U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.178, REVISION 2

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PLANT-SPECIFIC, RISK-INFORMED DECISIONMAKING FOR INSERVICE INSPECTIONS OF PIPING

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 developing risk-informed inservice inspection (RI-ISI) programs. This guide describes acceptable methods for using information from a probabilistic risk assessment (PRA) with deterministic engineering information in the development of RI-ISI programs.

This RG supplements the guidance provided in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Ref. 1), and includes precise terminology to ensure that the defense-in-depth philosophy is interpreted and implemented consistently.

Applicability

This RG applies to light-water reactor (LWR) licensees 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.
  - o 10 CFR 50.55a, "Codes and standards," requires, in part, that systems and components meet the requirements of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code (Ref. 4), as specified in 10 CFR 50.55a(b) and (g).

Written suggestions regarding this guide or development of new guides may be submitted through the NRC's public Web site in the NRC Library at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/, under Document Collections, in Regulatory Guides, at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/contactus.html.

Electronic copies of this RG, previous versions of RGs, and other recently issued guides are also available through the NRC's public Web site in the NRC Library at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/, under Document Collections, in Regulatory Guides. This RG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under ADAMS under Accession Number (No.) ML21036A105. The associated draft guide DG-1288 may be found in ADAMS under Accession No. ML20210M047. The Regulatory Analysis for DG-1288 may be found in ADAMS under Accession No. ML20210M044. No public comments were received on DG-1288; therefore, the staff has not prepared responses to public comments.

- 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," requires that applications for license amendments fully describe the changes desired.

- 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.

Related Guidance

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) (Ref. 5), provides guidance to the NRC staff in performing safety reviews of construction permit or operating license applications (including requests for amendments) under 10 CFR Part 50, and early site permit, design certification, combined license, standard design approval, or manufacturing license applications under 10 CFR Part 52 (including requests for amendments). SRP Section 3.9.8, "Risk-Informed Inservice Inspection of Piping," addresses the guidance for the NRC staff's review of RI-ISI programs.

- RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1" (Ref. 6), lists ASME BPV Code, Section XI, Code Cases that the NRC staff approved for use as voluntary alternatives to the mandatory ASME BPV Code provisions incorporated by reference into 10 CFR 50.55a.

- RG 1.174 provides guidance on an acceptable approach for developing risk-informed applications for a licensing-basis change that considers engineering issues and applies risk insights.

- RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Ref. 7), provides an approach for determining whether the base PRA, in total or the parts used to support an application, is acceptable for use in regulatory decisionmaking for LWRs. Also note that the NRC will update RG 1.200 periodically as it adopts new PRA standards.

Purpose of Regulatory Guides

The NRC issues RGs to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated events, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs will be deemed acceptable if they provide a sufficient basis for the findings required 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 Part 50 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), approval numbers 3150-001. Send comments regarding this information collection to the Information Services Branch (T6-A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0010), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW Washington, DC 20503; e-mail: oira_submission@omb.eop.gov."

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 Revision

This revision of the guide (Revision 2) provides updated guidance for considerations on the defense-in-depth philosophy to be consistent with the related guidance described in RG 1.174. The NRC staff revised RG 1.174 in 2018 (Revision 3) to expand the guidance on the meaning of, and the process for, assessing defense-in-depth considerations. Specifically, this revision of RG 1.178 references the defense-in-depth guidance in Revision 3 of RG 1.174 in several staff regulatory positions.

Additionally, the NRC staff revised this guide to (1) update Section C.2.2, "Evaluation of Risk Impact," of this RG to be consistent with Section C.2.3 in RG 1.174, which provides specific considerations with respect to determining the acceptability of the PRA used in risk-informed decisionmaking, and (2) add the reference to ASME Code Case N-716-1, "Alternative Classification and Examination Requirements, Section XI, Division 1," dated January 27, 2013 (Ref. 8), which describes an RI-ISI process as approved in RG 1.147.

Background

The NRC and the nuclear industry recognized that PRAs have evolved to be useful in supplementing deterministic engineering approaches in reactor regulation. On August 16, 1995, the NRC issued its policy statement on the increased use of PRA in nuclear regulatory activities, titled "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement" (Ref. 9). After the publication of the policy statement, the Commission directed the NRC staff to develop a regulatory framework that incorporated risk insights. The NRC articulated this framework in SECY-95-280, "Framework for Applying Probabilistic Risk Analysis in Reactor Regulation," dated November 27, 1995 (Ref. 10). This guide implements, in part, the Commission's policy statement and the NRC staff's framework for incorporating risk insights into the regulation of nuclear power plants, as further discussed below.

In support of the use of risk-informed decisionmaking, the NRC developed RG 1.174, which provides guidance on an acceptable approach for developing risk-informed applications for a licensing-basis change, considers engineering issues, and applies risk insights. As companion guidance to RG 1.174, SRP Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," describes the NRC staff's review plan to evaluate the technical acceptability of PRA results for risk-informed license amendment requests. This guide supplements the guidance provided in RG 1.174. Specifically, this guide addresses the guidance for developing the RI-ISI programs for piping, with its companion SRP Section 3.9.8. In comparison, RG 1.175, "Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing" (Ref. 11), and RG 1.177, "Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (Ref. 12), provide the guidelines for developing risk-informed inservice testing and risk-informed technical specifications, respectively.

In September 1998, the Commission published a version of this guide for trial use. As stated therein, that RG did not establish any final staff positions for purposes of 10 CFR 50.109, "Backfitting." Any changes to the RG before staff adoption in final form would not be considered backfits. This was intended to ensure that the lessons learned during the trial use period could be addressed, enhancing stability in the review, approval, and implementation of proposed RI-ISI programs.

Subsequently, the NRC staff approved two methods for developing and implementing RI-ISI programs. The Electric Power Research Institute (EPRI) developed one methodology (EPRI TR-112657, "Revise Risk-Informed Inservice Inspection Evaluation Procedure," issued December 1999 (Ref. 13)).

The Westinghouse Owners Group (WOG) developed the other methodology (WCAP-14572, "Westinghouse Owners Group Application of Risk Informed Methods to Piping Inservice Topical Report," issued February 1999 (Ref. 14)). The NRC used the trial RG 1.178 to support the review and approval of the two industry-developed methodologies. Based on the experience during the review and approval of the industry methodologies and of numerous plant-specific relief requests for inservice inspection (ISI) programs, the NRC staff issued Revision 1 of this RG in 2003. In addition, the NRC staff has periodically reviewed this RG and is now issuing this updated version. This revision incorporates the updated guidance on defense in depth and conforms to the latest RG program guidance.

The NRC has not approved the RI-ISI processes described in EPRI TR-112657 and WCAP-14572 for generic use. When licensees intend to use an RI-ISI process not approved for generic use, the NRC staff anticipates that the licensees will incorporate changes to their ISI programs by requesting NRC approval of alternative inspection programs that meet the criteria of 10 CFR 50.55a(z)(1) and provide an acceptable level of quality and safety. As always, licensees should identify how the chosen approach, methods, data, and criteria are appropriate for the decisions they need to make.

Licensees may use ASME Code Cases approved by the NRC as an alternative to compliance with ASME BPV Code provisions that the NRC has incorporated by reference into 10 CFR 50.55a. In 2014, the NRC approved ASME Code Case N-716-1 as an alternative RI-ISI process in RG 1.147. ASME Code Case N-716-1 indicates that a licensee may use the Code Case without requesting NRC approval unless the licensee holds a combined operating license issued after January 1, 2012. When a licensee intends to use Code Case N-716-1 for new reactors, the NRC staff will review operating experience to confirm the adequate use of the Code Case in the development of plant-specific RI-ISI programs. However, the status of ASME BPV Code and Code Cases continues to change, and the NRC staff updates 10 CFR 50.55a and RG 1.147 periodically. Accordingly, licensees should refer to the latest revision of 10 CFR 50.55a and RG 1.147 to identify the provisions of ASME BPV Code and Code Cases on ISI that the NRC has approved.

This guide's principal focus is on the use of PRA findings and risk insights for decisions on changes proposed to plant's ISI programs for piping. Such changes include (but are not limited to) license amendments under 10 CFR 50.90, requests for the use of alternatives under 10 CFR 50.55a, and exemptions under 10 CFR 50.12, "Specific exemptions." This guide describes methods acceptable to the NRC staff for integrating insights from PRA techniques with deterministic engineering analyses into ISI programs for piping.

The current ISI requirements for piping components appear in 10 CFR 50.55a and the general design criteria (GDC) listed in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. The NRC incorporated by reference ASME BPV Code, Section XI, in 10 CFR 50.55a, which addresses the codes and standards for design, fabrication, testing, and inspection of piping systems. The objective of the ISI program is to identify service-induced degradation that might lead to pipe leaks and ruptures, thereby meeting, in part, the requirements set in the GDC and 10 CFR 50.55a. ISI programs are intended to address all piping locations that are subject to degradation. Incorporating risk insights into the programs can focus inspections on the more important locations and reduce personnel exposure, while at the same time maintaining or improving public health and safety.

The justification for any reduction in the number of inspections should confirm that such a reduction would not result in an increase in leakage frequency or a degradation of defense in depth. When categorizing piping segments in terms of their contribution to risk, it is the responsibility of a licensee to ensure that the categorization of piping segments and the resulting inspection programs are consistent with the key principles and risk guidelines (e.g., core damage frequency (CDF) and large early release frequency (LERF)) addressed in RG 1.174). This guide augments the information presented in RG 1.174 by providing guidance specific to incorporating risk insights into ISI programs for piping.

The NRC has recommended additional augmented inspection programs, implemented by the industry, to address generic piping degradation problems to preclude piping failure. Notable examples of augmented programs for piping inspections include the following topics:

- intergranular stress-corrosion cracking of stainless steel piping in boiling-water reactors (BWRs) (Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping," issued January 1988 (Ref. 15))

- thermal fatigue (NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," issued June 1988 (Ref. 16); NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," issued December 1988 (Ref. 17); and NRC Information Notice 93-20, "Thermal Fatigue Cracking of Feedwater Piping to Steam Generators," issued March 1993 (Ref. 18))

- stress-corrosion cracking in pressurized-water reactors (PWRs) (NRC Bulletin 79-17, "Pipe Cracks in Stagnant Borated Water Systems at PWR Plants," Revision 1, issued October 1979 (Ref. 19))

- service water integrity program (NRC Generic Letter 89-13, "Service Water Systems Affecting Safety-Related Equipment," issued July 1989 (Ref. 20))

- flow-accelerated corrosion (FAC) in the balance of plant for both PWRs and BWRs (NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," issued May 1989 (Ref. 21))

Augmented programs have generally been developed to address observed degradation, and the inspections tend to be targeted at locations where the most severe effects are expected. A licensee may incorporate selected augmented programs, or parts of the programs, into its RI-ISI program, provided that it identifies, and the NRC staff approves, the specific programs and program changes.

This guide addresses risk-informed methods to develop, monitor, and update more efficient ISI programs for piping at a nuclear power facility. It does not preclude other approaches for incorporating risk insights into the ISI programs. Licensees may propose other approaches for NRC consideration. The staff intends that the methods presented in this guide be regarded as examples of acceptable practices; licensees should have some flexibility in satisfying the regulations on the basis of their accumulated plant experience and knowledge. This guide addresses risk-informed approaches that are consistent with the basic elements identified in RG 1.174. In addition, it provides guidance on the issues described below for the purposes of RI-ISI.

- estimating the probability of a leak that prevents the system from performing its function (disabling leak), and a rupture for piping segments

- identifying the structural elements for which ISI can be modified (reduced or increased), based on factors such as risk insights, defense in depth, and reduction of unnecessary radiation exposure to personnel

- determining the risk impact of changes to ISI programs

- capturing deterministic considerations in the revised ISI program

- developing an inspection program that monitors the performance of the piping elements for consistency with the conclusions from the risk assessment

This guide only addresses changes to the ISI programs for inspection of piping. To adequately reflect the risk implications of piping failure, both partial- and full-scope RI-ISI programs are acceptable to the NRC staff. A licensee may elect to limit its RI-ISI program to a subset of piping classes, such as ASME Class 1 piping only. Partial-scope applications should include the full population of piping within the selected subset of piping, such as ASME Class or plant systems. A full-scope RI-ISI includes the following:

- all Class 1, 2, and 3 piping[1] within the current ASME BPV Code, Section XI, programs

- all piping whose failure would compromise the following:

  - safety-related structures, systems, or components (SSCs) that are relied upon to remain functional during and following design-basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe-shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in 10 CFR Part 100, "Reactor Site Criteria"

  - nonsafety-related SSCs with the following characteristics:

    - relied upon to mitigate accidents or transients or are used in plant emergency operating procedures, or

    - whose failure could prevent safety-related SSCs from fulfilling their safety-related function, or

    - whose failure could cause a reactor scram or actuation of a safety-related system

For both the partial- and full-scope evaluations, the licensee is to ensure compliance with the acceptance guidelines and key principles of RG 1.174. In the evaluations for developing RI-ISI programs, the locations of concern include all weld and base metal locations at which degradation may occur, although pipe welds are the usual point of interest in the inspection program. Within this guide, references to "welds" are intended in a broad sense to address inspections of critical structural locations in general, including the base metal as well as weld metal. Inspections will often focus on welds because detailed evaluations will often identify welds as the locations most likely to experience degradation. Welds are most likely to have fabrication defects, welds are often at locations of high stress, and certain degradation mechanisms (stress-corrosion cracking) usually occur at welds. Nevertheless, other degradation mechanisms, such as FAC (e.g., erosion/corrosion) and thermal fatigue, occur independent of welds.

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[1] Generally, ASME Code Class 1 includes all reactor pressure boundary components. ASME Code Class 2 generally includes systems or portions of systems important to safety that are designed for postaccident containment and removal of heat and fission products. ASME Code Class 3 generally includes the system components or portions of systems important to safety that are designed to provide cooling water and auxiliary feedwater for the front-line systems.

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 considered IAEA Safety Requirements and Safety Guides[2] pursuant to the Commission's International Policy Statement and Management Directive and Handbook 6.6 (Ref. 22).

The following IAEA Safety Standards Series incorporate similar design and preoperational testing guidelines and are consistent with the basic safety principles considered in developing this Regulatory Guide:

- IAEA Safety Standard SSG-2, "Deterministic Safety Analysis for Nuclear Power Plants," issued 2019 (Ref. 23), provides general guidance on the adequacy of deterministic safety analysis, complemented by probabilistic safety assessments and engineering judgment.

- IAEA Safety Guide SSG-3, "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants—Specific Safety Guide," issued 2010 (Ref. 24), provides good practices on aspects of a risk-informed approach to ISI to help optimize the risk assessment for inservice testing programs.

- IAEA Safety Standard SSR-2/1, "Safety of Nuclear Power Plants: Design," issued 2016 (Ref. 25), provides assurance that defense in depth has been implemented in the design of the plant. The safety analysis for the design of the nuclear power plant should include both deterministic and probabilistic analyses to ensure that all safety requirements are met throughout all stages of the lifetime of the plant.

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[2] Such information related to this guide may be found at WWW.IAEA.Org/ or by writing the International Atomic Energy Agency, P.O. Box 100 Wagramer Strasse 5, A 1400 Vienna, Austria; telephone (+431) 2600-0; fax (+431) 2600-7; or e mail Official.Mail@IAEA.Org. It should be noted that some of the international recommendations do not correspond to the requirements specified in the NRC's regulations and the NRC's requirements take precedence over the international guidance.

C. STAFF REGULATORY GUIDANCE

When a licensee elects to incorporate risk insights into its ISI programs, the staff assumes that the licensee will build upon its existing PRA activities. RG 1.174 describes in detail the five key principles involved in the integrated decisionmaking process. In addition, RG 1.174 describes a four-element process for evaluating proposed risk-informed changes.

The key principles and the section of this guide that addresses each of these principles for RI-ISI programs are as follows.

a. The proposed licensing-basis change meets the current regulations unless it is explicitly related to a requested exemption. (See section C.2.1.1.)

b. The proposed licensing-basis change is consistent with the defense-in-depth philosophy. (See section C.2.1.2.)

c. The proposed licensing-basis change maintains sufficient safety margins. (See section C.2.1.3.)

d. When proposed licensing-basis changes result in an increase in risk, the increases should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants. (See section C.2.2.)

e. The impact of the proposed licensing-basis change should be monitored using performance measurement strategies. (See section C.3.)

Section C of RG 1.174 describes the four-element process for developing risk-informed regulatory changes: (1) define the proposed change, (2) perform an engineering analysis, (3) define the implementation and monitoring program, and (4) submit the proposed change. The order in which the elements are performed may vary, or they may occur in parallel, depending on the particular application and the preference of the program developers. The process is highly iterative. Thus, the final description of the proposed change to the ISI program as defined in Element 1 depends on both the analysis performed in Element 2 and the definition of the implementation of the ISI program performed in Element 3. While ISI is, by its nature, an inspection and monitoring program, it should be noted that the monitoring referred to in Element 3 is associated with making sure that the assumptions made about the impact of the changes to the ISI program are not invalidated. For example, if the inspection intervals are based on an allowable margin to failure, the monitoring is performed to make sure that these margins are not eroded. Element 4 involves preparing the documentation to be submitted to the NRC and to be maintained by the licensee for later reference.

This section provides detailed descriptions of the methods, approaches, or data that the staff considers acceptable for meeting the requirements of the applicable regulations cited in the introduction to this guide.

1. Element 1: Define the Proposed Change

1.1 Description of Proposed Change

The licensee should prepare a full description of the proposed changes to the ISI program. This description should do the following:

a. Identify the elements of the ISI program to be changed.

b. Identify the piping in the plant that is both directly and indirectly involved with the proposed changes. Identify and appropriately address any piping not presently covered in the plant's ISI program but categorized as safety significant (e.g., through an integrated decisionmaking process using PRA insights). In addition, identify the particular systems that are affected by the proposed changes, since this information is an aid in planning the supporting engineering analyses.

c. Identify the information that will be used to support the changes. This could include performance data, deterministic engineering analyses, and PRA information.

d. Include a brief statement describing how the proposed changes meet the intent of the Commission's PRA policy statement.

1.2 Changes to Approved Risk-Informed Inservice Inspection Programs

This section provides guidance on the need for licensees to report program activities and guidance for the formal NRC review of changes made to RI-ISI programs.

The licensee should implement a process for determining when RI-ISI program changes require formal NRC review and approval. It should evaluate the changes made to the NRC-approved RI-ISI program that could affect the process and results reviewed and approved by the NRC staff to ensure that the basis for the staff's approval has not been compromised. The licensee should evaluate all changes using the change mechanisms described in the applicable regulations (e.g., 10 CFR 50.55a, 10 CFR 50.59) to determine whether they require NRC review and approval before implementation. If there is a question on this issue, the licensee should seek NRC review and approval before implementation.

2. Element 2: Perform Engineering Analysis

As part of defining the proposed change to its ISI program, the licensee should conduct an engineering evaluation of the proposed change, using and integrating a combination of deterministic engineering methods and PRA. The major objective of this evaluation is to confirm that the proposed program change will not compromise defense in depth, safety margins, and other key principles described in this guide and in RG 1.174. RG 1.174 provides general guidance for performing this evaluation, which is supplemented by the RI-ISI guidance herein.

The discussions below summarize the regulatory issues and engineering activities that should be considered for RI-ISI programs, divided, for simplicity, into deterministic and PRA analyses. Section C.2.1 addresses the deterministic engineering analysis, Section C.2.2 addresses the PRA-related analysis, and Section C.2.3 describes the integration of the deterministic and PRA analyses. In reality, many facets of the deterministic and PRA analyses are iterative.

The engineering evaluations should do all of the following:

a. Demonstrate that the proposed change is consistent with the defense-in-depth philosophy.

b. Demonstrate that the proposed change maintains sufficient safety margins.

c. Demonstrate that, when proposed changes result in an increase in risk, the increases should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.

d. Support the integrated decisionmaking process.

The scope and quality of the engineering analyses performed to justify the changes proposed to the ISI programs should be appropriate for the nature and scope of the change. The following subsections present the decision criteria associated with each key principle identified above. The licensee can propose equivalent criteria if such criteria can be shown to meet the key principles set forth in Section C of RG 1.174.

2.1 Engineering Analysis

The licensee should consider the appropriateness of qualitative and quantitative analyses, as well as analyses using traditional engineering approaches and those techniques associated with the use of PRA findings. Areas to be evaluated from this viewpoint include meeting the regulations, defense-in-depth attributes, safety margins, assessment of the failure potential of piping segments, and assessment of primary and secondary effects (failures) that result from piping failures.

The engineering analysis for a RI-ISI piping program should achieve all of the following:

a. Assess compliance with applicable regulations.

b. Perform a defense-in-depth evaluation.

c. Perform a safety margin evaluation.

d. Define piping segments.

e. Assess the failure potential for the piping segment.

f. Assess the consequences (both direct and indirect) of piping segment failure.

g. Categorize the piping segments in terms of safety significance.

h. Develop an inspection program.

i. Assess the impact of changing the ISI program on CDF and LERF.

j. Demonstrate conformance with the key principles (e.g., maintaining sufficient safety margins, defense-in-depth consideration, Commission policy statement on safety goals).

2.1.1 Assess Compliance with Applicable Regulations

The engineering evaluation should assess whether the proposed changes to the ISI programs would compromise compliance with the regulations. The evaluation should consider the appropriate requirements in the licensing basis and applicable regulatory guidance. Specifically, the evaluation should consider the following:

a. 10 CFR 50.55a,

b. Appendix A to 10 CFR Part 50,

c. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50,

d. ASME BPV Code, Section XI (10 CFR 50.55a),

e. RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III"

(Ref. 26), and

f. RG 1.147.

In addition, the evaluation should consider whether the proposed changes have affected license conditions. A broad review of the licensing requirements and commitments may be necessary because proposed ISI program changes could affect issues not explicitly stated in the licensee's final safety analysis report or ISI program documentation.

The regulation in 10 CFR 50.55a(z)(1) allows the Director of the Office of Nuclear Regulation to authorize alternatives to its specific requirements, provided the proposed alternative will ensure an acceptable level of quality and safety. Thus, licensees may propose alternatives to the acceptable RI-ISI approaches presented in this guide, so long as they provide supporting information that demonstrates that they maintain the key principles discussed in this guide.

The licensee should include in its RI-ISI program submittal the exemption requests, alternative request, technical specification amendment requests (if applicable), and relief requests necessary to implement its RI-ISI program.

NRC-endorsed ASME Code Cases that apply risk-informed ISI programs are consistent with this RG in that they encourage the use of risk insights in the selection of inspection locations and the use of appropriate and possibly enhanced inspection techniques that are appropriate to the failure mechanisms that contribute most to risk.

2.1.2 Defense-in-Depth Evaluation

Defense in depth is an element of the NRC's safety philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The defense-in-depth philosophy has been applied in plant design and operation to provide multiple means to accomplish safety functions and prevent the release of radioactive material. It has been and continues to be an effective way to account for uncertainties in equipment and human performance and, in particular, to account for the potential for unknown and unforeseen failure mechanisms or phenomena that, because they are unknown or unforeseen, are not reflected in either the PRA or deterministic engineering analyses. The staff requirements memorandum on SECY-98-144, "Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulation," dated March 1, 1999 (Ref. 27), provides additional information on defense in depth as an element of the NRC's safety philosophy.

The engineering evaluation should demonstrate whether the implementation of the proposed ISI program change is consistent with the defense-in-depth philosophy. The intent of this key principle of risk-informed decisionmaking is to ensure that any impact of the proposed licensing-basis change on defense in depth is fully understood and addressed and that consistency with the defense-in-depth philosophy is maintained. The intent is not to prevent changes in the way defense in depth is achieved. The licensees should fully understand how the proposed licensing-basis change impacts plant design and operation from both risk and deterministic engineering perspectives.

RG 1.174 should be used to evaluate the impact of a proposed licensing-basis change on defense in depth to determine whether that consistency is achieved.

An important element of defense in depth for RI-ISI is maintaining the reliability of independent barriers to fission product release. Class 1 piping (primary coolant system) is the second boundary between the radioactive fuel and the general public. Therefore, even if the RI-ISI program categorized all segments in the hot and cold legs of the primary system piping as having low safety significance and calculated that, with no inspections, the frequency of leaks would not increase beyond existing performance history of the ASME BPV Code, the staff would continue to require some level of nondestructive examination inspection.

2.1.3 Safety Margins

In engineering programs that affect public health and safety, safety margins are applied to the design and operation of a system. These safety margins and accompanying engineering assumptions are intended to account for uncertainties, but in some cases, they can lead to operational and design constraints that are excessive and costly, or that could detract from safety (e.g., result in unnecessary radiation exposure to plant personnel). Insufficient safety margins may require additional attention. Before requesting relaxation of the existing requirements, the licensee must ensure that the uncertainties are adequately addressed. The quantification of uncertainties would likely require supporting sensitivity analyses.

The engineering analyses should address whether the impacts of the changes proposed to the ISI program are consistent with the key principle that adequate safety margins are maintained. The licensee is expected to select the method of engineering analysis appropriate for evaluating whether sufficient safety margins would be maintained if the proposed change were implemented. An acceptable set of guidelines for making that assessment is summarized below. Other equivalent decision criteria could also be found acceptable.

Sufficient safety margins are maintained when both of the following occurs:

a. Codes and standards or alternatives approved for use by the NRC are met. (See section C.2.1.1.)

b. Safety analysis acceptance criteria in the licensing basis (e.g., updated final safety analysis report, supporting analyses) are met, or proposed revisions provide sufficient margins to account for analysis and data uncertainty.

2.1.4 Piping Segments

A systematic approach should be applied when analyzing piping systems. One acceptable approach is to divide or separate a piping system into segments; different criteria or definitions can be applied to each piping segment. Another acceptable method is to identify segments of piping within the piping systems that have the same consequences of failure. Other methods could subdivide a segment that exhibits a given consequence into segments with similar degradation mechanisms or similar failure potential. The definition of a segment could encompass multiple criteria, as long as a sound engineering and accounting record is maintained and can be applied to an engineering analysis in a consistent and sound process. Consequences of failure may be defined in terms of an initiating event, loss of a particular train, loss of a system, or combinations thereof. The location of the piping in the plant, and whether inside or outside the containment or compartment, should be considered when defining piping segments.

The definition of a piping segment can vary with the methodology. Defining piping segments can be an iterative process. In general, an analyst may need to modify the description of the piping segments before they are finalized. This guide does not impose any specific definition of a piping segment, but the analysis and the definition of a segment must be consistent and technically sound.

2.1.5 Assess Piping Failure Potential

The engineering analysis includes evaluating the failure potential of a piping segment. Determining the failure potential of piping segments, either with a quantitative estimate or by categorization into groups, should be based on an understanding of such parameters as degradation mechanisms, operational characteristics, potential dynamic loads, flaw size, flaw distribution, inspection parameters, and experience data bases. The evaluation should state the appropriate definition of the failure potential (e.g., failure on demand or operating failures associated with the piping, with the basis for the definition) that will be needed to support the PRA or risk assessment. The failure potential used in or in support of the analysis should be appropriate for the specific environmental conditions, degradation mechanisms, and failure modes for each piping location. When data are analyzed to develop a categorization process relating degradation mechanisms to failure potential, the data should be appropriate and publicly available. When an elicitation of expert opinion is used in conjunction with, or in lieu of, probabilistic fracture mechanics analysis or operating data, a systematic process should be developed for conducting such an elicitation. In such cases, a suitable team of experts should be selected and trained (see NUREG/CR-5424, "Eliciting and Analyzing Expert Judgment—A Practical Guide," issued January 1990 (Ref. 28), and NUREG-1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program," issued November 1996 (Ref. 29)).

When implementing probabilistic fracture mechanics computer programs that estimate structural reliability and are used in risk assessment of piping, or other analytic methods for estimating the failure potential of a piping segment, some of the important parameters that the analysis needs to assess include the identification of structural mechanics parameters, degradation mechanisms, design limit considerations, operating practices and environment, and the development of a data base or analytic methods for predicting the reliability of piping systems. Design and operational stress or strain limits are assessed. This information is available to the licensee in the design information for the plant. The loading and resulting stresses or strains on the piping are needed as input to the calculations that predict the failure probability of a piping segment. The use of validated computer programs, with appropriate input, is strongly recommended in a quantitative RI-ISI program because it may facilitate the regulatory evaluation of a submittal. The analytic method should be validated with applicable plant and industry piping performance data.

To understand the impact of specific assumptions or models used to characterize the potential for piping failure, appropriate sensitivity or uncertainty studies should be performed. These uncertainties include, but are not limited to, design versus fabrication differences, variations in material properties and strengths, effects of various degradation and aging mechanisms, variation in steady-state and transient loads, availability and accuracy of plant operating history, availability of inspection and maintenance program data, applicability and size of the data base to the specific degradation and piping, and the capabilities of analytic methods and models to predict realistic results. Evaluation of these uncertainties provides insights to the input parameters that affect the failure potential and, therefore, require careful consideration in the analysis.

The methodology, process, and rationale used to determine the likelihood of failure of piping segments should be independently reviewed during the final classification of the risk significance of each segment. Referencing applicable generic topical reports approved by the NRC is one acceptable means to standardize the process. When new computer codes are used to develop quantitative estimates, the techniques should be verified and validated against established industry codes and available data. When data are used to evaluate the likelihood of piping failures, the data should be submitted to the NRC or referenced by an NRC-approved topical report. As stated in RG 1.174, "data, methods, and assessment criteria used to support regulatory decisionmaking should be well documented and available for public review." It is the responsibility of the licensee to provide the data, methods, and justification to support its estimation of the failure potential of piping segments.

2.1.6 Assess Consequences of Piping Segment Failures

When evaluating the risk from piping failures, the analyst needs to evaluate the potential consequences, or failures, that a piping failure can initiate. This can be accomplished by performing a detailed walkdown of a nuclear power facility's piping network. The consequences of the postulated pipe segment failure include direct and indirect effects of the failure. Direct effects include the loss of a train or system and associated possible diversion of flow or an initiating event such as a loss of coolant accident. Indirect effects include the spatial effects of flood, spray, pipe whip, or jet impingement that may affect adjacent SSCs or depletion of water sources and loss of associated systems.

2.2 Evaluation of Risk Impact

In accordance with the Commission's policy statement on PRA, the risk-informed application process is intended not only to support a reduction in the number of inspections but also to identify areas where increased resources should be allocated to enhance safety. Therefore, an acceptable RI-ISI process should not focus exclusively on areas in which reduced inspection could be justified. This section addresses ISI-specific considerations in the PRA to support relaxation of inspections, enhancement of inspections, and validation of component operability.

The PRA can be used to categorize the piping segments into safety-significant and low-safety-significant classifications (or more classifications, if a finer graded approach is desired) and to confirm that the change in risk caused by the change in the ISI program is in accordance with the guidance of RG 1.174.

RG 1.174 contains much of the general guidance that is applicable to this topic. A PRA used in a risk-informed regulation should be performed correctly, in a manner consistent with accepted practices. RG 1.200 describes one acceptable approach for determining whether the acceptability of the PRA, in total or the parts that support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for LWRs.

Section C.2.3.1 in RG 1.174 discusses, in general terms, issues related to the scope, level of detail, conformance with the technical elements needed in a PRA as described in RG 1.200, and plant representation of a PRA that is used for risk-informed applications. These aspects of the PRA should be commensurate with its intended use and the role the PRA results play in the integrated decision process. Sections C.2.3.2, C.2.3.3, and C.2.3.4 in RG 1.174 give more specific considerations and address the use of PRA in categorizing and assessing the impact on risk metrics, respectively. NRC safety evaluation dated January 18, 2012, "Final Safety Evaluation of Electric Power Research Institute Topical Report, 1021467, Revision 0, Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs" (Ref. 30), endorses EPRI TR-1021467-A, "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed Inservice Inspection Programs" (Ref. 31). This topical report is based on RG 1.200 and provides guidance on determining the technical acceptability of PRAs used to develop a RI-ISI program and identifies the specific supporting requirements of the PRA standard that are applicable to RI-ISI programs.

While a full-scope PRA that covers all modes of operation and initiating events is preferred, a lesser scope PRA can be used to provide useful risk information. However, it should then be supplemented by additional considerations, as discussed below.

2.2.1 Modeling Piping Failures in a Probabilistic Risk Assessment

Input from the deterministic engineering analysis addressed in Section C.2.1 includes the identification of piping segments from the point of view of the failure potential (degradation mechanisms) and consequences (resulting failure modes and consequential primary and secondary effects). The deterministic analysis identifies both the primary and secondary effects that can result from a piping failure. The assessment of the primary and secondary failures identifies the portions of the PRA that are affected by the piping failure.

Each piping segment failure may have one of three types of impacts on the plant:

a. Initiating event failures occur when the failure directly causes a transient and may or may not also fail one or more plant trains or systems.

b. Standby failures are those failures that cause the loss of a train or system but that do not directly cause a transient. Standby failures are characterized by train or system unavailability that may require shutdown because of the technical specifications or limiting conditions for operation.

c. Demand failures are failures to meet a demand for a train or system and are usually accompanied by the transient-induced loads on the segment during system startup.

The impact of the piping segment failure on risk should be evaluated with the PRA. Evaluation may involve a quantitative estimate derived from the PRA, a systematic technique to categorize the consequence of the pipe failure on risk, or some combination of quantification and categorization. If a segment failure were to lead to plant transients and equipment failures that are not at all represented in the PRA (a new and specific initiating event, for example), the evaluation process should be expanded to assess these events.

PRAs normally do not include events that represent failure of individual piping segments nor the structural elements within the segments. A quantitative estimate of the impact of segment failures can be made by modifying the PRA logic to systematically and explicitly include the impact of the individual piping segment failures. The impact of each segment's failure on risk can also be estimated without modifying the PRA's logic by identifying an initiating event, basic event, or group of events, already modeled in the PRA, whose failures capture the effects of the piping segment's failure (referred to as the surrogate approach). In either case, to assess the impact of a particular segment failure, the analyst sets the appropriate events to a failed state in the PRA and requantifies the PRA or the appropriate parts of the PRA as needed. The analysis should appropriately incorporate segment failures that only cause an initiating event, that only degrade or fail a mitigating system required to respond to an independent initiating event, and that simultaneously cause an initiating event and degrade or fail a mitigating system responding to the initiating event. The requantification should explicitly address truncation errors, since cut set or truncated sequences may not fully capture the impact of multiple failure events.

If a systematic technique is used to categorize the consequence of pipe failures, it should also be based on PRA results. In this case, however, the categories may be represented by ranges of conditional results, and instead of quantifying the impact of each segment failure, the process should provide for determining the range within which each segment's failure would lie. In general, the consequences would range from high, for those segments whose failure would have a high likelihood of leading to core damage or large early release, to low for those segments whose failure would likely not lead to core damage or large early release. The licensee should discuss and justify the ranges selected. The use of ranges instead of individual results estimates may require fewer calculations, but the categorization process and decision criteria should be justified, well defined, and repeatable.

2.2.1.1 Dependencies and Common-Cause Failures

The effects of dependencies and common-cause failures for ISI components need to be considered carefully because of the significance they can have on risk. Generally, the data is insufficient to produce plant-specific estimates based solely on plant-specific data. For common-cause failures, data from generic sources may be required.

2.2.1.2 Human Reliability Analyses to Isolate Piping Breaks

For ISI-specific analyses, the human reliability analysis methodology used in the PRA should account for the impact that the piping segment break would have on the operator's ability to respond to the event. In addition, the reliability of the inspection program (including both operator and equipment qualification), which factors into the probability of detection, should also be addressed.

2.2.2 Use of Probabilistic Risk Assessment for Categorizing Piping Segments

When the impact of each segment's failure on plant risk metrics has been determined, the safety significance of the segments is developed. The method of categorizing a piping segment can vary. For example, if the pipe failure event frequency or probability is estimated and the events are incorporated into the PRA logic model, importance measure calculations and the determination of safety significance, as discussed in RG 1.174 and SRP Chapter 19, "Severe Accidents," may be performed. Alternatively, if a conditional CDF, conditional LERF, conditional core damage probability, or conditional large early release probability (depending on the impact the segment failure has on the plant) is estimated for each segment from the PRA, a CDF or LERF (or both) caused only by pipe failures may be developed by combining the conditional consequences and segment failure probabilities or frequencies external to the PRA logic model. Importance measures can also be developed using these results and these measures compared to appropriate threshold criteria to support the determination of the safety significance of each segment. The calculations used in such a process should yield well-defined estimates of CDF, LERF, and importance measures. The licensee should discuss and justify the threshold criteria used.

As discussed in Section C.2.2.1, the consequence of segment failures may be represented by categories of consequences instead of quantitative estimates for each segment. In this case, the potential for pipe failure, as discussed in Section C.2.1.5, would also be developed as categories ranging from high to low, depending on the degradation mechanisms present and the corresponding likelihood that the segment will fail. These consequence and failure likelihood categories should be systematically combined to develop categories of safety significance. The licensee should discuss and justify how it relates the consequence and failure likelihood categories to the safety-significant category assigned to each combination.

The safety-significance category of the piping segment will help determine the level of inspection effort devoted to the segment. In general, safety-significant segments will receive more inspections and more demanding inspections than low-safety-significant segments. In any integrated categorization process, the licensee needs to address the principles in RG 1.174. Irrespective of the method used in the analysis, the licensee should justify the final categorization process as being robust and reasonable with respect to the analysis uncertainties.

2.2.3 Demonstrate Change in Risk Resulting from Change in Inservice Inspection Program

Any change in the ISI program has an associated risk impact. Evaluation of the change in risk may be a detailed calculation or it may be a bounding estimate supported by sensitivity studies, as appropriate. The change may be a risk increase, a risk decrease, or risk neutral. The change is evaluated and compared with the guidelines presented in RG 1.174. The staff expects that the RI-ISI program would lead to both risk reduction and reduction in radiation exposure to plant personnel.

The change in risk estimate should appropriately account for the change in the number of elements inspected and the effects of enhanced inspection. The methods used to determine the piping failure potential, the piping failure consequence, and the impact of the change in the number of inspections should together provide confidence that any increase in risk is small and acceptable in accordance with RG 1.174 guidelines and consistent with the intent of the Commission's safety goal policy statement.

2.3 Integrated Decisionmaking

Sections C.2.1 and C.2.2 address the elements of engineering analysis and PRA analysis of an RI-ISI program. These elements are part of an integrated decisionmaking process that assesses the acceptability of the program. The key principles of RG 1.174 are systematically addressed. Technical and operations personnel at the plant review the information and render a finding of the safety-significance category for each piping segment under review. Detailed guidelines for the categorization of piping segments should be developed and discussed with the group responsible for the determination (typically performed by the plant's expert panel).

The method for selecting the number of piping elements to be inspected should be justified.

3. Element 3: Implementation, Performance Monitoring, and Corrective Action Strategies

Integrating the information obtained from Elements 1 and 2 of the RI-ISI process (as described in Sections C.1 and C.2 of this guide), the licensee develops proposed RI-ISI implementation, performance monitoring, and corrective action strategies. The RI-ISI program should identify piping segments whose inspection strategy (i.e., frequency, number of inspections, methods, or all three) should be increased, as well as piping segments whose inspection strategies might be relaxed. The number of required inspections should be a product of the systematic application of the risk-informed process. The program should be self-correcting as experience dictates. The program should contain performance measures used to confirm the safety insights gained from the risk analyses.

3.1 Program Implementation

A licensee should have in place a schedule for inspecting all segments categorized as safety significant in its RI-ISI program. This schedule should include inspection strategies and inspection frequencies, inspection methods, and the sampling program (e.g., the number of elements or areas to be inspected, the acceptance criteria) for the safety-significant piping that is within the scope of the ISI program, including piping segments identified as safety significant that are not currently in the ISI program.

The analysis for the RI-ISI program will, in most cases, confirm the appropriateness of the inspection interval and scope requirements of the edition and addenda of ASME BPV Code, Section XI, committed to by a licensee in accordance with 10 CFR 50.55a. ASME BPV Code, Section XI, contains the requirements for these intervals. However, should active degradation mechanisms surface, the inspection interval would be modified as appropriate. Updates to the RI-ISI program should be performed at least periodically to coincide with the inspection program requirements contained in Section XI under Inspection Program B. The RI-ISI program should be evaluated periodically as new information becomes available that could impact the ISI program.

For example, if changes to the PRA impact the decisions made for the RI-ISI program, if plant design and operations change such that they impact the RI-ISI program, if inspection results identify unexpected flaws, or if replacement activities impact the failure potential of piping, the effects of the new information should be assessed. The periodic evaluation may result in updates to the RI-ISI program that are more restrictive than required by Section XI. As plant design feature changes are implemented, changes to the input associated with the RI-ISI program segment definition and element selections should be reviewed and modified as needed. Changes to piping performance or the plant procedures that can affect system operating parameters, piping inspections, component and valve lineups, equipment operating modes, or the ability of the plant personnel to perform actions associated with accident mitigation should be reviewed in any RI-ISI program update. Leakage and flaws identified during scheduled inspections should be evaluated as part of the RI-ISI update.

Piping segments categorized as safety significant that are not in the licensee's current ISI program should (wherever appropriate and practical) be inspected in accordance with applicable ASME Code Cases (or revised ASME BPV Code), including compliance with all administrative requirements. Where ASME BPV Code, Section XI, inspection is not practical or appropriate, or it does not conform to the key principles identified in this document, the licensee should develop alternative inspection intervals, scope, and methods to ensure piping integrity and to detect piping degradation. The licensee should provide a summary of the piping segments and their proposed inspection intervals and scope to the NRC before implementation of the RI-ISI program at the plant.

For piping segments categorized as safety significant that were the subject of a previous NRC-approved relief request or were exempt under existing ASME BPV Code, Section XI, criteria, the licensee should assess the appropriateness of the relief or exemption in light of the risk significance of the piping segment.

3.2 Performance Monitoring

3.2.1 Periodic Updates

The RI-ISI program should be updated at least on the basis of periods that coincide with the inspection program requirements contained in ASME BPV Code, Section XI, under Inspection Program B. These updates should be performed more frequently if dictated by any plant procedures to update the PRA (which may be more restrictive than a Section XI period-type update) or as new degradation mechanisms are identified.

3.2.2 Changes to Plant Design Features

As changes to the plant design are implemented, changes to the inputs associated with RI-ISI program segment definition and element selections may occur. It is important to address these changes to the inputs included in any assessment that may affect resultant pipe failure potentials used to support the RI-ISI segment definition and element selection. Some examples of these inputs include the following:

a. operating characteristics (e.g., changes in water chemistry control)

b. construction and preservice examination results

c. welding techniques and procedures

d. stress data (operation modes, pressure, and temperature changes)

In addition, plant design changes could result in significant changes to a plant's CDF and LERF, which in turn could result in a change in consequence of failure for system piping segments.

3.2.3 Changes to Plant Procedures

Licensees should include changes to plant procedures that affect ISI, such as system operating parameters, test intervals, or the ability of plant operations personnel to perform actions associated with accident mitigation, for review in any RI-ISI program update. Additionally, changes in those procedures that affect component inspection intervals, valve lineups, or operational modes of equipment should also be assessed for their impact on changes in postulated failure mechanism initiation or CDF/LERF contribution.

3.2.4 Equipment Performance Changes

Equipment performance changes should be reviewed with system engineers and maintenance personnel to ensure that the periodic evaluation of the RI-ISI program update includes changes in performance parameters such as valve leakage, increased pump testing, or identification of vibration problems. Specific attention should be paid to these conditions if they were not previously assessed in the qualitative inputs to the element selections of the RI-ISI program.

3.2.5 Examination Results

When scheduled RI-ISI program nondestructive examinations, pressure tests, and corresponding visual testing level 2 (VT-2) examinations for leakage have been completed, and if unacceptable flaws, evidence of service-related degradation, or indications of leakage have been identified, the existence of these conditions should be evaluated. This update of the RI-ISI program should follow the applicable elements of Appendix B to 10 CFR Part 50 to determine the adequacy of the scope of the inspection program.

3.2.6 Information on Individual Plant and Industry Failures

The review of individual plant maintenance activities associated with repairs or replacements, including identified flaw evaluations, is an important part of any periodic update, regardless of whether the activity is the result of the RI-ISI program examination. Evaluating this information as it relates to a licensee's plant provides failure information and trending information that may have a profound effect on the element locations currently being examined under the RI-ISI program. Industry failure data are just as important to the overall program as the owner's information. During the periodic update, industry data bases (including available international data bases) should be reviewed for applicability to the owner's plant.

3.3 Corrective Action Programs

Each licensee of a nuclear power plant is responsible for having a corrective action program, consistent with RG 1.174. Measures are to be established to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures must ensure that the cause of the condition is determined and corrective action is taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action are to be documented and reported to appropriate levels of management.

For ASME BPV Code piping categorized as safety significant, this corrective action program should be consistent with applicable provisions in ASME BPV Code, Section XI. For non-Code and Code-exempt piping categorized as safety significant, the licensee should also use appropriate Section XI provisions or should submit an alternative program based on the risk significance of the piping.

3.4 Acceptance Guidelines

The following acceptance guidelines are for the implementation, monitoring, and corrective action programs for the accepted RI-ISI program plan:

a. The evaluation of the implementation program will be based on the attributes presented in Sections C.3.1 through C.3.3 of this guide.

b. The corrective action program should provide reasonable assurance that a nonconforming component will be brought back into conformance in a timely fashion. The corrective actions required in ASME BPV Code, Section XI, should continue to be followed.

c. Evaluations within the corrective action program may also include the following:

   (1) Ensure that the root cause of the condition is determined and that corrective actions are taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action are to be documented and reported to appropriate levels of management.

   (2) Determine the impact of the failure or nonconformance on system or train operability since the previous inspection.

   (3) Assess the applicability of the failure or nonconforming condition to other components in the RI-ISI program.

   (4) Correct other susceptible RI-ISI components as necessary.

   (5) Incorporate the lessons in the plant data base and computer models, if appropriate.

   (6) Assess the validity of the failure rate and unavailability assumptions that can result from piping failures used in the PRA or in support of the PRA.

(7) Consider the effectiveness of the component's inspection strategy in detecting the failure or nonconforming condition. The inspection interval would be reduced, or the inspection methods adjusted, as appropriate, when the component (or group of components) experiences repeated failures or nonconforming conditions.

d. The corrective action evaluation should be provided to the licensee's PRA and RI-ISI groups so that any necessary model changes and regrouping are done, as appropriate.

e. The RI-ISI program documents should be revised to document any RI-ISI program changes resulting from the corrective actions taken.

f. A program should be in place that monitors industry findings.

g. Piping is subject to examination. The examination requirements include all piping evaluated by the risk-informed process and categorized as safety significant.

h. The inspection program is to be completed during each 10-year inspection interval with the following exceptions.

(1) If, during the interval, a reevaluation using the RI-ISI process is conducted and scheduled items are no longer required to be examined, these items may be eliminated.

(2) If, during the interval, a reevaluation using the RI-ISI process is conducted and items must be added to the examination program, those items should be added.

i. If additional examinations are needed following the identification of unacceptable flaws, additional examinations should be performed on the elements with the same root cause or degradation mechanisms as the identified flaw or relevant condition. The number of additional examinations should be equivalent to the number of elements required to be inspected during the current outage. If unacceptable flaws or relevant conditions are again found similar to the initial problem, the remaining elements identified as susceptible should be examined. All additional examinations should be performed during the current outage.

j. Pressure testing and VT-2 visual examinations are to be performed on Class 1, 2, and 3 piping systems in accordance with ASME BPV Code, Section XI, as specified in the licensee's ISI program. The pressure testing and VT-2 examinations are also to be performed on non-Code safety-significant piping. The non-Code safety-significant piping will be treated as ASME Code Class piping for the purposes of examination and pressure testing.

k. Examination methods, equipment qualification, personnel qualification, and procedure qualification are to be in accordance with the edition and addenda of the ASME BPV Code endorsed by the NRC in 10 CFR 50.55a.

l. Acceptance standards for identified flaws and repair or replacement activities are to be performed in accordance with the ASME BPV Code, Section XI requirements.

m. Records and reports should be prepared and maintained in accordance with the ASME BPV Code, Section XI, edition and addenda as specified in the licensee's ISI program.

4. Element 4: Documentation

The sections below present the recommended contents for a plant-specific RI-ISI submittal. This guidance will help ensure the completeness of the information provided and aid in minimizing the time needed for the review process.

4.1 Licensee's Risk-Informed Inservice Inspection Submittal Documentation

References to NRC-approved topical reports that address the methodology and issues requested in a submittal are acceptable. Documentation guidelines specified in approved topical reports may be considered instead of the following guidelines when the methodology from an approved topical report is used. Since topical reports could cover more issues than applied by a licensee, or the licensee may elect to deviate from the full body of issues addressed in the topical report, such distinctions should be clearly stated.

The application to implement an RI-ISI program should include the following items:

a. Make a request to implement an RI-ISI program as an authorized alternative to the current NRC-endorsed ASME BPV Code under 10 CFR 50.55a(z)(1). The licensee should also describe how the proposed change impacts any commitments made to the NRC.

b. Discuss each of the following five key principles of risk-informed regulations (see Section C.2 of RG 1.174 for more details):

   (1) The proposed change meets the current regulations unless it is explicitly related to an alternative requested under 10 CFR 50.55a(Z)(1), a requested exemption, or a rule change.

   (2) The proposed change is consistent with the defense-in-depth philosophy.

   (3) The proposed change maintains sufficient safety margins.

   (4) When proposed changes result in an increase in risk, the increases should be small and consistent with the guidance in RG 1.174.

   (5) The impact of the proposed change should be monitored using performance measurement strategies.

c. Identify the aspects of the plant's current requirements that would be affected by the proposed RI-ISI program. This identification should include all commitments and augmented programs (for example, the intergranular stress-corrosion cracking inspections and other commitments arising from generic letters affecting piping integrity) that the licensee intends to change or terminate as part of the RI-ISI program. The application of the RI-ISI methodology to incorporate and change the augmented program should be justified.

d. Identify the specific revisions to existing inspection schedules, locations, and methods that would result from implementation of the proposed program.

e. Submit plant procedures or documentation containing the guidelines for all phases of evaluating and implementing a change in the ISI program based on probabilistic and deterministic insights. These should include a description of the integrated decisionmaking process and criteria used for categorizing the safety significance of piping segments, a description of how the integrated decisionmaking was performed, a description and justification of the number of elements to be inspected in a piping segment, the qualifications of the individuals making the decisions, and the guidelines for making those decisions.

f. Present the results of the licensee's ISI-specific analyses used to support the program change with enough detail to be clearly understandable to the reviewers of the program. These results should include the following information:

(1) List the piping systems reviewed.

(2) List each segment, including the number of welds, weld type, and properties of the welding material and base metal, the failure potential, CDF, conditional CDF, LERF, conditional LERF, conditional core damage probability, conditional large early release probability, importance measure results (e.g., risk achievement worth, Fussel-Vesely) and justification of the associated threshold values, degradation mechanism, and test and inspection intervals used in or in support of the PRA. Results from other methods used to develop the consequences and categorization of each segment (or weld) should be documented with a similar level of detail.

(3) Describe the degradation mechanisms for each segment (if segments contain welds exposed to different degradation mechanism, for each weld) used to develop the failure potential of each segment. For the selected limiting locations, provide examples of the failure mode, failure potential, failure mechanism, weld type, weld location, and properties of the welding material and base metal.

(4) Provide a detailed description and justification for the number of elements to be inspected.

(5) List the equipment assumed to fail as a direct or indirect consequence of each segment's failure (if segments contain welds with different failure consequences, for each weld).

(6) Describe how the impact of the change between the current ASME BPV Code, Section XI, and the proposed RI-ISI programs is evaluated or bounded and how this impact compares with the risk guidelines in Section C.2.4 of RG 1.174.

g. Describe the means by which failure probabilities, frequencies, or potential were determined.

h. Describe the PRA used for the categorization process and for the determination of risk impact, in terms of the process to ensure acceptability of the scope, level of detail, conformance with the technical elements of the PRA standard, and plant representation of the PRA, and how limitations in these qualities are compensated for in the integrated decisionmaking process. Section C.6.3 of RG 1.174 provides detailed discussions of this.

i. If the submittal includes modified inspection intervals, submit the methodology and results of the analysis.

j. Describe the implementation, performance monitoring, and corrective action strategies and programs in sufficient detail for the staff to understand the new ISI program and its implications.

k. Provide the applicable documentation discussed under the cumulative risk documentation for submittal in Section C.6.3.2 of RG 1.174.

l. Refer to NRC-approved topical reports on implementing the RI-ISI program and supporting documents. Clearly identify any variations from the topical reports and supporting documents.

4.2 Archival Documentation

The licensee should maintain at its facility the technical and administrative records used in support of its submittal or should be able to generate the information on request. This information should be available for NRC review and audit. If changes are planned to the ISI program based on internal procedures and without prior NRC approval, the licensee should also place the following information in the plant's document control system so that the analyses for any given change can be identified and reviewed. The record should include, but not be limited to, the following information:

a. all the documentation discussed in Section C.4.1; although the documentation requirements in a submittal may be reduced when referring to NRC-approved topical reports, all the documentation included under Section C.4.1 should be available for onsite inspection

b. plant and applicable industry data used in support of the RI-ISI program, including all analyses and assumptions used in support of the RI-ISI program and communications with outside organizations supporting the RI-ISI program (e.g., use of peer and independent reviews, use of expert contractors)

c. detailed procedures and analyses performed by an expert panel, or other technical groups, if relied upon for the RI-ISI program, including a record of deliberations, recommendations, and findings

d. documentation of the plant's baseline PRA used to support the ISI submittal that is of sufficient detail to allow an independent reviewer to ascertain whether the PRA reflects the current plant configuration and operational practices commensurate with the role the PRA results play in the integrated decisionmaking process and, in addition to documentation on the PRA itself, analyses performed in support of the ISI submittal in a manner consistent with the baseline documentation, including such analyses as the following:

    (1) the process used to identify initiating events developed in support of the RI-ISI submittal and the results from the process

    (2) any event trees and fault trees developed during preparation of the RI-ISI submittal

    (3) documentation of the methods and techniques used to identify and quantify the impact of pipe failures using the PRA, or in support of the PRA, if different from those used during the development of the baseline PRA

    (4) the techniques used to identify and quantify human actions

    (5) the data used in any uncertainty calculations or sensitivity calculations, consistent with the guidance provided in RG 1.174.

    (6) the way the segment categorization accounted for uncertainty, as well as the sensitivity studies performed to ensure the robustness of the categorization

e. detailed results of the inspection program corresponding to the ISI inspection records described in the implementation, performance monitoring, and corrective action program accompanying the RI-ISI submittal

f. for each piping segment, information on weld type, weld location, and properties of welding material and base metal

g. for each piping segment, information on the process and assumptions used to develop the failure mode and failure potential (frequency/probability), in addition to the identification of the failure mechanism

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, "Backfitting," and as described in NRC Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests" (Ref. 32), nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52. The 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 RG 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 guidance in Management Directive 8.4.

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

3. *U.S. Code of Federal Regulations*, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter I, Title 10, "Energy."

4. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," New York, NY.[4]

5. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Washington, DC.

6. NRC, RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Washington, DC.

7. NRC, RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Washington, DC.

8. ASME BPV Code, Code Case N-716-1, "Alternative Classification and Examination Requirements, Section XI, Division 1," American Society of Mechanical Engineers, New York, NY, January 27, 2013.

9. NRC, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities: Final Policy Statement," *Federal Register*, Vol. 60, p. 42622, Washington, DC, August 16, 1995.

10. NRC, SECY-95-280, "Framework for Applying Probabilistic Risk Analysis in Reactor Regulation," Washington, DC, November 27, 1995.

11. NRC, RG 1.175, "Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," Washington, DC.

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[3] 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 online 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.

[4] Copies may be obtained from the American Society of Mechanical Engineers, 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/.

12. NRC, RG 1.177, "Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Washington, DC.

13. Electric Power Research Institute (EPRI) TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Revision B-A, Palo Alto, CA, December 1999.[5]

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15. NRC, Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless-Steel Piping," Washington DC, January 25, 1988.

16. NRC, Bulletin No. 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," Washington, DC, June 22, 1988.

17. NRC, Bulletin No. 88-11, "Pressurizer Surge Line Thermal Stratification," Washington, DC, December 20, 1988.

18. NRC, Information Notice 93-20, "Thermal Fatigue Cracking of Feedwater Piping to Steam Generators," Washington, DC, March 24, 1993.

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20. NRC, Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," Washington, DC, July 18, 1989.

21. NRC, Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," Washington, DC, May 2, 1989.

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24. International Atomic Energy Agency, Safety Guide SSG-3, "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants Specific Safety Guide," Vienna, Austria, 2010.[5]

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5 Copies of Electric Power Research Institute standards and reports may be purchased from EPRI, 3420 Hillview Avenue, Palo Alto, CA 94304; telephone (800) 313-3774; fax (925) 709 1310.

6 Copies of International Atomic Energy Agency (IAEA) documents may be obtained through the IAEA Web site: WWW.IAEA.Org/ or by writing the International Atomic Energy Agency, P.O. Box 100, Wagramer Strasse 5, A-1400 Vienna, Austria.

25. International Atomic Energy Agency Safety Standards SSR-2/1, "Safety of Nuclear Power Plants: Design," Revision 1, Vienna, Austria, 2016.[5]

26. NRC, RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," Washington, DC.

27. NRC, SRM-SECY-98-144, "Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulation," Washington, DC, March 1, 1999.

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29. NUREG-1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program." Washington, DC, November 1996.

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