTC 4.5.5, "Condition Monitoring Program," allowed the use of a condition monitoring process as an alternative to the exercising and testing requirements of Subsections ISTC 4.5.1 to 4.5.4. More recent OM Code editions include similar requirements in paragraphs ISTC-3520, "Exercising Requirements," and ISTC-3530, "Valve Obturator Movements," to continue bidirectional check valve testing. The condition monitoring process, defined in Appendix II, "Check Valve Condition Monitoring Program," allows licensees certain IST flexibility in establishing the types of test, examination, and preventive maintenance activities and their associated intervals, when justified based on the valve's performance and operating condition.

NRC Recommendation

The use of Appendix II of the OM Code for a condition monitoring program for check valves, with the regulatory conditions in 10 CFR 50.55a, provides the licensee with knowledge of the valve's operating condition, informed and verified expectations of the valve's performance over extended intervals, and a process to justify prudent IST interval extensions to reduce the burden of unnecessary IST. Therefore, the NRC staff recommends that licensees implement a condition monitoring program for check valves to justify extending the exercise test interval to the leak test frequencies specified in Option B of Appendix J.

Basis for Recommendation

The use of Appendix II of the OM Code as incorporated by reference in 10 CFR 50.55a, provides licensees with knowledge of the valve's operating condition, monitors and verifies valve performance over extended intervals, and provides a process to justify prudent IST interval extensions to reduce the burden of unnecessary IST.

4.4.8 Pyrotechnic-Actuated Valves in New Reactors

Some designs for new nuclear power plants include pyrotechnic-actuated (squib) valves that have more safety significance than squib valves in currently operating nuclear power plants. In addition, squib valves for new reactors might have different designs and be much larger than squib valves used in current plants. The OM Code in the 2012 through 2017 Editions has included provisions for PST and IST of squib valves in new reactors. The NRC has incorporated by reference those editions of the OM Code in 10 CFR 50.55a without conditions for squib valve testing.

To supplement 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 VEGP Units 3 and 4. 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 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, and 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 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 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 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 OM Code for explosively actuated valves, and sets forth requirements for both preservice 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 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 OM Code requirements.

The NRC staff considers the PST and IST provisions for squib valves in new reactors specified in the 2012 through 2017 Editions of the OM Code to incorporate the provisions in the license conditions in the COLs for VEGP Units 3 and 4. Applicants and licensees for new reactors will need to address the PST and IST provisions for squib valves consistent with the applicable regulatory requirements, license conditions, and FSAR provisions.

5 SUPPLEMENTAL GUIDANCE ON INSERVICE TESTING OF PUMPS

5.1 General Pump Inservice Testing Issues

In 1995, OM Code Subsection ISTB introduced a new approach to pump testing, in which pumps are divided into two basic groups with an enhanced baseline or preservice and three periodic tests (i.e., Group A, Group B, and Comprehensive). This modified pump testing program is commonly referred to as the Comprehensive Pump Test (CPT). The pump grouping criterion of Subsection ISTB is based on the way the pumps are operated at the plant.

The CPT allows less-rigorous pump testing to be performed for certain pumps on a quarterly frequency while requiring a pump test to be performed with more accurate pressure and differential pressure instrumentation every 2 years at ±20 percent of pump design flow. In the 2012 Edition of the OM Code, the CPT flow rate was changed. This CPT flow rate is defined as the flow rate established by the Owner that is effective for detecting mechanical and hydraulic degradation during subsequent testing. The best efficiency point, system flow rates, and any other plant specific flow rates are considered when determining this flow rate. The CPT was developed with the knowledge that some pumps, such as containment spray pumps, cannot be tested at the required high flow rates because of system design limitations. Subparagraph ISTB-3300(e)(1) of the OM Code requires licensees to establish reference values at the CPT flow rate for the CPT test.

ASME prepared Appendix V, “Pump Periodic Verification Test Program,” to the OM Code to provide justification for the relaxation of the upper end of the “Acceptable Range” and the “Required Action Range” for flow and differential or discharge pressure for comprehensive pump testing in Subsection ISTB. The NRC staff on the OM Code committee accepted these changes provided a licensee adds a pump periodic verification test program in the OM Code. The NRC included conditions in 10 CFR 50.55a(b)(3) to provide assurance that Appendix V was applied to Subsections ISTB and ISTF in specific editions and addenda of the OM Code.

5.1.1 Categories of Pumps for Inservice Testing

The OM Code (2017 Edition) requires that all pumps that the licensee identifies as part of an IST program must be categorized as either Group A or Group B pumps. Section ISTB-2000 defines Group A pumps as “pumps that are operated continuously or routinely during normal operation, cold shutdown, or refueling operation.” By contrast, ISTB-2000 defines Group B pumps as “pumps in standby systems that are not operated routinely except for testing.”

5.1.2 Testing Requirements and Frequency of Inservice Tests

The OM Code identifies various types of tests, such as Preservice, Group A, Group B, and Comprehensive tests. All pumps receive a Preservice test followed, on a quarterly basis, by the test associated with the pump category (Group A test for Group A pumps, and Group B test for Group B pumps), and at least once every 2 years by a CPT. A CPT may also be substituted for a Group A or Group B test. Similarly, a Group A test may be substituted for a Group B test, and a Preservice test may be substituted for any inservice test.

OM Code, ISTB-3410, “Pumps in Regular Use,” states:

> Group A pumps that are operated more frequently than every 3 months need not be run or stopped for a special test, provided the plant records show the pump was operated at least once every 3 months at the reference conditions, and the quantities specified were determined, recorded, and analyzed per section ISTB-6000.

OM Code, ISTB-3420, “Pumps in Systems Out of Service,” states:

> For a pump in a system declared inoperable or not required to be operable, the test schedule need not be followed. Within 3 months before the system is placed in an operable status, the pump shall be tested, and the test schedule followed in accordance with the requirements of this Subsection. Pumps that can only be tested during plant operation shall be tested within 1 week following plant startup.

OM Code, ISTB-3430, “Pumps Lacking Required Fluid Inventory,” states:

> Group B pumps lacking required fluid inventory (e.g., pumps in dry sumps) shall receive a comprehensive test at least once every 2 yr. except as provided in para. ISTB-3420. The required fluid inventory shall be provided during this test. A Group B test is not required.

NRC Recommendation

In 1995, OM Code, Subsection ISTB, introduced the new CPT, which allows licensees to perform less-rigorous pump testing for certain pumps on a quarterly frequency, while requiring licensees to perform a pump test with more accurate pressure/differential pressure instrumentation every 2 years at ±20 percent of pump design flow. The 2012 Edition of the OM Code changed the CPT flow rate. The CPT flow rate is established by the Owner to be effective for detecting mechanical and hydraulic degradation during subsequent testing. This section also discusses previously issued guidance and experience.

5.2 Use of Variable Reference Values for Flow Rate and Differential Pressure During Pump Testing

Some system designs do not allow for testing at a single reference point or a set of reference points. In such cases, it may be necessary to plot pump curves to use as the basis for variable reference points. Consequently, the OM Code Committee prepared Code Case OMN-9, Revision 0, “Use of a Pump Curve for Testing,” which the NRC staff subsequently included in RG 1.192. In particular, the NRC staff accepted Code Case OMN-9 with the conditions that (1) when the repair, replacement, or routine servicing of a pump may have affected a reference curve, the licensee must determine a new reference curve, or reconfirm an existing reference curve, in accordance with Section 3 of Code Case OMN-9; and (2) if it is necessary or desirable, for some reason other than that stated in Section 4 of Code Case OMN-9, to establish an additional reference curve or set of curves, the licensee must determine the new curves in accordance with Section 3 of Code Case OMN-9. The use of OMN-9 requires an alternative request because OMN-9 is only applicable to the OM Code-1990 through OMb Code 1992. It is not applicable to 1995 or later Code editions. In the OMb-2006 addenda, the ASME OM Code committee published a new Code Case OMN-16, “Use of a Pump Curve for Testing,” to replace Code Case OMN-9. Code Case OMN-16 incorporates all the NRC conditions imposed for the use of Code Case OMN-9. Code Case OMN-16 is endorsed in RG 1.192 and no alternative request is required for its use. Note that Figure 1 was inadvertently omitted from OMN-16, Revision 1, in the 2012 Edition of the OM Code. Therefore, Code Case OMN-16, Revision 1, is approved for use provided it is supplemented with Figure 1 of OMN-16 that is in the 2006 Addenda of the OM Code. OMN-16, Revision 2, which includes Figure 1, is endorsed in RG 1.192, Revision 3, and no alternative request is required to use OMN-16, Revision 2.

5.2.1 Reference Values

Licensees shall determine reference values from the results of preservice testing or the first inservice test. The resultant reference values shall be at points of operation that are readily duplicated during subsequent tests, and the licensee shall compare all subsequent test results to the initial reference values or the new reference values established in accordance with the OM Code. Licensees shall only establish reference values when the pump is known to be operating acceptably. If the particular parameter being measured or determined can be significantly influenced by other related conditions, these conditions shall be analyzed.

5.2.2 Reference Curves

If the establishment of specific reference values is impractical for a centrifugal or vertical line shaft pump, the licensee may establish reference curves. In so doing, the licensee needs to determine the reference curves from the data measured during preservice testing or the first inservice test. In addition, the licensee needs to establish a reference curve from a minimum of five data points for each 20 percent of the maximum pump curve range, and the range of the reference curve needs to be sufficient to bound the points of operation that are expected during subsequent tests. The licensee then compares all results to the initial reference curves or the new values established in accordance with Sections 5.2.3 and 5.2.4, below. In addition, the licensee only establishes reference curves when the pump is known to be operating acceptably. If vibration is relatively unaffected by changing differential pressure or flow over the reference curve range, the licensee may use a single reference value as the test quantity, provided it is at the minimum of the measured data. By contrast, if the licensee uses reference curves, the record of the test documents and justifies the reasons for doing so and the suitability of the methods used to develop the reference curves and acceptance criteria (See section ISTB-9000).

5.2.3 Effect of Pump Replacement, Repair, and Maintenance on Reference Values or Reference Curves

When the repair, replacement, or routine servicing of the pump may have affected a reference value, a set of reference values, or a reference curve, the licensee determines a new reference value, set of reference values, or a new reference curve, or reconfirms the previous value (or curve) by an inservice test run before declaring the pump operable. The licensee then identifies any deviation between the previous and new set of reference values (or reference curves), and the record of the tests documents the verification that the new values (or curves) represent acceptable pump operations (see section ISTB-9000).

5.2.4 Establishment of Additional Sets of Reference Values or Reference Curves

If it is necessary or desirable, for some reason other than discussed above, to establish an additional set of reference values or curves, the licensee runs an inservice test under the conditions of an existing set of reference values, or within the range of existing reference curves, and analyzes the results. If the operation is acceptable in accordance with Section 7 of Code Case OMN-9 (Section 16-6200 of OM Code Case OMN-16), a second test run under the new reference conditions needs to follow as soon as practicable, and the results of this test establish the additional set of reference values or reference curves. Whenever a licensee establishes an additional set of reference values or reference curves, the record of the tests documents and justifies the rationale for doing so (See section ISTB-9000).

NRC Recommendation

The NRC accepts the use of pump curves for reference values of flow rate and differential pressure if the licensee clearly demonstrates, in a relief request, that it would be impractical to establish a fixed set of reference values. A relief request must include a description of the methodology to be used in evaluating these pumps. To obtain approval for a proposed method of evaluating these pump parameters to detect hydraulic degradation and determine pump operability, the licensee must demonstrate that the acceptance criteria are equivalent to the Code requirements as specified under test acceptance criteria in ISTB Table ISTB-5121-1 or ISTB-5221-1 (depending on pump type), for allowable ranges using reference values and curves.

To use this test method, the licensee must plot a valid pump characteristic curve from empirical data or obtain one from the pump manufacturer and verify it with measurements taken when the pump was known to be in good operating condition. Additional guidance is given in Sections 5.2.2, 5.2.3, and 5.2.4 above; the OM Code; and Code Case OMN-16 including RG 1.192.

Basis for Recommendation

Where it is not practical to return to the same flow configuration for each subsequent inservice pump test, the licensee must establish a method for evaluating the operational readiness of pumps in variable flow systems. This may be the case for service water or component cooling water systems and other systems where temperature or flow is controlled at a variety of locations. During quarterly pump testing, the licensee may not be able to manually control each of these local stations and duplicate the overall system reference conditions, as required by the OM Code.

Using the manufacturer's pump-specific curves for flow and differential pressure, the licensee may be able to evaluate the pump in as-found system conditions. In implementing this guidance, the licensee would confirm these values by performing in situ testing. Another method would be to plot pump curves over the range of conditions expected during the system's normal operation. It is also important to develop a method of evaluating pump vibration measurements taken with the pump operating over the range of possible as-found conditions, since this is a variable pump parameter. By evaluating these measurements of pump vibration, the licensee will ensure that a pump that is severely degraded, either hydraulically or mechanically, is declared inoperable and appropriate action is taken to address the degradation.

5.3 Allowable Variance from Reference Points and Fixed-Resistance Systems

Certain designs do not allow for the licensee to set the flow at an exact value because of limitations in the instruments and controls for maintaining steady flow. The characteristics of piping systems in other designs do not allow for the licensee to adjust the flow to exact values. The OM Code does not allow for variance from a fixed reference value, stating only that "the resistance of the system shall be varied until either the measured differential pressure or the measured flow rate equals the corresponding reference value." Licensees have requested relief to establish a range of values similar to using a pump curve, but with a very narrow band. For example, one licensee proposed to use a reference curve with the tolerance around the selected value of flow to be ±2 percent. Plant implementing procedures may instruct operators to set the flow to 1,500 gallons per minute (gpm). When this step is performed, the operator would attempt to set the flow as close as possible to 1,500 gpm and maintain it steadily at approximately 1,500 gpm.

NRC Recommendation

The NRC staff has determined that, if the design does not allow for establishing and maintaining flow at an exact value, achieving a steady flow rate or differential pressure at approximately the set value does not require relief for establishing pump curves. The allowed tolerance for setting the fixed parameter must be established for each case individually, including the accuracy of the instrument and the precision of its display. This will necessitate verification of the effect of precision on accuracy as considered in the design of the instrument gauge. For Group A and Group B tests, a total tolerance of ±2 percent of the reference value (including instrument accuracy) is allowed without prior NRC approval; for Preservice tests and CPTs, the allowable total tolerance is ±1/2 percent (including instrument accuracy) for pressure and differential pressure, ±2 percent (including instrument accuracy) for flow. For a tolerance greater than the allowed percent (which may be necessary depending on the precision of the instrument), the licensee may make a corresponding adjustment to acceptance criteria to compensate for the uncertainty, or may perform and document an evaluation to justify a greater tolerance. In using this guidance, the IST program document or implementing procedures must document the variance and the method for establishing it.

The intent is that the variance in the reference value setting may be ±2 percent for Group A and Group B tests and ±1/2 percent for Preservice tests and CPTs, ±1/2 for pressure and differential pressure, and ±2 percent for flow without requiring relief. Nonetheless, any variance in the setting will have an impact on test results.

Basis for Recommendation

The OM Code does not address the likelihood that it may not be possible to control a flow rate or differential pressure to an exact value. When the OM Code specifies that the system resistance must be varied until either the flow or differential pressure equals the corresponding reference value, it does not intend the set value to have an acceptable range as stated in the ISTB test acceptance criteria, including ISTB Tables ISTB-5121-1 and ISTB-5221-1. The acceptance criteria apply only to the parameter being determined after the resistance is varied. Licensees should recognize that the reference value for certain pumps can only be achieved within a specified tolerance. Licensees may set the repeatable parameter as close as possible to the reference value during each test, rather than treating any variance in the value with a pump curve. If, upon establishing trends in data, the licensee determines that the parameter varies such that the readings are outside the accuracy of the instrument, the licensee may need to establish pump curves and propose an alternative to the OM Code requirements for the applicable pumps. (See Section 5.2)

Paragraph ISTB-3500 specifies the requirements for instrument fluctuations and describes the basis for allowing a variance from the reference value of ±2 percent for Group A and Group B tests, and ±1/2 percent for pressure and differential pressure, ±2 percent for flow for Preservice tests and CPTs. In addition, paragraph ISTB-3500 allows the use of symmetrical damping devices or averaging techniques to reduce instrument fluctuations to within ±2 percent or ±1/2 percent (as applicable) of the observed reading for values specified in the implementing procedures. Greater variances must be justified and acceptance criteria adjusted as necessary.

If an analog gauge is used, the required accuracy is percent full scale. For a digital gauge, the required accuracy is over the calibrated range. For a combination of gauges, the required accuracy is loop accuracy.

ASME developed Code Case OMN-21, “Alternate Requirements for Adjusting Hydraulic Parameters to Specified Reference Points,” to address this issue as follows:

The allowed tolerance for setting the fixed parameter must be established for each case individually, including evaluation of throttling capability. Licensees should consider improvements in throttling methods where system control is especially poor. A total throttling tolerance of + 2 / -1 percent of the flow rate reference value is considered as meeting the requirements of the OM Code sections.

For a tolerance greater than + 2 / -1 percent of the flow rate reference value, a corresponding adjustment to acceptance criteria shall be made to compensate for the uncertainty, or an evaluation would be performed and documented justifying a greater tolerance. The variance and the method for establishing the variance must be documented in the IST program documents or implementing procedures.

Code Case OMN-21 is endorsed in RG 1.192, Revision 3, with no conditions. Code Case OMN-21 states that it is applicable to the 1995 through 2011 Addenda of the OM Code. Therefore, a relief request is required to use Code Case OMN-21 if the licensee’s Code of Record is later than the 2011 Addenda of the OM Code.

The basis for the Code change is as follows:

The OM Code does not address the possibility that a flow rate or differential pressure may not be controllable to an exact value. When the OM Code specifies that the system resistance be varied until either the flow or differential pressure equals the corresponding reference value, it does not literally intend that the “set value” be precisely attained without any fluctuations. Licensees recognize that the reference value for certain pumps can only be achieved within a specified tolerance. Licensees shall attempt to set the repeatable parameter as close as possible to the reference value during each test.

The basis for allowing a variance of + 2 / -1 percent from the flow rate reference value deals with instrument fluctuations and system stability issues. The OM Code allows symmetrical damping devices or averaging techniques to be used to reduce instrument fluctuations to within 2 percent of the observed reading for values specified in the implementing procedures. Greater variances must be justified and acceptance criteria adjustments made as necessary. The limitation of 1 percent in the negative direction reduces the non-conservative impact on the variable parameter. The total 3 percent allowable variance provides for a reasonable throttling control range while minimizing the impact on trendability of the variable parameter.

Licensees should ensure that performance trending of pumps is capable of detecting degradation as early as possible. Larger variances in the reference parameter will induce scatter in the variable parameter data. Techniques such as data normalization, where recorded test data is corrected by the known pressure to flow relationship, should be used when necessary to provide for accurate short-term trending.

5.4 Monitoring Pump Vibration in Accordance with ISTB

Subsection ISTB of the OM Code allows licensees to monitor pump vibration in units of either pump displacement (peak-to-peak) or pump velocity (peak), and includes acceptance criteria for both units of measurement. As specified in Table ISTB-3000-1, the measurement of pump vibration is required for Preservice tests, Group A tests, and CPTs. However, the OM Code does not require vibration measurements for Group B tests.

NRC Recommendation

The NRC staff has determined that if the licensee uses Subsection ISTB of the OM Code as the basis for monitoring vibration in the IST program, the program must include all of the requirements for such monitoring. Licensees may update their programs in accordance with this position without further relief if they meet all related requirements for monitoring vibration in paragraph ISTB-3540, and sections ISTB-5000 and ISTB-6000. However, Commission approval to use a later OM Code edition or addenda is still required pursuant to 10 CFR 50.55a(f)(4)(iv). See Sections 1.1 and 2.1 for further guidance.

In following this guidance, the frequency response range of the instrumentation must be as specified in subparagraph ISTB-3510(e) for both low-speed and high-speed pumps, unless the licensee demonstrates that the information gained at the low-frequency response does not apply for the bearing design of the pumps. In that event, the licensee must still provide an acceptable alternative to the required testing. Although the instruments in the low-frequency response ranges may not be widely used, the unavailability of instruments does not constitute sufficient justification for either obtaining relief from the frequency response range requirements of Subsection ISTB, or obtaining authorization of an alternative to the requirements.

Basis for NRC Recommendation

As shown in the OM Code, Figure ISTB-5223-1, “Vibration Limits,” licensees may choose to use units of velocity, rather than displacement, in measuring vibration in pumps that operate above 600 revolutions per minute (rpm). Such an approach would enable the licensee to more rapidly detect wear in the anti-friction bearing and other types of pump degradation and, thus, to perform repairs in a more timely manner.

Pump bearing degradation results in increased vibration at frequencies of 5 to 100 times the rotational speed of the pump. These high-frequency bearing vibrations may not significantly increase the measured displacement of pump vibration and could go undetected. However, the high-frequency vibrations would significantly increase the measured velocity of pump vibration, which could indicate the need for corrective action before the bearing fails. Because pump bearings vibrate at high frequencies, the measured vibration velocity indicates the mechanical condition of the pumps and reveals pump bearing degradation much more accurately than does measured vibration displacement.

Advantages of measuring vibration velocity, rather than displacement, to monitor the mechanical condition of pumps (with the exception of low-speed pumps) are widely acknowledged in the industry. Many licensees measure pump vibration velocity to detect pump degradation and obtain advance warning of incipient pump bearing failure. Upon obtaining this advance warning, the licensee can plan and prepare for maintenance during scheduled outages instead of suffering losses resulting from unplanned outages to repair failed critical equipment. Subsection ISTB of the OM Code includes a set of allowable ranges for inservice pump vibration velocity and measured pump vibration displacement. These ranges are based on an evaluation of empirical data and various acceptance criteria for pump vibration velocity established by U.S. industries, academia, international industry, and foreign agencies.

The minimum frequency response range requirement is established from one-third of the minimum pump shaft rotational speed to at least 1000 Hz in order to encompass all noise contributors that could indicate degradation. Instruments with a frequency response range that meets these requirements for slow-speed pumps may not be widely used. However, the unavailability of instruments, alone, does not constitute adequate justification for obtaining relief or authorization of an alternative; however, it may be a significant element in the justification. The NRC has observed that, because of technology advancement and research in the field of instrumentation, vibration measuring transducers meeting the OM Code requirements can now be procured from various suppliers at reasonable costs. Therefore, licensee to provide additional justification for alternative request related to instruments. Additionally, frequencies less than running speed may not be indicative of problems for certain types of bearings; however, subharmonic frequencies may be indicative of rotor rub, seal rub, loose seals, or coupling damage. The type of bearings and other subharmonic concerns would typically be discussed in the justification for relief.

Subsection ISTB of the OM Code requires licensees to measure vibration in units of either pump displacement (peak-to-peak) or pump velocity (peak). Digital equipment can measure directly in peak units. The 10-year update of the ISI and IST programs reflects the need for licensees to incorporate new technologies that have been incorporated into the codes and standards. The OM Code Committee responded to an inquiry (Interpretation 95-4, File OMI 94-2) explaining that the intent of the OM Code is to allow vibration to be measured in root mean square (rms) and mathematically converted to peak readings. Licensees are cautioned that the OM Code vibration acceptance criteria are in peak or peak-to-peak units, and the use of rms is not acceptable without a mathematical conversion. To comply with the requirements, licensees that use rms values for recording data must adjust the limits of Subsection ISTB, or convert the data to peak values.

Several plants have requested an alternative to the vibration acceptance criteria of Subsection ISTB for smooth-running pumps, and the NRC has authorized such requests. However, licensees with such approval must continue to assess the vibration data and monitor increases that may be indicative of a change. In one reported incident, a pump with very low vibration experienced an increase in vibration levels over three successive tests, although the levels remained below the criteria for smooth-running pumps. Upon investigating the cause of the increase, the licensee determined that the bearing had degraded and required replacement.

5.5 Pump Flow Rate and Differential Pressure Instruments

The NRC has received requests for alternative to continue using instruments that do not meet either the range or accuracy requirements of the OM Code. The OM Code requires each analog instrument to have a full-scale range that is three times the reference value or less, while each digital instrument must be such that the reference values do not exceed 70 percent of the calibrated range of the instrument. The NRC has accepted Code Case OMN-6, “Alternate Rules for Digital Instruments,” as specified in RG 1.192, which allows each digital instrument to be such that the reference values do not exceed 90 percent of the calibrated range of the instrument. Code Case OMN-6 has been incorporated into the 2012 through 2017 Editions of the OM Code. For Group A and Group B pumps, paragraph ISTB-3510 requires an accuracy of ±2 percent of full-scale for analog instruments, ±2 percent of total loop accuracy for a combination of instruments, or ±2 percent of reading over the calibrated range for digital instruments. For Preservice tests and CPTs, the required instrument accuracy is ±1/2 percent for pressure and differential pressure instruments.

5.5.1 Range and Accuracy of Analog Instruments

NRC Recommendation

When the range of a permanently installed analog instrument is greater than three times the reference value, but the accuracy of the instrument is more conservative than that required by the OM Code, the NRC staff may grant relief or authorize an alternative when the combination of the range and accuracy yields a reading that is at least equivalent to that achieved using instruments that meet the OM Code requirements (i.e., up to ±6 percent for Group A and B tests, and ±1.5 percent for pressure and differential pressure instruments for Preservice tests and CPTs). The use of a test gauge (in lieu of a permanent instrument) is acceptable if the reading is at least equivalent to that required by the OM Code. When using temporary instruments, the NRC staff recommends that the licensee's IST records should include an instrument number for use in tracing each instrument and a calibration data sheet for use in verifying that the instruments are accurately calibrated. The licensee need not obtain relief if the temporary instruments meet the range and accuracy requirements of the OM Code. If relief is requested, the licensee would typically describe the effect on each group of applicable pumps and would typically discuss adjustment of acceptance limits to account for the inaccuracies.

Basis for Recommendation

Because the IST requirements originally specified an instrument range of 4 times the reference values or less, the permanent instruments in many early-licensed plants do not meet the current requirements of the OM Code for an instrument range of three times the reference values or less. The NRC does not generally consider instrument installation or replacement an undue burden, and compliance with the instrument requirements in later editions of the OM Code does not constitute a backfit.

This position applies to the early-licensed plants, but not for the purchase of replacement instruments that can be procured to meet the current requirements of the OM Code; therefore, for new instrument installations, licensees must meet the accuracy and range requirements (although the OM Code does not prohibit like-for-like instrumentation for the existing installation).

The licensee is not relieved of its responsibility to make modifications to comply with changes to IST provisions as a result of changes to the OM Code. Instrument modifications are considered practical in the context of 10 CFR 50.55a(f)(4). However, the use of any available instruments that meet the intent of the OM Code requirements for the actual reading would yield an acceptable level of quality and safety for testing. Licensees are required to submit a relief request in this case.

When the licensee submits a relief request, it should separately address each group of affected pumps if the instruments are permanently installed. By contrast, a general relief request may be granted for temporary instrumentation. If the instruments do not meet the intent of the OM Code requirements, the NRC may require the licensee to adjust acceptance limits to account for the instrument inaccuracies.

Licensees are cautioned that the CPT requires more accurate instruments than those specified in earlier editions of the OM Code. As a result, licensees must verify that instrument accuracy is appropriate for the type of test being performed (Group A or Group B test versus a CPT). Licensees should also note that previously acceptable instruments may no longer be acceptable when updating to a more recent edition of the OM Code.

5.5.2 Use of Tank Level to Calculate Flow Rate for Positive Displacement Pumps

The NRC has received requests for relief to use the tank level to calculate the flow rate in a system with a positive displacement pump when the system was not designed with a flow meter in the flow loop.

Paragraph ISTB-3550 of the OM Code requires licensees to measure the pump flow rate using a rate or quantity meter installed in the pump test circuit. If the meter does not directly indicate the flow rate, the record of the test shall identify the method used to reduce the flow data. In addition, subparagraph ISTB-5300(a) requires, for Group A tests and CPTs, a 2-minute run time after achieving stable pump conditions before recording data during the test.

NRC Recommendation

When flow meters are not installed in the flow loop of a system with a positive displacement pump, it is impractical to directly measure flow rate for the pump. The NRC staff has determined that, if the licensee uses the tank level to calculate the flow rate as described in paragraph ISTB-3550, the implementing procedure must include the calculational method and any test conditions needed to achieve the required accuracy. Specifically, the licensee must verify that the reading scale for measuring the tank level and the calculational method yield an accuracy within ±2 percent for Group A and B tests, Preservice tests, and CPTs. If the meter does not directly indicate the flow rate, the record of the test shall identify the method used to reduce the flow data.

Basis for Recommendation

The OM Code requires licensees to measure the pump flow rate in order to determine the extent of any pump degradation. A minimum pump run time of 2 minutes is required in order to achieve stable performance parameters before recording data during the test.

Requiring licensees to install a flow meter to measure the flow rate and to guarantee the test tank size, such that the pump flow rate will stabilize and then be run for 2 minutes before recording data, would be a burden because of the design and installation changes to be made to the existing system. Therefore, compliance with the OM Code requirements would be a hardship.

The average flow rate is calculated by measuring the change in test tank level over a period of time and converting it to flow rate using the following standard formula:

$$Q\ (\text{GPM}) = X\ \Delta L\ (\text{inch}) / \Delta t\ (\text{Second})$$

Where: Q is flow rate
X is a constant which reflects tank dimensions and unit conversions
ΔL is the measured change in level in the tank in time Δt

Pump discharge pressure will match system pressure up to the shutoff head of the positive displacement pump. Because of the characteristics of a positive displacement pump, there should be virtually no change in pump discharge flow rate as a result of the rising tank level. Therefore, rising tank level will not have an impact on test results. By having approximately the same level in the tank at the beginning of each test, licensees can achieve repeatable results. In addition, the suction would be from a large source at a constant pressure, which will allow pump performance parameters to stabilize quickly. This method would provide reasonable assurance of operational readiness, provided that the licensee measures the test tank level in accordance with the accuracy requirements of Table ISTB-3500-1. The implementing procedures should document the calculational method and test conditions required to achieve this accuracy. Therefore, the proposed alternative of using the tank level to calculate the flow rate provides reasonable assurance of operational readiness of the pump. Licensees must submit a request to implement this proposed alternative.

5.5.3 Use of Tank or Bay Level to Calculate Differential Pressure

The NRC has received requests for relief to use the tank or bay level to calculate differential pressure when a direct measurement of inlet pressure or differential pressure is not available.

NRC Recommendation

When inlet pressure gauges are not installed in the inlet of a vertical line shaft pump, it is impractical to directly measure inlet pressure for use in determining differential pressure for the pump. The NRC staff has determined that, if the licensee uses the bay level to calculate the suction (inlet) pressure as described in subparagraph ISTB-3520(b), the implementing procedure must include the calculation. The licensee must also verify that the reading scale for measuring the level and the calculational method yield an accuracy within ±2 percent for Group A and B tests, and ±1/2 percent for Preservice tests and CPTs. If direct measurements are impractical for other types of pumps with suction from a tank, the licensee must apply similar controls. The OM Code allows the licensee to determine differential pressure by obtaining the information from a differential pressure gauge or differential pressure transmitter, or by determining the difference between the pressure at a point in the inlet pipe and the pressure at a point in the discharge pipe (subparagraph ISTB-3520(b)). Therefore, the licensee may implement a calculational method without obtaining relief because the OM Code allows for the determination of differential pressure from the discharge pressure and the pressure in the pump inlet.

Basis for Recommendation

The method is in accordance with a determination of differential pressure allowed by the OM Code. Although the inlet pressure is not directly measured, it is “measured” for the purpose of determining the pressure at a point in the inlet. By including the calculation in the implementing procedures, the licensee can determine the differential pressure in a manner that is consistent and repeatable from test to test. This method will yield the information needed for monitoring the hydraulic condition of the applicable pumps without the need to install suction (inlet) pressure gauges, which may not be practical, depending on the design limitations in the inlet of the pump.

5.5.4 Accuracy of the Flow Rate Instrument Loop

As clarified in OM Code Interpretations 95-7 (1990 Edition, Subsection ISTB 4.6.1 and Table ISTB 4.6.1.1; 1987 Edition with 1988 Addenda, Part 6, paragrah 4.6.11 and Table 1, Instrument Accuracy) and Inquiries IN 91-3 (1987 Edition through 1990 Edition, Part 6, paragraph 4.6.1.1) and IN 91-037 (ASME BPV Code Section XI, 1977 Edition Through Later Editions and Addenda through the 1987 Addenda, Table IWP-4110-1, Instrument Accuracy-Flowrate), the accuracy requirements of analog instruments that are used to measure process flow apply only to the reference calibration of the instrument, such as that supplied by the instrument manufacturer, in determining loop accuracy. In determining instrument accuracy, the OM Code does not explicitly require the licensee to consider physical attributes (such as orifice plate tolerances), tap locations, environmental effects (such as temperature, radiation or humidity), vibration effects (such as seismic) or process effects (such as temperature). However, factors associated with attributes that could affect the measurements include the effects of wear, accumulation of dirt or grease on an annubar flow coefficient, and the reversed installation of a one-direction orifice plate.

NRC Recommendation

The OM Code requirements for instrument accuracy ensure that the instrument loop accuracy is adequate for monitoring pumps for degrading conditions. The accuracy for analog instruments specified in paragraph ISTB-3500 of the OM Code applies only to the calibration of the instruments. The NRC staff recommends that, when test results indicate that conditions in the pump or the test circuit have changed, licensees should consider corrective action for other attributes that could affect the overall loop accuracy of the measurements.

Basis for Recommendation

In ASME Code Interpretation 95-7 and ASME Code Inquiries IN 91-3 and IN 91-037, the ASME Code Committee states that the requirements for the final indication of flow rate on an analog instrument to be within 2 percent of full scale of actual process flow rate applies only to the calibration of the instrument and does not take into account physical attributes, environmental effects, vibration effects, or process effects.

5.6 Operational Readiness of Pumps

The OM Code establishes the requirements for preservice and inservice testing of pumps to assess their operational readiness to perform the specified functions.

5.7 Duration of Tests

Subparagraphs ISTB-5100(a), ISTB-5200(a), and ISTB-5300(a), "Duration of Tests," requires that, for measuring parameters as specified in Table ISTB-3000-1, each pump shall be run for at least 2 minutes after pump conditions are stable as the system permits. This duration is applicable to Group A tests and CPTs.

NRC Recommendation

The NRC staff recommends, if practicable, that this duration also be applied to Group B pump tests.

Basis for Recommendation

The 2-minute run time is adequate after pump operation becomes stable. This 2-minute run time minimizes overheating of pumps that are tested using the minimum flow recirculation line. The NRC recommends that licensees should minimize the time pumps are operated on the minimum flow recirculation line. (See NRC Bulletin 88-04, “Potential Safety-Related Pump Loss”).

5.8 Adjustments for Instrument Inaccuracies

If the accuracy of plant instrumentation used for IST activities for pumps is not well understood, the test results may not be adequate to meet the licensee’s safety analysis, even if they meet the OM Code requirements. For example, TS or the FSAR might require a pump to produce 1,000 gpm at 500 pounds per square inch differential (psid), but the IST reference values are 1,000 gpm (fixed) and 550 psid. The low end of the acceptable range for differential pressure from Table ISTB-5121-1 for Group A and Group B tests (0.90) would be 495 psid, although conservatively set at 500 psid. If this test is also to prove operability of the pump in addition to meeting IST requirements, and the ±2 percent instrument inaccuracies were taken into account for flow rate and differential pressure, there is the possibility that the pump is putting out less than the required values. In this example, the instrument accuracies would need to be taken into account if they were not already considered when the design parameters were developed.

When pump test procedures are developed, limits in the safety analysis cannot be ignored. The IST requirements are written generally. If specific plant limits are more conservative, to ensure compliance with design-basis assumptions, such limits must be clearly indicated as the “operability” limits and used for acceptance criteria of IST. For example, when obtaining values using instrumentation that meets the accuracy requirements specified for IST activities, the value as-read would be used. If a licensee is attempting to perform a critical test, more accurate instrumentation may be necessary; however, the value recorded would be the value read if the accuracy of the instrumentation met the specified accuracy. Only when instruments are used that cannot meet the specified accuracy for a test would an adjustment be necessary to meet the OM Code. Design analyses might not account for instrument accuracy readings; however, when the pump selection is made, the designer generally selects from a catalog of available sizes and chooses one with margin above the analyses numbers.

In the determination of loop accuracy, it is intended that only the instrument manufacturer’s reference accuracy be considered. It is not necessary to consider all uncertainties (such as environmental effects, process effects, vibration effects, etc.).

5.9 Pump Testing Using Minimum Flow Return Lines With or Without Flow Measuring Devices

The NRC has received relief requests from licensees requesting approval of pump testing by using minimum flow return lines with or without measuring devices as an alternative to the IST requirements as specified in the Subsection ISTB of the OM Code.

As specified in Section 5.1.2 above, the OM Code identifies various types of tests, such as Preservice tests, Group A tests, Group B tests, and CPTs. All pumps receive a Preservice test followed on a quarterly basis by the test associated with the pump category (Group A test for Group A pumps, and Group B test for Group B pumps), and at least once every 2 years by a

CPT. A CPT may also be substituted for a Group A or Group B test. Similarly, a Group A test may be substituted for a Group B test, and a Preservice test may be substituted for any inservice test.

Subparagraphs ISTB-5100(b), ISTB-5200(b), and ISTB-5300(b) of the OM Code allow the use of a bypass test loop for Group B tests, provided that it is designed to meet the pump manufacturer's operating specifications (e.g., flow rate, and time limitations) for minimum flow operation. The bypass test loop may be used for Group A tests or CPTs, provided that the flow rate through the loop meets the requirements specified in paragraph ISTB-3300.

An inservice pump test requires that the pump parameters shown in Table ISTB-3000-1 must be measured and evaluated to determine pump condition and detect degradation. Pump differential pressure and flow rate are two parameters that are measured and evaluated together to determine pump hydraulic performance.

In cases where only the minimum-flow return line is available for pump testing, regardless of the test interval, the NRC staff's position is that flow instrumentation that meets the requirements of paragraph ISTB-3500 should be installed in the mini-flow return line. Installation of this instrumentation is necessary to provide flow rate measurements during pump testing so that this data can be evaluated with the measured pump differential pressure to monitor for pump hydraulic degradation.

When testing a pump using a minimum flow recirculation line, the guidance provided in NRC Bulletin 88-04 applies. Licensees should review operating conditions to ensure that a pump is not subject to dead head conditions and that the minimum flow line is adequately sized and that operation will not damage the pump.

5.10 Alternative to ASME OM Code Comprehensive Pump Testing Requirements

The NRC has received alternative requests from licensees requesting authorization of alternatives to the Comprehensive Pump Testing (CPT) requirements specified in paragraph ISTB-3300, "Reference Values," of the OM Code.

The CPT was developed with the knowledge that some pumps, such as containment spray pumps, cannot presently be tested at the required high flow rates because of system design limitations. Consequently, subparagraph ISTB-3300(e)(1) requires licensees to establish reference values within ±20 percent of the design flow for the CPT.

Some designs do not allow a CPT at a pump design flow of ±20 percent because of the original system design configuration. In such cases, it may be necessary to use the pump's recirculation line for IST; however, recirculation lines are not typically designed to accommodate ±20 percent of the design flow.

Starting with the 2012 Edition of the OM Code, the CPT must be performed at the comprehensive test flow rate. This is defined as the flow rate established by the Owner that is effective for detecting mechanical and hydraulic degradation during subsequent testing. As required by the OM Code, the best efficiency point, system flow rates, and any other plant specific flow rates shall be considered when establishing the comprehensive test flow rate.

NRC Recommendation

For licensees with a Code of Record before the 2012 Edition of the OM Code, the NRC will accept the use of lower flow (reference values) other than ±20 percent of the design flow, as specified by subparagraph ISTB-3300(e)(1) for CPT, if the licensee's alternative request clearly demonstrates the impracticality of establishing a reference value within ±20 percent of the design flow for the CPT. For licensees with the Code of Record as 2012 Edition of the OM Code and later, the Code requires licensees to establish a CPT flow rate that is effective for detecting mechanical and hydraulic degradation as defined in ISTB-2000.

To obtain authorization for a proposed alternative method of performing the CPT with a flow other than as specified in subparagraph ISTB-3300(e)(1) and measuring pump parameters to detect hydraulic degradation and determine pump operability, the licensee must demonstrate that the acceptance criteria are equivalent to the OM Code requirements for CPTs in subparagraph ISTB-3300(e)(1).

To show the impracticality, the licensee should include (as a minimum, but not limited to) the following information in the submitted relief request:

1. Provide reason(s) for not performing the CPT at the required flow of ±20 percent of pump design flow.
2. Specify the maximum flow at which the CPT can be performed.
3. Provide the estimated cost of any temporary or permanent system modification required to enable a CPT to be performed at ±20 percent of pump design flow, along with any difficulty associated with implementing the modification.
4. Provide all details (e.g., temporary modifications of piping, containment sump, etc.), including pump performance curves, if a full-flow test was performed during preservice or service of the plant.
5. Provide pump performance curves and any other data associated with the pump's shop testing provided by the manufacturer.
6. Provide the records and history of maintenance and repairs performed on the pump.
7. Provide any appropriate compensatory actions being proposed to supplement the alternative testing, such as (but not limited to) the following examples:
    a. testing at the best efficiency point (BEP) on a longer interval; BEP is defined as the capacity and head at which the pump efficiency is at its maximum
    b. commitment to perform additional performance monitoring
    c. adjustment of acceptance criteria
    d. continuation of the previous OM Code testing, including taking overall vibration data quarterly
    e. periodic sampling and analysis of the lube oil.

Additional guidelines are included in NUREG/CP-0152, Volume 4, "Proceedings of the Seventh NRC/ASME Symposium on Valve and Pump Testing," in a paper titled "Comprehensive Pump Testing Based on ASME OM Code Requirements and its Alternative and Related Relief Requests."

This guidance requires relief because the OM Code does not allow for a reference value of flow for CPTs, other than at flow rates within ±20 percent of pump design flow. The NRC will review any relief requests on a case-by-case basis.

5.11 Waterleg Pumps

The NRC has received proposed alternatives from licensees of boiling-water reactor (BWRs) for Group A tests for waterleg pumps. Paragraph ISTB-3400 and Table ISTB-3400-1 of the OM Code specify that a Group A test must be performed quarterly for Group A pumps. The waterleg pumps are low flow pumps that are required to operate whenever their respective ECCS trains are in the operable condition. As such, the pumps perform continuous duty on a recirculation line and provide makeup as needed. There is typically no flow instrumentation on the recirculation line, and the flow instrumentation on the main ECCS header is not sufficiently accurate to measure the low flow of the pumps. When requesting an alternative Group A test for a waterleg pump, a licensee should explain how the pump discharge pressure is monitored, the main ECCS header is verified to be full of water, the pump vibration monitoring frequency, and any other maintenance or testing activity performed to ensure the pump will continue to meet its intended function.

5.12 Smooth-Running Pumps

Pumps that have very low vibration reference values (less than or equal to 0.05 inch per second) are called smooth-running pumps. A small increase in smooth-running pump vibration during the OM Code-required IST causes the pump to exceed OM Code vibration acceptance criteria, which normally results in unnecessary corrective action. The NRC has authorized alternative vibration acceptance criteria for smooth-running pumps on a case-by-case basis in accordance with 10 CFR 50.55a(z).

Alternative requests for smooth-running pumps should specify a minimum vibration reference value (≤ 0.05 inch per second), and these smooth-running pumps must be included in a predictive maintenance (PdM) program. The importance of the PdM program for smooth-running pumps was demonstrated when a plant using NRC-authorized alternative vibration acceptance criteria noted a bearing failure that was not detected by the IST program, but was detected through enhanced vibration monitoring as part of the plant's PdM program. During IST, corrective action was not required because the measured vibration was below the alert range as specified by the OM Code. After the pump bearing failed, it became clear that a simple minimum vibration reference value alone is not sufficient to identify degradation of a smooth-running pump. PdM programs normally include bearing temperature trending, oil sampling and analysis, thermographic analysis, and enhanced vibration monitoring. The objective of the PdM program should be to detect and correct problems involving the mechanical condition of the pump before the pump reaches its overall vibration alert limit.

ASME has issued Code Case OMN-22, "Smooth Running Pumps," related to smooth running pumps to eliminate the requirement for an alternative request. Until Code Case OMN-22 is published by ASME and endorsed in RG 1.192, an alternative request is required to implement this Code Case for smooth running pumps.

5.13 Vibration-Measuring Transducers

Subsection ISTB of the OM Code requires that the frequency response range of vibration-measuring transducers and their readout system be from one-third of the minimum pump shaft rotational speed to at least 1,000 hertz (Hz). Licensees have proposed alternatives to this OM Code requirement in accordance with 10 CFR 50.55a(z) for pumps with low shaft rotational speeds. Similar alternative requests submitted by licensees have been withdrawn following discussion with the NRC. The proposed alternatives state that the procurement and calibration of vibration-measuring transducers and their readout systems for the lower end of the OM Code-specified range were hardships because of the limited number of vendors supplying such equipment, the level of equipment sophistication, and equipment cost. The NRC typically authorized these alternative requests in the past. However, vibration-measuring transducers and their readout system can now be procured from various suppliers at a reasonably low cost due to technology advancement and research work performed in the field of instrumentation. Therefore, licensee requests to use this alternative are generally no longer authorized by the NRC.

5.14 Motor Drivers for Pumps

Pump drivers are outside the scope of the OM Code, with the exception of vibration testing for vertical line shaft pumps where the driver is an integral part of the pump. Most pumps are driven by electric motors, which are connected via coupled shafts. Motor vibration attributable to coupling misalignment may not be realized or measured at the pump, and small changes in the vibration of a motor can have significant effects on pump operation and the operational readiness.

Institute of Electrical and Electronics Engineers (IEEE) Standard 741-2007, "IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations," briefly address the vibration issue, and refers to IEEE C37.96-2000, "IEEE Guide for AC Motors Protection," for motors. IEEE 741-2007 includes the testing and the surveillance requirements and lists several standards in the reference section for testing. IEEE C37.96-2000 extensively addresses the vibration issue on electric motors because of its significant impact on bearings, lubricants, protective devices, etc.

5.15 Pumps in New Reactors

Subsection ISTF, "Inservice Testing of Pumps in Water-Cooled Reactor Nuclear Power Plants – Post-2000 Plants," in the OM Code provides IST requirements for pumps in new reactors. The Subsection ISTF provisions for pumps in new reactors specify an inservice test on a quarterly frequency. The NRC has incorporated by reference Subsection ISTF in 10 CFR 50.55a with a condition to ensure that the provisions of Appendix V are implemented.

In Commission papers SECY-90-016 and 93-087, the NRC indicated that new nuclear power plants should be designed to accommodate full flow testing of pumps within the scope of the IST program. The capability to perform full flow testing should be reflected in the development of the IST activities for pumps in new reactors.

The NRC staff will conduct inspections of the development and implementation of the IST program (including full flow testing of pumps) during construction and operation of new nuclear power plants.

6 STANDARD TECHNICAL SPECIFICATIONS

Standard Technical Specifications are contained in NUREG-1430 through NUREG-1434.

The Administrative Controls Technical Specification 5.5 includes a requirement to establish, implement, and maintain a program entitled “Inservice Testing Program.” This program provides controls for properly applying test frequencies associated with inservice testing of components activities under 10 CFR 50.55a(f) to Surveillance Requirements under 10 CFR 50.36.

7 IDENTIFICATION OF CODE NONCOMPLIANCE

7.1 Nonconforming Conditions

For details, see Section 2.1.4, "Identification of Code Noncompliance," in this NUREG.

7.2 Starting Point for Technical Specification Required Action Completion Times

For details, see Standard Technical Specification NUREG-1430 through NUREG-1434, "Limiting Condition for Operation (LCO) 3.0.2."

8 RISK INFORMED INSERVICE TESTING

8.1 Introduction

The NRC regulations at 10 CFR 50.55a, paragraph (f), "Preservice and Inservice Testing Requirements," require, in part, that certain pumps and valves must meet the requirements of the ASME OM Code.

RG 1.175 describes an acceptable alternative approach for applying risk insights from probabilistic risk assessment (PRA), in conjunction with established traditional engineering information, to make changes to a nuclear power plant's IST program. The approach described in RG 1.175 addresses the high-level safety principles specified in RG 1.174 and attempts to strike a balance between defining an acceptable process for developing risk-informed IST programs without being overly prescriptive. The resultant risk-informed IST programs will have improved effectiveness with regard to the utilization of plant resources, while still maintaining acceptable levels of quality and safety. However, licensees may propose other approaches for consideration by the NRC staff. It is intended that the approaches presented in RG 1.175 should be regarded as examples of acceptable practices, and that licensees should have some degree of flexibility in satisfying regulatory requirements on the basis of their accumulated plant experience and knowledge. As discussed in RG 1.175, licensees proposing to implement a risk-informed IST program are required to submit a request to implement an alternative to the OM Code in accordance with 10 CFR 50.55a(z).

OM Code, Subsection ISTE, "Risk-Informed Inservice Testing of Components in Water-Cooled Reactor Nuclear Power Plants," describes the development of an overall risk-informed IST program. As indicated in 10 CFR 50.55a(b)(3)(viii), licensees may not implement the risk-informed approach for inservice testing of pumps and valves specified in Subsection ISTE in the 2009 Edition through the 2017 Edition of the OM Code, without first obtaining NRC authorization in accordance with 10 CFR 50.55a(z)(1). In *Federal Register* notice 82 FR 32934 (July 18, 2017), the NRC discusses the aspects that need to be addressed when submitting a request to implement Subsection ISTE in the 2009 through the 2017 Edition of the OM Code. The NRC staff is reviewing Subsection ISTE in the 2020 Edition of the OM Code as part of the incorporation by reference of that edition of the OM Code in 10 CFR 50.55a with any applicable conditions.

A nuclear power plant licensee or applicant may propose implementation of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants," in a license amendment request (LAR) submitted for NRC review. Implementation of 10 CFR 50.69 allows for risk-informed treatment of SSCs as an alternative to certain special treatment requirements (STRs) in the NRC regulations. The IST provisions required by 10 CFR 50.55a are among the STRs that are removed for safety-related pumps, valves, and dynamic restraints that are categorized by a PRA as having low safety significance (RISC-3), if an LAR requesting implementation of 10 CFR 50.69 is approved by the NRC for a specific nuclear power plant. Appendix B, "Guidance for Treatment of Pumps, Valves, and Dynamic Restraints during Implementation of 10 CFR 50.69," to this NUREG provides information related to the treatment of pumps, valves, and dynamic restraints for the implementation of 10 CFR 50.69 at nuclear power plants.

8.2 Discussion

Until such time that a risk-informed alternative to the current OM Code requirements is incorporated by reference into the regulations, or a licensee is implementing 10 CFR 50.69 for RISC-3 SSCs, the alternative approach described in RG 1.175 must be authorized by the NRC pursuant to 10 CFR 50.55a(z)(1) on a plant-specific basis prior to implementation. Because 10 CFR 50.55a(z)(1) places no restrictions on the scope of alternatives that may be authorized, licensees may propose risk-informed alternatives to their entire IST program, or may propose alternatives that are more limited in scope (e.g., for a particular system or group of systems, or for a particular group of components). In either case, the NRC staff expects the licensee's proposal to address the principles described in RG 1.175, including those related to implementation and monitoring.

If a licensee proposes a risk-informed alternative to the OM Code test requirements, the application should contain a summary description of the proposed alternative. The summary description should specify the key technical and administrative aspects necessary to describe and control the risk-informed alternative. The NRC staff will review and approve this summary description pursuant to 10 CFR 50.55a(z)(1) and, as such, the summary description will serve as the framework within which the licensee may make future changes to its risk-informed alternative without having to resubmit it for NRC approval.

8.3 Online Inservice Testing

In an effort to shorten refueling outages, many licensees are trying to perform as much maintenance, testing, and surveillance as possible with the nuclear power plant on line.

For example, several licensees have submitted requests to obtain NRC authorization for an alternative to conduct IST activities once per refueling cycle, rather than during the refueling outage as prescribed by the OM Code. In preparing (and evaluating) such alternative requests, licensees (and the NRC staff) should consider several factors to ensure that the proposed alternative provides an acceptable level of quality and safety.

If a licensee is testing a particular pump or valve during RFOs, the licensee may have determined that it is impractical to test the pump or valve quarterly during operation. The licensee's IST program document should, therefore, discuss the basis for deferring the testing from quarterly (and during cold shutdowns) to RFOs. Alternative requests to perform testing once each refueling cycle with the nuclear power plant on line should be prepared in light of the RFO justification for each affected valve or group of valves. If necessary, the licensee should revise the RFO justification to be consistent with the alternative request.

Licensees (and the NRC staff) should also consider whether the testing can be accomplished within the allowed outage time permitted by any applicable TS. In general, the time necessary to complete the testing should be significantly less than the allowed outage time. This is to preclude TS violations or the need to issue exigent TS amendments or notices of enforcement discretion (NOEDs). In addition, licensees should not conduct non-corrective maintenance/testing activities at power if the associated post-maintenance testing cannot reasonably be accomplished until the next outage.

Sometimes, there is a tradeoff between testing these components at power (e.g., when they could be needed to mitigate the consequences of an accident) and testing them during outages (e.g., when there may be greater reliance on shutdown cooling or when other equipment is necessarily out-of-service). Licensees should quantitatively or qualitatively address the risks associated with testing components on line, rather than testing during the RFO. If the proposed testing could have a significant risk impact, or if its justification includes risk-related arguments, the alternative request should be prepared and reviewed in accordance with RG 1.174, as applicable. Licensees should also identify any compensatory measures to be established as a means to reduce the impact (e.g., risk and operational worker safety) of testing with the nuclear power plant at power.[2] If relevant, licensees should also provide information on how testing at power (rather than testing during RFOs) will affect scheduled maintenance work windows for the applicable system (i.e., whether the testing can be completed within the work windows or whether it will extend either the shutdown or at-power work windows). In addition, licensees will need to develop a new estimate of the maintenance unavailabilities that reflects the increased maintenance activities at power, and will need to document the basis for the new estimate (e.g., use plant logs or maintenance data to include in the current estimate of the maintenance unavailabilities those activities that were being performed during shutdown that will now be performed at power).

At times, testing (or the disassembly and inspection of components) during RFOs can be more advantageous than at-power operations from a worker safety perspective (for example, systems may be cold and depressurized). When requesting NRC authorization to perform testing with the nuclear power plant on line, licensees should consider worker safety and should discuss whether the applicable components can be adequately isolated and restored.

In Section 11.2.3 of NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” (NUMARC), now NEI, provided additional guidance for conducting online maintenance and testing. It states, in part:

> Online maintenance [and testing] should be carefully managed to achieve a balance between the benefits and potential impacts on safety, reliability or availability. For example, the margin of safety could be adversely impacted if maintenance is performed on multiple equipment or systems simultaneously without proper consideration of risk, or if operators are not fully cognizant of the limitations placed on the plant due to out of service equipment. Online maintenance should be carefully evaluated, planned and executed to avoid undesirable conditions or transients, and to thereby ensure a conservative margin of core safety.

8.4 ASME Risk-Informed Code Cases

Over the past several years, ASME has developed a series of risk-informed OM Code Cases related to testing pumps and valves that include risk-informed provisions, including the following examples:

- OMN-1, “Alternative Rules for Preservice and Inservice Testing of Active Electric Motor-Operated Valve Assemblies in Light-Water Reactor Power Plants”
- OMN-3, “Requirements for Safety Significance Categorization of Components Using Risk Insights for Inservice Testing of LWR Power Plants”
- OMN-4, “Requirements for Risk Insights for Inservice Testing of Check Valves at LWR Plants”
- OMN-7, “Alternative Requirements for Pump Testing”

- OMN-11, “Risk-Informed Testing for Motor-Operated Valves”
- OMN-12, “Alternative Requirements for Inservice Testing Using Risk Insights for Pneumatically and Hydraulically Operated Valve Assemblies in Light-Water Reactor Power Plants”

Certain OM Code Cases are listed as approved in Tables 1 and 2 of the specific revision to RG 1.192that has been incorporated by reference into 10 CFR 50.55a. Licensees may voluntarily use these Code Cases, without additional NRC staff approval, as an alternative to complying with the OM Code provisions that have been incorporated by reference into 10 CFR 50.55a, provided that the licensee uses the Code Cases with the conditions specified in RG 1.192 (i.e., the Code Case is generally acceptable, but the NRC staff has determined that the alternative requirements must be supplemented in order to provide an acceptable level of quality and safety).

When using ASME’s risk-informed OM Code Cases, licensees must perform the testing and performance monitoring of individual components as specified in the risk-informed component Code Cases (e.g., OMN-1, OMN-3, OMN-4, OMN-7, OMN-11, and OMN-12), as modified by any conditions specified in RG 1.192.

9 REFERENCES

9.1 American Society of Mechanical Engineers. NQA-1, “Quality Assurance Requirements for Nuclear Facilities Application,” New York, NY, 2008 Edition through 2009 Addenda.

9.2 American Society of Mechanical Engineers *Boiler and Pressure Vessel Code*, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” New York, NY.

9.3 Nuclear Energy Institute, NEI-96-07, “Guidelines for 10 CFR 50.59 Implementation,” Revision 1, Washington, DC, dated November 2000, ADAMS Accession No. ML003771157.

9.4 Nuclear Energy Institute White Paper, “Standard Format for Requests from Commercial Reactor Licensees Pursuant to 10 CFR 50.55a,” Revision 1, Washington, DC, June 7, 2004, ADAMS Accession No. ML070100400.

9.5 The National Technology Transfer and Advancement Act of 1995, Pub. L. No. 104–113, 1995. NRC Inspection Manual Para 9900, Technical Guidance.”

9.7 *U.S. Code of Federal Regulations*, “Domestic Licensing of Production and Utilization Facilities,” Part 50 and Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” Chapter 1, Title 10, “Energy.”

9.8 American Society of Mechanical Engineers/American National Standards Institute (ANSI) *Operation and Maintenance of Nuclear Power Plants*, Division 1, OM Code: Section IST, 1995 Edition through 2017 Edition, New York.

9.9 NRC Office Instruction LIC-102, Revision 3, “Review of Relief Requests, Proposed Alternatives, and Requests to Use Later Code Editions and Addenda,” ADAMS Accession No. ML18351A218.

9.10 American Society of Mechanical Engineers Standard QME-1, “Qualification of Active Mechanical Equipment used in Nuclear Power Plants,” 2007 and 2017 Editions.

9.11 Performance Test Code, PTC 25.3-1976, “Safety and Relief Valves,” New York, 1976.

9.12 American Society of Mechanical Engineers/American National Standards Institute Operations and Maintenance Standards, New York, NY.

9.12.1 ANSI/ASME, Part 1 (OM-1), “Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices,” New York, NY, 1981 and 1987.

9.12.2 ANSI/ASME, Part 6 (OM-6),” Inservice Testing of Pumps in Light-Water Reactor Power Plants,” New York, NY, 1988 and 1989 Addenda.

9.12.3 ANSI/ASME Part 10 (OM-10), “Inservice Testing of Valves in Light-Water Reactor Power Plants,” New York, NY, 1988 and 1989 Addenda.

9.13 Reserved

9.14 Reserved

9.15 Reserved

9.16 Reserved

9.17 Reserved

9.18 Reserved

9.19 Nuclear Management and Resources Council (NUMARC, now NEI), NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 2, April 1996, ADAMS Accession No. ML101020415.

9.19 Reserved

9.20 SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” January 12, 1990, ADAMS Accession No. ML003707849.

9.21 SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” April 2, 1993, ADAMS Accession No. ML003708021.

9.22 SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs,” March 28, 1994, ADAMS Accession No. ML003708068.

9.23 SECY-95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084),” May 22, 1995, ADAMS Accession No. ML003708005.

9.24 ASME *Boiler and Pressure Vessel Code*, New York, NY.

9.25 American Society of Mechanical Engineers/American National Standards Institute *Operation and Maintenance of Nuclear Power Plants*, New York.

9.26 *Federal Register*

- 9.26.1 *Federal Register*, Vol. 41, No. 30, “Codes and Standards for Nuclear Power Plants (10 CFR Part 50),” February 12, 1976, p. 6256.
- 9.26.2 *Federal Register*, Vol. 57, No. 3152, “Codes and Standards for Nuclear Power Plants (10 CFR Part 50),” August 6, 1992, p. 34666.
- 9.26.3 *Federal Register* Notice (56 FR 36175), July 31, 1991.
- 9.26.4 *Federal Register* Notice (64 FR 51370), September 22, 1999.
- 9.26.5 *Federal Register* Notice (66 FR 40626), August 3, 2001.
- 9.26.6 *Federal Register* Notice (69 FR 58804), October 1, 2004.
- 9.26.7 *Federal Register Notice* (76 FR 36232), June 21, 2011.
- 9.26.8 *Federal Register Notice* (82 FR 32934), July 18, 2018.

9.27 Generic Issues and Generic Letters

- 9.27.1 “Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications,” January 1980.
- 9.27.2 Generic Issue 105, “Interfacing Systems LOCAs at Light-Water Reactors.”
- 9.27.3 Generic Letter 87-06, “Periodic Verification of Leak-Tight Integrity of Pressure Isolation Valves,” March 13, 1987.
- 9.27.4 Generic Letter 87-09, “Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements,” May 4, 1987.
- 9.27.5 Generic Letter 88-14, “Instrument Air Supply System Problems Affecting Safety-Related Equipment,” August 8, 1988.
- 9.27.6 Generic Letter 89-04, “Guidance on Developing Acceptable Inservice Testing Programs,” April 3, 1989.
- 9.27.7 Generic Letter 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” June 28, 1989, and supplements.
- 9.27.8 Generic Letter 90-06, “Resolution of Generic Issue 70, Power-Operated Relief Valve and Block Valve Reliability, and Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors, Pursuant to 10 CFR 50.54(f),” June 20, 1990.
- 9.27.9 Reserved

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

9.27.11 Generic Letter 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,” September 30, 1996.

9.28 Regulatory Issue Summaries

9.28.1 Regulatory Issue Summary 2000-03, “Resolution of GSI 158 Performance of Safety-Related Valves Under Design Basis Conditions,” March 15, 2000, including NRC staff comments on JOG Program on AOV periodic testing.

9.28.2 Regulatory Issue Summary 2004-12, “Clarification on Use of Later Editions and Addenda to the ASME OM Code and Section XI,” July 28, 2004, ADAMS Accession No. ML042090436.

9.28.3 Regulatory Issue Summary 2004-16, “Use of Later Editions and addenda to the ASME Code Section XI for Repair/Replacement Activities,” dated October 19, 2004.

9.28.4 Regulatory Issue Summary 2012-08, Revision 1, “Developing Inservice Testing and Inservice Inspection Programs under 10 CFR Part 52,” dated July 17, 2013 (ADAMS Accession No. ML13122A365).

9.29 Information Notices and Bulletins

9.29.1 Information Notice 82-08, “Check Valve Failures on Diesel Generator Engine Cooling Systems,” March 26, 1982.

9.29.2 Bulletin No. 83-03, “Check Valve Failures in Raw Water Cooling System of Diesel Generators,” March 10, 1983.

9.29.3 Information Notice 83-54, “Common Mode Failure of Main Steam Isolation Non-Return Check Valves,” August 11, 1983.

9.29.4 Information Notice 85-84, “Inadequate Inservice Testing of Main Steam Isolation Valves,” October 30, 1985.

9.29.5 Information Notice 86-50, “Inadequate Testing to Detect Failures of Safety-Related Pneumatic Components or Systems,” June 18, 1986.

9.29.6 Information Notice 87-01, “RHR [Residual Heat Removal] Valve Misalignment Causes Degradation of ECCS in PWRs,” January 6, 1987.

9.29.7 Information Notice 87-40, “Back Seating Valves Routinely to Prevent Packing Leakage,” August 31, 1987.

9.29.8 Information Notice 88-70, “Check Valve Inservice Testing Program Deficiencies,” August 29, 1988.

9.29.9 Information Notice 89-32, “Surveillance Testing of Low-Temperature Overpressure-Protection Systems,” March 23, 1989.

9.29.10 Information Notice 89-62, “Malfunction of Borg-Warner Pressure Seal Bonnet Check Valves Caused by Vertical Misalignment of Disk,” August 31, 1989.

9.29.11 Information Notice 91-56, “Potential Radioactive Leakage to Tank Vented to Atmosphere,” September 19, 1991.

9.29.12 Information Notice 91-74, “Changes in Pressurizer Safety Valve Setpoints Before Installation,” November 25, 1991.

9.29.13 Information Notice 94-08, “Potential for Surveillance Testing To Fail To Detect an Inoperable Main Steam Isolation Valve,” February 1, 1994.

9.29.14 Information Notice 94-44, “Main Steam Isolation Valve Failure To Close on Demand Because of Inadequate Maintenance and Testing,” June 16, 1994.

9.29.15 Information Notice 94-56, “Inaccuracy of Safety Valve Set Pressure Determination Using Assist Devices,” August 11, 1994.

9.29.16 Information Notice 96-02, “Inoperability of PORVs Masked by Downstream Indications During Testing,” January 5, 1996.
9.29.17 Information Notice 96-48, “Motor-Operated Valve Performance Issues,” August 21, 1996.
9.29.18 Information Notice 97-16, “Preconditioning of Plant Structures, Systems, and Components before ASME Code Inservice Testing or Technical Specification Surveillance Testing,” April 4, 1997.
9.29.19 Information Notice 97-90, “Use of Non-Conservative Acceptance Criteria in Safety Related Pump Surveillance Tests,” December 30, 1997.
9.29.20 Information Notice 2000-21, “Detached Check Valve Disk Not Detected by the Use of Acoustic and Magnetic Non-Intrusive Test Technique,” December 15, 2000.
9.29.21 Information Notice 2003-01, “Failure of a Boiling Water Reactor Main Steam Safety/Relief Valve,” January 15, 2003.
9.29.22 Information Notice 2003-15, “Importance of Followup Activities in Resolving Maintenance Issues,” September 5, 2003.
9.29.23 Information Notice 2006-03, “Motor Starter Failures due to Mechanical-Interlock Binding,” January 25, 2006.
9.29.24 Information Notice 2006-15, “Vibration-Induced Degradation and Failure of Safety-Related Valves,” July 27, 2006.
9.29.25 Information Notice 2006-29, “Potential Common Cause Failure of Motor-Operated Valves as a Result of Stem Nut Wear,” December 14, 2006.
9.29.26 Information Notice 2008-20, “Failures of Motor-Operated Valve Actuator Motors with Magnesium Alloy Rotors,” December 8, 2008.
9.29.27 Information Notice 2010-03, “Failures of Motor-Operated Valves Due to Degraded Stem Lubricant,” February 3, 2010.
9.29.28 Information Notice 2012-06, “Ineffective Use of Vendor Technical Recommendations,” April 24, 2012.
9.29.29 Information Notice 2012-14, “Motor-Operated Valve Inoperable Due To Stem-Disc Separation,” July 24, 2012.
9.29.30 Information Notice 2013-14, “Potential Design Deficiency in Motor-Operated Valve Control Circuitry,” August 23, 2013.
9.29.31 Information Notice 2014-11, “Recent Issues Related to the Qualification and Commercial Grade Dedication of Safety-Related Components,” September 19, 2014.
9.29.32 Information Notice 2015-13, “Main Steam Isolation Valve Failure Events,” December 10, 2015.
9.29.33 Information Notice 2016-05, “Operating Experience Regarding Complications from a Loss of Instrument Air,” April 27, 2016.
9.29.34 Information Notice 2016-09, “Recent Issues Identified When Using Reverse Engineering Techniques in the Procurement of Safety-Related Components,” July 15, 2016.
9.29.35 Information Notice 2017-03, “Anchor/Darling Double Disc Gate Valve Wedge Pin and Stem-Disc Separation Failures,” June 15, 2017.
9.29.36 Bulletin No. 88-04, “Potential Safety-Related Pump Loss,” May 5, 1988.

9.30 Reserved

9.31 Inspection Manual (sections published periodically)

9.31.1 NRC Inspection Manual, Chapter 0326, “Operability Determinations,” September 30, 2019, ADAMS Accession No. ML19273A878.

9.31.2 Part 9900, “Technical Guidance: Maintenance – Preconditioning of Structures, Systems, and Components before Determining Operability,” September 28, 1998.

9.31.3 Inspection Procedure 61726, “Surveillance Observations.”

9.31.4 Inspection Procedure 62706, “Maintenance Rule.”

9.31.5 Inspection Procedure 62708, “Motor-Operated Valve Capability,” October 23, 2013, ADAMS Accession No. ML13142A123.

9.31.6 Attachment 22, “Surveillance Testing,” to NRC Inspection Manual IP 71111, “Reactor Safety: Initiating Events, Mitigating Systems, Barrier Integrity,” January 1, 2020, ADAMS Accession No. ML19197A106.

9.31.7 Inspection Procedure 71111.21N.02, “Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements,” July 26, 2019, ADAMS Accession No. ML19067A240.

9.31.8 Inspection Procedure 73756, “Inservice Testing of Pumps and Valves,” July 27, 1995.

9.31.9 Inspection Procedure 73758, “Part 52, Functional Design and Qualification, and Preservice and Inservice Testing Programs for Pumps, Valves and Dynamic Restraints,” February 6, 2020, ADAMS Accession No. ML19364A004.

9.32 Letters and Public Meetings

9.32.1 Memorandum from Frederick J. Hebdon, Director, Project Directorate II-3, Division of Reactor Projects I/II, NRC Office of Nuclear Reactor Regulation, to Jon R. Johnson, Acting Director, Division of Reactor Projects, Region II, in response to Technical Assistance Request TIA 96-007: Regulatory Acceptability of Lubricating Valves Prior to Surveillance Testing, dated July 2, 1996 (ADAMS Accession No. ML17228B551).

9.32.2 Minutes of the Public Meetings on Generic Letter 89-04, October 25, 1989, ADAMS Accession No. ML19262F381.

9.32.3 Supplement to Minutes of Public Meetings on Generic Letter 89-04, January 22, 1992, ADAMS Accession No. ML20139A075.

9.33 NUREGs

9.33.1 NUREG-0737, “Clarification of TMI Action Plan Requirements,” November 1980.

9.33.2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.”

9.33.3 NUREG-1275, “Operating Experience Feedback Report: Solenoid-Operated Valve Problems, Vol. 6,” September 1991, ADAMS Accession No. ML063550260.

9.33.4 NUREG-1350, “NRC Information Digest,” issued annually.

9.33.5 NUREG-1430, Revision 4, “Standard Technical Specifications — Babcock and Wilcox Plants.”

9.33.6 NUREG-1431, Revision 4, “Standard Technical Specifications — Westinghouse Plants.”

9.33.7 NUREG-1432, Revision 4, “Standard Technical Specifications — Combustion Engineering Plants.”

9.33.8 NUREG-1433, Revision 4, “Standard Technical Specifications — General Electric Plants (BWR/4).”

9.33.9 NUREG-1434, Revision 4, “Standard Technical Specifications — General Electric Plants (BWR/6).”

9.33.10 NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," Revision 0 (April 1995), Revision 1 (January 2005), and Revision 2 (October 2013), ADAMS Accession Nos. 9705030476, ML050550290, and ML13295A020.

9.34 NUREG/CPs

9.34.1 NUREG/CP-0111, "Proceedings of the Symposium on Inservice Testing of Pumps and Valves," October 1990, ADAMS Accession No. ML19252A244.
9.34.2 NUREG/CP-0123, "Proceedings of the Second NRC/ASME Symposium on Valve and Pump Testing," July 1992, ADAMS Accession No. ML19100A252.
9.34.3 NUREG/CP-0123, "Proceedings of the Second NRC/ASME Symposium on Valve and Pump Testing, Supplement 1," November 1992, ADAMS Accession No. ML19100A333.
9.34.4 NUREG/CP-0137, "Proceedings of the Third NRC/ASME Symposium on Valve and Pump Testing," July 1994, ADAMS Accession Nos. ML19275E537, and ML19275E594.
9.34.5 NUREG/CP-0152, Volume 3, "Proceedings of the Sixth NRC/ASME Symposium on Valve and Pump Testing," July 17–20, 2000.
9.34.6 NUREG/CP-0152, Volume 4, "Proceedings of the Seventh NRC/ASME Symposium on Valve and Pump Testing," July 15–18, 2002.
9.34.7 NUREG/CP-0152, Volume 5, "Proceedings of the Eighth NRC/ASME Symposium on Valve and Pump Testing, "July 12–14, 2004.
9.34.8 NUREG/CP-0152, Volume 6, "Proceedings of the Ninth NRC/ASME Symposium on Valves, Pumps and Inservice Testing," July 17–19, 2006.
9.34.9 NUREG/CP-0152, Volume 7, "Proceedings of the Tenth NRC/ASME Symposium on Valves, Pumps and Inservice Testing," July 14–16, 2008.
9.34.10 NUREG/CP-0152, Volume 8, "Proceedings of the Eleventh NRC/ASME Symposium on Valves, Pumps, and Inservice Testing," August 15–16, 2011.
9.34.11 NUREG/CP-0152, Volume 9, "Proceedings of the Twelfth NRC/ASME Symposium on Valves, Pumps, and Inservice Testing," June 23–25, 2014.
9.34.12 NUREG/CP-0152, Volume 10, "Proceedings of the Thirteenth NRC/ASME Symposium on Valves, Pumps, and Inservice Testing," July 17–19, 2017

9.35 NUREG/CRs

9.35.1 NUREG/CR-5102, "Interfacing Systems LOCA: Pressurized-Water Reactors," February 1989, ADAMS Accession No. ML19276D424.
9.35.2 NUREG/CR-5124, "Interfacing Systems LOCA: Boiling-Water Reactors," February 1989, ADAMS Accession No. ML19276F798.
9.35.3 NUREG/CR-5775, "Quantitative Evaluation of Surveillance Test Intervals Including Test-Caused Risks," February 1992, ADAMS Accession No. ML19172A254.
9.35.4 NUREG/CR-6644, "Generic Issue 158: Performance of Safety-Related Power-Operated Valves Under Operating Conditions," ADAMS Accession No. 9910190167.
9.35.5 NUREG/CR-6654, "A Study of Air-Operated Valves in U.S. Nuclear Power Plants."
9.35.65 NUREG/CR-6396, "Examples, Clarifications, and Guidance on Preparing Request for Relief from Pump and Valve Inservice Testing Requirements," prepared by Idaho National Engineering Laboratory and published February 1996, ADAMS Accession No. ML19275F829.

9.36 Regulatory Guides (RGs)

9.36.1 Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” Revision 5, February 2017, ADAMS Accession No. ML16082A501.

9.36.2 Regulatory Guide 1.58, “Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel,” Revision 1, September 1980. (Withdrawn - See 56 FR 36175, July 31, 1991)

9.36.3 Regulatory Guide 1.147, “Inservice Inspection Code Case Acceptability ASME Section XI, Division 1,” Revision 19, March 2017, ADAMS Accession No. ML19128A244.

9.36.4 Regulatory Guide 1.174, “An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 3, January 2018 ADAMS Accession No. ML17317A256.

9.36.5 Regulatory Guide 1.175, “An Approach for Plant-Specific, Risk-Informed Decision-Making: Inservice Testing,” August 1998.

9.36.6 Regulatory Guide 1.192, Revision 3 “Operation and Maintenance Code Case Acceptability: ASME OM Code,” October 2019, ADAMS Accession No. ML19128A261.

9.36.7 Regulatory Guide 1.193, Revision 6, “ASME Code Cases Not Approved for Use,” October 2019, ADAMS Accession No. ML19128A269.

9.36.8 Regulatory Guide 1.187, Revision 1, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” Washington, DC, May 2019, ADAMS Accession No. ML17195A655.

9.36.9 Regulatory Guide 1.100 (Revision 4), “Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” May 22, 2020, ADAMS Accession No. ML19312C677.

9.37 SECY-77-439, “Single-Failure Criterion,” August 17, 1977, ADAMS Accession No. 7812180291.

9.38 SECY-92-223, “Resolution of Deviations Identified during the Systematic Evaluation Program,” June 19, 1992, ADAMS Accession No. 9206300320.

9.39 WASH-1400 (NUREG-75/014), “Reactor Safety Study: An Assessment of Accident Risk in U.S. Commercial Nuclear Power Plants,” 1975, ADAMs Accession No. ML053290245.

9.40 Commission’s Staff Requirements Memorandum (SRM), dated September 11, 2002, for Commission Paper SECY-02-0067, “Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Operational Programs (Programmatic ITAAC),” ADAMS Accession No. ML020700641.

9.41 Commission’s 2014, SRM, dated May 14, 2014, for SECY-04-0032, “Programmatic Information Needed for Approval of a Combined License Without Inspections, Tests, Analyses and Acceptance Criteria,” ADAMS Accession No. ML041350440.

9.42 SECY-05-0197, “Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria,” ADAMS Accession No. ML052770257.

9.43 JOG AOV Program Document, Revision 1, December 13, 2000, ADAMS Accession No. ML010950310, and NRC comment letter dated October 8, 1999, to Nuclear Energy Institute, ADAMS Accession No. ML020360077.

APPENDIX A
GUIDELINES FOR INSERVICE EXAMINATION, TESTING, AND SERVICE LIFE MONITORING PROGRAM FOR DYNAMIC RESTRAINTS (SNUBBERS) AT NUCLEAR POWER PLANTS

TABLE OF CONTENTS

TABLE OF FIGURES

LIST OF TABLES

ABBREVIATIONS

ASME	American Society of Mechanical Engineers
BPV	Boiler and Pressure Vessel
BWR	boiling-water reactor
CFR	*Code of Federal Regulations*
DBD	design-basis document
DTPG	defined test plan group
EGM	enforcement guidance memorandum
FMG	failure mode grouping
FR	*Federal Register*
FSAR	final safety analysis report
GDC	General Design Criterion
GL	generic letter
GSI	generic safety issue
IN	information notice
IP	inspection procedure
ISI	inservice inspection
IST	inservice testing
ITAAC	inspection, tests, analyses, and acceptance criteria
LBHS	large bore hydraulic snubbers
LCO	limiting condition operation
NEI	Nuclear Energy Institute (formerly NUMARC)
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation (NRC)
NUMARC	Nuclear Management and Resources Council (now NEI)
OM	Operations and Maintenance
P&ID	piping and instrument diagram
PWR	Pressurized-water reactor

RCP	reactor coolant pump
RG	regulatory guide
RIS	Regulatory Issue Summary
SAR	safety analysis report
SG	steam generator
SLM	service-life monitoring
STS	Standard Technical Specifications
TS	Technical Specification(s)
TRM	Technical Requirement Manual
UFSAR	updated final safety analysis report
WGC	Working Group Committee

A.1 INTRODUCTION

A.1.1 Regulatory Basis

Title 10, Section 50.55a, "Codes and standards," of the *Code of Federal Regulations* (10 CFR 50.55a) defines the requirements for applying industry codes and standards to boiling- or pressurized-water-cooled nuclear power facilities. Each of these facilities is subject to the conditions in paragraphs (a), (b), (f), and (g) of 10 CFR 50.55a, as they relate to inservice inspection (ISI) and inservice testing (IST). Originally, before September 8, 1992, all ISI and IST requirements were specified in paragraph (f) of 10 CFR 50.55a. By rulemaking effective September 8, 1992 (see 57 FR 34666: August 6, 1992), the U.S. Nuclear Regulatory Commission (NRC) established paragraph (f) of 10 CFR 50.55a to separate the IST requirements from the ISI requirements in paragraph (g). The IST requirements for pumps and valves fall under paragraph (f), whereas ISI of components (including supports) falls under paragraph (g). Snubbers inservice examination and testing requirements falls under paragraph (g) of the 10 CFR 50.55a.

The NRC regulations in 10 CFR 50.55a list all the American Society of Mechanical Engineers (ASME) Codes that have been incorporated by reference including the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) and the ASME *Boiler and Pressure Vessel Code* (BPV Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," with conditions, that the NRC has approved for use.

On June 3, 2020, the NRC regulations in 10 CFR 50.55a(a)(i) incorporate by reference the 1970 Edition through the 2017 Edition of the OM Code and the ASME BPV Code, Section XI. Originally, the ASME BPV Code, Section XI, Article IWF-5000, specified the requirements of inservice examination and testing of snubbers, which has been deleted from the 2006 Addenda of ASME BPV Code, Section XI. Similarly, as of June 3, 2020, the NRC regulations in 10 CFR 50.55a(a)(iv) incorporate by reference the 1995 Edition through the 2017 Edition of the OM Code promulgated by the ASME, in which Subsection ISTA of the OM Code provides general requirements for IST of pumps and valves, and inservice examination and testing of snubbers. Subsections ISTB, ISTC, and ISTD specify the inservice requirements for pumps (pre-2000 plants), valves, and dynamic restraints, respectively. Subsection ISTF specifies requirements for pumps (post-2000) in new reactors. Based on those requirements, each nuclear power plant licensee must establish a snubber program, specify the snubbers included in the program as well as the examination and test plans in accordance with the applicable ASME BPV Code, Section XI or OM Code.

The regulations in 10 CFR 50.55a(g)(4) state, "Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions and addenda of the ASME BPV Code (or the ASME OM Code for snubber examination and testing)." The ASME BPV Code, Section XI, provides the rules for ISI of nuclear power plant components. The ASME OM Code provides the rules for IST of nuclear power plant pumps, valves, and snubbers.

The regulations in 10 CFR 50.55a(g)(4)(ii) require the use of the latest edition of the Code and addenda that has been incorporated by reference 18 months prior to the beginning of each 120-month inspection interval. This Code is considered to be the “Code of Record” for the inspection interval.

The regulations in 10 CFR 50.55a(g)(4)(iv) specify that inservice examination of components [including supports and snubbers] and system testing of components may meet the requirements set forth in subsequent editions and addenda of the ASME BPV Code, Section XI (or the ASME OM Code for snubber examination and testing) that are incorporated by reference in 10 CFR 50.55a(a), subject to the conditions listed in 10 CFR 50.55a(b), and subject to NRC approval. Portions of editions or addenda may be used, provided that all related requirements of the respective editions or addenda are met. Licensees must request approval to use later Code editions and addenda via a letter to the NRC; they may not simply update the inservice examination and test requirements of their snubber program. When requesting to use editions and addenda of the ASME Code that have not yet been incorporated by reference, licensees must request authorization to use these later editions and addenda as an alternative to the regulations pursuant to 10 CFR 50.55a(z). For further clarification, see NRC Regulatory Issue Summary (RIS) 2004-12, “Clarification on Use of Later Editions and Addenda to the ASME OM Code and ASME BPV Code Section XI.”

Where a snubber examination and test requirement of the ASME BPV Code, Section XI, or OM Code is determined to be impractical for a facility, the NRC’s regulations allow the licensee to submit an alternative request from the given requirement, along with information to support the determination. Alternative requests generally detail the reasons for deviating from the Code requirements and propose alternative testing methods or frequencies. The Commission is authorized to evaluate licensees’ alternative requests and may grant the requested relief or impose alternative requirements, considering the burden that the licensee might incur if the Code requirements were enforced for the given facility. Pursuant to 10 CFR 50.55a(z)(1) and (2), the Commission may also authorize the licensee to implement an alternative to the Code requirements, provided that the alternative ensures an acceptable level of quality and safety or the Code requirement presents a hardship without a compensating increase in the level of quality and safety.

10 CFR 50.55a(b)(3)(v) states, in part, that Article IWF-5000, “Inservice Inspection Requirements for Snubbers,” of the ASME BPV Code, Section XI, must be used when performing inservice inspection examinations and tests of snubbers at nuclear power plants, excepts conditioned in paragraphs 10 CFR 50.55a(b)(3)(v)(A) and (B) as follows:

(A) Snubbers: First provision: Licensee may use Subsection ISTD, “Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Water-Cooled Reactor Nuclear Power Plants,” of the ASME OM Code, 1995 Edition, through the latest edition and addenda incorporated by reference in 10 CFR 50.55a(a)(1)(iv), in place of requirements for snubbers in the editions and Addenda through the 2005 Addenda of the ASME BPV Code, Section XI, Articles IWF-5200(a) and (b) and IWF-5300(a) and (b), by making appropriate in the licensee-controlled documents.

(B) Snubbers: Second provision: Licensees must comply with the provisions for examining and testing snubbers in Subsection ISTD of the ASME OM Code and make appropriate changes to their technical specifications or licensee-controlled documents when using the 2006 Addenda and later editions of Section XI of the ASME BPV Code.

Therefore, if a plant's "Code of Record," is the 2006 Addenda or later Editions of the ASME BPV Section XI, they must use the OM Code requirements for snubber inservice examination and testing.

A.1.2 Regulatory History of NRC Staff Guidance on Examination, Testing and SLM of Snubbers

Since the start of commercial operation of the nuclear power plants, inservice visual examination and functional testing of snubbers have been regulatory requirements. Originally, these requirements were imposed by the plant TS surveillance requirements (SRs). There are three SRs for snubbers: (1) visual inspection or examination; (2) functional or operational testing, and (3) Snubber Service Life Monitoring (SLM). The visual inspection is the observation of the condition of installed snubbers to identify those that are damaged, degraded, or suspected as inoperable caused by physical means, leakage, corrosion, or environmental exposure. TS surveillance testing utilizes statistical sampling to validate the functionality of the tested population within an assumed quality confidence level. Only a sample of the installed snubbers is actually tested. Functional testing is performed to verify operational readiness by verifying that the tested snubber operates within the specified performance limits. The functional test typically involves removing the snubber from service and testing it on a specially-designed test machine. The performance of visual examinations is a separate process that complements the functional testing program and provides additional confidence in snubber population reliability. The TS specifies a schedule for snubber visual examinations and functional testing, which is usually based on refueling outage intervals. The typical original TS specified SLM as follows:

> A record of the service life of each snubber: the date at which the designated service life commences and the installation and maintenance record on which the designated service life is based shall be maintained.
>
> Concurrent with the first inservice visual inspection and at least once per 18 months (which is now 24 months) thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled service life review, the snubber service life shall be reevaluated, or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled life review (during the refueling outage). This reevaluation, replacement or reconditioning shall be indicated in the records.

Improved TS for various boiling and pressurized water-cooled nuclear power plants (NUREG-1430 thru 1434, Revision 4) allow relocating inservice examination and testing requirements of snubbers from the TS to a plant's Technical Requirements Manual (TRM). Relocating snubber ISI and testing requirements from the TS to TRM, however, does not eliminate the need to comply with the 10 CFR 50.55a requirements.

Until 1990, the ASME Code requirements addressing inservice examination and testing of snubbers were contained in ASME BPV Code, Section XI, Article IWF-5000. IWF-5000 referenced ASME/American National Standards Institute Standard for Operation and Maintenance of Nuclear Power Plants, Part 4 (OM-4), 1987 Edition with OMa-1988 Addenda, for preservice and inservice examinations and testing of snubbers requirements. In 1990, the

ASME published the initial edition of the OM Code which provided rules for inservice examination and testing of snubbers by incorporating most of the requirements of OM-4 (1987/1988 addenda).

The 1990 Edition of the OM Code consisted of one section (Section IST) entitled “Rules for Inservice Testing of Light-Water Reactor Power Plants.” This section was divided into four subsections: ISTA, “General Requirements,” ISTB, “Inservice Testing of Pumps in Light-Water Reactor Power Plants,” ISTC, “Inservice Testing of Valves in Light-Water Reactor Power Plants,” and ISTD, “Preservice and Inservice Examination and Testing of Nuclear Power Plant Dynamic Restraints (Snubbers).” At that time, the inservice examination and testing of snubbers was governed (under rulemaking) by the ISI requirements of Section XI of the ASME BPV Code, 1989 Edition, Article IWF-5000. Therefore, Subsection ISTD was not incorporated by reference in 10 CFR 50.55a at that time.

In 1990, the NRC issued Generic Letter (GL) 90-09, “Alternate Requirements for Snubber Visual Inspection Intervals and Corrective Actions.” The alternative visual inspection schedule is based on the number of unacceptable snubbers found during the previous inspection interval in proportion to the sizes of the various snubber populations or categories. This reduces future occupational radiation exposure and is highly cost effective. The alternative inspection interval is based on a fuel cycle of up to 24 months. GL 90-09 allowed licensees to extend snubber inservice visual examination interval up to 48 months.

At that time, most of the licensees revised their snubber examination and testing documents such as TS, TRM or licensee-controlled documents, based on GL 90-09. Similarly, in the 1995 Edition of the OM Code, Table ISTD 6.5.2-1 (now OM Code-2017, Table ISTD-4252-1) incorporated all the visual examination requirements provided in GL 90-09.

In 2000, for the first time, the regulations in 10 CFR 50.55a(b)(3)(v) stated that licensees may use the guidance in Subsection ISTD, OM Code, 1995 Edition with the 1996 Addenda, for examination and testing of snubbers, in lieu of the ASME BPV Code, Section XI. In 2012, the regulations in 10 CFR 50.55a provided information about the use of the ASME BPV Code, Section XI, and OM Code for snubber examination and testing in 10 CFR 50.55a(b)(3)(v). 10 CFR 50.55a(b)(3)(v) states, in part, in 10 CFR 50.55a(b)(3)(v)(A) that licensees may use Subsection ISTD of the 1995 through the latest edition and addenda of the OM Code incorporated by reference in 10 CFR 50.55a(a), in place of the snubber requirements in the editions and addenda up to the 2005 Addenda of the ASME BPV Code, Section XI, and in 10 CFR 50.55a(b)(3)(v)(B) that licenses must comply with the provisions for examining and testing snubbers in Subsection ISTD of the OM Code when using the 2006 Addenda and later editions and addenda of the Section XI of the ASME BPV Code.

Snubber inservice examination and testing provisions are specified in the editions and addenda of ASME BPV Code, Section XI, up through the 2005 Addenda. Snubber inservice examination and testing provisions were removed from ASME BPV Code, Section XI, in the 2006 Addenda. The ASME BPV Code, Section XI, option will no longer exist when using the 2006 addenda and later editions and addenda of ASME BPV Code, Section XI, because these editions and addenda of Section XI do not provide inservice examination and testing provisions for snubbers. When using the 2006 addenda or later editions of ASME BPV Code, Section XI, snubber examination and testing must be in accordance with the OM Code, Subsections ISTA and ISTD, or relief or an alternative must be obtained from the NRC.

A.1.3 <u>NRC Recommendations and Guidance</u>

The recommendations herein supplement the guidance provided in the OM Code and ASME BPV Code, Section XI, for inservice examination and testing of snubbers. This document is written based on the requirements as specified in the 2017 Edition of the OM Code and ASME BPV Code, Section XI, which are the latest edition of the OM Code and ASME BPV Code, Section XI, incorporated into paragraph (a) of 10 CFR 50.55a. To the extent practical, this document reflects the applicable section, subsection, or paragraph of the applicable documents (subsections of 10 CFR 50.55a, OM Code, ASME BPV Code, Section XI, regulatory guides, etc.).

The guidance presented herein is voluntary and may be used for requesting for requesting relief under 10 CFR 50.55a(g), or for authorization of an alternative under 10 CFR 50.55a(z). Licensees may also request relief or the use of an alternative which is not in conformance with this guidance. The NRC may grant relief or authorize the alternative if the licensee has addressed all of the aspects of the relief or alternative in an acceptable manner.

This document specifically discusses applicable portions of Article IWA-1000 and IWA-2000, IWA-4000, IWF-5000, and IWF-6000 of the ASME BPV Code, Section XI, and Subsections ISTA and ISTD, Nonmandatory Appendices A (with Supplement) through H of the OM Code, which licensees should implement pursuant to 10 CFR 50.55a(g)(4). It also provides guidance for licensees to use 10 CFR 50.55a(g)(4)(ii) and (iv) in updating their snubber inservice examination and testing program to the requirements of the ASME BPV Code, Section XI, or OM Code, as applicable.

A.1.4 <u>Synopsis of Report</u>

This appendix follows the format of a typical inservice examination and testing snubber program plan, including Development and Implementation, General Guidance, and Code Noncompliance.

Section A.2, “Developing and Implementing an Inservice Examination and Testing Program of Snubbers,” describes existing inservice examination and testing requirements, discusses the scope of the snubber program, and describes guidance for presenting information in snubber programs. This also provides specific recommendations on snubber and large bore (steam generator and reactor coolant pump) snubber related issues.

Sections A.5 and A.6 present a list of references related to snubbers.

These discussions are intended to clarify the existing requirements of the Code or the regulations and, as such, may provide recommendations to ensure that licensees continue to meet the Code and other regulatory requirements.

A.2 DEVELOPING AND IMPLEMENTING AN INSERVICE EXAMINATION AND TESTING PROGRAM OF SNUBBERS

Licensees may use the following guidance for developing and implementing snubber inservice examination and testing programs. This guidance supplements existing requirements and previously approved guidance on inservice examination and testing.

A.2.1 Compliance Considerations

The NRC regulations at 10 CFR 50.55a(g) state that ISI of components (including supports) which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions and addenda of the ASME BPV Code (or ASME OM Code for snubbers). This ISI along with testing is intended to assess the reliability and operational readiness of the snubbers.

Paragraph 50.55a(g)(4)(ii) requires that an inservice examination activity conducted during successive 120-month inspection must be conducted in compliance with the requirements of the latest edition and addenda of the ASME Code incorporated by reference in the version of 10 CFR 50.55a(a) that is in effect 18 months before the start of the interval and subject to the conditions listed in 10 CFR 50.55a(b). This Code Edition is called the "Code of Record."

The ASME OM Code up to the 2017 Edition specifies in ISTA-3200, "Administrative Requirements," that the Code user comply with the latest edition and addenda of the ASME OM Code adopted by the regulatory authority 12 months prior to the issuance of an Operating License or 12 months before initial fueling under a COL, and 12 months prior to the start of successful IST Program intervals. In that this NUREG cannot authorize alternatives to the ASME OM Code, licensees will need to submit an alternative request to follow the 18-month provisions in 10 CFR 50.55a(g)(4)(ii) rather than complying with the 12-month provisions in ISTA-3200, until the ASME OM Code is updated to reflect the 10 CFR 50.55a requirement.

10 CFR 50.55a(g)(4)(iv) allows inservice inspection of components (including supports) to meet the requirements set forth in editions and addenda subsequent to the "Code of Record," that are incorporated by reference in 10 CFR 50.55a(a), subject to conditions listed in 10 CFR 50.55a(b) and subject to NRC approval. Request for approval to use later editions and addenda previously incorporated by reference in 10 CFR 50.55a may be made via a letter to the NRC. See RIS 2004-12 for further clarification.

However, when requesting to use editions and addenda of the ASME Code (or portion thereof) that have not been incorporated by reference, licensees must first request authorization to use these later editions and addenda as an alternative to the regulations pursuant to 10 CFR 50.55a(z). When licensees choose to use any or all portions of a more recent edition, they must meet all related requirements of the respective editions or addenda, and such requests are subject to NRC approval in accordance with 10 CFR 50.55a(g)(4)(iv).

The NRC may authorize alternatives to Code examination and testing requirements submitted as alternative requests or in a similar format that includes a discussion of the requirements, a description of the proposed alternative, and the justification for authorization of the alternative. 10 CFR 50.55a includes the following provisions for authorizing alternatives or granting relief:

- 10 CFR 50.55a(z)(1) allows the NRC to authorize alternatives if the proposed alternatives would provide an acceptable level of quality and safety. The NRC will normally authorize an alternative pursuant to this provision only if the licensee proposes a method of examination or testing that is equivalent to, or an improvement of, the method specified by the Code, or if the examination or testing will comply or is consistent with later Code editions approved by the NRC in 10 CFR 50.55a(a).
- 10 CFR 50.55a(z)(2) allows the NRC to authorize an alternative if compliance with the Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The NRC may authorize an alternative pursuant to this provision if, although the proposed alternative testing does not comply with the Code, the increase in overall plant safety and quality attained by complying with the Code requirement is not justified in light of the difficulty associated with compliance.
- 10 CFR 50.55a(g)(6)(i) includes the following provision:

  The Commission will evaluate determinations that Code requirements of "Code of Record," per 10 CFR 50.55a(g) are impractical. The Commission may grant relief and may impose such alternative requirements as it determines is authorized by law, giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

The NRC may grant relief pursuant to this provision or may authorize alternatives if the licensee demonstrates that the design or access limitations make the Code requirement impractical. Thus, the NRC staff's evaluation considers the burden created by imposing the Code requirements on the licensee.

For plants using their TS to govern inservice examination and testing of snubbers, 10 CFR 50.55a(g)(5)(ii) requires that if a revised ISI program for a facility conflicts with the TS, the licensee shall apply to the NRC for amendment of the TS to conform the TS to the revised program. Therefore, when performing 120-month ISI program updates in accordance with 10 CFR 50.55a(g)(4), licensees must submit any required amendments to ensure their TS remain consistent with the new Code of Record or NRC-authorized alternative used in lieu of the Code requirements. The TS governing the snubber ISI and test program do not eliminate the 10 CFR 50.55a requirements to update the program at 120-month intervals or to request and receive NRC authorization for alternatives to the Code requirements when appropriate.

The NRC expects licensees to ensure that their snubber examination and testing programs are in compliance with 10 CFR 50.55a(g) (which points to ASME OM Code for snubber inservice examination and testing) or authorized alternatives. If licensees discover that their programs are not meeting 10 CFR 50.55a(g) requirements or authorized alternatives, they shall take appropriate actions to bring their programs back into compliance and ensure that non-compliant systems, structures and components are operable. In certain circumstances involving snubber programs at nuclear power plants that are not in compliance with NRC requirements, enforcement discretion has been provided by the NRC. The NRC's Office of Enforcement issued Enforcement Guidance Memorandum (EGM)-10-001, "Dispositioning Violation of Inservice Examination and Testing Requirements for Dynamic Restraints (Snubbers)," on June 1, 2010, to provide NRC staff guidance for the disposition of certain 10 CFR 50.55a violations and the potential of granting enforcement discretion for the affected requirements.

The NRC expects that licensees of nuclear power plants, who were not meeting the 10 CFR 50.55a requirements for snubber inservice examination and testing as described in RIS 2010-06, shall have entered any noncompliance into their corrective action program and corrected the noncompliance by meeting the applicable ASME Code requirements or by submitting for relief to the NRC.

A.2.1.1 ASME Code Case Applicability

Code Cases represent alternative or additions to the ASME Code. A Code Case is the official method of the ASME for handling a reply to an inquiry or alternative when study indicates that the Code wording needs clarification, or when the reply modifies the existing requirements of the Code, or grants permission to use alternative methods. ASME develops Code Cases through a consensus process to clarify the intent of existing Code requirements or to provide an alternative to a specific Code requirement. A Code Case may be issued for the purpose of providing alternative rules when justified, to permit early implementation of an approved revision when the need is urgent, or to provide rules not covered by existing provisions of the ASME Code.

The NRC reviews new or revised Code Cases to determine their acceptability for incorporation by reference in 10 CFR 50.55a through the regulatory guides. Accordingly, the NRC staff developed RG 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code,” as well as RG 1.193, “ASME Code Cases Not Approved for Use.”

The regulations at 10 CFR 50.55a(a)(3)(iii) incorporate by reference RG 1.192. Licensees may implement the Code Cases listed in RG 1.192 without obtaining further NRC review or approval if the Code Cases are used in their entirety with any supplemental conditions specified in RG 1.192 and the licensee’s ISI or IST Code of Record is applicable to the Code Case. RG 1.193 lists Code Cases not approved for general use. Licensees requesting the NRC’s approval to implement a Code Case listed in the RG 1.193 must show, at minimum, that adequate protection to public health and safety is provided if the Code Case is applied by the licensee/applicant.

If a licensee would like to use a Code Case with an Edition or Addendum of the Code to which it is not applicable, the licensee has the following options:

a. Request to use the Code Case as an alternative, beyond its stated applicability, and receive authorization by the NRC pursuant to 10 CFR 50.55a(z), or

b. If the Code Case is applicable to an Edition or Addendum of the Code later than the version of the Code being used by the licensee, the licensee could update their inservice examination and testing of snubber program to the later version of the Code pursuant to 10 CFR 50.55a(g)(4)(iv) and then use the Code Case, provided the Code Case has been approved for use in the appropriate RG 1.192 and incorporated by reference into 10 CFR 50.55a. Note that the later version of the Code must also have been incorporated by reference into 10 CFR 50.55a, the licensee must update the snubber program to all related requirements of the respective Edition or Addenda, and the use of the later Code must be specifically approved by the Commission.

The NRC may authorize the use of a Code Case that it has not yet been approved for use in RG 1.192 if a licensee requests the use of the Code Case under 10 CFR 50.55a(z). The authorization for a specific licensee to use a Code Case that is not listed in RG 1.192 does not authorize any other licensee to use the Code Case.

If RG 1.193 identifies a Code Case as being unacceptable, the NRC is unlikely to approve a licensee request to use that specified Code Case (whether by exemption, authorization of alternatives, or granting of relief). Licensees requesting the NRC's approval to implement a Code Case listed in the RG 1.193 must show, at minimum, that adequate protection to public health and safety is provided if the Code Case is applied by the licensee/applicant.

A.2.1.2 Conditions to the ASME OM Code

A.2.1.2.1 10 CFR 50.55a(b)(3)(v)—OM Condition: Subsection ISTD

Article IWF-5000, "Inservice Inspection Requirements for Snubbers," of the ASME BPV Code, Section XI, must be used when performing inservice examinations and tests of snubbers at nuclear power plants, except conditioned in paragraphs 10 CFR 50.55a(b)(3)(v)(A) and (B) as follows:

(A) Snubbers: First provision: Licensee may use Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants," of the ASME OM Code 1995 Edition through the latest edition and addenda incorporated by reference in 10 CFR 50.55a(a)(1)(iv), in place of requirements for snubbers in the editions and Addenda through the 2005 Addenda of the ASME BPV Code, Section XI, Articles IWF-5200(a) and (b) and IWF-5300(a) and (b), by making appropriate changes to technical specification (TS) or licensee-controlled documents. The regulation at 10 CFR 50.55a(b)(3)(v)(A) also states that, when using Subsection ISTD as an alternative "Preservice and inservice examination must be performed using the VT-3 visual examination method described in IWA-2213."

(B) Snubbers: Second provision. Licensees must comply with the provisions for examining and testing snubbers in Subsection ISTD of the ASME OM Code and make appropriate changes to their technical specifications or licensee-controlled documents when using the 2006 Addenda and later editions of Section XI of the ASME BPV Code.

This condition in 10 CFR 50.55a(b)(3)(v) allows licensees using 1995 Editions through the latest edition and addenda up to the 2005 Addendum of the ASME BPV Code, Section XI, to optionally use Subsection ISTD of the OM Code in place of the requirements for snubbers in Section XI. This condition also states that snubber preservice and inservice examinations must be performed using the VT-3 visual examination method (as described in IWA-2213) when using Subsection ISTD of the OM Code. The NRC imposed the VT-3 visual examination requirement to ensure that licensees use an appropriate visual examination method/scope for the examination of integral and nonintegral snubber attachments, such as lugs, bolting, and clamps, when using Subsection ISTD as an alternative to ASME BPV Code, Section XI.

Licensees that use the 2006 Addenda and later editions and addenda to Section XI of the ASME BPV Code must follow the requirements of Subsection ISTD of the OM Code for snubbers because snubber inservice examination and testing requirements have been deleted from the scope of ASME BPV Code, Section XI, in the 2006 Addenda. The condition in 10 CFR 50.55a(b)(3)(v) does not invoke the VT-3 visual examination requirement when licensees use the 2006 Addendum and later editions and addenda to Section XI because ASME revised Figure IWF-1300-1, "Illustrations of Typical Support Examination Boundaries," in the 2006 Addenda to Section XI to clarify that integral and nonintegral snubber attachments are within the scope of Section XI and not within OM Code scope. Therefore, the visual examination method specified in the 2006 Addendum and in later editions and addenda to ASME BPV Code, Section XI, applies to the examination of integral and nonintegral snubber attachments.

A.2.1.3 Snubber Program Controls

There are various types of supports used in a nuclear power plant to support piping systems and equipment. A snubber is considered a type of component standard support. Whereas, some licensees have incorrectly interpreted that the inservice examination and testing of snubbers is not a 10 CFR 50.55a requirement because (1) 10 CFR 50.55a(g) addresses components (including supports) without mentioning snubbers, (2) snubber examination and testing was historically covered by TS, and (3) Improved TS allow snubber examination and testing requirements to be relocated from the TS to the TRM. To clarify this, snubbers were mentioned for the first time in 10 CFR 50.55(g) in 2012. Licensees have the option to control the inservice examination and testing of snubbers through their TS or other licensee-controlled documents. For plants using their TS to govern the inservice examination and testing of snubbers, 10 CFR 50.55a(g)(5)(ii) requires that if a revised ISI program for a facility conflicts with the TS, the licensee shall apply to the Commission for the amendment of the TS to conform the TS to the revised program. When performing 120-month program updates in accordance with 10 CFR 50.55a(g)(4), licensees must submit any required amendments or any alternative requests to ensure that their TS remain consistent with the new ISI program. The TS, TRM, or other licensee-controlled documents governing the snubber inservice examination and testing program do not eliminate the 10 CFR 50.55a requirements to update the program at 120-month intervals in accordance with 10 CFR 50.55a(g)(4) or to request and receive NRC authorization for alternatives to the Code requirements when appropriate. The NRC issued RIS 2010-06, "Inservice Inspection and Testing Requirements of Dynamic Restraints (Snubbers)," to inform the licensees, and clarify NRC's rules and regulations regarding snubber inservice examination and testing, in accordance with 10 CFR 50.55a, at nuclear power facilities.

A.2.1.4 ASME/ANSI OM Part 4 and ASME OM Code

The 2005 Addenda and earlier Editions of the ASME BPV Code, Section XI, Article IWF-5000, provides "Inservice Inspection Requirements for Snubbers." IWF-5000 states that snubber examination and testing shall be performed in accordance with ASME/ANSI *Operation and Maintenance of Nuclear Power Plants,* Part 4 (OM Part 4). In 2006 addenda of the ASME BPV Code, Section XI, deleted snubbers examination and testing requirements. The current ASME BPV Code, Section XI, IWF-1220, Endnote 39, states that snubber examination and test requirements can be found in the ASME OM Code. The OM Part 4 and OM Code are two different ASME Codes. OM Part 4 and OM Code are not interchangeable. While using ASME BPV Code, Section XI, licensees must use OM Part 4 and while OM Code licensees must use Subsection ISTD.

A.2.1.5 Voluntary Use of the Later Edition and Addenda of the ASME Code

The regulations in 10 CFR 50.55a(f)(4) and (g)(4) establish the effective edition and addenda of the ASME OM Code or ASME BPV Code, as applicable, to be used by licensees in performing inservice testing of pumps and valves and inservice inspection of components (including supports). A snubber is a type of component standard support.

10 CFR 50.55a(f)(4)(ii) and (g)(4)(ii) require the use of the latest edition and addenda of the applicable ASME Code that have been incorporated by reference 18 months prior to the

beginning of each 120-month IST and ISI interval. The ASME Code edition or addenda identified as applicable to the 120-month IST and ISI interval is considered to be the "Code of Record" for the IST and ISI interval.

10 CFR 50.55a(f)(4)(iv) and (g)(4)(iv) allow IST of pumps and valves and ISI of components (including supports) to meet the requirements set forth in subsequent editions and addenda to the "Code of Record" that are incorporated by reference in 10 CFR 50.55a(a), subject to conditions listed in 10 CFR 50.55a(b) and subject to NRC approval. The fact that these ASME Code editions and addenda have been incorporated by reference into the NRC regulations does not imply that NRC approval has already been granted for their use in lieu of the "Code of Record" for the specific licensee. If a licensee plans to apply a later edition or addenda of the ASME Code either at the outset or during its 120-month IST or IST interval, the licensee must request NRC approval to use the later ASME Code editions and addenda via a letter to the NRC. The licensee may not update the IST and ISI programs without NRC approval to use the later ASME Code editions and addenda. When proposing to use any or all of a later edition or addenda, the licensee must meet all related requirements of the respective editions or addenda, and any deviations are subject to NRC approval in accordance with 10 CFR 50.55a(f)(4)(iv) and (g)(4)(iv). The regulations do not specify when the licensee should submit the request to apply a later edition or addendum of the ASME Code. The regulations only specify that the licensee must receive NRC approval before use of a later ASME Code edition or addendum.

The amount of written documentation needed to support a request to use a later ASME Code edition and addendum that 10 CFR 50.55a(a) incorporates by reference is significantly less than that necessary for other types of requests for relief from or alternatives to the ASME Codes. For example, licensees are not required to provide specific justification for requests to use later ASME Code editions and addenda that 10 CFR 50.55a(a) incorporates by reference. This is because the NRC has reviewed and accepted the provisions of those ASME Code editions or addenda, with any appropriate conditions, as part of the process for incorporation of the edition and addenda by reference in the regulations.

The purpose of the NRC review and approval for the use of later ASME Code editions and addenda is to provide and monitor consistency in the use of the appropriate ASME Code editions and addenda in IST and ISI activities of nuclear plant components.

When requesting to use editions and addenda of the ASME Code that have not been incorporated by reference into 10 CFR 50.55a, licensees must first request authorization to use these later editions and addenda as an alternative to the regulations pursuant to 10 CFR 50.55a(z).

The NRC staff issued RIS 2004-12 and RIS 2004-16, "Use of Later Editions and addenda to the ASME Code Section XI for Repair/Replacement Activities," dated October 19, 2004, to clarify this matter.

A.2.2 Scope of Inservice Examination and Testing Programs

General Design Criterion (GDC) 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Plants," to 10 CFR Part 50 requires that all structures, systems, and components that are necessary for safe operation must be tested to demonstrate that they will perform satisfactorily in service. Among other things, GDC 1 requires that components that are important to safety must be tested to quality standards that are commensurate with the importance of the safety function(s) to be performed. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 describes the requisite quality assurance (QA) program, which includes testing, for safety-related components. In addition, 10 CFR 50.55a(g) requires that licensees must use the ASME BPV Code, Section XI, or the OM Code for inservice examination and testing of components that are covered by the Code. Each licensee has the responsibility to demonstrate the continued capability of all components within the scope of their snubber inservice examination and testing program. Regulatory guides provide additional NRC guidance regarding scope and classification. In short, the ASME Code defines the scope, 10 CFR 50.55a incorporates by reference the Code with conditions, and regulatory guides provide additional guidance. For pre-GDC plants, the licensee should verify the scope of the ISI and IST program is consistent with the applicable licensing requirements.

Each licensee has the responsibility to demonstrate the continued functionality of all snubbers within the scope of their snubber examination and testing program. The NRC staff considers the ASME Code to apply to dynamic restraints (snubbers) installed in piping systems. The snubber program, including implementing procedures, is subject to the requirements of 10 CFR Part 50, Appendix B, and OM Code, Subsection ISTA, or ASME BPV Code, Section XI, Article IWA. Changes to the scope, test methods, or acceptance criteria shall be reviewed under the requirements of 10 CFR 50.59, "Changes, Tests, and Experiments," 10 CFR 50.55a, and 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (Maintenance Rule), as appropriate.

A.2.2.1 Basis for Scope Requirements

The requirements for the scope of components to be included in an inservice examination and testing of snubber program are addressed in 10 CFR 50.55a(g)(4), and OM Code.

The NRC regulations in 10 CFR 50.55a(g)(4), "Inservice testing standards requirement for operating plants," state the following:

> Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) that are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements OM Code must meet the inservice test requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions and addenda of the ASME BPV Code (or ASME OM Code for snubber exanimation and testing) that become effective subsequent to editions and addenda specified in paragraphs (g)(2) and (3) of this section and that are incorporated by reference in paragraph (a)(1)(ii) or (iv) for snubber examination and testing of this section, to the extent practical within the limitations of design, geometry, and materials of construction of the components. Components that classified as Class MC pressure retaining components and their integral attachments, and components that are classified as Class CC pressure retaining components and their integral attachments, must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of the ASME BPV Code and addenda that are incorporated by reference in paragraph (a)(1)(ii) of this section, subject to condition listed in paragraph (b)(2)(iv) of this section and conditions listed in paragraphs (b)(2)(viii) and (ix) of this section, to the extend within the limitation of design, geometry, and materials of construction of the components.

The OM Code defines the scope by stating that IST programs (snubber programs) shall include snubbers that are required to perform a specific function in (1) shutting down the reactor to a safe shutdown condition, (2) maintaining the safe shutdown condition, (3) mitigating the consequences of an accident, and (4) ensuring the integrity of the reactor pressure boundary. Therefore, the scope of components to be included in a snubber inservice examination and testing program must encompass the components covered in Subsection ISTA of the OM Code.

Paragraph ISTA-1100 of the OM Code refers to components that are “needed to mitigate the consequences of an accident.” The NRC regulations in 10 CFR 50.2, “Definitions,” define safety-related SSCs as follows:

> Safety-related structures, systems and components means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:
>
> (1) The integrity of the reactor coolant pressure boundary,
>
> (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
>
> (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

The language of ISTA-1100 and 10 CFR 50.2 is consistent with respect to mitigating the consequences of accidents with the exception that ISTA-1100 does not include specific references to exposure guidelines provided in the NRC regulations. Therefore, the NRC staff interprets the paragraph in ISTA-1100 regarding snubbers that are required to perform a specific function in (1) shutting down the reactor to a safe shutdown condition, (2) maintaining the safe shutdown condition, (3) mitigating the consequences of an accident, or (4) ensuring the integrity of the reactor coolant pressure boundary to be consistent with the definition of safety-related as specified in the NRC regulations. Licensees and applicants could use their identification of the safety-related snubbers with respect to design basis events in the FSAR for their nuclear power plants to identify the snubbers within the scope of this paragraph in ISTA-1100. Similarly, other events such as anticipated transient without scram (ATWS), station blackout (SBO), and low temperature overpressure protection (LTOP), might have NRC requirements for components relied upon to mitigate their events that are safety-related and not within the scope of the OM Code.

Snubbers within the scope of 10 CFR 50.55a are included in the scope of 10 CFR 50.65. Licensees may elect to consolidate testing for snubbers, designating any non-Code components as such in the snubber program.

Tables A-1 and A-2 (which appear at the end of Section A.2) provide examples of systems with snubbers that licensees typically include in their IST program. These tables may be used for developing a licensee’s snubber examination and testing program. These tables are not intended to be all-inclusive, but they may form the basis for the initial review of a licensee’s snubber program scope.

Figure A2.1, “Flow Chart – Development of Preservice and Inservice Examination and Testing Program for Snubbers,” (which appears at the end of this chapter) provides a quick reference to regulatory requirements for development of the inservice examination and testing program for snubbers. For complete details, see 10 CFR 50.55a.

Figure A2-2, “Various ASME Section XI Code Versions Flowchart – Use of ASME OM Code for Snubber Program,” (which appears at the end of this chapter) provides a quick reference to the regulatory requirements of ASME BPV Code, Section XI, and OM Code for the inservice examination and testing program for snubbers. For complete details, see 10 CFR 50.55a.

A.2.2.2 Snubber Attached to Steam Generator and Reactor Coolant Pumps

There are special requirements for pressurized water reactor (PWR) plants with regard to the testing of snubbers attached to the Steam Generators (SGs) and Reactor Coolant Pumps (RCPs). These are generally large bore hydraulic snubbers (LBHSs). Large bore hydraulic snubbers are defined as those units with rated capacities of 50 kips or greater. Unlike smaller hydraulic snubbers, LBHSs were exempt from inservice functional testing prior to 1980, primarily due to a lack of available test equipment of sufficient size. In 1980 and 1984, the NRC issued Generic Letters to all licensees requesting modification of plant TS to include LBHSs testing provisions. The results of initial tests revealed numerous cases where LBHSs were either out of specified tolerance or completely inoperable. Subsequently, the NRC developed Generic Issue (GI) 113), “Dynamic Qualification and Testing of Large Bore Hydraulic Snubbers (LBHSs),” with the objective of evaluating the reliability of LBHSs in operating commercial nuclear power plants.

NUREG/CR-5416, “Technical Evaluation of Generic Issue 113: Dynamic Qualification and Testing of Large Bore Hydraulic Snubbers,” dated August 1992, provided major recommendations for LBHSs. This effort was in coordination with the industry, vendors, and snubber manufacturers. As a result, the NRC established specific IST recommendations for LBHSs installed on PWR steam generators or RCPs. The NRC recommendation was that these snubbers be tested as a separate test population, and a requirement was eventually incorporated into the OM Code to ensure this was accomplished. Although the Generic Issue focused on hydraulic snubbers, the resulting requirement does not specify a snubber size or type to which it applies.

Therefore, all licensees are reminded that 10-year inservice examination and test snubber programs shall include their SG snubbers and RCP snubbers, regardless of size or type. The Code requirement is that snubbers attached to SGs and those attached to RCPs be designated as at least one, separate Defined Test Plan Group (DTPG) for testing purposes as specified in ISTD-5353. Large bore snubbers (greater than 50 kips) located on piping or other components may be included in the general snubber population for testing and examination purposes.

A.2.2.3 Testing of non-Code Snubbers

Licensees are required to test safety-related components to demonstrate that they will perform satisfactorily in service in accordance with 10 CFR Part 50, Appendices A and B. Regulations in 10 CFR 50.55a address the inservice inspection and inservice testing program for components within the scope of the ASME Code.

An inservice examination and testing program is also a reasonable vehicle to periodically demonstrate the operational readiness of snubbers that are not covered by the Code, but are within the scope of 10 CFR Part 50, Appendices A and B. Thus, if a licensee chooses to include non-Code snubbers in its ASME Code inservice examination and testing program (or some other licensee-developed examination and testing program) and is unable to meet certain Code provisions for the non-Code components, the regulations (10 CFR 50.55a) do not require the licensee to submit a relief request to the NRC. Nonetheless, the licensee should maintain documentation that provides assurance of the continued functionality of the non-safety components through the performed tests, and such documentation should be available for NRC staff inspection at the plant site.

Therefore, while 10 CFR 50.55a delineates the examination and testing requirements for snubbers, licensees should not limit their inservice examination and testing to only those snubbers that are covered by the scope of OM Code. For example, 10 CFR Part 50, Appendices A and B, contain requirements for examination and testing of components that may not be specifically addressed by 10 CFR 50.55a or OM Code. Rather than implement separate programs for the Code versus non-Code snubbers, licensees may choose to incorporate the non-Code snubbers into a single program along with the Code snubbers.

A.2.2.4 Use of ASME BPV Code, Section XI, IWF-1230, "Support Exempt from Examination," to Exempt Snubbers from Inservice Examination and Testing in Snubber Program

ASME BPV Code, Section XI, IWF-1230, "Supports Exempt from Examination," states that supports exempt from the examination requirements of Article IWF-2000, "Examination and Inspection," are those connected to piping and other items exempt from volumetric, surface, or VT-1 or VT-3 visual examination by IWB-1220, IWC-1220, IWD-1220, and IWE-1220. In addition, the portion of supports that are inaccessible by being encased in concrete, buried underground, or encapsulated by guard pipe are also exempt from the examination requirements of Article IWF-2000. The 2005 Addenda and earlier Editions of the ASME BPV Code, Section XI, IWF-2100, states that "The requirements of this Article IWF-2000 apply to the examination and inspection of component supports, but not to the inservice test requirements of IWF-5000 (snubbers)."

Therefore, the use of IWF-1230 for exemptions within the snubber program without considering other regulatory requirements might exempt certain snubbers intended to be in the scope of IWF-5000 requirements when using 2005 Addenda and earlier Edition of the ASME BPV Code, Section XI (i.e., some safety-related snubbers or non-safety-related snubbers which are important to safety). NRC staff concludes that IWF-1230 shall not be used to exempt snubbers. Similarly, IWF-1230 shall not be used to exempt snubbers when using the OM Code for inservice examination and testing. Any deviation of snubber scope of Snubber Program from ISTA-1100 of the OM Code and other regulatory requirements must be approved as an alternative request by NRC.

A.2.2.5 Reserved

A.2.3 Code Class Systems Containing Safety-Related Snubbers

The plant safety analysis report (SAR), technical specifications (TS), and other documents list the systems and components (i.e., snubbers) that must function to support the safe operation and shutdown of the plant. Tables A-1 and A-2 (which appear at the end of this Appendix A chapter) provide examples of systems with snubbers that licensees typically include in their IST program for PWRs and boiling-water reactors (BWRs). These tables may be used for developing licensee's snubber examination and testing program. These tables are not intended to apply to all plants. The listed systems and components are not considered safety-related at every plant, and are not necessarily classified as Code Class 1, 2, or 3. (For information on quality group and Code classifications, see RG 1.26 and Section 3.2.2 of NUREG-0800.) The licensee's safety analysis generally contains a section describing the Code classification of components. The snubber inservice examination and testing program scope must be consistent with the SAR.

A.2.4 Snubber Inservice Examination and Testing Programs and their Documentation

10 CFR 50.55a(g)(4) states, in part, that throughout the service life of a boiling or pressurized water-cooled nuclear power facility, ASME Code Class 1, 2, and 3 components (including supports) must meet the ISI and testing requirements of the ASME BPV Code, Section XI (or ASME OM Code for snubber inservice examination and testing) as incorporated by reference in 10 CFR 50.55a(a). In order to meet the requirements of the ASME OM Code, licensees shall establish a snubber inservice examination, testing and service life monitoring program to verify the operational readiness of snubbers. This snubber program shall include (1) Visual examination, (2) Operational readiness functional testing, and (3) SLM requirements as specified in the OM Code.

A.2.4.1 Snubber Program while using the 2005 Addenda and Earlier Edition of the ASME BPV Code, Section XI

Licensees using ASME BPV Code, Section XI, for the snubber inservice examination and testing program need to consider the following rules for inservice examination and testing of snubbers:

- Subsection IWA addresses the general requirements for inservice examination and testing of snubbers.
- Article IWF-5000 addresses inservice examination and testing requirements for snubbers.
- Subarticle IWA-1400(c) requires that the Owner shall submit certain plans and reports to the enforcement and regulatory authorities.
- Article IWA-4000 provides the requirements for Repair/Replacement activities, including snubbers.
- Article IWA-6000 addresses the records and reports that are required for the examination and testing programs of snubbers.
- Paragraph IWA-6210 states that the Owner shall prepare plans for preservice and inservice examinations and tests to meet the requirements of the ASME BPV Code, Section XI, requirements. Article IWF-1000 addresses the scope and responsibility for inservice inspection of snubbers as a subset of components support examination.
- OM Code, Subsections ISTA and ISTD, may be used in lieu of the ASME BPV Code, Section XI, Article IWF-5000, as allowed by 10 CFR 50.55a(b)(v)(A).

A.2.4.2 Snubber Program while using OM Code

Licensees using the OM Code for the snubber inservice examination and testing program needs to consider the following rules for inservice examination and testing of snubbers.

- Subsection ISTD addresses the “Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Water-Cooled Reactor Nuclear Power Plants.”
- Subsection ISTA includes general requirements (including scope) for inservice examination and testing of snubbers.
- Paragraph ISTA-3200 states that IST plans shall be filed with the regulatory authorities. This includes snubber examination and testing plans.
- Section ISTA-9000 addresses the records and reports that are required for these examination and testing programs.
- Paragraph ISTA-9210 states that the Owner shall prepare plans for preservice and inservice examinations and tests to meet the requirements of the OM Code.
- Paragraph ISTA-9220 states that licensees shall prepare examination, and test records in accordance with the requirements of this article in conjunction with the applicable Subsection of the OM Code (i.e., ISTD).
- Nonmandatory Appendix A, and the Supplements to Nonmandatory Appendix A to the OM Code describe voluntary guidance for licensees to use in preparing their examination and test plans.

A.2.4.3 Snubber Program while using NRC Authorized Alternative or Relief

Licensees not using the ASME BPV Code, Section XI, or OM Code requirements for their snubber program, must submit a request for relief from or an alternative to the ASME BPC Code, Section XI, and OM requirements. NRC-authorized alternatives to use TRM or other licensee-controlled documents, in lieu of the ASME BPV Code, Section XI, or OM Code requirements for inservice examination and testing of snubbers, do not preclude the need for licensees to submit snubber examination and test plans to the regulatory authorities as defined in IWA-1400(c) and ISTA-3200. An NRC-authorized alternative is only applicable for the 10-year program interval for which it is requested and only to the requested sections or paragraphs of the ASME Code, not from the entire ASME Code.

A.2.4.4 Snubber Programs and Their Bases

Some licensees are using TRMs or other licensee-controlled documents for snubber inservice examination and testing in lieu of the ASME BPV Code, Section XI, or OM Code requirements. TRMs or other licensee-controlled documents serve as bases for most snubber programs and most of the snubber programs have similarities across the industry. Many licensees are in the process of updating their snubber programs to comply with and incorporate the OM Code. Many licensees have already updated their programs to use the OM Code. As a minimum, the updated snubber program documentation shall contain sufficient information to verify alignment with the OM Code requirements. Bases documents have typically included a description of the methodology used in preparing the snubber program plans. The bases document shall clearly state where and how a list of program snubbers is kept and maintained. Although not required by the regulation, the bases documents will help licensees ensure the consistent implementation of their snubber programs throughout the course of typical organizational (including personnel) changes. A good bases document will also enable the plant staff to clearly understand the snubber categorization process, as well as the basis for examination and testing requirements. The bases document can also serve as a useful reference for reviews performed under 10 CFR 50.59 when changes are made to a facility.

As a minimum, the following three elements as described in the OM Code should be addressed in a typical snubber program bases document:

1. Visual Examination Requirements
2. Functional Testing Requirements
3. Service Life Monitoring Requirements

Individual aspects of each element should be detailed as outlined in the following sections in order to provide clear definitions and descriptions of the program elements. Information clarifying the basis for inclusion in the program should be provided, including references to applicable Code sections or licensing commitments where appropriate.

To help ensure consistency throughout the industry, licensees are encouraged to use these guidelines when developing snubber programs and their bases.

A.2.4.4.1 Inservice Visual Examination (description, definitions, and basis of each item)

Licensees should address at least the snubber inservice visual examination items as follows:

1. Addressing integral and nonintegral attachments to snubbers (see Section A.2.8)
2. Snubber Categorization
3. Subsequent Examination Intervals
4. Inservice Examination Sample Size
5. Method of Visual Examination (Owner's procedure of optional VT-3)
6. Inservice Examination Failure Evaluation
7. Inservice Examination Corrective Action
8. NRC authorized alternative (or relief request), if applicable
9. Code Case used for visual examination, if applicable

A.2.4.4.2 Inservice Operational Readiness or Functional Test (description, definitions, and basis of each item)

Licensees should address at least the snubber inservice operational readiness or functional test items as follows:

1. Functional Test Frequency
2. Defined Test Plan Group (DTPG)
3. Testing Sample Plan (10% testing sample, or 37 testing sample plan) used and Initial Snubber Sample size(s) anticipated for each DTPG (It is recognized that 10% Plan samples may vary slightly over the course of an interval due to ongoing station

modifications or replacement. The bases document may be periodically updated for significant changes but is not expected to be a day-to-day “living” document.)

4. Additional Sample Size for each plan (based on selected sample plan on 10% testing sample plan, or 37 testing sample plan)
5. Failure Evaluation requirements and methods (reference applicable document)
6. Test Failure Mode Groupings (FMG) methodology (reference applicable document)
7. Corrective Actions for each sample plan and FMG identified (reference applicable document)
8. NRC authorized alternative (or relief request), if applicable
9. Code Case used for functional testing of snubbers, if applicable

A.2.4.4.3 Service Life Monitoring Program

Licensees are responsible to establish, maintain, and implement a service life monitoring (SLM) program for all the snubbers in the scope of 10 CFR 50.55a(g) to optimize and supplement (1) ISTD-4200, “Visual Examination,” which is performed only once in 24 months, or 48 months or 10 years based upon plant performance, and (2) ISTD-5200, “Operational Testing,” which is performed on only 10 percent of selected snubbers or 37 snubbers. The SLM program shall be based on the snubber manufacturer’s recommendation, operating environment, snubber design limits, snubber type (mechanical or hydraulic), modification and maintenance history, and test records of snubbers. The SLM program is strengthened by use of ISTD-6100 through ISTD-6500 requirements, and additional visual examination and operational readiness testing of snubbers beyond the requirements of ISTD-4200 and ISTD-5200, and/or optional condition monitoring activities (such as stroking and/or greasing of mechanical snubbers).

The licensees must develop the SLM program as defined in the OM Code, section ISTD-6000, or in accordance with an approved alternative or relief request. SLM must consider all the maintenance record data available for snubbers while evaluating or reevaluating the service life. Nonmandatory Appendix F of Subsection ISTD provides additional guidance in developing a SLM Program. The SLM program is the primary instrument for assuring continued reliability of a snubber population at a plant. The statistical method of sample testing provides point-in-time assessment of population functionality, but in general does not serve as an effective tool to either maintain or improve reliability. This is due to the fact that such functional testing is based on small samples (10% or 37 snubbers) on a periodic basis, and is not predictive in nature. Based on snubber aging study information, discussed in NUREG/CR-5870, “Results of LWR Snubber Aging Research,” dated May 1992, the NRC recommended the inclusion of SLM of snubbers in addition to the statistical testing process. Most licensees have included some reference to SLM in their existing programs. Subsection ISTD of the OM Code also included SLM along with snubber examination and inservice testing requirements. Many of the licensees have updated their snubber programs to incorporate OM Code requirements. The updated snubber programs often simply reference plant procedures for snubber examinations and testing without providing any references to applicable sections of the OM Code. Program documentation is expected to provide information regarding specific SLM requirements and how the requirements are satisfied.

The records of all activities (i.e., repair, replacements, maintenance, corrective action work, failures, etc.) related to all snubbers must be documented and considered for the SLM program.

A.2.4.5 Snubber List or Snubber Controlled Data Bases

In preparing and maintaining a snubber list or data base, licensees are expected to consider the ability to produce reports providing adequate information to both implement and assess the program. Reports generated to provide a snubber listing might include the following suggested headings, which are shown along with a description of the information that licensees might produce under each heading.

Title: Report name, including the applicable plant and unit.

Reference information: Include references to maintaining and location of controlled snubber data. This may be a controlled station data base from which reports and lists are generated as needed.

Program/Report revision or revision date: List the revision number and date/or date (on each page).

System, Code class, and group: List applicable information such as plant system, ASME Code Class, and type of snubber (hydraulic or mechanical).

Snubber identification: List a unique identifier for each snubber; this identifier should be used consistently in all snubbers' inservice examination and testing program documentation and design information such as system piping and instrument diagrams (P&IDs), isometric, test procedures, and relief requests.

Drawings number: List the applicable isometrics, support drawings or figures that depict the snubber.

The following items might not normally be included in a list of individual snubbers, as they typically apply to entire populations. This information should be included as annotations for any individual snubbers for which a specific frequency or relief request applies on a unique basis.

Test frequency: List the actual frequency for each examination and test to be performed.

Relief request(s): List any applicable relief requests in the snubber list.

A.2.4.6 Snubber Program Plan Documentation and Their Submittal to NRC

10 CFR 50.55a(g)(4) requires that, throughout the service life of a boiling or pressurized water-cooled nuclear power facility, ASME Code Class 1, 2, and 3 components (including supports) meet the requirements of the ASME BPV Code, Section XI (or ASME OM Code for snubbers examination and testing) as incorporated by reference in 10 CFR 50.55a(a). The applicable OM Code, Subsection ISTA, provide the documentation and submittal requirements for inservice testing and examination of certain components in light-water nuclear power plants. Therefore, based on these requirements, licensees are required to submit their snubber examination and testing plans and their updates every 120 months.

(a) Documentation requirements for snubber program plan when using the 2005 Addenda and earlier Edition of the ASME BPV Code, Section XI

IWA-1400(c) notes that Owners have the responsibility to prepare plans, schedules, and inservice inspection summary reports, and submit these plans and reports to the enforcement and regulatory authorities having jurisdiction at the plant site.

Article IWA-6000, “Record and Reports,” provides the requirements for preparation, submittal, and retention of records and reports.

(b) Documentation requirements for snubber program plan when using the OM Code

ISTA-3200(a) requires that IST plans for inservice examination and testing of snubbers shall be filed with the regulatory authorities having jurisdiction at the plant site. This requirement is currently in the OM Code, but the ASME is removing this requirement beginning with the 2020 Edition of the OM Code. The NRC staff will propose a regulatory requirement for submittal of the IST Program Plan by nuclear power plant licensees in a 10 CFR 50.55a rulemaking

ISTA-9000, “Records and Reports,” provides the requirements for preparation, submittal, and retention of records and reports.

Nonmandatory Appendix A and the Supplement to Nonmandatory Appendix A describe voluntary guidance for licensees to use when developing snubber inservice examination and testing plans.

(c) Documentation requirements for snubber programs when using NRC authorized alternative to use TRM or other licensee controlled documents in lieu of the ASME BPV Code, Section XI, or ASME OM Code.

NRC-authorized alternatives to use TRMs or other-licensee-controlled documents, in lieu of the ASME BPV Code, Section XI, or OM Code requirements for inservice examination and testing of snubbers, do not provide relief from submitting snubber examination and test plans and reports to the regulatory authorities. Submittal is required by the applicable ASME BPV Code, Section XI, or OM Code as noted in (a) and (b) above.

A.2.5 Relief Requests and Proposed Alternatives

Relief Requests

Licensees can request that the NRC grant relief from an ASME BPV Code, Section XI, or OM Code requirement in accordance with 10 CFR 50.55a(g)(5)(iii), 10 CFR 50.55a(g)(5)(iv), 10 CFR 50.55a(f)(5)(iii), and 10 CFR 50.55a(f)(5)(iv). Such requests made under what are called "relief."

The regulations at 10 CFR 50.55a(g)(4) require licensees to examine and test snubbers in the snubber program to the "extent practical" within the limitations of the design, geometry, and materials of construction. The regulations at 10 CFR 5.55a(f)(5)(iii)–(iv) and 10 CFR 50.55a(f)(6)(i) use the term "impractical" instead of "extent practical." The terms "extent practical" and "impractical" apply to inservice examination and test requirements in the ASME BPV Code, Section XI, or OM Code that licensees cannot perform because of the design, geometry, and materials of construction.

In accordance with the regulations, when updating a program to a later edition of the ASME BPV Code, Section XI, or OM Code, licensees must implement the updated program at the beginning of a 120-month interval. The regulations state that in cases in which a licensee determines that an ASME BPV Code, Section XI, or OM Code-specified snubber examination and test is impractical and is not included in the revised snubber program, it must submit a relief request demonstrating the basis for its determination to the NRC no later than 12 months after the previous 120-month interval ends, or 12 months after the current interval starts. However, experience has shown that licensees also identify impractical test provisions throughout the interval. In such cases, licensees may request relief as soon as they identify the condition. Because the ASME BPV Code, Section XI, or OM Code requirements are impractical, the licensee would examine and test the applicable snubbers using the method proposed in the relief request in the period of time from the beginning of the new interval or time of discovery (or from the time of identification) using the method granted by the NRC for the applicable 10-year snubber program interval.

Alternative Requests

Licensees can request that the NRC authorize an alternative to an ASME BPV Code, Section XI, or OM Code, requirement in accordance with 10 CFR 50.55a(z). Requests made under 10 CFR 50.55a(z) are called "alternatives."

The ASME BPV Code, Section XI, and OM Code establish the requirements for inservice examination and testing of snubbers to assess their operational readiness in light-water reactor nuclear power plants. The requirements are constantly being reviewed and improved in order to meet the basic function of maintaining the safe and reliable operation and maintenance of nuclear power plants.

It is understood that not all plants are designed the same. It is also understood that the general requirements developed in the ASME BPV Code, Section XI, or OM Code may not be applicable or that complying with these requirements may be difficult. Licensees may propose alternatives to the ASME Code provided that (1) the alternative would provide an acceptable level of quality and safety under 10 CFR 50.55a(z)(1); or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety under 10 CFR 50.55a(z)(2). Hardships generally involve reductions in radiation exposure to as low as reasonably achievable (ALARA), challenges to operators or plant equipment, components that are somewhat unique in design.

Licensees must not implement proposed alternatives to the ASME BPV Code, Section XI, or OM Code requirements under 10 CFR 50.55a(z) until the NRC staff completes its evaluation and authorizes the alternative.

On rare occasions, the NRC may grant verbal authorizations as an alternative under 10 CFR 50.55a(z) when, because of unforeseen circumstances, licensees need NRC authorization before the agency is able to issue its written safety evaluation as described in NRC Office Instruction LIC-102, Revision 3 "Review of Relief Requests, Proposed Alternatives, and Requests to Use Later Code Editions and Addenda," (ADAMS Accession No. ML18351A218). For more details, see Section 2.5.5, "NRC Temporary Verbal Authorization of an Alternative Request", of the main section of the NUREG-1482, Revision 3.

Currently, most of the licensees are using the OM Code to meet the requirements of 10 CFR 50.55a for snubber inservice examination and testing, whereas few licensees are using a variety of licensee-controlled documents or procedures in lieu of the applicable ASME Code requirements. These licensee-controlled documents or procedures include the following:

1. Technical Specification (TS)
2. Technical Requirement Manual (TRM)
3. Final Safety Analysis Report (FSAR)
4. Updated Final Analysis Report (UFSAR)
5. Selected Licensee Commitment (SLC)
6. Licensee-Controlled Specification (LCS)
7. Equipment Control Guidelines (ECG)
8. Other Licensee-Controlled Procedures.

In 2008, the NRC staff identified several instances in which nuclear power plant licensees have used a TRM, or other licensee-controlled documents and procedures, which do not meet requirements of their "Code of Record" for the ISI and testing of snubbers. These licensees have not requested approval to use these alternatives from the NRC. The NRC issued RIS 2010-06 on June 1, 2010, to inform licensees of the NRC's rules and regulations regarding snubber inservice examination and testing, in accordance with 10 CFR 50.55a(g), at nuclear power plants.

For more details about contents and format, see Nuclear Energy Institute (NEI) document, "Standard Format for Requests from Commercial Reactor Licensees Pursuant to 10 CFR 50.55a," Revision 1, dated June 2004 (ADAMS Accession No. ML070100400).

In addition, it may be inappropriate to make substantive changes to licensing documents under the 10 CFR 50.59 process. Any changes to TRMs or other licensing documents must be evaluated in view of the original relief request and assessed as to the appropriate change process.

Approved alternative and relief requests are applicable only for the 10-year interval for which alternative or relief request was submitted and approved. If a licensee desires to continue using the alternative or relief request beyond that interval, a new request must be submitted for the new interval.

A.2.6 Snubber Program Plan and its Update Documents

As described in Section A.2.4.6 above, the inservice examination and testing program plans for snubbers required by the ASME Code must be submitted to the NRC. Snubber inservice examination and testing program plans submitted to the NRC are used to prepare for NRC inspections and to address other licensing actions that may arise. To facilitate these regulatory activities, the NRC requests to receive up-to-date program plan documents when the licensee makes significant changes to the snubber inservice examination and testing program plan in the interim period between the required 10-year interval plan submittals. These interim informational submittals are generally considered "good faith" and not a regulatory requirement. As long as the snubber inservice examination testing program plan remains consistent with the regulations, ASME Code relief or alternative requests are not required for these interim updates. That is, deletions from or additions to the snubber program do not necessarily require NRC approval, unless commitments to obtain such approval exist as a result of prior approved requests or similar commitments. The burden is on each licensee to verify that its snubber program is complete and includes all snubbers that require inservice examination and testing. If a licensee deletes a particular snubber from its snubber program plan, the NRC staff recommends that the licensee document the basis in an appropriate manner. Such changes do not require approval, unless cumulative changes affect the examination or testing plans in a significant manner.

The NRC staff expects each licensee to maintain its snubber examination and testing program plan up-to-date and ensure that it remains consistent with changes in plant configuration. Conversely, if a system modification results in the addition of a snubber to the snubber program plan, the licensee needs to ensure that it is incorporated into the program to satisfy the Code and licensing requirements or that a relief or alternative request is submitted for NRC review and approval, as appropriate.

As required by 10 CFR 50.55a(g), the Snubber Program Plan and its update should be developed and aligned with the plant's 10-year ISI program interval. Currently, the industry is transitioning and updating their snubber programs from the ASME BPV Code, Section XI, to the OM Code. Therefore, the Snubber Program Plan may be aligned with 10-year IST program interval. Licensees should provide details about this in the snubber program and the applicable plant's 10-year IST program interval or 10-year ISI program interval document with the start and end dates.

A.2.7 Repair and Replacement of Snubbers

The OM Code, paragraphs ISTD-1500 and ISTD-1600, state that snubber repair and replacement activities shall be performed in accordance with the applicable Section XI of the ASME BPV Code. Therefore, the repair and replacement of snubbers are to be performed using applicable edition and addenda of the ASME BPV Code, Section XI, Article IWA-4000. The NRC-approved Inservice Inspection Code Cases in RG 1.147, such as N-508-3, "Rotation of Serviced Snubbers and Pressure Relief Valves for the purpose of Testing, Section XI, Division 1," may be used.

A.2.8 ISI of the Integral and Non-Integral Attachments of Supports Containing Snubbers

Inservice inspection of integral and non-integral attachments, such as lugs, bolts, pins, and clamps, must be performed by use of ASME BPV Code, Section XI, Subsection IWF, or other applicable Subsections. Subsection ISTD of the OM Code covers inservice examination and testing of snubbers (pin to-pin) inclusive, and does not address the inspection of integral and non-integral attachments, such as lugs, bolting, pins, and clamps.

A.2.9 Developing Snubber Program for New Nuclear Power Plants

Under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," the development of a plant-specific IST program (or snubber examination and testing program) is the responsibility of the applicant for a combined construction and operating license (COL) for a nuclear power plant. The Commission's Staff Requirements Memorandum (SRM), dated September 11, 2002, for Commission Paper SECY-02-0067, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Operational Programs (Programmatic ITAAC)," stated that ITAAC for an operational program are unnecessary if the program and its implementation are fully described in the COL application and found to be acceptable by the NRC. The Commission also stated that the burden is on the COL applicant to provide the necessary and sufficient programmatic information for approval of the COL without ITAAC. In its May 14, 2004, SRM for SECY-04-0032, "Programmatic Information Needed for Approval of a Combined License Without Inspections, Tests, Analyses and Acceptance Criteria," the Commission defined "fully described" as meaning that the program is clearly and sufficiently described in terms of the scope and level of detail to allow a reasonable assurance finding of acceptability. The Commission also noted that required programs should always be described at a functional level and at an increasing level of detail where implementation choices could materially and negatively affect the program effectiveness and acceptability. SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," summarizes the NRC position regarding the full description of operational programs to be provided by COL applicants.

COL applicants provide a description of the snubber examination and testing program for NRC review as part of their COL applications. In some cases, the COL applicant incorporates by reference in its FSAR the description of the snubber examination and testing program provided in the Design Certification documentation, such as the Design Control Document (DCD) or Design Certification FSAR. To date, COL applicants have described their snubber examination and testing program based on the requirements in the OM Code, Subsection ISTD. The guidance in this NUREG may be used in developing and implementing the IST program including the snubber inservice examination and testing program for new nuclear power plants.

The NRC staff reviews the description of the IST program including snubber examination and testing in the COL application, with its incorporation by reference of IST provisions in the applicable design certification documentation, as part of the safety evaluation for the COL application. The NRC staff will conduct inspections of the development and implementation of the IST program following COL issuance.

A.2.10 **<u>Technical Specification (TS) Improvement to Modify Requirements Regarding the Addition of Limiting Condition for Operation (LCO) 3.0.8 on the Inoperability of Snubbers</u>**

Limiting condition for operation (LCO) 3.0.8 provides a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber. The NRC staff issued a *Federal Register* Notice (69 FR 68412) on November 24, 2004, that provided a Model Safety Evaluation and Model Application related to the addition of LCO 3.0.8 for inoperable snubbers on supported systems in TS. The purpose of this model is to permit the NRC to efficiently process amendments that propose to modify requirements by adding LCO 3.0.8 to TS to provide a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, provided risk is assessed and managed, as generically approved by this notice. Licensees of nuclear power reactors to which the model applies could request amendments utilizing the model application.

Table A-1 Typical Systems or Portions of Systems in the Scope of 10 CFR 50.55a Where Snubbers are included in the Inservice Examination and Testing Snubber Program for a Pressurized-Water Reactor (Non-Inclusive)

Typical safety-related, Code-class system in pressurized-water reactors
Reactor coolant system and flowpaths for establishing natural circulation
Main steam system
High-pressure safety injection system
Chemical and volume control or makeup system
Low-pressure safety injection system
Shutdown cooling, residual heat removal, or decay heat removal systems
Containment spray system
Main feedwater system
Auxiliary feedwater system
Primary containment system
Component cooling water system
Spent fuel pool/pit cooling system
Service water system
Emergency diesel generator system (if within the scope of 10 CFR 50.55a)
Ventilation systems
Instrument air system (if within the scope of 10 CFR 50.55a)

TableA-2 Typical Systems or Portions of Systems in the Scope of 10 CFR 50.55a Where Snubbers are included in the Inservice Examination and Testing Snubber Program for a Boiling-Water Reactor (Non-Inclusive)

Typical safety-related, Code-class system in boiling-water reactors
Nuclear boiler and reactor recirculation system
Main steam system
High-pressure core coolant injection (HPCI) system
High-pressure core spray system
Reactor core isolation cooling (RCIC) system (if safety-related)
Reactor water cleanup system
Residual heat removal (RHR) system
Spent fuel pool cooling system
Feedwater coolant injection and isolation condenser system (if applicable)
Standby liquid control (SBLC) system
Main feedwater system
Primary containment system
Closed cooling or component cooling water system
Service water system
Control rod drive system (portions within the scope of 10 CFR 50.55a)
Emergency diesel generator systems (if within the scope of 10 CFR 50.55a)
Ventilation systems
Instrument air system (if within the scope of 10 CFR 50.55a)
Traversing incore probe system (if within the scope of 10 CFR 50.55a)

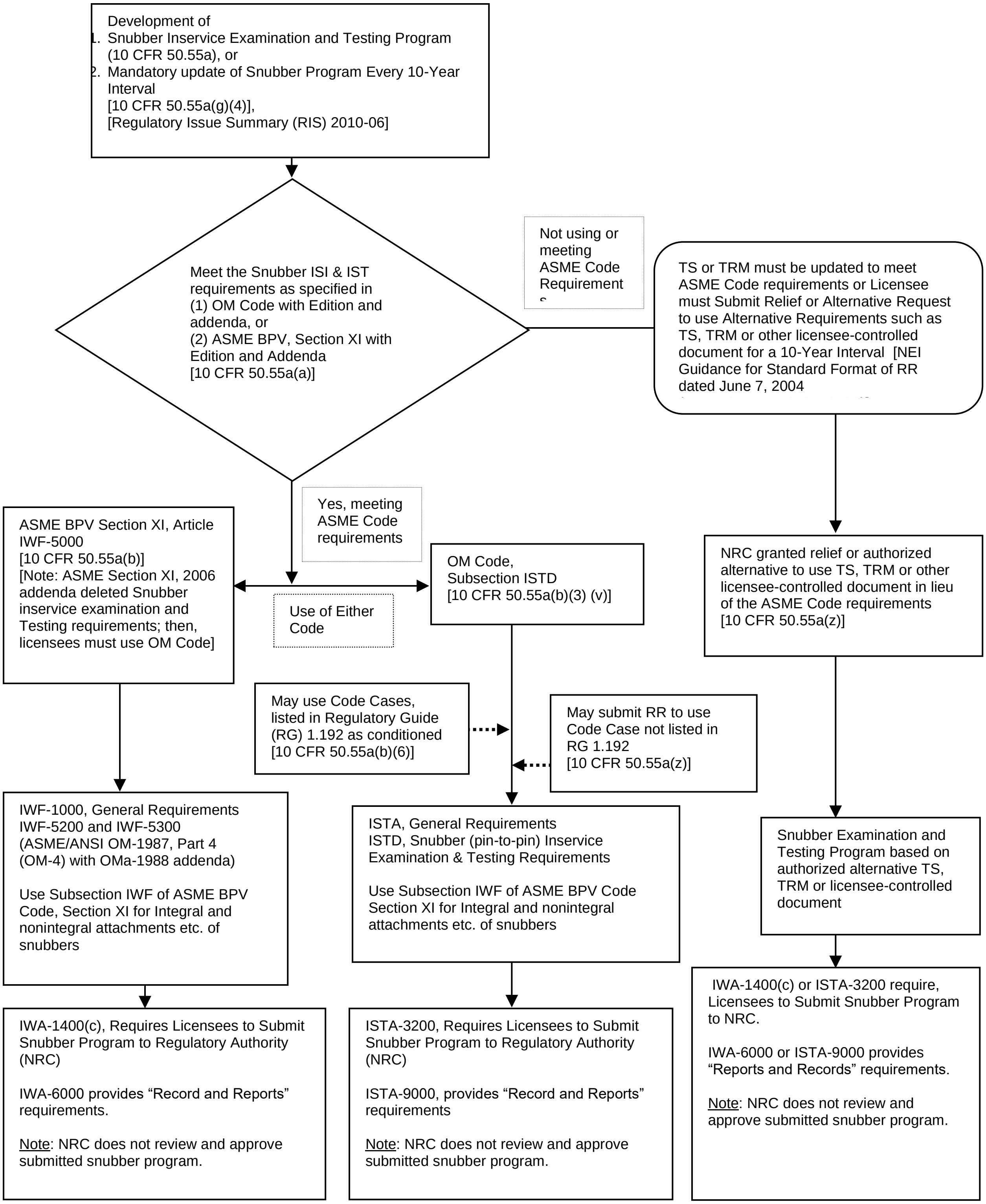

Figure A-1 Development of Preservice and Inservice Inservice Examination and Testing Program for Snubbers*

* Note: Flow chart provided for guidance only. For complete details, see 10 CFR 50.55a.

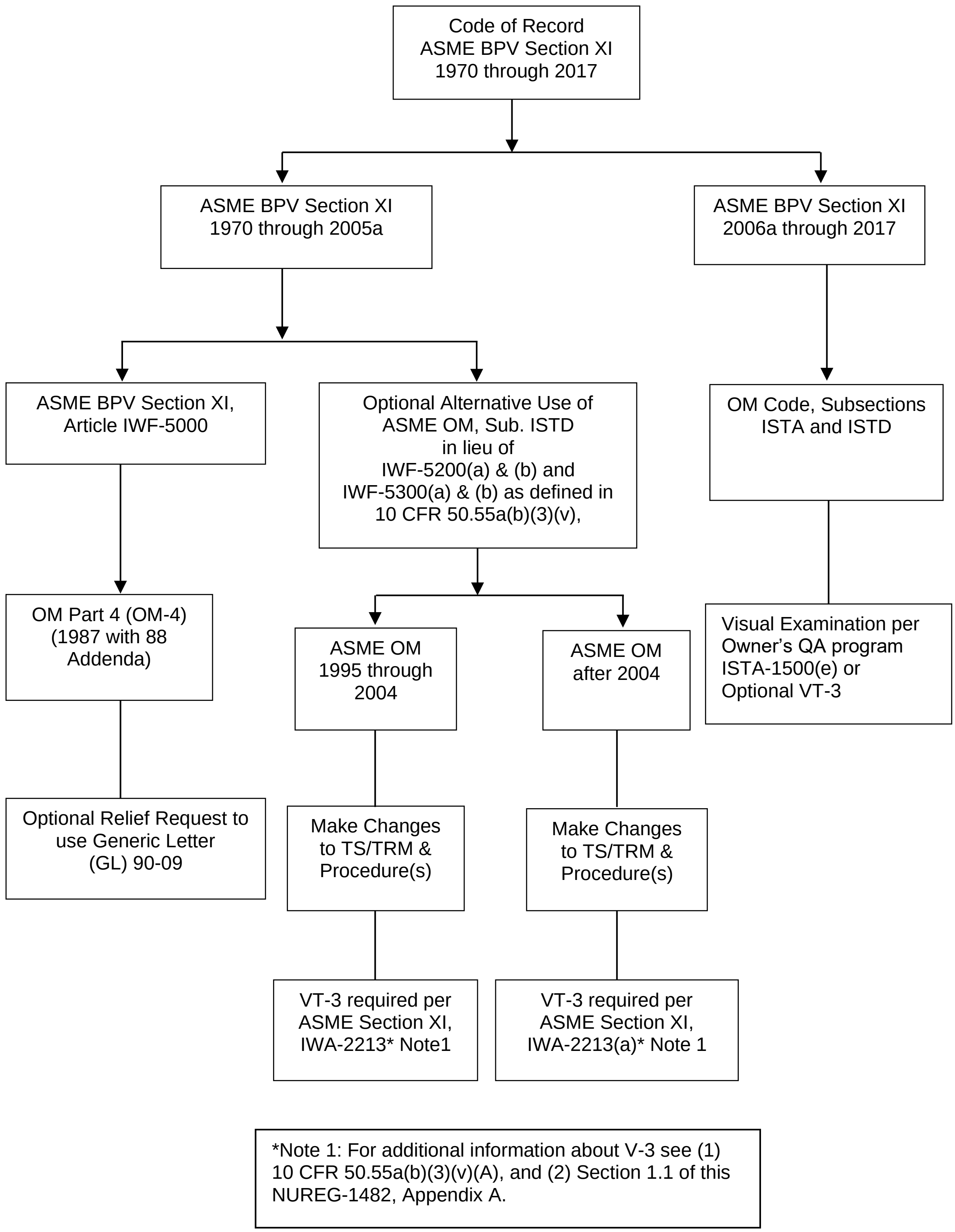

Figure A-2 Use of ASME OM Code for Snubber Program Addressing the Various ASME Section XI Code Versions (more details, see 10 CFR 50,55a)

A.3 SUPPLEMENTAL GUIDANCE ON INSERVICE EXAMINATION AND TESTING OF SNUBBERS PROGRAM

A.3.1 Use of Code Case OMN-13, Revision 2, Performance-Based Requirements for Extending Snubber Inservice Visual Examination Interval at LWR Power Plants.

- Code Case OMN-13, Revision 2, may be used to extend the visual examination interval beyond 48 months, if applicable conditions specified in Code Case OMN-13 are met.
- Code Case OMN-13, Revision 2, Section 3.7(b) requires that if the cumulative number of unacceptable snubbers exceeds the applicable values in the Table ISTD-4252-1, the use of Code Case OMN-13 ends.
- Code Case OMN-13, Revision 2, is endorsed in RG 1.192, Revision 3, with conditions. Code Case OMN-13, Revision 2, states that it is applicable to the 1995 Edition though the 2011 Addenda of the OM Code. The condition in the RG 1.192, Revision 3, allows licensees to implement Code Case OMN-13, Revision 2, in the OM Code Editions and Addenda that are incorporated by reference in 10 CFR 50.55a.

A.3.1.1 Applicability

- Code Case OMN-13, Revision 2, establishes specific requirements that must be met in order to allow extension of the visual examination interval beyond the maximum interval allowed by Table ISTD-4252-1.

A.3.1.2 General Requirements

- Code Case OMN-13, Revision 2, can only be implemented after the requirements of paragraph ISTD-4251 and subparagraphs ISTD-4252(a) through ISTD-4252(c) have been satisfied, and previous examination per Table ISTD-4252-1 has been performed at a maximum interval of two fuel cycles (48 months).
- While using Code Case OMN-13, Revision 2, all General Requirements must remain implemented during the extension of the visual examination interval of 10 years.
- While using Code Case OMN-13, Revision 2, if the total number of failures exceeds the requirement of Column B of Table-4252-1, the use of Code Case OMN-13 for the current examination interval (10 years) will end. For details, see Section 3.7(b) of the Code Case OMN-13, Revision 2.
- To use Code Case OMN-13 again, the General Requirements must be reestablished and implemented as specified in Section 2.0, “General Requirements.”

A.3.1.3 Limitations:

- OM Subsection ISTD shall be implemented prior to the use of Code Case OMN-13.
- Code Case OMN-13 shall not be used in conjunction with Code Case OMN-15, Revision 2.

A.3.2 Use of Code Case OMN-15, Revision 2, "Performance-Based Requirements for Extending the Snubber Operational Readiness Testing Interval at LWR Power Plants."

- Code Case OMN-15, Revision 2, may be used to extend the snubber functional testing interval up to 10 years.
- During any test campaign using Code Case OMN-15, if the entire DTPG population requires testing, use of this Code Case shall be discontinued. The testing requirements of Subsection ISTD shall apply for that DTPG.
- Code Case OMN-15, Revision 2, is endorsed in RG 1.192, Revision 3, with no conditions. Code Case OMN-15, Revision 2, states that it is applicable to the 1998 through 2009 Edition of the OM Code. Therefore, a relief request is required to use Code Case OMN-15, Revision 2, if the licensee's Code of Record is later than the 2009 Edition of the OM Code.

A.3.2.1 Applicability:

- Code Case OMN-15, Revision 2, establishes specific conditions and requirements that must be met for extending the test requirements specified in paragraphs ISTD-5200 and ISTD-5240, and replacing the test plan specified in paragraphs ISTD-5260, ISTD-5300, and ISTD-5400.

A.3.2.2 Limitations:

- OM Subsection ISTD shall be implemented prior to the use of Code Case OMN-15. For additional limitations, see Section 3.0, "Limitations," of Code Case OMN-15, Revision 2.
- Code Case OMN-15, Revision 2 shall not be used in conjunction with Code Case OMN-13.

A.3.3 Compensating Strut

The term "Compensating Strut" is a proprietary trade name which is often used generically to address a type of snubber that locks completely and does not allow displacement under load. Compensating struts are being used to restrain the piping systems and equipment during seismic events and various abnormal dynamic or transient loads at some nuclear power plants. OM Code, Subsection ISTD, does not address compensating struts as separate devices since they are considered to be a category of mechanical snubbers. Therefore, all the regulatory requirements of 10 CFR 50.55a and OM Code, Subsection ISTD and Subsection ISTA, are applicable to compensating struts which are used to restrain piping and equipment in plants. Licensees shall include compensating struts in their snubber inservice examination and testing programs.

The following are commonly understood definitions of mechanical snubbers and compensating struts for clarification:

Mechanical Snubber – An acceleration or velocity limiting device which provides restraint to a component or system during a sudden application of abnormal forces but allows free motion during normal thermal movement. Acceleration or velocity is maintained at the design threshold without locking; hence, the mechanical snubber will not become locked in a fixed position by a sustained, uninterrupted force.

Compensating Strut – An acceleration sensitive device which provides restraint to a component or system during a sudden application of abnormal forces but allows free motion during normal thermal movement. When the predetermined threshold acceleration is exceeded, the compensating strut locks up becoming a rigid restraint and allows no further free motion until the applied force drops to or near zero.

A.3.4 Development of a Fleet's Combined Snubber Program for Multiple Power Plants

While developing a nuclear power plant fleet's Combined Snubber Program for various power plants, the licensee should use the guidelines provided in Section A.2 above and the following:

- Applicable 10-year ISI or IST intervals including start and end dates should be included , and identified for each plant and unit.
- Various Editions and Addenda of ASME BPV Code, Section XI, and OM Code that are used to develop a common document should be specified for each plant and unit.
- Test Plan to be used (10 percent plan or 37 snubber plan, etc.) should be specified clearly stating the plan used for each plant and unit.
- A separate DTPG for large bore snubbers attached to steam generator and reactor coolant pumps shall be specified, as required by the OM Code, paragraph ISTD-5253.
- Use of Code Case OMN-13 shall be specified for each applicable plant and unit. For plants intending to implement the Code Case at a later date, a revised plan must be submitted at the time of implementation (use of terminology such as "Code Case OMN-13 may be used" is inappropriate).
- While using Code Case OMN-13, the cumulative number of unacceptable snubbers must be documented and, if the number of unacceptable snubbers exceeds the applicable values in Table ISTD-4252-1, the current examination interval and use of Code Case OMN-13 ends.
- Use of Code Case OMN-15 shall be specified for each applicable plant and unit. For plants intending to implement the Code Case at a later date, a revised plan must be submitted at the time of implementation (use of terminology such as "Code Case OMN-15 may be used" is inappropriate).

A.3.5 Importance of Lubricant (Grease) in Mechanical Snubbers for Service Life Monitoring

The design of a mechanical snubber allows free thermal movement of a component during normal operating conditions but restrains the component during abnormal dynamic conditions. Adequate lubrication is essential for the proper functioning of mechanical snubbers.

Industry operating experience (OE) with snubber failures has demonstrated that the failure modes of mechanical snubbers might not be identified by visual examination. The functional testing is more effective than visual examination in confirming snubber operational readiness. As required by OM Code or ASME BPV Code, Section XI, only a small sample of the snubber population (10% testing sample plan or 37 snubbers testing sample plan) is selected for functional testing. With the small sample of snubbers selected for functional testing each refueling outage, it might take decades before all of an nuclear power plant's snubbers are tested. Furthermore, some snubbers might never be tested during their service life.

Based on operating experience, most of the snubber failures were determined to be caused by grease degradation, such as: (1) oil separation from grease; (2) dried or "caked" grease; (3) excessive grease; (4) sticky and tacky grease; and (5) hardened or missing grease. A well-planned SLM program for snubbers can minimize the number of snubber failures due to degradation of grease. An effective SLM program would include provisions for preventive maintenance (such as regreasing, manual exercising, partial disassembly for an internal inspection, or additional functional testing for SLM for mechanical snubbers) based on the results of performance monitoring and the evaluation of the service conditions for snubbers.

Potential grease degradation due to high temperature. High grease temperature might result from the installation area environmental conditions or internal conditions caused by vibration. In addition to high temperature, the NRC staff has observed that additional factors might contribute to degradation of the snubber lubricant (grease). For example, grease degradation might result from the following:

- piping or equipment vibration
- elevated temperature for extended periods of time (e.g., high room temperature, high fluid temperature, and snubber installed in close proximity to high-temperature components)
- hazardous environment at the snubber location
- aging of the snubber and grease
- snubber installed from spare inventory without a recent preservice test
- shelf life of the grease exceeding the manufacturer's recommendation.

A key factor in the determination of the effective service life of mechanical snubbers during SLM is the condition of the grease and consideration of the shelf life for the replacement grease. For more details, see Information Notice (IN) 2015-09, "Mechanical Dynamic Restraint (Snubber) Lubricant Degradation Not Identified Due to Insufficient Service Life Monitoring." Therefore, adequate grease is very important for proper functioning of mechanical snubbers.

A.4 GUIDANCE FOR TREATMENT OF SNUBBERS DURING IMPLEMENTATION OF 10 CFR 50.69

A nuclear power plant licensee or applicant may request implementation of 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants,” for risk-informed treatment of structures, systems, and components (SSCs) as an alternative to certain special treatment requirements (STRs) in the NRC regulations. The detailed guidance for 10 CFR 50.69 categorization of various components and snubbers important to the pressure boundary or structural integrity is provided in Appendix B.

Licensees and applicants must consider the piping stress calculation boundaries (anchor-to-anchor) along with system boundaries during the categorization process. See Figure A-3 for details.

- The categorization boundary for piping and associated components (including snubbers) is based on system isolation valves, as well as the piping stress analysis (anchor-to-anchor).
- In addition, there may be II/I piping in plants, where during an initiating event non-categorized piping system may damage the categorized piping system.

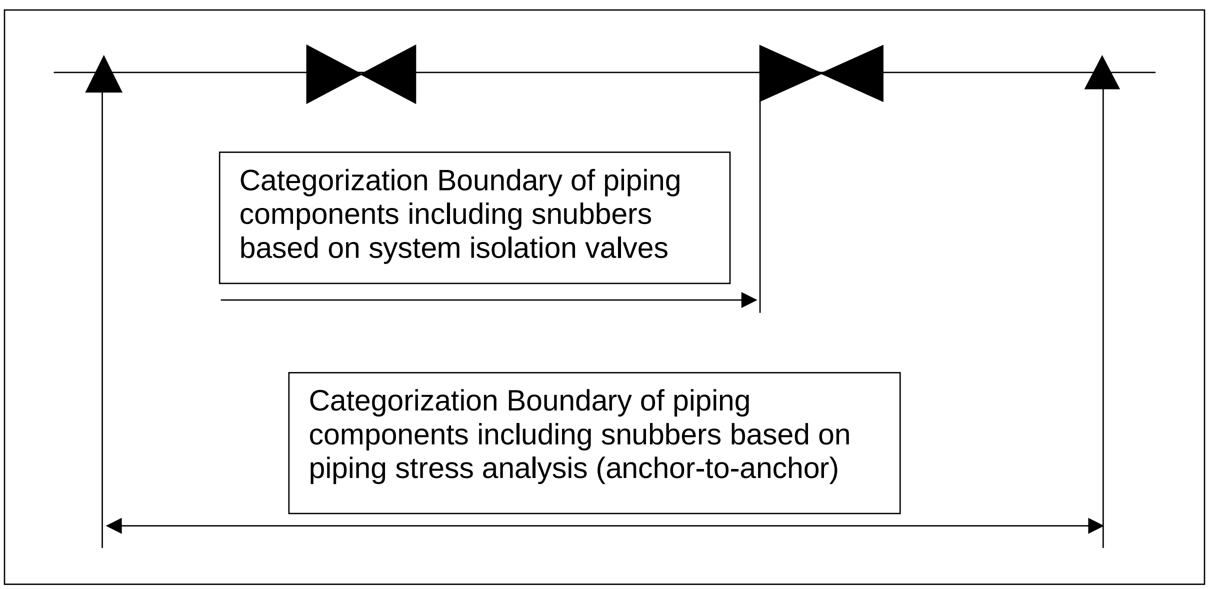

Figure A-3 10 CFR 50.69 Categorization of Components Based on System Isolation Valves and System Piping Stress Analysis

A.5 REFERENCES

A.5.1 U.S. *Code of Federal Regulations*, Domestic Licensing of Production and Utilization Facilities, Title 10, “Energy,” Chapter I – Nuclear Regulatory Commission, Part 50, Section 50.55a, Codes and standards.

A.5.2 *American Society of Mechanical Engineers Boiler and Pressure Vessel Code,* Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” 2004 Edition thru 2008 Addenda.

A.5.3 *American Society of Mechanical Engineers Operation and Maintenance Code,* “Rules for Inservice Testing of Light-Water Reactor Power Plants,” 2017 Edition.

A.5.4 *American Society of Mechanical Engineers Operation and Maintenance Code,* Operation and Maintenance of Nuclear Power Plants, Part 4 (OM-4), 1987 Edition with 1988 Addenda.

A.5.5 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.192, “Operation and Maintenance Code Case Acceptability: ASME OM Code,” Revision 3, dated October 2019, ADAMS Accession No. ML19128A261.

A.5.6 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.193, “ASME Code Cases Not Approved for Use,” Revision 6, dated October 2019, ADAMS Accession No. ML19128A269.

A.5.7 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.147, “Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1,” Revision 19, dated October 2019, ADAMS Accession No. ML19128A244.

A.5.8 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.187, Revision 1, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” dated May 2019, ADAMS Accession No. ML17195A655.

A.5.9 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.100, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” Revision 4, dated May 22, 2020, ADAMS Accession No. ML19312C677.

A.5.10 U.S. Nuclear Regulatory Commission, Regulatory Issue Summary 2004-12, “Clarification on use of Later Editions and Addenda to the ASME OM Code and Section XI,” dated July 22, 2004, ADAMS Accession No. ML042090436.

A.5.11 U.S. Nuclear Regulatory Commission, Regulatory Issue Summary 2012-08, “Developing Inservice Testing and Inservice Inspection Programs Under 10 CFR Part 52,” dated July 17, 2013, ADAMS Accession No. ML13122A365.

A.5.12 U.S. Nuclear Regulatory Commission, Generic Letter (GL) 90-09, “Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions,” December 11, 1990.

A.5.13 Nuclear Energy Institute (NEI), White Paper, “Standard Format for Requests from Commercial Reactor Licensees Pursuant to 10 CFR 50.55a, Revision 1,” June 2004, ADAMS Accession No. ML070100400.

A.5.14 U.S. Nuclear Regulatory Commission, Regulatory Issue Summary (RIS) 2010-06, “Inservice Inspection and Testing Requirements of Dynamic Restraints (Snubbers),” dated June 1, 2010, ADAMS Accession No. ML101310338.

A.5.15 U.S. Nuclear Regulatory Commission, Enforcement Guidance Memorandum (EGM) 10-001, “Dispositioning Violation of Inservice Examination and Testing Requirements for Dynamic Restraints (Snubbers),” dated June 1, 2010, ADAMS Accession No. ML101390020.

A.5.16 U.S. Nuclear Regulatory Commission, Generic Issue (GI-113), “Dynamic Qualification and Testing of Large Bore Hydraulic Snubbers (LBHSs),” NUREG-0933, Main Report with Supplement 1-34.

A.5.17 U.S. Nuclear Regulatory Commission, “Results of LWR Snubber Aging Research,” NUREG/CR-5870, May 1992, ADAMS Accession No. ML040340438.

A.5.18 U.S. Regulatory Commission, Information Notice (IN) 2015-09, “Mechanical Dynamic Restraint (Snubber) Lubricant Degradation Not Identified Due to Insufficient Service Life Monitoring,” dated September 2015, ADAMS Accession No. ML15126A53.

A.5.19 U.S. Regulatory Commission, Regulatory Issue Summary 2004-16, “Use of Later Editions and addenda to the ASME Code Section XI for Repair/Replacement Activities,” dated October 19, 2004.

A.6 HISTORICAL REFERENCES

A.6.1 U.S. Nuclear Regulatory Commission (NRC) Generic Letters:

A.6.1.1 GL 84-13, “Technical Specification for Snubbers”

A.6.1.2 GL 89-09, “ASME III Component Replacement”

A.6.1.3 GL 90-09, “Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions”

A.6.1.4 GL 80-99, “Technical Specification Revisions for Snubbers Surveillance” (ADAMS Accession No. ML093430363)

A.6.2 NRC Inspection Enforcement (IE) Bulletins

A.6.2.1 BL-73-03, “Defective Hydraulic Shock Suppressors and Restraints”

A.6.2.2 BL-73-04, “Defective Bergen-Patterson Hydraulic Shock Suppressors Absorbers”

A.6.2.3 BL-75-05, “Operability of Category I Hydraulic Shock and Sway Suppressors”

A.6.2.4 BL-78-10, “Bergen-Paterson Hydraulic Shock Suppressor Accumulator Spring Coil”

A.6.2.5 BL-79-02, “Pipe Support Base Plate Designs Using Concrete Expansion Anchors Bolts”

A.6.2.6 BL-79-14, Revision 1, Supplement 2, “Seismic Analysis for As-Build Safety-Related Piping”

A.6.2.7 BL-81-01, Revision 1, “Surveillance of Mechanical Snubbers”

A.6.3 NRC Information Notices (IN):

A.6.3.1 IN 79-01, “Bergen-Paterson Hydraulic Shock Arrestors”

A.6.3.2 IN 79-05, “Improper Material in Safety-Related Components”

A.6.3.3 IN 83-13, “Design Misapplication of Bergen-Paterson Resistance Clamp”

A.6.3.4 IN 83-20, “ITT Grinnell Figure 306/307 Mechanical Snubber Attachment Interference”

A.6.3.5 IN 83-47, “Failure of Hydraulic Snubbers as a Result on Contaminated Hydraulic Fluid”

A.6.3.6 IN 84-67, “Recent Snubber Inservice Testing with High failure Rates”

A.6.3.7 IN 84-73, “Down Rating of Self-Aligning Ball Bushing in Snubbers”

A.6.3.8 IN 86-102, “Repeated Multiple Failure of Steam Generator Hydraulic Snubbers due to Control Valve Sensitivity”

A.6.3.9 IN 89-30, “High Temperature Environment at Nuclear Power Plants”

A.6.3.10 IN 94-48, “Snubber Lubrication Degradation in High Temperature Environments”

A.6.3.11 IN 95-09, “Use of Inappropriate Guidelines and Criteria for Nuclear Piping and Pipe Support Evaluation and Design”

A.6.3.12 IN 97-16, “Preconditioning of Plant Structures, Systems, and Components before ASME Code Inservice Testing or Technical Specification Surveillance Testing”

A.6.3.13 IN 2015-09, “Mechanical Dynamic Restraint (Snubber) Lubricant Degradation Not Identified Due to Insufficient Service Life Monitoring”

A.6.4 NRC IE Circulars (Cr):

A.6.4.1 Cr 76-05, “Hydraulic Shock and Sway Suppressors – Maintenance of Bleed and Lock-Up Velocities on ITT Grinnell’s Model Nos. - Figure 200 and Figure 201, Catalog PH-74-R,” dated October 8, 1976.

A.6.4.2 Cr 78-07, “Damaged Components of Bergen Paterson Series 25000 Test Stand,” dated May 31, 1978.

A.6.4.3 Cr 79-25, “Shock Arrestor Strut Assembly Interference”

A.6.4.4 Cr 81-05, “Self-Aligning Rod End Bushing for Pipe Supports”

A.6.5 NUREGs:

A.6.5.1 NUREG/CR-5386, “Basis for Snubber Aging Research: Nuclear Plant Aging Research Program, Vol. 1,” dated January 1990, ADAMS Accession No. ML040360184.

A.6.5.2 NUREG/CR-4279, “Aging and Service Wear of Hydraulic and Mechanical Snubbers Used Safety-Related Piping and Components of Nuclear Power Plant - Phase I Study,” dated February 1986, ADAMS Accession No. ML040230419.

A.6.5.3 NUREG/CR-5870, “Results of LWR Snubbers Aging Research,” dated May 1992, ADAMS Accession No. ML040340438.

A.6.6 Other Snubbers Related Documents:

A.6.6.1 PNL-SA-20219, “ASME Subsection ISTD Recommendation Based upon NAPR Snubber aging Research Results,” dated December 1991, Pacific Northwest Laboratory, Richland, WA-99352, (Work Supported by U. S. Department of Energy), ADAMS Accession No. ML040290285.

A.6.6.2 Resolution of Generic Safety Issues: Issue 113: Dynamic Qualification Testing of Large Bore Hydraulic Snubbers (Rev-2), NUREG-0933, Main Report with Supplements 1-34.

APPENDIX B
GUIDANCE FOR TREATMENT OF PUMPS, VALVES, AND DYNAMIC RESTRAINTS DURING IMPLEMENTATION OF 10 CFR 50.69

A nuclear power plant licensee or applicant may request implementation of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants." Implementation of 10 CFR 50.69 allows for risk-informed treatment of structures, systems, and components (SSCs) as an alternative to certain special treatment requirements (STRs) in the NRC regulations.

10 CFR 50.69 defines risk-informed safety class (RISC) of SSCs as follows:

> RISC-1 SSCs: safety-related SSCs that perform safety significant functions.
>
> RISC-2 SSCs: nonsafety-related SSCs that perform safety significant functions.
>
> RISC-3 SSCs: safety-related SSCs that perform low safety significant functions.
>
> RISC-4 SSCs: nonsafety-related SSCs that perform low safety significant functions.

Paragraph (a) of 10 CFR 50.69 defines a safety significant function as a function whose degradation or loss could result in a significant adverse effect on defense-in-depth, safety margin, or risk.

Paragraph (b) of 10 CFR 50.69 indicates that if approved, a licensee may voluntarily comply with the requirements in 10 CFR 50.69 as an alternative to compliance with specific requirements for RISC-3 and RISC-4 SSCs. The listed requirements that may be omitted for RISC-3 and RISC-4 SSCs include the inservice testing (IST) requirements and inservice inspection (ISI) requirements, and repair and replacement provisions in 10 CFR 50.55a.

Paragraph (d)(1) of 10 CFR 50.69 specifies the following requirements for the treatment of RISC-1 and 2 SSCs:

> The licensee or applicant shall ensure that RISC-1 and RISC-2 SSCs perform their functions consistent with the categorization process assumptions by evaluating treatment being applied to these SSCs to ensure that it supports the key assumptions in the categorization process that relate to their assumed performance.

Paragraph (d)(2) of 10 CFR 50.69 specifies the following requirements for the treatment of RISC-3 SSCs:

> The licensee or applicant shall ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design-basis conditions, including seismic conditions and environmental conditions and effects throughout their service life. The treatment of RISC-3 SSCs must be consistent with the categorization process. Inspection and testing, and corrective action shall be provided for RISC-3 SSCs.

(i) Inspection and testing. Periodic inspection and testing activities must be conducted to determine that RISC-3 SSCs will remain capable of performing their safety-related functions under design-basis conditions; and

(ii) Corrective action. Conditions that would prevent a RISC-3 SSC from performing its safety-related functions under design-basis conditions must be corrected in a timely manner. For significant conditions adverse to quality, measures must be taken to provide reasonable confidence that the cause of the condition is determined and corrective action taken to preclude repetition.

Paragraph (e) of 10 CFR 50.69 specifies requirements for feedback and process adjustment based on the specific RISC-1, 2, 3, or 4 classifications. Paragraph (f) of 10 CFR 50.69 specifies requirements for program documentation, change control and records. Paragraph (g) of 10 CFR 50.69 specifies reporting requirements for RISC-1 and RISC-2 SSCs.

In the *Federal Register* notice (69 FRN 68008), dated November 22, 2004, for 10 CFR 50.69, the Commission provides guidance for implementing 10 CFR 50.69. In the FRN, the Commission states that prescriptive requirements as to how licensees are to treat specific SSCs (e.g., safety-related) are referred to as “special treatment requirements.” As noted in the FRN, the STRs are developed to provide greater assurance that SSCs will perform their functions under particular conditions with high quality and reliability. The STRs include examination techniques, testing strategies, documentation requirements, personnel qualification requirements, and independent oversight.

In the FRN, the Commission provided an overview of the 10 CFR 50.69 requirements. In particular, 10 CFR 50.69 represents an alternative set of requirements whereby a licensee or applicant may voluntarily undertake categorization of its SSCs consistent with the requirements in 10 CFR 50.69(c), remove the STRs listed in 10 CFR 50.69(b) for SSCs that are determined to be of low individual safety significance, and implement alternative treatment requirements described in 10 CFR 50.69(d). The regulatory requirements not removed by 10 CFR 50.69(b) continue to apply as well as the requirements specified in 10 CFR 50.69. The rule contains requirements by which a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant.

As stated at 69 FRN 68008, 68011, the Commission notes that 10 CFR 50.69, while intended to ensure that the scope of STRs imposed on SSCs is risk-informed, is not intended to allow for the elimination of SSC functional requirements or to allow equipment that is required by the deterministic design basis to be removed from the facility (i.e., changes to the design of the facility must continue to meet the current requirements governing design change; most notably 10 CFR 50.59). Instead, the rule should enable licensees and the NRC staff to focus their resources on SSCs that make a significant contribution to plant safety by restructuring the regulations to allow an alternative risk-informed approach to special treatment. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, this approach should allow an acceptable, though reduced, level of confidence (i.e., ‘‘reasonable confidence’’) that these SSCs will satisfy functional requirements. However, continued maintenance of the health and safety of the public will depend on effective implementation of 10 CFR 50.69 by the licensee or applicant applying the rule at its nuclear power plant.

Items to consider as described in 69 FRN 68008, 68011-68016 include:

1. Licensees can establish the treatment for RISC-3 SSCs without prior NRC review.

2. The NRC plans to conduct inspections of 10 CFR 50.69 implementation. The principal focus of the inspection will be on the safety significant aspects of 10 CFR 50.69 implementation such as categorization and treatment of RISC-1 and RISC-2 SSCs, but the inspection will also consider the implementation of RISC-3 treatment focusing on programmatic and common-cause issues, which could undermine the categorization process and its results.

3. The treatment of RISC-3 SSCs must be consistent with the categorization process. One way to achieve this could be the application of consensus standards where the application of such standards meets the 10 CFR 50.69(d)(2) requirements for RISC-3 SSCs.

4. RISC-3 SSCs will be exempt from STRs for qualification methods for environmental conditions and effects and seismic conditions. Nevertheless, RISC-3 SSCs continue to be required to be capable of performing their safety-related functions under applicable environmental conditions and effects and seismic conditions, albeit at a lower level of confidence as compared to RISC-1 SSCs. A licensee implementing 10 CFR 50.69 must consider operating life (aging) and combinations of operating life parameters (synergistic effects) in the design of RISC-3 electrical equipment. This is particularly important if the equipment contains materials which are known to be susceptible to significant degradation due to thermal, radiation, and/or wear (cyclic) aging including any known synergistic effects that could impair the ability of the equipment to meet its design-basis function.

Treatment of RISC-1 and 2 Pumps, Valves, and Dynamic Restraints

For SSCs categorized as RISC-1 or RISC-2, all existing applicable requirements continue to apply (i.e., no STRs are removed by 10 CFR 50.69).

10 CFR 50.69(d)(1) requires licensees to have a valid technical basis to ensure that RISC-1 and RISC-2 SSCs perform their functions consistent with the categorization process assumptions (availability, capability, and reliability). These functions may be beyond the design basis for individual SSCs. For example, if a relief valve was being credited with capability to relieve water (as opposed to its design condition of steam), such an evaluation would look at whether the component has been determined to be able to perform as assumed. If an existing technical basis does not exist (e.g., the supporting test program does not test the SSC at the beyond design-basis conditions), the licensee is required to develop a technical basis for the credit taken in the PRA, potentially including a treatment program for the SSC that validates the credited capability. 10 CFR 50.69(d)(1) does not extend existing STRs to RISC-1 beyond design-basis functions or to RISC-2 SSCs; for example, the beyond design-basis capabilities of RISC-1 SSCs credited in the PRA are not subject to the requirements of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Reprocessing Plants."

Treatment of RISC-3 Pumps, Valves, and Dynamic Restraints

Through the application of 10 CFR 50.69, RISC-3 SSCs are no longer required to follow the specific STRs listed in 10 CFR 50.69(b)(1). 10 CFR 50.69(d)(2) contains alternate inspection, testing, and corrective action requirements for RISC-3 SSCs. The alternative treatment requirements are performance-based and give licensees the flexibility to implement alternative treatment provided the treatment satisfies 10 CFR 50.69(d)(2).

As stated at 69 FRN 68008, 68020, in implementing the rule requirements, licensees will need to obtain data or information sufficient to make a technical judgment that RISC-3 SSCs will remain capable of performing their safety-related functions under design-basis conditions, and to enable the licensee to take actions to restore equipment performance consistent with corrective action requirements included in the rule. Effective implementation of the treatment requirements should result in reasonable confidence that RISC-3 SSCs will perform their safety-related function under normal and design-basis conditions.

Note that two of the STRs listed in 10 CFR 50.69(b)(1) are 10 CFR Part 21 and 10 CFR Part 50 Appendix B. RISC-3 SSCs might have been purchased or dedicated as safety-related components and subject to the full range of STRs. Other components of the same nameplate and/or treatment might be categorized as RISC-1 at the plant or other plants, or might be considered safety-related at plants that have not implemented 50.69. Licensees should ensure that operating experience from RISC-3 SSCs is shared, as applicable, to support the 10 CFR Part 21 requirements.

10 CFR 50.69 places two requirements on the alternate treatment of RISC-3 SSCs: (1) treatment must be consistent with the categorization process, and (2) treatment must result in reasonable confidence that RISC-3 SSCs will perform their safety-related function under normal and design-basis conditions for the life of the component. “Reasonable confidence” is less than that associated with RISC-1 SSCs in order to be consistent with the low safety significance of the individual RISC-3 SSCs. The rule language “consistent with the categorization process” means that, when establishing the treatment for RISC-3 SSCs, the licensee must take into account the assumptions in the categorization process regarding reliability and design-basis capability. Any regulatory requirements applicable to RISC-3 SSCs not removed by 10 CFR 50.69(b)(1) continue to apply. For example, 10 CFR 50.69 does not provide relaxations for making changes to the design-basis functional requirements of SSCs.

10 CFR 50.69(b)(2)(iv) removes RISC-3 SSCs from the scope of certain provisions of 10 CFR 50.55a, “Codes and standards.” The provisions being removed include those that relate to inspection and testing in 10 CFR 50.55a(f) (10 CFR 50.55a(g) for snubbers). 10 CFR 50.69 allows licensees to use alternative treatment instead of the OM Code. Nevertheless, the NRC staff considers the ASME Code Cases endorsed in 10 CFR 50.55a and listed in Regulatory Guide (RG) 1.192 to be one acceptable method of establishing treatment of RISC-3 SSCs, where applicable, in that those endorsed Code Cases adjust treatment based on the safety significance of the components.

Other considerations described in 69 FRN 68008 include:

- Licensees’ evaluation of the consistency between the treatment of RISC-3 SSCs and the risk categorization process may be qualitative so long as it provides reasonable confidence of the reliability and design-basis capability of RISC-3 SSCs. (69 FRN 68008, 68041)

- The licensee is responsible for addressing applicable vendor recommendations and operational experience, such as may be described in NRC information notices or identified in responses to NRC bulletins, generic letters, or other licensee commitment documents. (69 FRN 68008, 68041)

- The treatment applied to RISC-3 SSCs must support the assumptions used in justifying the removal of requirements applicable to those SSCs. For example, where a licensee or applicant intends as part of implementing 10 CFR 50.69 to eliminate leakage testing required in 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," for containment isolation valves, the treatment applied to those valves must support the assumption that they are capable of closing under design-basis conditions. (69 FRN 68008, 68027)

- As described in NUREG/CR-6752, ''A Comparative Analysis of STRs for Systems, Structures, and Components (SSCs) of Nuclear Power Plants with Commercial Requirements of Non-Nuclear Power Plants,'' Section 5.2.4, "Use of Commercial Codes, Standards, and Practices for RISC-3 SSCs," commercial standards by themselves are not adequate to provide reasonable confidence of functionality. Therefore, a simple reference to these practices may not provide a basis to satisfy the rule's requirements. (69 FRN 68008, 68041)

- Operating experience and research shows that exercising a valve or pump, by itself, does not provide reasonable confidence of design-basis capability if that exercising does not detect service-induced aging or degradation. (69 FRN 68008, 68042)

As stated at 69 FRN 68008, 68042, under 10 CFR 50.69, most STRs will be removed from RISC-3 SSCs, which will typically comprise a large percentage of safety-related SSCs in a nuclear power plant. These STRs will be replaced with the high-level treatment requirements in 10 CFR 50.69(d)(2) that will allow significant reduction in the treatment applied to RISC-3 SSCs. This reduction in treatment can introduce common-cause concerns and weaken defenses against them. Therefore, 10 CFR 50.69(d)(2) requires that inspection, testing and corrective action be provided for RISC-3 SSCs. The inspection and testing requirement in 10 CFR 50.69(d)(2)(i) is to provide sufficient performance data for RISC-3 SSCs to determine if the reduction in treatment has adversely affected their design-basis capability and to provide reasonable confidence that the SSCs can perform its safety function throughout their service life, consistent with the reliability assumptions in the risk categorization.

As stated at 69 FRN 68008, 68042, a licensee or applicant may not simply assume that a sensitivity study that increases the failure probability for all RISC-3 SSCs simultaneously, with no additional basis to support it, would necessarily bound the potential change in risk that could result due to implementation of 10 CFR 50.69. There is a potential that risk due to implementation of 10 CFR 50.69 could increase as a result of the reduction in treatment due to common-cause interactions or degradation, and this impact might not be uniform across the population of RISC-3 SSCs. For example, if a licensee were to simply eliminate maintenance, testing, or lubrication of pumps or valves, it could significantly impact performance of those specific components and the impact might exceed the cumulative impact of individually reducing the reliability of all RISC-3 SSCs by a few percent or less. In satisfying 10 CFR 50.69, the licensee or applicant must consider potential common-cause interactions and degradation mechanisms in establishing treatment for RISC-3 SSCs so there is a reasonable basis to support the assumptions made for the risk sensitivity study.

As stated at 69 FRN 68008, 68042, in some cases, a licensee or applicant implementing 10 CFR 50.69 might apply more rigorous test methods than previously applied to satisfy the ASME

OM Code IST provisions because 10 CFR 50.69 does not specify restrictive time limits on test intervals that were provided in the ASME OM Code.

As stated at 69 FRN 68008, 68042, 10 CFR 50.69(d)(2)(i) requires the licensee to conduct periodic inspection and testing activities to determine whether RISC-3 SSCs will remain capable of performing their safety-related functions under design-basis conditions. The prescriptive STRs in 10 CFR 50.55a and 50.65 for inspection, testing, and surveillance have been removed for RISC-3 SSCs. In lieu of those prescriptive requirements, the rule requires the licensee or applicant to implement inspection and testing of RISC-3 SSCs sufficient to provide reasonable confidence that RISC-3 SSCs remain capable of performing their safety-related functions under design-basis conditions throughout their service life. The licensee or applicant may apply industrial practices for the treatment of RISC-3 SSCs if those practices maintain the capability of the RISC-3 SSCs to perform their design-basis safety functions. 10 CFR 50.69(d)(2)(i) means that the licensee or applicant must implement periodic testing or inspection sufficient to provide reasonable confidence that RISC-3 pumps and valves will be capable of performing their safety-related functions under design-basis conditions. To determine that the pump or valve will remain capable of performing its safety-related function, the licensee or applicant will need to obtain sufficient operational information or performance data to provide with reasonable confidence that the RISC-3 pumps and valves will be capable of performing their safety-related functions if called upon to function under operational or design-basis conditions over the interval between periodic testing or inspections. In addition, the operational information and Performance data must be sufficient to satisfy the requirements of 10 CFR 50.69(d)(2)(i) for use in identifying the need for corrective action under 10 CFR 50.69(d)(2)(ii) and in providing information for feedback to the categorization and treatment processes under 10 CFR 50.69(e)(3).

As stated at 69 FRN 68008, 68043, 10 CFR 50.69(d)(2)(ii) requires that conditions that would prevent a RISC-3 SSC from performing its safety-related functions under design-basis conditions must be corrected in a timely manner. In the case of significant conditions adverse to quality, the rule requires that measures be taken to provide reasonable confidence that the cause of the condition is determined and corrective action taken to preclude repetition. Significant conditions adverse to quality include common-cause concerns for multiple RISC-3 SSCs or concerns related to the validity of the categorization process or its results. For example, if measuring and test equipment is found to be in error or defective, the licensee or applicant will be responsible for determining the functionality of safety-related SSCs checked using that equipment to prevent the occurrence of common-cause problems that might invalidate the categorization process assumptions and results. Effective implementation of the corrective action process would include timely response to information from plant SSCs, overall plant operations, and industry generic activities that might reveal performance concerns for RISC-3 SSCs on both an individual and common-cause basis. The corrective action process alone is insufficient to monitor the effects of reduced treatment on RISC–3 SSCs, and therefore the Commission has incorporated feedback requirements into 10 CFR 50.69.

<u>Treatment of RISC-4 Pumps, Valves, and Dynamic Restraints</u>

10 CFR 50.69 does not impose any new treatment requirements on RISC-4 SSCs. Instead, RISC-4 SSCs are simply removed from the scope of any applicable STRs identified in 10 CFR 50.69(b)(1). The Commission states that this is justified in view of their non-safety-related classification and risk information. Requirements applicable to RISC-4 SSCs not removed by 10 CFR 50.69(b)(1) continue to apply.

Feedback, Documentation, and Reporting Requirements

10 CFR 50.69(e) discusses feedback and process adjustments. The validity of the categorization process relies on ensuring that the performance and condition of SSCs continue to be maintained consistent with applicable assumptions. Changes in the level of treatment applied to an SSC might result in changes in the reliability of the SSCs credited in the categorization process. Additionally, plant changes, changes to operational practices, and plant and industry operational experience may impact categorization process results. Consequently, the rule contains requirements for updating the categorization and treatment processes when conditions warrant to assure that continued SSC performance is consistent with the categorization process and results. Specifically, the rule requires licensees to review the changes to the plant, operational practices, applicable plant and industry operational experience, and, as appropriate, update the PRA and SSC categorization. The review must be performed in a timely manner but no longer than once every two refueling outages. In addition, licensees are required to obtain sufficient information on SSC performance to verify that the categorization process and its results remain valid.

For RISC-1 SSCs, much of the performance information may be obtained from present programs for inspection, testing, surveillance, and maintenance. However, for RISC-2 SSCs and for RISC-1 SSCs credited for beyond design-basis accidents, licensees need to ensure that sufficient information is obtained. For RISC-3 SSCs, there is a relaxation of the requirements for obtaining information when compared to the applicable STRs. However, sufficient information still needs to be obtained. The rule requires considering performance data, determining if adverse changes in performance have occurred, and making the necessary adjustments so that desired performance is achieved so that the evaluations conducted to meet 10 CFR 50.69(c)(1)(iv) remain valid.

For both RISC-1 and RISC-2 SSCs, requirements are included in 10 CFR 50.69(e)(2) for monitoring and for taking action when SSC performance degrades. When a licensee or applicant determines that a RISC-3 SSC does not meet its established acceptance criteria for performance of design-basis functions, the rule requires that a licensee perform timely corrective action under 10 CFR 50.69(d)(2)(ii). Further, as part of the feedback process, the review of operational data may reveal inappropriate credit for reliability or performance, and a licensee would need to re-visit the findings made in the categorization process or modify the treatment for the RISC-3 SSCs under 10 CFR 50.69(e)(3). These provisions would then restore the facility to the conditions that were considered in the categorization process and would also restore the capability of the SSCs to perform their functions.

10 CFR 50.69(f) requires the licensee or applicant to document the basis for its categorization of SSCs before removing STRs. 10 CFR 50.69(f) also requires the licensee or applicant to update the final safety analysis report to reflect which systems have been categorized.

10 CFR 50.69(g) requires reporting of events or conditions that prevented, or would have prevented, a RISC-1 or RISC-2 SSC from performing a safety significant function.

Implementation of 10 CFR 50.69 and Technical Specification

In Section III.4.10.2, “Section 50.36 Technical Specifications,” of the *Federal Register* notice, dated November 22, 2004 (69 FR 68008, 68028-68029), the Commission stated that the 10 CFR 50.69 rule does not include 10 CFR 50.36 in the list of special treatment requirements that may be replaced by the alternative 10 CFR 50.69 requirements for RISC-3 and RISC-4 SSCs when implementing a 10 CFR 50.69 license amendment. As a result, the NRC staff does not consider the TSs (including Improved Technical Specifications (ITS) and the associated Technical Requirements Manual (TRM)) to be part of the 10 CFR 50.69 rule. Therefore, licensees should continue to follow their TSs (including the ITS and TRM, as applicable) when implementing a 10 CFR 50.69 license amendment.

References

B.1 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants.”

B.2 *Federal Register* notice (69 FRN 68008), “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors; Final Rule,” dated November 22, 2004.

B.3 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.”

B.4 NUREG/CR-6752, ‘‘A Comparative Analysis of STRs for Systems, Structures, and Components (SSCs) of Nuclear Power Plants with Commercial Requirements of Non-Nuclear Power Plants,’’ dated January 2002, ADAMS Accession No. ML0203300051.

B.5 NRC Regulatory Guide 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.”

B.6 NRC Regulatory Guide 1.147, “Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1.”

B.7 NRC Regulatory Guide 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code.”

B.8 NRC Regulatory Guide 1.201, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants according to their Safety Significance,” Revision 1.

B.9 Nuclear Energy Institute 00-04, “10 CFR 50.69 Categorization Guideline,” dated July 2005 (ADAMS Accession No. ML052910035).

B.10 NRC Inspection Procedure 37060, “10 CFR 50.69 Risk-Informed Categorization and Treatment of Structures, Systems, and Components Inspection,” dated September 2011.

NRC FORM 335 (12-2010) NRCMD 3.7	U.S. NUCLEAR REGULATORY COMMISSION	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)
	BIBLIOGRAPHIC DATA SHEET *(See instructions on the reverse)*	**NUREG-1482, Revision 3**

2. TITLE AND SUBTITLE

Guidelines for Inservice Testing at Nuclear Power Plants

Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants

3. DATE REPORT PUBLISHED

MONTH	YEAR
July	**2020**

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Thomas Scarbrough, Robert Wolfgang, Michael Farnan, Nicholas Hansing, and Gurjendra Bedi

6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Engineering and Hazards
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Same as above

10. SUPPLEMENTARY NOTES

Gurjendra S. Bedi, Project Manager

11. ABSTRACT (200 words or less)

In Revision 2 of NUREG-1482, the staff of the U.S. Nuclear Regulatory Commission (NRC) discusses the applicable regulations for the inservice testing of pumps and valves, and the examination and testing of dynamic restraints (snubbers) at commercial nuclear power plants. The information in NUREG-1482, “Guidelines for Inservice Testing at Nuclear Plants,” Revision 0, issued April 1995, and Revision 1, issued January 2005, and Revision 2 issued October 2013, has described this topics in the past, This NUREG report replaces Revision 0, Revision 1, and Revisions 2 to NUREG-1482, and is applicable, unless stated otherwise, to all editions and addenda of the American Society of Mechanical Engineers *Code of Operation and Maintenance of Nuclear Power Plants* (OM Code). And ASME *Boiler and Pressure Vessel Code* (BPV), which Titles 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a) incorporates by reference (*Federal Register*, Vol. 85, No. 86, page 26540-26581, dated May 4, 2020). NUREG-1482, Revision 3 incorporates the public feedback, as appropriate, received against draft NUREG-1482, Revision 3 (ADAMS Accession No. ML19310D440) The public feedback and comments evaluation are available in document (ADAMS Accession No. ML20153A761).

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

NUREG-1482	Valves
Inservice Testing	Pumps
IST	Valve Testing
10 CFR 50.55a	Pump testing
ASME OM Code	Dynamic Restraints
ASME BPV	Snubbers
ASME Section XI	Examination
Relief Requests	Visual examination
Alternative	Snubber testing

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page) unclassified

(This Report) unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS

@NRCgov

NUREG-1482, Revision 3 **Guidelines for Inservice Testing at Nuclear Power Plants** **July 2020**