U.S. NUCLEAR REGULATORY COMMISSION

DRAFT REGULATORY GUIDE DG-1383

Proposed new Regulatory Guide 1.246

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ACCEPTABILITY OF ASME CODE, SECTION XI, DIVISION 2, "REQUIREMENTS FOR RELIABILITY AND INTEGRITY MANAGEMENT (RIM) PROGRAMS FOR NUCLEAR POWER PLANTS," FOR NON-LIGHT WATER REACTORS

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes an approach that is acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for the development and implementation of a preservice inspection (PSI) and inservice inspection (ISI) program for non-light water reactors (non-LWRs). It endorses, with conditions, the 2019 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 2, "Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants" (Ref. 1), hereafter referred to as ASME Code, Section XI, Division 2, for non-LWR applications. This RG also describes a method that applicants can use to incorporate PSI and ISI programs into a licensing basis.

Applicability

This RG applies to nuclear power reactor applicants and licensees for combined or operating licenses for reactors with non-LWR designs subject to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 2), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 3).

Applicable Regulations

- 10 CFR Part 50 provides regulations for licensing production and utilization facilities.
    - 10 CFR 50.34(b)(6)(iv) requires an application for an operating license to include, in the final safety analysis report, plans for conducting normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components (SSCs).

This RG is being issued in draft form to involve the public in the development of regulatory guidance in this area. It has not received final staff review or approval and does not represent an NRC final staff position. Public comments are being solicited on this DG and its associated regulatory analysis. Comments should be accompanied by appropriate supporting data. Comments may be submitted through the Federal rulemaking Web site, http://www.regulations.gov, by searching for draft regulatory guide DG-1383. Alternatively, comments may be submitted to the Office of Administration, Mailstop: TWFN 7A-06M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Program Management, Announcements and Editing Staff. Comments must be submitted by the date indicated in the *Federal Register* notice.

Electronic copies of this DG, previous versions of DGs, and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC Library at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/. The DG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML21120A185. The regulatory analysis may be found in ADAMS under Accession No. ML21120A192.

- o 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," establishes quality assurance requirements for the design, manufacture, construction, and operation of those SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to public health and safety.

- 10 CFR Part 52 governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities.

  - o 10 CFR 52.79(a)(5), in part, requires an application for a combined license to include an analysis and evaluation of the design and performance of SSCs with the objective of assessing both the risk to public health and safety from facility operation, including determining the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs intended to prevent accidents and mitigate the consequences of accidents.

  - o 10 CFR 52.79(a)(29)(i) requires an application for a combined license to include plans for conducting normal operations, including maintenance, surveillance, and periodic testing of SSCs.

Related Guidance

- 10 CFR Part 50, Appendix A, "General Design Criteria [(GDCs)]," contains the GDCs that establish the minimum requirements for the principal design criteria for water-cooled nuclear power plants. Appendix A also indicates that the GDCs are generally applicable to other types of nuclear power units and are intended to provide guidance in determining the principal design criteria (PDC) for such other units.

  - o 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records," requires, in part, that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Where generally recognized codes and standards are used, GDC 1 requires that they be identified and evaluated to determine their applicability, adequacy, and sufficiency and be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.

- 10 CFR 50.55a incorporates by reference ASME Code Sections III, "Rules for Construction of Nuclear Facility Components," (Ref. 4), Section XI, and the ASME Operation and Maintenance (OM) Code (Ref. 5), mandates that systems and components of boiling and pressurized water-cooled nuclear power reactors meet these codes, and provides conditions on their use. Although 10 CFR 50.55a is not currently applicable to non-LWRs, these codes and the NRC's conditions on their use may be relevant guidance for non-LWRs, similar to the GDCs in Appendix A to 10 CFR Part 50.

- RG 1.28, "Quality Assurance Program Criteria (Design and Construction)" (Ref. 6), provides guidance for the establishment and execution of Quality Assurance programs for nuclear power plants during their design and construction.

- RG 1.193, "ASME Code Cases Not Approved for Use" (Ref. 7), lists the Code Cases for ASME Code, Section III, Section XI, and the ASME OM Code that the NRC has not approved for generic use in 10 CFR 50.55a.

- RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors" (Ref. 8), describes the NRC's proposed guidance on how the GDCs in 10 CFR Part 50, Appendix A, may be adapted for non-LWR designs.

- RG 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors” (Ref. 9), describes the NRC’s proposed guidance on using a technology-inclusive, risk-informed, and performance-based methodology to inform the licensing basis and content of applications for non-LWRs, including, but not limited to, molten salt reactors, high-temperature gas-cooled reactors, and a variety of fast reactors at different thermal capacities.

Purpose of Regulatory Guides

The NRC issues RGs to describe methods that are acceptable to the staff for implementing specific parts of the agency’s regulations, to explain techniques that the staff uses in evaluating specific issues or postulated events, and to describe information that the staff needs in its review of applications for permits and licenses. Regulatory guides are not NRC regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs are acceptable if supported by a basis for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Parts 50 and 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), under control numbers 3150-0011 and 3150-0151, respectively. Send comments regarding this information collection to the FOIA, Library, and Information Collections Branch ((T6-A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555 0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011 and 3150-0151), Office of Management and Budget, Washington, DC, 20503.

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.

B. DISCUSSION

Reason for Issuance

The NRC staff is issuing this RG to provide applicants and licensees of non-LWRs an acceptable method for developing and implementing a PSI and ISI program. This RG endorses with conditions ASME Code, Section XI, Division 2 for use by non-LWR licensees.

Background

NRC regulations in 10 CFR 50.34(b)(6)(iv) and 52.79(a)(29)(i) require all applicants for operating and combined licenses to include plans for conducting normal operations, including maintenance, surveillance, and periodic testing of SSCs. However, the regulations prescribe specific preservice and inservice inspection program requirements only for boiling and pressurized water-cooled nuclear power reactors.[1] Nevertheless, as described below, the GDCs in 10 CFR Part 50, Appendix A, as applicable to non-LWR designs, indicate the importance of an adequate preservice and inservice inspection program.

RG 1.232 provides guidance on how the GDCs in 10 CFR Part 50, Appendix A may be adapted for non-LWR designs.[2] Appendix A to RG 1.232 contains the advanced reactor design criteria (ARDC). These criteria are generally applicable to six different types of non-LWR technologies (i.e., sodium-cooled fast reactors, lead-cooled fast reactors, gas-cooled fast reactors, modular high-temperature gas-cooled reactors, fluoride high-temperature reactors, and molten salt reactors).

Within Appendix A to RG 1.232 are several ARDC that relate to SSC testing:

- ARDC-1 states that SSCs important to safety are to be tested to quality standards commensurate with the importance of the safety functions to be performed.

- ARDC-14 states that the reactor coolant boundary shall be tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- ARDC-30 indicates that the components that are part of the reactor coolant boundary shall be tested to the highest quality standards practical.

- ARDC-32 provides that the components that are part of the reactor coolant boundary shall be designed to permit periodic inspection and functional testing of important areas and features to assess their structural and leaktight integrity.

- ARDC-36 indicates that a system that provides emergency core cooling shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.

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1 The regulations in 10 CFR 50.55a require that systems and components of boiling- and pressurized-water-cooled nuclear power reactors meet the requirements for PSI and ISI in ASME Code, Section XI, Division 1, "Rules for Inspection and Testing of Components of Light-Water-Cooled Plants" (Ref. 10).

2 While the GDCs establish minimum requirements for the PDC for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission, the GDCs are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the PDC for such other units.

- ARDC-39 states that the containment heat removal system shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.

- ARDC-42 provides that the containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping, to assure the integrity and capability of the systems.

- ARDC-45 indicates that the structural and equipment cooling systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to ensure the integrity and capability of the systems.

- ARDC-53 states that the containment structure shall be designed to permit appropriate periodic inspection of all important areas, such as penetrations.

The above ARDC indicate the need for inspection activities, and as mentioned above, the NRC requires applicants for operating and combined licenses to include plans for maintenance, surveillance, and periodic testing of SSCs, which boiling and pressurized water-cooled nuclear power reactors satisfy by implementing ASME Code, Section XI, Division 1 as incorporated in 10 CFR 50.55a. However, neither the ARDC nor the regulations of 10 CFR Part 50 or 52 indicate the appropriate standards to which a PSI/ISI program is to be developed and implemented for non-LWRs.

ASME Code, Section XI, Division 2, provides a process for developing a RIM program similar to a traditional PSI and ISI program under ASME Code, Section XI, Division 1, for all types of nuclear power plants. The RIM program contains provisions beyond a traditional program, such as significant use of probabilistic risk assessment (PRA) to develop reliability targets for SSCs within the scope of the program. It also relies on establishing such practices as monitoring, nondestructive examination and repair and replacement to maintain the reliability of components based on the degradation mechanisms that may exist throughout the life of the plant.

ASME Code, Section XI, Division 2, also provides a process for the identification of the scope, degradation mechanisms, and reliability targets for the in-scope SSCs; identification and evaluation of RIM strategies and uncertainties; program implementation; performance monitoring; and program updates to be applied for passive components to give assurance that the reliability will meet preestablished targets (developed from the PRA information for the facility). ASME Code, Section XI, Division 2, does not stipulate any specific strategies to be employed but calls for these to be developed by expert panels, considering types of examinations currently used for ASME Code, Section XI, Division 1, and known or potential degradation mechanisms for typical materials used in the construction of nuclear facilities. The acceptance criteria used in Appendix VII to ASME Code, Section XI, Division 2, are based on the acceptance criteria in ASME Code, Section XI, Division 1.

The NRC staff reviewed the 2019 Edition of ASME Code, Section XI, Division 2, to determine whether the code is an acceptable approach to developing a RIM program for non-LWRs. Because ASME Code, Section XI, Division 1, provides requirements for a PSI and ISI program for an LWR, the scope of this RG focuses on non-LWRs. The discussions below summarize the NRC staff's observations about ASME Code, Section XI, Division 2, and the potential issues or concerns that may arise through its implementation. These observations led to conditions that are identified in Section C of this RG that should be applied to the use of ASME Code, Section XI, Division 2 for non-LWRs.

Bases for NRC Staff Regulatory Guidance Positions

The following items discuss the basis for the NRC staff's positions stated in Section C of this RG on potential issues or concerns when implementing ASME Code, Section XI, Division 2 for non-LWRs.

Basis for Regulatory Guidance Position 1

Since current regulations specifying standards to be used for preservice and inservice inspection programs are only applicable to light water reactors, an applicant for a non-LWR operating license should propose a license condition to implement an PSI/ISI program. If ASME Code, Section XI, Division 2 is to be used, the license condition should state that the program will follow this RG. For example, an applicant should propose a license condition similar to the following or with appropriate modifications:

> [Licensee] must implement and maintain in effect all provisions of a Reliability and Integrity Management program that satisfy the requirements of ASME Code, Section XI, Division 2, 2019 Edition, as further conditioned by the NRC's endorsement in RG 1.246, "Acceptability of ASME Code, Section XI, Division 2, 'Requirements For Reliability And Integrity Management (RIM) Programs For Nuclear Power Plants,' For Non-Light Water Reactors," Revision 0. Any departure from ASME Code, Section XI, Division 2, 2019 Edition, or the conditions and limitations in RG 1.246 must be submitted to the NRC for review and approval in accordance with 10 CFR 50.90.

As discussed above, the regulations in 10 CFR 50.34(b)(6)(iv) and 52.79(a)(29)(i) require all applicants for operating and combined licenses to include plans for conducting normal operations, including maintenance, surveillance, and periodic testing of SSCs. Within Appendix A to RG 1.232 are several ARDC that relate to SSC testing. ARDC 1 states that SSCs important to safety are to be tested to quality standards commensurate with the importance of the safety functions to be performed.

The license condition is necessary to ensure the program exists and will be implemented to ensure the safe operation of the SSCs throughout the life of the facility. The license condition will support the staff's finding that 10 CFR 50.34(b)(6)(iv) and 52.79(a)(29)(i) are met and the program is commensurate with the safety function provided by the SSC in accordance with the guidelines of RG 1.232. To make the required findings, the staff will need to review a summary of the RIM program, which should include the following:

- A summary of the information described in ASME Section XI, Division 2, Article RIM-2;
- The bases for the scope of the program;
- The methodology for establishing the reliability targets;
- The methodology for determining that the reliability targets will be satisfied by the RIM strategies;
- Reliability targets;
- The results of the degradation assessments;
- Flaw evaluation acceptance criteria; and
- RIM strategies selected to achieve the reliability targets.

In addition to the summary of the RIM program, the application should include the following information for the reasons described in the following sections:

- Any information the applicant developed to replace provisions of the 2019 Edition of ASME Code, Section XI, Division 2 described as "in the course of preparation" or otherwise under development;

- Considerations specified by ASME Section XI, Division 2, as discussed in Regulatory Guidance Position 3;
- Justification for adequacy of the PRA in terms of scope, level of detail, and technical adequacy for use with RIM and the development of reliability targets and the documented results of a PRA peer review, if one has been conducted;
- For plants that do not have regularly scheduled refueling outages at frequencies of 5 years or less, proposed frequencies for the submission of owner's activity reports (OARs) and owner's repair/replacement certification (NIS-2) forms;
- Appropriate justification for flaw evaluation acceptance criteria for any components that exceed the temperature ranges in ASME Code, Section III, Division 1;
- Any alternate examination methods developed under ASME Code, Section XI, Division 2, Appendix A that the applicant wishes to use;
- Any alternatives to Section XI, Division 2, as endorsed with conditions by this RG, that the applicant wishes to use and justification supporting the use of the alternatives.

Basis for Regulatory Guidance Position 2

The regulations in 10 CFR 50.55a(b) contain conditions on the use of ASME Code, Section XI, Division 1. These conditions should also be applied where ASME Code, Section XI, Division 2 references sections of Division 1 with applicable conditions in 10 CFR 50.55a(b). When an applicant or licensee references ASME Code, Section XI, Division 1, the edition that is being referenced should correspond to the edition of ASME Code, Section XI, Division 2, that is being used. This RG endorses the 2019 Edition of Division 2, therefore, applicants and licensees applying this RG should reference the 2019 Edition of Division 1.

Basis for Regulatory Guidance Position 3

ASME Code, Section XI, Division 2, specifies that considerations be made for areas including, but not limited to, those listed in the table below. The summary of the RIM program included in the application should contain sufficient information for the NRC staff to determine how these aspects are considered within the RIM program to assure that the process within ASME Code, Section XI, Division 2, has been followed appropriately.

RIM Paragraph	**Code Provision**
RIM-2.3	Conditions shall be considered in the degradation mechanism assessment.
RIM-2.3	Screening criteria found in Mandatory Appendix VII are minimum requirements to be considered.
RIM-2.4.2	The allocation of Reliability Targets shall consider the uncertainty inherent in the prediction of SSC reliability.
RIM-2.7.4	The RIM program shall consider the design requirements of SSCs that are identified as part of the RIM strategies established in RIM-2.5.
RIM-2.7.7	The examination volumes and methods that shall be considered by the monitoring and nondestructive examination (MANDE) expert panel (MANDEEP) in establishing MANDE criteria.
RIM-2.10	Additional considerations for RIM program implementation.
App V	The referenced Division 1 figures in this appendix are solely used to denote typical geometries for various SSCs containing welds and related examination volumes that the MANDEEP should consider.

App VII	Tables VII-1.2-1 and VII-3.2-1: Note (1) indicates restricted inspection access and spatial phenomena must be considered in the development or application of MANDE criteria developed by the MANDEEP.

Basis for Regulatory Guidance Position 4

ASME Code, Section XI, Division 2, contains instructions to establish the scope of the RIM program along with determination of degradation mechanisms and reliability targets for SSCs that are within the scope of the RIM program. This is followed by the development of RIM strategies such as ISI examinations; testing; MANDE; and maintenance, repair, and replacement practices for the SSCs. The NRC staff will review the RIM program summary to verify that the provisions of ASME Code, Section XI, Division 2, for establishing reliability targets have been properly implemented and to ensure that reliability targets are established using appropriate PRA inputs and that the RIM strategies will ensure the reliability targets can be achieved.

Paragraph RIM-2.7.1 of ASME Code, Section XI, Division 2, lists the information RIM program should contain. The NRC staff will review and approve the program summary to ensure that RIM-2.7.1 has been met. The staff's review will focus on the appropriateness of: (1) the bases for the scope of the program; (2) the method for establishing the reliability targets; (3) the reliability targets; (4) degradation assessments; (5) flaw acceptance criteria; (6) RIM strategies; and (7) the method used to demonstrate the RIM strategies will be able to satisfy the reliability targets. After approval, based on the license condition described in Regulatory Position 1, the licensee will be able to make changes to the RIM program in accordance with ASME Code, Section XI, Division 2 without prior NRC approval, except that changes to the methods to establish reliability targets and to demonstrate the reliability targets will be met will require NRC review and approval.

If alternatives to ASME Code, Section XI, Division 2 are desired, these are also to be submitted to the NRC for review and approval.

Paragraph RIM-4.2.4 permits the use of volumetric and surface examination in lieu of leak testing for areas affected by repair/replacement activities. The NRC staff does not consider NDE to be an acceptable substitute for leak testing and does not endorse the use of paragraph RIM-4.2.4 without prior NRC approval.

Subsequent to NRC approval, licensees should document changes to the RIM program and provide notification to the staff via the owner's activity report (OAR) or in a letter to the NRC. Licensees should submit the notification prior to the next refueling outage or within 3 years, whichever is less. The staff considers three years to be an adequate timeframe for the licensee to respond without unnecessary burden for the licensee. The NRC will review changes as appropriate under the regional inspection program.

Use of nonmandatory Appendix A, "Alternate Requirements for NDE and Monitoring," is not endorsed by the NRC, and use of Appendix A will need to be submitted to the NRC for review and approval. Use of Appendix A may involve complex analysis and warrants review by the NRC to ensure that NDE and monitoring continue to provide adequate safety.

The OARs and NIS-2 forms that are generated through implementation of the RIM program, along with any analytical evaluations of inspection results that are conducted in accordance with ASME Code, Section XI, Division 2, should be submitted to the NRC. Although ASME Code, Section XI, Division 2 specifies in nonmandatory Appendix B, "Regulatory Administrative Provisions for Nuclear Plants Using RIM Program," to submit the OAR 90 days after completion of an outage, the staff finds120 days to be acceptable as the 120-day frequency is consistent with ASME Code, Section XI, Division 1. This is an administrative process that should be the same for non-LWR as for LWR licensees and providing this information within 120 days after an outage has been found by the NRC staff to be acceptable for licensees implementing Division 1 without placing undue burden on the licensee.

For plants that do not have regularly scheduled refueling outages at frequencies of 5 years or less, applicants should propose frequencies for the submission of OARs and NIS-2 forms. If the RIM program includes online MANDE, then the proposed frequencies for the submission should be based on the frequency of the inspections and any indications or anomalies identified during the online MANDE. Also, the results of any analytical evaluations of inspection results that are conducted in accordance with ASME Code, Section XI, Division 2, from online inspections should be provided to the NRC within 120 days of the inspection.

The flaw evaluation acceptance criteria in Appendix VII to ASME Code, Section XI, Division 2 are based on acceptance criteria in ASME Code, Section XI, Division 1. The use of these flaw evaluation acceptance criteria is acceptable provided the temperature ranges do not exceed those in ASME Code, Section III, Division 1. Although non-LWR licensees are not required to meet Section III, the NRC staff has not reviewed the flaw evaluation acceptance criteria for use in temperature ranges above those described in Section III, Division 1, and cannot endorse their use at such temperatures. Should the applicable temperatures for a component exceed these temperature ranges, the applicant or licensee will need to establish the acceptance criteria and submit to the NRC for review and approval.

Basis for Regulatory Guidance Position 5

Qualification of NDE personnel is not dependent on the type of plant. Therefore, previous NRC staff positions on qualification of NDE personnel also apply to non-LWRs. Use of ASME NDE (ANDE)-1-2015, “Nondestructive Examination and Quality Control Central Qualification and Certification Program” (Ref. 11), was not approved by the NRC when proposed under ASME Code Case N-788-1 “Third Party NDE Certification Organizations Section XI, Division 1” (Ref. 12). RG 1.193 documents that this Code Case is not generally acceptable for use. The NRC has been following and participating in the development of ANDE for several years and is reviewing ANDE as it progresses. Code Case N-788-1 and the ANDE standard do not contain sufficient specificity for use as a qualification or certification program. Several important sections of ASME ANDE-1-2015 are not defined and are to be determined in the future by specific industry sector committees. It is not possible for the NRC to evaluate a certification and qualification program that has not been defined. For this reason, Code Case N-788-1 and the referenced ANDE-1-2015 are not sufficient on their own as a qualification and certification program able to be used as an alternative to ASME Code, Section XI, Division 1, Subarticle IWA-2300. The American National Standards Institute/American Society for Nondestructive Testing (ANSI/ASNT)-CP-189, “Standard for Qualification and Certification of Nondestructive Testing Personnel” (Ref. 13), may be used as provided in ASME Code, Section XI, Division 1.

Basis for Regulatory Guidance Position 6

Section RIM-1.9 addresses standards and specifications referenced within ASME Code, Section XI, Division 2. Table RIM-1.9.1, column “Revision Date or Indicator,” specifies multiple editions or “latest edition” for some standards, methods, or specifications. For the NRC to find the ASME Code, Section XI, Division 2 to be acceptable for use, applicants should cite specific editions for use with the code. An edition of ASME NQA-1, “Quality Assurance Requirements for Nuclear Facility Applications” (Ref. 144), endorsed by the NRC in RG 1.28, should be used. With respect to PRA standard ASME/ANS RA-S-1.4, licensees or applicants should reference Trial Use RG 1.247, “Acceptability of Probabilistic

Risk Assessment Results for Advanced Non-Light Water Reactor Risk-Informed Activities,"[3] which endorses, with conditions, ASME/ANS RA-S-1.4-2021 edition, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants." (Ref. 15).

Basis for Regulatory Guidance Position 7

The applicant or licensee should justify why the PRA is adequate in terms of scope, level of detail, and technical adequacy for use with RIM and the development of reliability targets. If a PRA peer review was conducted prior to the application, the staff would examine the documented results. An acceptable peer review is one which is performed according to an established process and by qualified personnel to document the results and to identify strengths and weaknesses of the PRA. The staff may, under certain circumstances, decide to perform an audit to verify the technical adequacy of the PRA and the disposition of any PRA peer review findings.

Basis for Regulatory Guidance Position 8

ASME Code, Section XI, Division 2 would permit using alternate examination methods in lieu of the examination requirements specified in the construction code. The NRC does not approve the use of alternate NDE in lieu of NDE requirements in a licensee's construction code,[4] e.g., ASME Code Section III. ASME Code, Section XI, Division 2 is intended to work in conjunction with the construction code to ensure reliability targets are met. When it is determined that a modification to the design is necessary to achieve appropriate reliability targets, the modification is governed by the construction code for the plant. When alternate examination methods are to be substituted for the methods specified in the construction code, that change must be made in accordance with the construction code and may need NRC staff review and approval.

Basis for Regulatory Guidance Position 9

ASME Code, Section XI, Division 2 provides appropriate time constraints for completing PSIs for components initially selected for examination in accordance with the RIM program but lacks information related to the timing for completion of preservice examinations that may be needed due to activities such as repair and replacement, modifications that may add components, or changes that may add existing components into the scope of the RIM program that were not initially included. Paragraph RIM-2.7.3 specifies that preservice examinations shall be documented using encoded equipment and

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3 Trial Use RG 1.247 had not been issued as of September 2021, but is expected to be issued by December 2021, prior to this regulatory guide being issued in final form. Plans to issue Trial Use RG 1.247 were discussed at the August 26, 2021, Advanced Reactor Stakeholders Public Meeting (see the meeting presentation at ADAMS Accession No. ML21237A463). This draft RG (DG-1383) does not take a position on the acceptability of ASME/ANS RA-S-1.4-2021 edition, it merely notes that it is the edition that the NRC staff has endorsed for use. Trial Use RG 1.247, endorses, with conditions, ASME/ANS RA-S-1.4-2021 edition; however, applicants and licensees should be aware that the trial use nature of the endorsement means that the NRC staff may depart from it in licensing, make changes to the endorsement in the future, or withdraw it altogether and that such actions would not constitute backfitting under 10 CFR 50.109, "Backfitting" or forward fitting as described in Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests" (Ref. 16), or affect the issue finality of an approval under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

4 Under 10 CFR 50.55a, systems and components of boiling and pressurized water-cooled nuclear power reactors meet ASME Code Section III. Section III provides standards for the design and construction of nuclear facility components and supports. Non-LWRs are not required to meet ASME Code Section III, however, the NRC staff anticipates that most applications for construction permits or combined licenses for non-LWRs will reference a standard for design and construction—a "construction code"—such as Section III, Division 5, which the NRC staff is currently considering for endorsement for use by non-LWRs.

scanners. It is the NRC staff's understanding that this is intended for volumetric examinations and is not applicable to other preservice examinations, such as visual and surface examinations.

Basis for Regulatory Guidance Position 10

Areas within the 2019 Edition of ASME Code, Section XI, Division 2, that are "in the course of preparation" or otherwise under development are not being endorsed in this RG. These areas are:

- examination requirements/figure number and acceptance standards for item 2.90, tubes and tubesheet, and item 2.100, tubes, in Table V-1.1-2, "Examination Category B, Pressure -Retaining Welds in Vessels Other Than Reactor Vessels," in Mandatory Appendix V, "Catalog of NDE Requirements and Areas of Interest," and
- the following Articles in Mandatory Appendix VII, "Supplements for Types of Nuclear Plants":
    - Article VII-2, "Supplement for Liquid Metal Reactor-Type Plants;"
    - Article VII-4, "Supplement for Molten Salt Reactor-Type Plants;"
    - Article VII-5, "Supplement for Generation 2 LWR Reactor-Type Plants;" and
    - Article VII-6, "Supplement for Fusion Machine-Type Plants."

Where ASME Code, Section XI, Division 2 states that provisions are "in the course of preparation" or otherwise under development, applicants may need to develop that information. Such information should be submitted to the NRC for review and approval in the application.

Basis for Regulatory Guidance Position 11

Mandatory Appendix V includes tables for examination categories. These tables mimic comparable tables in ASME Code, Section XI, Division 1, which were written for plants that use pressurized components. Several of the tables include pressure-retaining in the title of the table. Although non-LWRs may operate at atmospheric or near atmospheric pressure, these ASME Code, Section XI, Division 1, tables should be considered in the development of RIM strategies and MANDE requirements regardless of the design pressure of the components. This is because some pertinent inspections may be gleaned from considering these examinations. If the operating and accident pressures are very low and will not challenge the integrity of the component, this should be considered in the process for establishing RIM strategies and MANDE requirements.

Basis for Regulatory Guidance Position 12

Mandatory Appendix IV, "Monitoring and NDE Qualification," Article IV-5, "Records," instructs the user that the MANDEEP shall establish record retention requirements. Records generated under a PSI/ISI program are quality records that come under the record retention requirements of the quality assurance program established under 10 CFR 50 Appendix B for the facility and should be retained in accordance with requirements established in the quality assurance program.

Basis for Regulatory Guidance Position 13

Mandatory Appendix VII contains tables of degradation mechanism attributes and attribute criteria in the supplements for various reactor types. The NRC staff noted that the tables did not identify stress relaxation cracking (sometimes referred to as reheat cracking) as a degradation mechanism. Stress relaxation cracking should be considered in addition to the degradation mechanisms in Mandatory Appendix VII degradation mechanism tables.

Basis for Regulatory Guidance Position 14

RIM-4.2.3 calls for liquid leak tests to be conducted in accordance with ASME Code, Section V, “Nondestructive Examination” (Ref. 17), Article 10, Mandatory Appendix IV, “Helium Mass Spectrometer Test,” or Mandatory Appendix VI, “Pressure Change Test.” A helium mass spectrometer test is not a liquid test and would not be consistent with the remaining paragraphs in RIM-4.2.3. Also, ASME Code, Section V, Article 10, does not provide instructions for conducting a pressure test and performing a visual test to identify leakage. The NRC staff believes that a pressure drop test or a pressure test in conjunction with a visual test are appropriate means for liquid leak tests. RIM-4.2.3 establishes a hold time of 10 minutes for components not required for normal plant operation. The NRC staff views this as insufficient for ensuring small leaks are detected on insulated components. It is the NRC staff’s position that the hold time should be 4 hours for components that are insulated.

Basis for Regulatory Guidance Position 15

The 2019 Edition of ASME Code, Section XI, Division 2, included a few inadvertent errors in paragraphs VII-1.4.2.3 and VII-3.4.2.3 and in article references in nonmandatory Appendix A. These are inconsequential errors. These items have been identified to the applicable ASME Code committees to address either by errata or by changes to the code.

Paragraphs VII-1.4.2.3 and VII-3.4.2.3 for linear flaws in Category D, for full penetration welds of nozzles in vessels refers to figures IWC-2500-1 and IWC-2500-2. These figures relate to vessel welds, not nozzle welds in vessels. Figures IWB-2500-7(a) through IWB-2500-7(d) or IWC-2500-3 and Figure IWC-2500-4 should be referred to in paragraphs VII-1.4.2.3 and VII-3.4.2.3 rather than figures IWC-2500-1 or IWC-2500-2. IWC-2500-3 and IWC-2500-4 are for nozzle welds in vessels.

Paragraphs A-2.4 and A-3.4.1 refer to Article A-6, “Records and Report,” for probability of failure calculations and structural evaluations. Article A-5, “Procedure for Structural Reliability Evaluation for Passive Components,” is the appropriate article to reference rather than Article A-6. Paragraph A-3.3 refers to Article A-7, “References.” Article A-6 is the appropriate article to reference when discussing documentation requirements.

Consideration of International Standards

The International Atomic Energy Agency (IAEA) works with member states and other partners to promote the safe, secure, and peaceful use of nuclear technologies. The IAEA develops Safety Requirements and Safety Guides for protecting people and the environment from harmful effects of ionizing radiation. This system of safety fundamentals, safety requirements, safety guides, and other relevant reports, reflects an international perspective on what constitutes a high level of safety. To inform its development of this RG, the NRC has considered IAEA Safety Requirements and Safety Guides pursuant to the Commission’s International Policy Statement (Ref. 18) and Management Directive and Handbook 6.6, “Regulatory Guides” (Ref. 19).

The following IAEA Safety Guide incorporates similar design and preoperational testing guidelines and is consistent with the basic safety principles considered in developing this RG:

- IAEA Safety Guide NS-G-2.6, “Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants,” issued 2002 (Ref. 20).

Documents Discussed in Staff Regulatory Guidance

This RG endorses, in part, the use of one or more codes or standards developed by external organizations, and other third party guidance documents. These codes, standards and third party guidance documents may contain references to other codes, standards or third party guidance documents ("secondary references"). If a secondary reference has itself been incorporated by reference into NRC regulations as a requirement, then licensees and applicants must comply with that standard as set forth in the regulation. If the secondary reference has been endorsed in a RG as an acceptable approach for meeting an NRC requirement, then the standard constitutes a method acceptable to the NRC staff for meeting that regulatory requirement as described in the specific RG. If the secondary reference has neither been incorporated by reference into NRC regulations nor endorsed in a RG, then the secondary reference is neither a legally-binding requirement nor a "generic" NRC approved acceptable approach for meeting an NRC requirement. However, licensees and applicants may consider and use the information in the secondary reference, if appropriately justified, consistent with current regulatory practice, and consistent with applicable NRC requirements.

C. STAFF REGULATORY GUIDANCE

The NRC staff finds the use of ASME Code, Section XI, Division 2, 2019 Edition, as an acceptable means to develop a RIM program for non-LWRs with the conditions noted below.

1. An applicant that intends to use ASME Code, Section XI, Division 2 to implement a RIM program for preservice and inservice inspections for a non-LWR should propose a license condition to use ASME Code, Section XI, Division 2, as conditioned in this RG, in their application.

To support the NRC's review, the application should also include a summary of the RIM program implementing this RG, which should include the following:

- A summary of the information described in ASME Section XI, Division 2, Article RIM-2;
- The bases for the scope of the program;
- The methodology for establishing the reliability targets;
- The methodology for determining that the reliability targets will be satisfied by the RIM strategies;
- Reliability targets;
- The results of the degradation assessments;
- Flaw evaluation acceptance criteria; and
- RIM strategies selected to achieve the reliability targets.

In addition to the proposed license condition and summary of the RIM program, the application should include the following information:

- Any information the applicant developed to replace provisions of the 2019 Edition of ASME Code, Section XI, Division 2 described as "in the course of preparation" or otherwise under development;
- Considerations specified by ASME Section XI, Division 2, as discussed in Regulatory Guidance Position 3;
- Justification for adequacy of the PRA in terms of scope, level of detail, and technical adequacy for use with RIM and the development of reliability targets and the documented results of a PRA peer review, if one has been conducted;
- For plants that do not have regularly scheduled refueling outages at frequencies of 5 years or less, proposed frequencies for the submission of OARs and NIS-2 forms;
- Appropriate justification for flaw evaluation acceptance criteria for any components that exceed the temperature ranges in ASME Code, Section III, Division 1;
- Any alternate examination methods developed under ASME Code, Section XI, Division 2, Appendix A that the applicant wishes to use;
- Any alternatives to Section XI, Division 2, as endorsed with conditions by this RG, that the applicant wishes to use and justification supporting the use of the alternatives.

2. The 2019 Edition of ASME Code, Section XI, Division 2, should be used in conjunction with the 2019 Edition of ASME Code, Section XI, Division 1, and the licensee's applicable construction code. When using the 2019 Edition of ASME Code, Section XI, Division 1, and the applicable construction code, applicants and licensees should follow any applicable conditions on those codes in 10 CFR 50.55a.

3. When ASME Code, Section XI, Division 2, indicates that an aspect should be considered in the development of the RIM program, the licensee should document how it considered that aspect.

4. Following issuance of a license with the condition identified in Regulatory Position 1 above, licensees may make changes to the RIM program in accordance with the procedures in ASME Code,

Section XI, Division 2. Changes to the RIM program that are not in accordance with ASME Code, Section XI, Division 2, as endorsed in this RG, should be submitted to the NRC for review and approval. In addition, regardless of any applicable provisions to the contrary in ASME Code, Section XI, Division 2, the following information should be submitted to the NRC for review and approval prior to implementation:

- Changes to the methodologies for establishing reliability targets and for demonstrating the RIM strategies will be capable of satisfying the reliability targets should be submitted to the NRC for review and approval. Inputs to the methodology (e.g., different inspection techniques, changes in inspection frequency, etc.) are not considered methodology changes.

- Alternatives to ASME Code, Section XI, Division 2 as endorsed with conditions by this RG, should be submitted to the NRC for review and approval.

- Use of volumetric and surface examination in lieu of leak testing for areas affected by repair/replacement activities, under paragraph RIM-4.2.4, is not endorsed by the NRC and should be submitted for review and approval obtained before placing systems into service.

- RIM program changes involving alternate examination methods developed under ASME Code, Section XI, Division 2, Appendix A, should be submitted for review and approval before use.

- If the applicable temperatures for a component exceed the temperature ranges allowed in ASME Code, Section III, Division 1, the applicant or licensee will need to establish the flaw evaluation acceptance criteria and submit to the NRC for review and approval.

- Changes to the schedule for submitting OARs and NIS-2 forms.

The following should be submitted to the NRC for information as described below:

- After NRC review and approval of the program, the licensee should notify the NRC for information of any RIM program changes. Notification is to be provided prior to the next refueling outage or within 3 years of making the change, whichever is less.

- OARs and NIS-2 forms should be submitted for information either within 120 days of completion of an outage or according to an alternate schedule reviewed and approved in the license application.

- Analytical evaluations of examination results should be submitted for information within 120 days of completion of an outage or within 120 days of the examination, as appropriate.

5. Use of ANDE-1-2015 is not endorsed by the NRC staff for the qualification of inspection personnel. Standard ANSI/ASNT-CP-189 should be used as provided in ASME Code, Section XI, Division 1.

6. Licensees should use an edition of ASME NQA-1 endorsed by the NRC in RG 1.28 and the ASME/ANS RA-S-1.4-2021 edition endorsed with conditions in Trial Use RG 1.247.

7. The owner should justify, in the RIM program, acceptability of the PRA for use in developing a RIM program and establishing reliability targets.

8. The provisions of ASME Code, Section XI, Division 2, should not be used to depart from matters governed by the construction code of the plant.

9. PSI requirements should be completed before placing a component in service after repair or replacement activities and for new components added as a result of modification activities. If components are added to the scope of the RIM program due to program changes, the components should receive a preservice examination at the next outage or at the earliest opportunity. Paragraph RIM 2.7.3, for encoding preservice examinations, applies only to volumetric examinations.

10. Where ASME Code, Section XI, Division 2 states that provisions are "in the course of preparation" or otherwise under development, applicants should develop any necessary information. The NRC will review this information on a case-by-case basis.

11. Mandatory Appendix V tables should be considered in the development of MANDE requirements and RIM strategies for components designed to very low or atmospheric pressures.

12. Regardless of the provisions in ASME Code, Section XI, Division 2, any records that would be considered quality records should be retained in accordance with the applicable provisions of the quality assurance program for the facility.

13. Stress relaxation cracking should be considered in addition to the degradation mechanisms in Mandatory Appendix VII degradation mechanism tables.

14. Liquid leak tests should be conducted using ASME Code, Section V, Article 10, Mandatory Appendix VI or the leak test instructions in the remaining paragraphs in RIM-4.2.3. Hold times for insulated components not required for normal plant operations should be 4 hours after attaining pressure.

15. When applying the acceptance standards to linear flaws in Category D, for full penetration welds in vessels, the figures in IWB-2500-7(a) thru IWB-2500-7(d) or IWC-2500-3 and IWC-2500-4 should be used. Article A 5 should be used where Article A 6 is referenced in nonmandatory Appendix A, paragraphs A 2.4 and A 3.4.1. Article A 6 should be used where Article A 7 is referenced in nonmandatory Appendix A, paragraph A 3.3.

D. IMPLEMENTATION

The NRC staff may use this RG as a reference in its regulatory processes, such as licensing, inspection, or enforcement. However, the NRC staff does not intend to use the guidance in this RG to support NRC staff actions in a manner that would constitute backfitting as that term is defined in 10 CFR 50.109, and as described in NRC Management Directive 8.4, nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52. The NRC staff also does not intend to use the guidance to support NRC staff actions in a manner that constitutes forward fitting as that term is defined and described in Management Directive 8.4. If a licensee believes that the NRC is using this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfitting or forward fitting appeal with the NRC in accordance with the process in Management Directive 8.4.

GLOSSARY

Analytical Evaluation	A quantitative process to determine the acceptability of postulated flaws, or flaws that exceed the applicable acceptance standards, including predicted future growth, to determine whether a component is acceptable for continued service without a repair or replacement activity. See ASME Code, Section XI, Division 1.
Boiling-Water-Cooled Nuclear Power Reactor	A reactor type that used water that is converted to steam in the reactor core to cool the core and turn a turbine generator.
Inservice Inspection (ISI) Program	The plan and schedule for performing actions for assuring the structural and pressure-retaining integrity of safety-related nuclear power plant components in accordance with the rules of ASME Code, Section XI.
MANDE	Monitoring and nondestructive examination (MANDE) is a term used by ASME Code, Section XI, Division 2, that includes the activities of monitoring, nondestructive examination (NDE), and use of surveillance specimens, as established by the Monitoring and NDE Expert Panel.
MHTGR	Modular high-temperature gas-cooled reactor. This reactor type operates with a cooling medium, typically helium. The reactor operates at higher temperatures than current light water reactors. The core is generally supported by graphite, which conducts heat away from the core if the normal cooling medium is lost.
NIS-2 Form	Form used by the ASME to document repair and replacement activities on ASME Code-designed components.
Non-Light Water Reactor	Nuclear reactor design that cools the reactor core without ordinary water.
Owner's Activity Report	The owner's activity report, OAR-1, is a form in ASME Code, Section XI, Division 2, to document the results of RIM activities that have taken place during a plant outage, or from online inspections, tests, MANDE, and maintenance, repair, and replacement practices that support the RIM program.
Passive Component	A component that has no moving parts and does not depend on an external input such as actuation, mechanical movement, or supply of power. It can experience changes in pressure, temperature, or fluid flow in performing its functions. Some examples of passive components include heat exchangers, pipes, vessels, electrical cables, and structures.
Probabilistic Risk Assessment	A quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of frequency of occurrence and consequences of event sequences, event sequence families, or release categories.
Preservice Inspection (PSI)	Inspections that are conducted on a component after construction examinations but before it is placed into service or after a repair or replacement activity. The intent of a PSI is to establish a baseline to compare future inspection results to determine whether damage is occurring. A preservice examination also provides assurance that no significant flaws from construction were missed and left in the component.

Pressurized-Water-Cooled Nuclear Power Reactor	A nuclear reactor type that pressurizes the water coolant to prevent boiling of the coolant.
Qualification or Certification Program	A program to ensure NDE personnel meet a minimum level of performance to conduct NDE or nondestructive tests.
Reheat Cracking/Creep Stress Cracking	Cracking that occurs in high-strength, low-alloy steels, particularly chromium, molybdenum, and vanadium steels, during post-heating and that may be caused by the poor creep ductility of the heat-affected zone.
Reliability Target	A performance goal established for the probability that an SSC will complete its specified function to achieve plant-level and reliability goals.
RIM Program	A program that defines, evaluates, and implements strategies to ensure that reliability targets for SSCs are defined, achieved, and maintained throughout the plant lifetime. The RIM program is outlined in Article RIM-2 of ASME Code, Section XI, Division 2. A RIM program would be similar to a conventional ISI program but may be extensive, including activities such as ISI; testing; MANDE; and maintenance, repair, and replacement practices.
RIM Strategy	Methods of ensuring a component meets its reliability target. The methods may include, but are not limited to, design strategies; material selection; fabrication procedures; operating practices; PSI; ISI; testing; MANDE; and maintenance, repair, and replacement practices.
Sodium Fast Reactor	A nuclear reactor that is essentially unmoderated and is cooled by liquid sodium.

REFERENCES [5]

1. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 2, "Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants," 2019 Edition, New York, NY, July 1, 2019.[6]

2. *U.S. Code of Federal Regulations* (CFR), "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter 1, Title 10, "Energy."

3. CFR, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter 1, Title 10, "Energy."

4. ASME, Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," Division 1, New York, NY, 2019.

5. ASME, Code for Operation and Maintenance of Nuclear Power Plant, New York, NY, 2020.

6. U.S. Nuclear Regulatory Commission (NRC), Regulatory Guide (RG) 1.28, "Quality Assurance Program Criteria (Design and Construction)," Washington, DC.

7. NRC, 1.193, "ASME Code Cases Not Approved for Use," Washington, DC.

8. NRC, RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," Washington, DC.

9. NRC, RG 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," Washington, DC.

10. ASME, Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1, "Rules for Inspection and Testing of Components of Light-Water-Cooled Plants," 2019 Edition, New York, NY, July 1, 2019.

11. ASME, ANDE-1-2015, "Nondestructive Examination and Quality Control Central Qualification and Certification Program."

12. ASME Code Case, N-788-1, "Third Party NDE Certification Organizations" New York, NY, approved March 6, 2017.

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[5] Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html.The documents can also be viewed on line or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

[6] Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Two Park Avenue, New York, NY 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web-based store at http://www.asme.org/Codes/Publications/.

13. American National Standards Institute/American Society for Nondestructive Testing (ANSI/ASNT) CP-189, "Standard for Qualification and Certification of Nondestructive Testing Personnel," Columbus, OH.[7]

14. ASME, NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," New York, NY.

15. ASME/American Nuclear Society (ANS), RA-S-1.4, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants," New York, NY.

16. NRC, MD 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," Washington, DC.

17. ASME, Boiler and Pressure Vessel Code, Section V, "Nondestructive Examination," New York, NY.

18. NRC, "Nuclear Regulatory Commission International Policy Statement," *Federal Register*, Vol. 79, No. 132, July 10, 2014, pp. 39415-39418.

19. NRC, Management Directive (MD) 6.6, "Regulatory Guides," Washington, DC, May 2, 2016. (ADAMS Accession No. ML18073A170)

20. International Atomic Energy Agency, Safety Guide NS-G-2.6, "Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants," Vienna, Austria, 2002.[8]

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7 Copies of ASNT standards may be purchased from the ASNT Web site (https://www.asnt.org/Store/) or by contacting ASNT customer service at (800) 222-2768 or (614) 274-6003 or e-mail at customersupport@asnt.org.

8 Copies of International Atomic Energy Agency (IAEA) documents may be obtained through its Web site: www.iaea.org or by writing the International Atomic Energy Agency, P.O. Box 100, Wagramer Strasse 5, A-1400 Vienna, Austria.