U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE RG 1.175, Revision 1

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PLANT-SPECIFIC, RISK-INFORMED DECISIONMAKING: INSERVICE TESTING

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 testing (RI-IST) programs. It 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). This guide describes acceptable methods for using information from a probabilistic risk assessment (PRA) with deterministic engineering information in the development of a RI-IST program to be submitted by a nuclear power plant licensee for review and authorization by the NRC.

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).

This RG provides guidance for proposing a risk-informed approach to implement an RI-IST program in accordance with 10 CFR 50.55a(z)(1) in lieu of the IST requirements in the American Society of Mechanical Engineers (ASME) *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) (Ref. 4) as incorporated by reference in 10 CFR 50.55a. If a licensee has been authorized to implement 10 CFR 50.69, the risk-informed approach described in this RG is not applicable because the IST requirements in 10 CFR 50.55a for certain pumps, valves, and snubbers are replaced by the alternative treatment requirements in 10 CFR 50.69.

Applicable Regulations

- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," provides

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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.) ML21140A055. The associated draft guide DG-1286 may be found in ADAMS under Accession No. ML19240B371. The Regulatory Analysis for DG-1286 may be found in ADAMS under Accession No. ML19240B374. No public comments were received on DG-1286; therefore, the staff has not prepared responses to public comments.

regulations for licensing production and utilization facilities.

- o 10 CFR 50.55a, "Codes and Standards," requires, in part, that systems and components must meet the IST requirements of the OM Code, as specified in 10 CFR 50.55a(b) and (f). This RG describes the information to be submitted by a licensee as part of an alternative request in accordance with 10 CFR 50.55a(z), "Alternatives to codes and standards requirements," subparagraph (1), "Acceptable level of quality and safety," to demonstrate that the proposed RI-IST program would provide an acceptable level of quality and safety.

- o 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," which states, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected."

- 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," provides regulations for 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). Note that the SRP Section 3.9.7, "Risk-Informed Inservice Testing," addresses RI-IST, consistent with the guidance in this RG (Ref. 6).

- NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants—Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants—Final Report," (Ref. 7), provides guidance for OM Code inquiries, the inservice examination and testing of snubbers, pump and valve IST, the use of ASME Code Cases, conditions on the use of the OM Code, guidance for OM Code noncompliance, requests for alternatives to the OM Code at operating commercial nuclear power plants, and the development of IST programs for new reactors.

- NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking—Final Report" (Ref. 8), provides guidance on how to treat uncertainties associated with PRA in risk-informed decisionmaking. This guidance is intended to foster an understanding of the uncertainties associated with PRA and their impact on the results of PRA.

- 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. 9), provides an approach for determining whether the base PRA, in total or the parts that are used to support an application, is acceptable for use in regulatory decisionmaking for LWRs. RG 1.200 endorses ASME/American Nuclear Society (ANS) RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic

Risk Assessment for Nuclear Power Plant Applications" (Ref. 10), which addresses PRA for core damage frequency (CDF) and large early release frequency (LERF) for internal and external hazard groups during at-power operations.

Purpose of Regulatory Guides

The NRC staff 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), approval 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 OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0011, 3150-0151), 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 1) updates guidance for consideration of the defense-in-depth philosophy to be consistent with the considerations described in RG 1.174. RG 1.174 was revised in 2018 to expand the guidance on the meaning of, and the process for, assessing defense-in-depth considerations. Specifically, this revision of RG 1.175 references the defense-in-depth guidance in RG 1.174 in several staff regulatory positions.

Additionally, the staff revised this guide to (1) adopt the term “PRA acceptability,” and related phrasing variants, instead of terms such as “PRA quality,” “PRA technical adequacy,” and “technical adequacy” to describe the appropriateness of the PRA used to support risk-informed licensing submittals; (2) update Regulatory Position C.2.2.3, “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 (3) incorporate guidance related to the OM Code for the inservice testing of pumps and valves at commercial nuclear power plants.

Background

Both the NRC and the nuclear industry recognize that PRA has evolved to the point that it can be used increasingly as a tool in regulatory decisionmaking. After the publication of 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. 11), the Commission directed the NRC staff to develop a regulatory framework that incorporated risk insights. The staff articulated that framework in SECY-95-280, “Framework for Applying Probabilistic Risk Analysis in Reactor Regulation,” November 27, 1995 (Ref. 12). This RG, which addresses IST of pumps and valves, and its companion regulatory documents implement, in part, the Commission’s policy statement and the staff’s framework by allowing risk insights to be incorporated into the operation of nuclear power plants. Licensees may submit requests to use RI-IST programs as alternatives to existing IST programs under 10 CFR 50.55a(z)(1).

The Commission’s policy statement on PRA encourages greater use of this analysis technique to improve safety decisionmaking and regulatory efficiency. In response to the policy statement, the staff and industry have used PRA in support of decisions to modify an individual plant’s IST program. NRC staff normally evaluate licensee-initiated IST program changes that are consistent with currently approved staff positions (e.g., RGs, standard review plans, branch technical positions) using deterministic engineering analyses. In such cases, the licensee would not be expected to submit risk information in support of the proposed change. For IST program change requests that go beyond current staff positions, in which a licensee elects to use risk information in support of the proposed IST change, the staff’s evaluation may use deterministic engineering analyses and the risk-informed approach set forth in this RG. The staff will review the information provided by the licensee to determine whether it can approve the application based on both deterministic and risk-informed considerations, as applicable, and will either approve or reject the application based upon the review.

This RG provides application specific details of a method acceptable to the NRC staff for developing RI-IST programs and supplements the information given in RG 1.174. This guide identifies acceptable approaches for utilizing PRA information, along with established deterministic engineering information in the development of RI-IST programs that can lead to more efficient use of plant resources while still maintaining acceptable levels of quality and safety. RG 1.174 provides overall guidance on the technical aspects that are common to developing acceptable risk-informed programs for all applications such as IST (this guide), inservice inspection, and technical specifications. However, licensees may propose other approaches for consideration by the NRC staff.

This revision does not provide guidance on preparing a request to use an RI-IST program for snubbers; however, the NRC staff will consider requests such requests if submitted.

Key Principles of Risk-Informed Integrated Decisionmaking

In risk-informed decisionmaking, risk-informed changes to an IST program are expected to meet a set of key principles as applicable to an alternative request in accordance with 10 CFR 50.55a(z)(1). Some of these principles are written in the terms typically used in deterministic engineering decisions (e.g., defense-in-depth), which are not directly applicable to a 10 CFR 50.55a(z)(1) alternative request. Although the principles are written in these terms, the use of risk analysis is encouraged to help ensure and to show that these principles are met. These principles include the following:

- Principle 1: The proposed licensing-basis change meets the current regulations unless it is explicitly related to a requested exemption (i.e., a specific exemption under 10 CFR 50.12, "Specific Exemptions").

- Principle 2: The proposed licensing-basis change is consistent with the defense-in-depth philosophy.

- Principle 3: The proposed licensing-basis change maintains sufficient safety margins.

- Principle 4: When proposed licensing-basis changes result in an increase in risk, the increase should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.

- Principle 5: The impact of the proposed licensing-basis change should be monitored using performance measurement strategies.

Each of these principles should be considered in the risk-informed, integrated decisionmaking process, as illustrated in Figure 1. RG 1.174 gives additional guidance on the key principles applicable to all risk-informed applications.

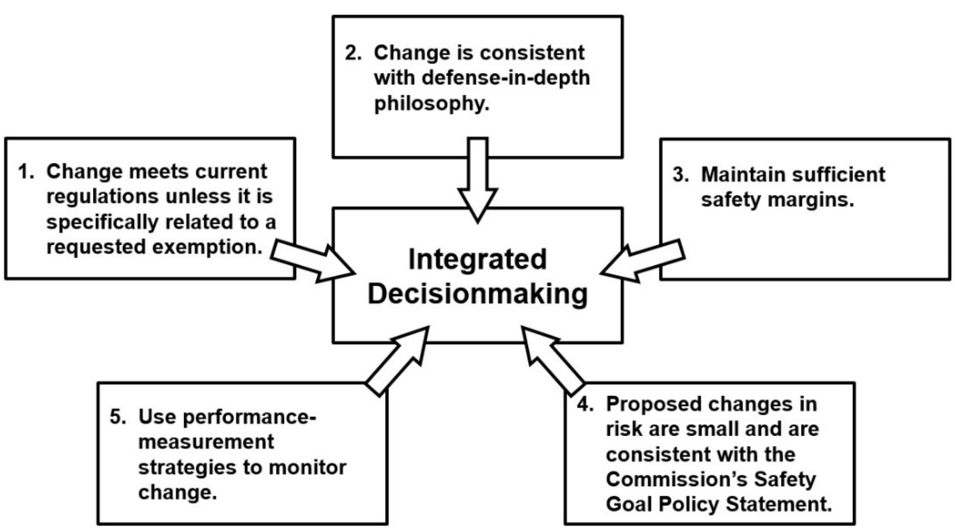

Figure 1. Principles of risk-informed integrated decisionmaking

Four-Element Approach to Risk-Informed Decisionmaking for Inservice Testing

RG 1.174 describes a four-element process for developing risk-informed regulatory changes. This approach, shown in Figure 2, illustrates the relationships among the elements. This approach is iterative rather than sequential. The discussion below provides an overview of this process specifically related to RI-IST programs. The order in which the elements are performed may vary or occur in parallel, depending on the particular application and the preference of the program developers.

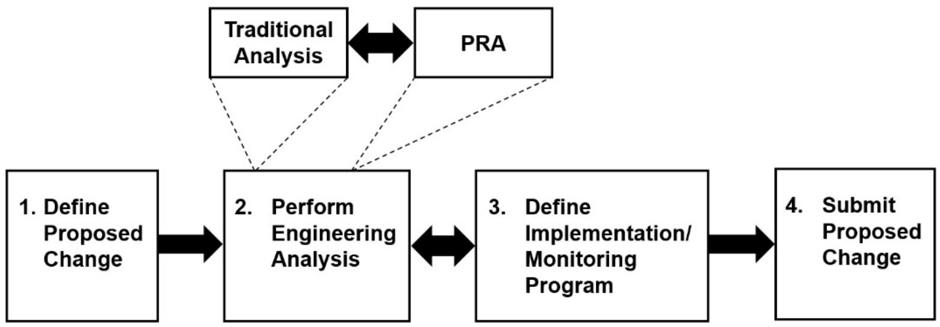

Figure 2. Principal elements of risk-informed, plant-specific decisionmaking

Element 1: Define Proposed Changes to the Inservice Testing Program. The purpose of this element is to identify (1) the particular components that would be affected by the proposed changes in testing practices, including those currently in the IST program and possibly some that are not (if it is determined through new information and insights such as the PRA that these additional components are important in terms of plant risk) and (2) specific revisions to testing schedules and methods for the chosen components. Plant systems and functions that rely on the affected components should be identified. Regulatory Position C.1 gives a more detailed description of Element 1.

Element 2: Perform Engineering Analysis. This element uses both deterministic engineering and PRA methods to help define the scope of the changes to the IST program and to evaluate the impact of the changes on the overall plant risk. Areas that are to be evaluated include the expected effect of the proposed RI-IST program on the design basis and severe accidents, defense-in-depth attributes, and safety margins. This evaluation considers the results of deterministic engineering and PRA methods together in an integrated decision process that will be carried over into the implementation phase (see Element 3). PRA results should be used to provide information for categorizing components into groupings of low safety-significant components (LSSCs) and high safety-significant components (HSSCs). Components in the LSSC group would then be candidates for less rigorous testing when compared with those in the HSSC group. When the revised IST plan has been developed, the plant-specific PRA should be used to evaluate the effect of the planned program changes on the overall plant risk, as measured by CDF and containment LERF.

During the integration of all the available information, many issues will need to be resolved using judgment, often involving a combination of different engineering skills. Industry documents typically refer to this activity as being performed by an “expert panel.” As discussed at the end of this section and in the appendix to this RG, this important process is the licensee’s responsibility and may be accomplished by means other than a formal panel. In any case, the key principles discussed in this guide should be addressed and shown to be satisfied regardless of the approach used for RI-IST program decisionmaking.

Regulatory Position C.2 contains additional application-specific details concerning RI-IST programs and Element 2.

Element 3: Define Implementation and Monitoring Program. The purpose of this element is to develop the implementation plan for the IST program. This involves determining both the methods to be used and the frequency of testing. The frequency and method of testing for each component is commensurate with the component's safety significance. To the extent practicable, the testing methods should address the relevant failure mechanisms that could significantly affect component reliability. In addition, a monitoring and corrective action program is established to ensure that the assumptions upon which the testing strategy has been based continue to be valid, and that no unexpected degradation in performance of the HSSCs and LSSCs occurs as a result of the change to the IST program. Regulatory Position C.3 gives specific guidance for Element 3.

Element 4: Submit Proposed Change. The final element involves preparing the documentation to be included in the submittal and the documentation to be maintained by the licensee for later reference, as needed. Regulatory Position C.4 gives guidance on documentation requirements for RI-IST programs.

The process is highly iterative. Thus, the final description of the proposed change to the IST program as defined in Element 1 depends on both the analysis performed in Element 2 and the definition of the implementation of the IST program performed in Element 3. The regulatory positions in Section C of this RG provide guidance on each element.

Although IST is, by its nature, a monitoring program, the monitoring that Element 3 refers to ensures that the assumptions made about the impact of the changes to the IST program are not invalidated. For example, if the test intervals are based on an allowable margin to failure, the monitoring is performed to make sure that these margins are not eroded.

In carrying out this process, the licensee will make a number of decisions based on the best available information. Some of this information will derive from deterministic engineering practice and some will be probabilistic in nature, resulting from PRA studies. The licensee is responsible for ensuring that it develops its RI-IST program using a well-reasoned and integrated decision process that considers both forms of input information (deterministic engineering and probabilistic). This important decisionmaking process may require participation of special combinations of licensee expertise (licensee staff), depending on technical and other issues, and require the support of outside consultants. Industry documents have generally referred to the use of an expert panel for such decisionmaking. The appendix to this RG discusses a number of IST-specific issues that might arise in expert panel deliberations.

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 staff considered IAEA Safety Requirements and Safety Guides[1]

[1] 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.

pursuant to the Commission's International Policy Statement (Ref. 13) and Management Directive and Handbook 6.6, "Regulatory Guides" (Ref. 14).

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

- IAEA Safety Standard SSG-2, "Deterministic Safety Analysis for Nuclear Power Plants" (Ref. 15), provides general considerations of the adequacy of the provisions by deterministic safety analysis, complemented by probabilistic safety assessment and engineering judgement.

- IAEA Safety Guide SSG-3, "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants Specific Safety Guide" (Ref. 16), provides good practices in aspects of risk-informed approach to inservice testing to help optimize the risk assessment on inservice testing program.

- IAEA Safety Standards SSR-2/1, "Safety of Nuclear Power Plants: Design" (Ref. 17), provides assurance that defense-in-depth has been implemented in the design of the plant. A safety analysis of the plant design should be performed in which methods of both deterministic and probabilistic analysis are included to assure the safety requirements are met throughout all stages of plant.

These documents provide recommendations for performing or managing a probabilistic safety assessment for nuclear power plants and using the assessment to support safe design and operation. This RG discusses some of the same principles with respect to changes to a plant's licensing basis. The NRC encourages licensees to consult these and other international documents noted throughout this regulatory guide and implement the good practices, where applicable, that are consistent with NRC regulations.

C. STAFF REGULATORY GUIDANCE

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 of this guide.

This RG provides a brief overview of a four-element approach, described in RG 1.174, as applied to the development of an RI-IST program for submittal as part of an alternative request in accordance with 10 CFR 50.55a(z)(1). This approach is iterative rather than sequential. Element 1 of the approach is to define the proposed changes to the IST program, which will help determine what supporting information is needed and define how subsequent reviews will be performed. Element 2 of the approach is to perform an engineering analysis to justify the proposed changes to the IST program. Element 3 of the approach is to define the implementation and monitoring program to ensure that the conclusions drawn from the risk-informed evaluation remain valid. Element 4 of the approach is to address the documentation requirements for licensee submittals to the NRC and to identify additional information that the licensee should maintain in its records in case later review or reference is needed. Appendix A to this RG contains additional guidance for dealing with certain IST-related issues that might arise during the deliberations of the licensee in carrying out integrated decisionmaking.

Licensees submitting risk information to the NRC should address each of the principles of risk-informed decisionmaking discussed in RG 1.174 and repeated in this guide and identify the appropriateness of chosen approaches and methods (whether they are quantitative or qualitative, deterministic or probabilistic), data, and criteria for considering risk for the decision to be made.

1. Element 1: Define Proposed Changes to the Inservice Testing Program

This first element of the process defines the proposed changes to the IST program. This involves describing what IST components (e.g., pumps and valves) will be involved and how their testing would be changed. This element also identifies supporting information and a proposed plan for the licensee's interactions with the NRC throughout the implementation of the RI-IST.

1.1 Define the Proposed Change

The licensee should prepare a full description of the proposed changes in the IST program. This description would include the following:

- Identify the aspects of the plant's design, operations, and other activities that the licensee proposes to modify together with implementation of the proposed RI-IST program such that the licensee can submit any necessary license amendment requests in addition to the alternative request to implement the proposed RI-IST program in accordance with 10 CFR 50.55a(z)(1). This will provide a basis from which the staff can evaluate the proposed RI-IST program in the context of any proposed changes that are outside the scope of the RI-IST program alternative request.

- Identify the specific revisions to existing testing schedules and methods resulting from implementation of the proposed program.

- Identify the components in the plant that are directly and indirectly involved with the proposed testing changes, including any components that are not presently covered in the plant's IST program but are determined to be important to safety (e.g., through PRA insights), and the particular systems that would be affected by the proposed changes. This information will aid in planning the supporting engineering analyses.

- Identify the information that will be used in support of the changes to the IST program, including performance data, deterministic engineering analyses, and PRA information.

- Provide a brief statement describing how the proposed changes to the IST program meet the objectives of the Commission's policy statement on the increased use of PRA in regulatory matters.

1.2 Inservice Testing Program Scope

IST requirements for certain safety-related pumps and valves are specified in 10 CFR 50.55a. These components are to be tested according to the requirements of the applicable OM Code.

The licensee's RI-IST program should include all components in the current OM Code-prescribed IST program. In addition, the RI-IST program should include those non-Code components that the licensee's integrated decisionmaking process categorized as HSSC.

1.3 Risk-Informed Inservice Testing Program Changes After Initial Approval

This section provides guidance on RI-IST program changes. The licensee should describe the proposed changes to any previously approved RI-IST programs in the alternative request.

The licensee should implement a process for determining when proposed RI-IST program changes require authorization as a new 10 CFR 50.55a(z)(1) alternative. Changes made to the authorized RI-IST program that could affect the process and results reviewed and authorized by the NRC staff should be evaluated to ensure that the basis for the NRC staff's authorization has not been compromised. All changes should be evaluated against the change mechanisms in accordance with the regulations (e.g., 10 CFR 50.55a, 10 CFR 50.59) to determine whether NRC review and authorization as a new 10 CFR 50.55a(z)(1) alternative request is required before implementation.

Licensees can change their RI-IST programs, consistent with the process and results that were reviewed and authorized by the NRC staff (i.e., as defined in the authorized RI-IST program description). Before implementation, licensees should have a process or procedures in place before making such changes to the previously authorized RI-IST program.

The cumulative impact of all RI-IST program changes (i.e., initial authorization plus subsequent changes) should be in accordance with Regulatory Position C.2.2.3.3.

Examples of changes to RI-IST programs that would require authorization as part of a new 10 CFR 50.55a(z)(1) alternative request include, but are not limited to, the following:

a. changes to the RI-IST program that involve programmatic changes (e.g., changes in the acceptance guidelines used for the licensee's integrated decisionmaking process), and

b. component test method changes that deviate from the methods described in the original 10 CFR 50.55a(z)(1) alternative request.

Examples of changes to RI-IST programs that would not require NRC review and authorization include, but are not limited to, the following:

a. changes to component groupings, test intervals, and test methods that do not involve a change to the overall RI-IST approach reviewed and authorized by the NRC staff;

b. component test method changes that involve the implementation of the OM Code of Record for the current 120-month IST program interval or an NRC-approved Code Case in RG 1.192 (Ref. 18) as incorporated by reference in 10 CFR 50.55a; and

c. recategorization of components because of experience, PRA insights, or design changes, but not programmatic changes, when the process used to recategorize the components is consistent with the RI-IST process and results reviewed and authorized by the NRC staff.

2. Element 2: Perform Engineering Analysis

As part of defining the proposed changes to the licensee's IST program, the licensee should conduct an engineering evaluation of the proposed change using 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 and other key principles described in this guide. RG 1.174 provides general guidance for the performance of this evaluation, supplemented by this RG.

2.1 Alternative Request

To implement an RI-IST program in lieu of the IST program required by the ASME OM Code as incorporated by reference in 10 CFR 50.55a, the licensee must submit an alternative request in accordance with 10 CFR 50.55a(z)(1) to demonstrate that the proposed RI-IST program will provide an acceptable level of quality and safety. In addition, the licensee should determine whether any additional requests need to be submitted in order to implement the proposed RI-IST program, such as license amendment requests, exemption requests, technical specification amendment requests, and other alternative or relief requests.

Following NRC authorization of the RI-IST program, the licensee is not required to submit additional requests to adjust the test interval of individual components that are categorized as having low safety significance (because the NRC staff reviewed and authorized the licensee's implementation plans for extending specific component test intervals as part of the licensee's RI-IST program submittal). However, licensees will need to submit alternative requests to use the RI-IST program in future 120-month IST program intervals.

2.2 Perform Engineering Evaluation

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 the potential effect of the proposed RI-IST program on defense-in-depth attributes and safety margins. In addition, defense-in-depth and safety margin should also be evaluated, as feasible, using risk techniques (e.g., PRA).

2.2.1 Defense-in-Depth Evaluation

RG 1.174 provides guidance on how to evaluate the impact of a proposed licensing-basis change on defense-in-depth to ensure that any impact is fully understood and addressed and that consistency with the defense-in-depth philosophy is maintained. The guidance in RG 1.174 should be used to evaluate the impact of a proposed RI-IST change on defense-in-depth to determine whether consistency with the defense-in-depth philosophy is maintained.

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 traditionally 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 (SRM)-SECY-98-144, "Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulation," dated March 1, 1999 (Ref. 19), 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 IST program change is consistent with the defense-in-depth philosophy as described in Section 2.1.1 of RG 1.174. The intent of this key principle of risk-informed decisionmaking with respect to the proposed RI-IST program is to ensure that any impact of the proposed 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 licensee should fully understand how the proposed change impacts the plant from both risk and deterministic engineering perspectives.

In addition, RG 1.174 provides detailed guidance on how to evaluate the impact of a proposed change on defense-in-depth to determine whether that consistency is achieved. The seven defense-in-depth considerations and four layers of defense that are addressed in Section C.2.1.1 of RG 1.174 should be used to evaluate the impact of a proposed IST change on defense-in-depth to determine whether it maintains consistency with the defense-in-depth philosophy.

2.2.2 Safety Margin Evaluation

The maintenance of safety margins is also a very important part of ensuring continued reactor safety and is included as one of the key principles in Regulatory Position C.2 of RG 1.174.

Regulatory Position C.2.1.2 of RG 1.174 states "[w]ith sufficient safety margins, (1) the codes and standards or their alternatives approved for use by the NRC are met and (2) safety analysis acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) are met or proposed revisions provide sufficient margin to account for uncertainty in the analysis and data."

The categorization process might identify components currently not included in the IST program, and their addition as HSSCs will clearly improve safety margins. It is also important that the performance-monitoring program be capable of quickly identifying significant degradation in performance so that, if necessary, corrective measures can be implemented before the margin to failure is significantly reduced. The improved understanding of the relative importance of plant components to risk resulting from the development of the RI-IST program should promote an improved understanding of how the components in the IST program contribute to a plant's margin of safety, and this should be discussed in the application.

2.2.3 Evaluation of Risk Impact

This section discusses issues specific to the RI-IST process. RG 1.174 contains much of the general guidance that is applicable to this topic. The licensee may use its PRA to address the principle that proposed increases in CDF and LERF are small and are consistent with the intent of the Commission's Safety Goal Policy Statement. A PRA used in a risk-informed application 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.

In an RI-IST program, information obtained from a PRA should be used in two ways: (1) to provide input to the categorization of SSCs into HSSC groupings, and (2) to assess the impact of the proposed RI-IST program on CDF and LERF. Regulatory Position C.2.2.3.1 discusses, in general terms, issues related to the scope, level of detail, conformance with the technical elements of the ASME/ANS PRA standard, and plant representation of a PRA that is used for RI-IST applications. Regulatory Positions C.2.2.3.2 and C.2.2.3.3 give more specific considerations and address the use of PRA in categorization and in the assessment of the impact on risk metrics, respectively.

2.2.3.1 Acceptability of Probabilistic Risk Assessments for Inservice Testing Applications

For the quantitative results of the PRA to play a major and direct role in decisionmaking, such results should be derived from "acceptable" analyses, and the extent to which the results apply should be well understood. Both Regulatory Position C of RG 1.200 and the Regulatory Position C.2.3 of RG 1.174 address, in general terms, acceptability of the PRA. The PRA analysis used to support an application is measured in terms of its appropriateness with respect to scope, level of detail, conformance with the technical elements, and plant representation. These aspects of the PRA should be commensurate with its intended use and the role the PRA results play in the integrated decision process.

While a full-scope PRA that covers all modes of operation and initiating events is preferred, a lesser scope PRA can provide useful risk information. If less than a full-scope PRA is used to support the proposed RI-IST program, supplemental information (deterministic and qualitative) should be considered during the integrated decisionmaking process. Regulatory Position C.2.2.3.2 provides more specific considerations to address this condition.

For the PRA to be useful in the development of an RI-IST program, the PRA model should be developed to the component level for the systems, including nonsafety systems, considered important for the prevention of core damage and release of radioactivity.

A PRA used for an RI-IST program should be performed correctly and in a manner that is consistent with accepted practices. The PRA should reflect the actual design, construction, operating practices, and operating experience of the plant. The acceptability of the PRA used for an RI-IST program should be commensurate with the role it plays in determining test intervals or test methods and with the role the integrated decisionmaking panel plays in compensating for the limitations of the PRA. RG 1.174 further discusses the requirements of PRA acceptability.

2.2.3.2 Categorization of Components

The categorization of components is important in the implementation of the RI-IST program because it is an efficient and risk-informed way of providing insights in the areas in which certain requirements can be relaxed with assurance of continued adequate protection. Thus, categorization of components, in addition to the engineering evaluation described in Regulatory Position C.2.2 and the calculation of change in overall plant risk described in Regulatory Position C.2.2.3.3, will provide significant input to the determination of whether the IST program is acceptable or not.

The establishment of the safety significance of components by using PRA-determined importance is valuable for the following reasons:

a. When performed with a series of sensitivity evaluations, the use of PRA to determine importance can identify potential risk outliers by identifying IST components that could dominate risk for various plant configurations and operational modes, PRA model assumptions, and data and model uncertainties.

b. Importance measure evaluations can provide a useful means to identify improvements to current IST practices during the risk-informed application process.

c. System- or functional-level importance results can provide a high-level verification of component-level results and can provide insights into the potential risk significance of RI-IST program components that are not modeled in the PRA.

Appendix A of RG 1.174 provides general guidelines for risk categorization of components using importance measures and other information. These general guidelines address acceptable methods for carrying out categorization and some of the limitations of this process. This section gives guidelines specific to the RI-IST application. As used here, risk categorization refers to the process for grouping RI-IST program components into LSSC and HSSC categories.

Components are initially categorized into HSSC and LSSC groupings based on threshold values for the importance measures. Depending on whether the PRA is performed using the fault tree linking or event tree linking approach, importance measures can be provided at the component or train level. In either case, the importance measures are applicable to the items taken one at a time; therefore, as discussed in RG 1.174, while a licensee is free to choose the threshold values of importance measures, it will be necessary to demonstrate that the integrated impact of the change meets Principle 4. The next section of this RG discusses one acceptable approach.

PRA systematically takes credit for non-ASME OM Code components as providing support, acting as alternatives, and acting as backups to those components that are within the scope of the current ASME OM Code. Accordingly, to ensure that the proposed RI-IST program will provide an acceptable level of quality and safety, the licensee should include these additional risk-important components in its RI-IST program proposal. Specifically, the licensee’s RI-IST program should include those ASME Code Class 1, 2, and 3 and non-Code components that the licensee’s integrated decisionmaking process categorized as HSSC and thus determined to be appropriate additional candidates for the RI-IST program.

Although PRAs model many of the SSCs involved in the performance of plant safety functions, other SSCs are not modeled for various reasons. However, this should not imply that nonmodeled components are not important in terms of contributions to plant risk. For example, some components are not modeled because certain initiating events may not be modeled (e.g., low-power and shutdown events, or some external events); in other cases, components may not be directly modeled because they are grouped together with events that are modeled (e.g., initiating events, operator recovery events, or within other system or function boundaries); and in some cases, components are screened out from the analysis because of their assumed inherent reliability. Failure modes may be screened out because of their insignificant contribution to risk (e.g., spurious closure of a valve). When feasible, the licensee should consider adding missing components or missing initiators or plant operating states to the PRA. When not feasible, the licensee may use information based on engineering analyses and judgment to determine whether a component should be treated as an LSSC or HSSC. One approach to combining these different pieces of information is to use what has been referred to as an expert panel. Appendices B and C to SRP Section 19.2, “Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance,” contain staff expectations on the use of expert panels in integrated decisionmaking and SSC categorization, respectively.

In classifying a component not modeled in the PRA as LSSC, the expert panel should determine the following:

a. The component does not perform a safety function, does not perform a support function to a safety function, and does not complement a safety function.

b. The component does not support operator actions credited in the PRA for either procedural or recovery actions.

c. The failure of the component will not result in the eventual occurrence of a PRA initiating event.

d. The component is not a part of a system that acts as a barrier to fission product release during severe accidents.

e. The failure of the component will not result in unintentional releases of radioactive material even in the absence of severe accident conditions.

When using risk importance measures to identify components that are low-risk contributors, the potential limitations of these measures have to be addressed. Therefore, the licensee's integrated decisionmaking process (e.g., expert panel) should include evaluations that demonstrate the sensitivity of the risk-importance results to the important PRA modeling techniques, assumptions, and data. Issues that the licensee should consider and address when determining low-risk contributors include truncation limit used, different risk metrics (i.e., CDF and LERF), different component failure modes, different maintenance states and plant configurations, multiple component considerations, defense-in-depth, and analysis of uncertainties (including sensitivity studies to component data uncertainties, common-cause failures (CCFs), and recovery actions). RG 1.174 and NUREG-1855 provide additional guidance on the treatment and analysis of uncertainties.

While the categorization process can be used to highlight areas in which the testing strategy can be improved and areas in which sufficient safety margins exist to the point that the testing strategy can be relaxed, it is the determination of the change in risk from the overall changes associated with the RI-IST program that is of concern in demonstrating that Principle 4 has been met. Therefore, this RG in general does not provide the threshold values of importance measures used to categorize components as HSSC or LSSC. Instead, the licensee should demonstrate that the overall impact of the change on plant risk is small, as discussed in Regulatory Position C.2.2.3.3.

As part of the categorization process, licensees should also address the initiating events and plant operating modes missing from the PRA evaluation. The licensee can do this either by providing qualitative arguments that the proposed change to the proposed RI-IST program will not result in an increase on risk, or by demonstrating that the component's significant to risk in these missing contributors are maintained as HSSC.

2.2.3.3 Use of a Probabilistic Risk Assessment to Evaluate the Risk Increase from Changes in the Inservice Testing Program

One of the important uses of the PRA is to evaluate the impact of the proposed RI-IST program with respect to the changes in CDF and LERF, as discussed in Sections C.2.4 and C.2.5 of RG 1.174. In addition, the PRA can provide a baseline risk profile of the plant, and the extent of analysis of the baseline CDF and LERF depends on the proposed change in CDF and LERF. When the PRA is not full scope, the licensee should address the significance of the out-of-scope items, as discussed in Regulatory Positions C.2.5 of RG 1.174 and C.2.2.3.1 of this guide.

2.2.3.3.1 Modeling the Impact of Changes resulting from the Proposed Risk-Informed Inservice Testing Program

In order for the PRA to support the decision appropriately, the proposed RI-IST program should have a functional mapping between the components associated with the RI-IST program and the PRA basic event probability quantification. Part of the basis for the acceptability of the proposed RI-IST program is a quantitative demonstration using a PRA that established risk measures are not significantly increased by the proposed changes to the IST for selected components. To establish this demonstration, the PRA should include models that appropriately account for the change in reliability of the components as a function of the RI-IST program implementation. In general, this will include not only changes to the test interval but also the effects of testing method modifications. Enhanced testing might be shown to improve or maintain component reliability, even if the interval is extended. That is, a better test might compensate for a longer interval between tests. Licensees that apply for substantial increases in test intervals are should address this area; that is, as appropriate, licensees should consider improvements in testing that would compensate for the increased intervals under consideration.

As discussed in Section 5 of NUREG/CR-6141, "Handbook of Methods for Risk-Based Analyses of Technical Specifications," issued December 1994 (Ref. 20), the total failure probability of a component consists of a standby time-related contribution and a cyclic demand-related contribution. The cyclic demand-related contribution is associated with failures that are caused by stresses imposed on the component by demanding, starting, or cycling the component, such as electrical and mechanical stresses occurring when the component is demanded. The standby time-related contribution is associated with failures occurring while the component is in standby between tests related to deterioration and events that occur over time, such as from corrosion or erosion. In Section 5 of NUREG/CR-6141, the standby time-related contribution of the component failure probability is expressed in terms of a constant standby failure rate for transition to the failed state between tests, test interval length, and test strategy (e.g., independent testing, sequential testing, staggered testing). The total failure probability of the component is assumed to be the standby time-related contribution (i.e., no cyclic demand-related contribution); and therefore, the total failure probability scales linearly with test interval length (i.e., doubles when test interval doubles). This assumption is conservative in that it tends to overstate the effect of test interval extension leading to an overestimate of the associated change in risk. This assumption is therefore acceptable to the NRC staff. It should be noted that the total failure probability has a cyclic demand-related contribution, such that more realistic modeling in this area (i.e., dividing the total failure probability into a standby time-related and cyclic demand-related contribution) could be used to support development of the proposed RI-IST program. In this case, such a breakdown of the failure rate should be justified through data analysis or engineering analyses.

This approach for evaluating test interval extensions assumes that failures are random occurrences and that the frequency of these occurrences does not increase as the test interval is increased (i.e., constant standby failure rate). This assumption is based on data from current IST test intervals, and therefore, may not include insidious effects that arise from extended test intervals (e.g., corrosion or erosion, intrusion of foreign material into working parts, adverse environmental exposure, or breakdown of lubrication) that could significantly degrade the component if test intervals become excessively long. Therefore, there is some concern that the standby failure rate may increase as test intervals are extended. Unless it can be demonstrated that either degradation is not expected to be significant or that the test would identify degradation before failures are likely to occur, use of the constant failure rate model could be nonconservative.

One way to address this uncertainty is to use the PRA insights to help design an appropriate implementation and monitoring program; for example, to approach the interval increase in a stepwise fashion rather than going to the theoretically allowable maximum in a single step, or to stagger the testing of redundant components (test different trains on alternating schedules) so that the population of components is being sampled relatively frequently, even though individual members of the population are not. By using such approaches, the existence of the above effects can be detected, and compensatory measures taken, to correct the testing of the remaining population members. However, it is important that the monitoring includes enough tests to ensure that these effects will be detected early enough to be addressed, and that the tests are capable of detecting the time-related degradation. Regulatory Position C.3.3 discusses performance monitoring.

A check should also be performed to determine whether non-IST manipulations have been credited either in IST basic events or in compensating component basic events. If each component is stroked or challenged between instances of IST, and if these activities are capable of revealing component failure, the effective fault exposure time can be less than the RI-IST interval and credited in the risk evaluation. In addition, some instances of revealing a component fault through challenge have adverse consequences, including functional failure. Thus, if credit is taken for shortening fault exposure time through functional challenges, the quantification of risk should account for such adverse consequences.

2.2.3.3.2 Evaluating the Change in Core Damage Frequency and Large Early Release Frequency

Once the impact of the proposed change on the individual basic event probabilities has been determined, the change in CDF and LERF can be evaluated. Careful consideration should be given to issues that become more important as the change in basic event probabilities becomes larger. When using a fault tree linking approach to PRA, it is preferable that the model be re-solved rather than simply requantifying the CDF and LERF cut set solutions. In addition, it is important to pay attention to the parametric uncertainty analysis, especially if the change is dominated by cut sets that have multiple LSSCs. The "state of knowledge correlation effect" could be significant if there are a significant number of cut sets with similar SSCs contributing to the change in risk. RG 1.174 discusses the parametric uncertainty analysis in more detail.

In addition, model and completeness uncertainties should be addressed as discussed in RG 1.174. In particular, initiating events and modes of plant operations whose risk impact are not included in the PRA need additional analyses or justification that the proposed changes do not significantly increase the risk from those nonmodeled contributors.

2.2.3.3.3 Acceptance Guidelines

The change in risk from implementation of the proposed RI-IST program should be consistent with the guidelines discussed in Regulatory Positions C.2.4 and C.2.5 of RG 1.174. The licensee should address the model and completeness uncertainty, as discussed in Regulatory Position C.2.5 of RG 1.174. In addition, the licensee should address parameter uncertainty either by propagating the uncertainty during sequence quantification or by demonstrating that the "state-of-knowledge correlation" effect is not significant, especially in cut sets in which the proposed RI-IST program changes might affect multiple components that are similar.

In evaluating the change in plant risk from the proposed changes from the RI-IST program, the licensee should perform the following:

a. Evaluate the risk significance of extending the test interval on affected components. This requires that the licensee address the change in component availability as a function of test interval. The analysis should include either a quantitative consideration of the degradation of the component failure rate as a function of time, supported by appropriate data and analysis, or arguments that support the conclusion that no significant degradation will occur.

b. Consider the effects of enhanced testing to the extent needed to substantiate the change.

Other issues that should be addressed in the quantification of the change in risk include the following:

a. The impact of the IST change on the frequency of event initiators (those already included in the PRA and those screened out because of low frequency) should be determined. For applications of an RI-IST program, potentially significant initiators include valve failure that could lead to interfacing system loss-of-coolant accidents or to other sequences that fail the containment isolation function.

b. The impact of the implementation of the proposed RI-IST program on the CCF contributions of affected component(s) should be addressed either by the use of sensitivity studies or by the use of qualitative assessments that show that the CCF contribution would not become significant under the proposed RI-IST program (e.g., by use of phased implementation, staggered testing, and monitoring for common-cause effects).

c. Justification of IST relaxations should not be based on credit for post-accident recovery of failed components (repair or ad-hoc manual actions, such as manually forcing stuck valves to open). However, credit may be taken for proceduralized implementation of alternative success strategies. For each human action that compensates for an increase in basic event probability as a result of IST relaxation, the human action should be feasible and proceduralized, including appropriate operator training, to ensure performance of the human action at the level credited in the quantification.

d. The failure rates and probabilities used for components affected by implementation of the proposed RI-IST program should appropriately consider both plant-specific and generic data. The licensee should determine whether individual components affected by the change are performing more poorly than the average associated with their class; the licensee should avoid relaxing IST for those components to the point that the unavailability of the poor performers would be appreciably worse than that assumed in the risk analysis. In addition, the licensee should review those components that have experienced repeated failures to see whether the testing scheme (interval and methods) would be considered adequate to support the performance credited to them in the risk analysis.

e. The evaluation should consider the truncation of LSSCs. It is preferred that solutions be obtained from a re-solution of the model rather than a requantification of CDF and LERF cut sets.

f. The cumulative impact of all RI-IST program changes (initial approval plus later changes) should comply with the guidance given in this section.

2.3 Integrated Decisionmaking

In accordance with Section C.2.6 of RG 1.174, the results of the evaluations under Sections C.1, C.2, and C.3 should be considered in an integrated manner to determine whether the licensee considers it acceptable to submit an alternative request to implement a proposed RI-IST program in accordance with 10 CFR 50.55a(z)(1). PRA results are compared to numerical acceptance guidelines, along with other deterministic considerations, operating experience, lessons learned from previous changes, and the implementation and monitoring program. The licensee's determination of the acceptability of submitting the proposed change to implement an RI-IST program in accordance with 10 CFR 50.55a(z)(1) should be based on all these considerations and not solely on the numerical results of the PRA. Numerical PRA results should be just one input into the decisionmaking and help in building an overall picture of the risk implications of the proposed change. As discussed previously, the numerical guidelines are used to ensure that any increase in risk is within acceptable limits, deterministic considerations are used to ensure that the proposed RI-IST program will satisfy the applicable rules and regulations, practical considerations are taken into account to judge the acceptability of proposing an RI-IST program, lessons learned from past experience ensure that mistakes are not repeated, and monitoring ensures that the proposed change will not degrade operational safety over time. RG 1.174, Section C.2.6, provides additional guidance on the integrated decisionmaking process. Furthermore, Section C of RG 1.174 identifies a set of expectations that licensees should follow in addressing the key principles. Because of the importance of these expectations, they are repeated here.

a. All safety impacts of the proposed licensing basis changes are evaluated in an integrated manner. The evaluation is part of an overall risk management approach in which the licensee is using risk analysis to improve operational and engineering decisions broadly by identifying and taking advantage of opportunities to reduce risk and not just to eliminate requirements the licensee sees as undesirable. For those cases in which risk increases are proposed, the benefits should be described and should be commensurate with the proposed risk increases. The approach used to identify changes in requirements should also be used to identify areas in which requirements should be increased, as well as those in which they can be reduced.

b. The engineering analyses (including deterministic and probabilistic analyses) conducted to justify the proposed licensing basis change should (1) be appropriate for the nature and scope of the change, (2) be based on the as-built and as-operated and maintained plant, and (3) reflect operating experience at the plant. The ASME/ANS PRA standard endorsed by RG 1.200 defines "as-built, as-operated" as a concept that reflects the degree to which the PRA matches the current plant design, plant procedures, and plant performance data, relative to a specific point in time (see Section C.2.3 of RG 1.174 for additional information on the relationship between RG 1.174 and the ASME/ANS PRA standard). Acceptability of the engineering analyses is determined by assessing the scope, level of detail, supporting technical analyses, and plant representation.

c. The plant-specific PRA supporting the licensee's proposals has been demonstrated to be acceptable.

d. Uncertainty receives appropriate consideration in the analyses and interpretation of findings, including use of a program of monitoring, feedback, and corrective action to address key sources of uncertainty. NUREG-1855 provides acceptable guidance for the treatment of uncertainties in risk-informed decisionmaking.

e. The use of CDF and LERF as bases for PRA is an acceptable approach for addressing Principle 4. Use of the Commission's Safety Goal qualitative health objectives in lieu of CDF and LERF is acceptable in principle, and licensees may propose their use. However, in practice, implementing such an approach would require an extension to a Level 3 PRA, in which case the methods and assumptions used in the Level 3 analysis, and associated uncertainties, would require additional attention. Later parts of RG 1.174 present guidance on risk metrics for plants licensed under 10 CFR Part 52.

f. Increases in CDF and LERF resulting from proposed licensing basis changes should be limited to small increments. The decision process should track and consider the cumulative effect of such changes, whether they result in an increase or a decrease, if available, in risk. For purposes of this guide, a proposed licensing basis change that meets the acceptance guidelines discussed in Section C.2.4 of RG 1.174 is considered to meet the intent of the policy statement.

g. The licensee should evaluate the acceptability of the proposed licensing basis changes in an integrated fashion that ensures that all principles are met.

h. Data, methods, and assessment criteria used to support regulatory decisionmaking should be well documented and available for public review.

These expectations apply to both probabilistic and deterministic engineering considerations, which are addressed in more detail in this section and in RG 1.174.

Licensees are expected to review commitments related to outage planning and control to verify that they are appropriately reflected in the licensee's component grouping. This should include components necessary to maintain adequate defense-in-depth as well as components that might be operated as a result of contingency plans developed to support the outage.

Licensees are also expected to review licensing basis documentation to ensure that the deterministic engineering-related factors mentioned above are adequately modeled or otherwise addressed in the PRA analysis.

When making final programmatic decisions, choices should be made based on all the available information. There may be cases where information is incomplete or where conflicts appear to exist between the deterministic engineering data and the PRA-generated information. The licensee should use well-reasoned judgment to resolve the issues in the best manner possible, including due consideration to the safety of the plant. Various industry documents (Refs. 21, 22 and 23) discuss the process of integrated decisionmaking, with reference to the use of an expert panel. The appendix to this RG includes detailed guidance on certain aspects of integrated decisionmaking specific to RI-IST programs. This RG is not intended to suggest that the licensee always form a body, such as an expert panel, to fulfill this function. This section covers some general guidance for this important activity; the appendix to this RG provides more specific details.

In summary, acceptability of the proposed change should be determined by using an integrated decisionmaking process that addresses three major areas: (1) an evaluation of the proposed change in light of the plant's licensing basis; (2) an evaluation of the proposed change relative to the key principles and the acceptance guidelines; and (3) the proposed plans for implementation, performance monitoring, and corrective action. As stated in the Commission's policy statement on the increased use of PRA in regulatory matters, the PRA information used to support the RI-IST program should be as realistic as possible, with reduced unnecessary conservatisms, yet include a consideration of uncertainties. These factors are very important when considering the cumulative plant risk and accounting for possible risk increases as well as risk benefits. The licensee should carefully document all of these considerations in the RI-IST program description, including those areas that have been quantified through the use of PRA, as well as qualitative arguments for those areas that cannot readily be quantified in the following:

a. The licensee's proposed RI-IST program should be supported by both a deterministic engineering analysis and a PRA analysis.

b. The licensee's RI-IST program submittal should be consistent with the guidance contained throughout this RG, specifically with the expectations listed in this section, or the submittal should justify why an alternative approach is acceptable.

3. Element 3: Define Implementation and Monitoring Program

Upon approval of an RI-IST program, the licensee should have in place an implementation schedule for testing all HSSCs and LSSCs identified in its program. This schedule should include test strategies and testing frequencies for HSSCs and LSSCs that are within the scope of the licensee's IST program and components identified as HSSCs that are not currently in the IST program.

3.1 Inservice Testing Program Changes

This section discusses the test strategy changes (i.e., component test frequency and methods changes) that licensees should make as part of an RI-IST program.

The RI-IST program should identify components on which the test strategy (i.e., frequency, methods, or both) should be more focused as well as components for which the test strategy might be relaxed. The information contained in, and derived from, the PRA should be used to help construct the testing strategy for components. To the extent practicable, components with high safety significance should be tested in ways that are effective at detecting their risk-important failure modes and causes (e.g., ability to detect failure, to detect conditions that are precursors to failure, and predict end of service life). Components categorized as LSSC may be tested less rigorously than components categorized as HSSC (e.g., less frequent or informative tests).

In some situations, an acceptable test strategy for components categorized as HSSC may be to conduct the existing approved OM Code IST test at the prescribed frequency. In some situations, an acceptable test strategy for components categorized as LSSC may be to conduct the existing approved OM Code IST test at an extended interval.

NRC-approved ASME risk-informed Code Cases may define an acceptable strategy for testing components categorized as HSSC and LSSC. Licensees that choose to pursue RI-IST programs should consider adopting ASME-developed and NRC-approved test strategies and should justify deviations from approved ASME Code Cases as part of the RI-IST program request.

In establishing the test strategy for components, the licensee should consider component design, service condition, and performance, as well as risk insights. The proposed test strategy should be supported by data that are appropriate for the component. The omission of either generic or plant-specific data should be justified. The proposed test interval should be significantly less than the PRA-assumed expected time to failure of the components in question (e.g., an order of magnitude less). For example, the motor-operated valve (MOV) exercise requirement (which is comparable to the current stroke time test) should be performed at intervals considerably shorter than the expected time to failure. In addition, the licensee should demonstrate that adequate component capability (margin) exists, above that required during design basis conditions, such that component operating characteristics over time do not result in reaching a point of insufficient margin before the next scheduled test activity.

The IST interval should generally not be extended beyond once every 6 years or three refueling outages (whichever is longer) without specific, compelling, documented justification. Extensions beyond 6 years or three refueling outages (whichever is longer) will be considered as component performance data at extended intervals are acquired.

Components categorized as HSSC that are not in the licensee's current IST program should be considered for testing as described by NRC-approved ASME risk-informed Code Cases. If the licensee does not use an OM Code testing method, the licensee should develop alternative test methods to ensure operational readiness and to detect component degradation (i.e., degradation associated with failure modes identified as important in the licensee's PRA). At a minimum, the RI-IST program should summarize these components and their proposed testing.

For components categorized as HSSC that were the subject of a previous NRC-approved relief request or an authorized alternative test, the RI-IST alternative request should discuss the appropriateness of the test method specified in the previous relief or alternative request in light of the component's safety significance.

If practical, IST components (with the exception of certain check valves and relief valves) should, at a minimum, be exercised or operated at least once every refueling cycle. More frequent exercising should be considered for components in any of the following categories, if practical:

a. components with high risk significance,

b. components in adverse or harsh environmental conditions, and

c. components with any abnormal characteristics (operational, design, or maintenance conditions).

The RI-IST program description should include the testing strategy for each component (or group of components) in the licensee's RI-IST program. The RI-IST program description should summarize all testing to be performed on a group of components (e.g., MOV testing in response to NRC Generic Letter 96-05). The licensee's RI-IST program plan should delineate the specific testing to be done on each component (or group of components).

3.2 Implementation Program

The applicable OM Code generally requires that safety-related components within the program scope, as defined in the current OM Code, be tested on a specific frequency regardless of safety significance. The RI-IST program if authorized in accordance with 10 CFR 50.55a(z)(1) will allow the extension of certain component testing intervals and modification of certain component testing methods based on the determination of individual component importance. The implementation of the program will involve scheduling test intervals based on the results of probabilistic analysis and deterministic evaluation of each individual component.

The RI-IST program should distinguish between HSSCs and LSSCs for testing intervals. All components in the RI-IST program should be individually specified in the 10 CFR 50.55a(z)(1) alternative request with their testing methods and intervals. The RI-IST program, as well as the implementing and test procedures should appropriately reference plant corrective action and feedback programs to ensure that testing failures are reevaluated for possible adjustment to the component's grouping and test strategy.

It may be acceptable to implement RI-IST programs on a phased approach. Subsequent to the authorization of an RI-IST program, the licensee implements interval extensions for LSSCs, and this may take place on a component, train, or system level. However, the NRC will generally not authorize immediately adjusting the test intervals of LSSCs to the maximum proposed test interval. Normally, test interval increases will be done stepwise, with gradual extensions permitted, consistent with cumulative performance data for operation at the extended intervals. The actual testing intervals for each component in the RI-IST program should be available at the plant site for inspection.

The tests described in the current OM Code assure that components relied on to safely shutdown the reactor, maintain it in a safe shutdown condition, and mitigate the consequences of an accident are able to perform their specified safety function(s) when required by plant conditions. However, enhanced tests, even at an extended test interval, may be more effective for detecting the important failure modes and causes of a component or group of components.

HSSCs that are not in the current IST program should be tested, where practical, in accordance with the OM Code, including compliance with all administrative requirements. When OM Code testing is not practical, the licensee should develop alternative test methods to ensure operational readiness and to detect component degradation (i.e., degradation associated with failure modes identified as important in the licensee's PRA).

An acceptable method to extend the test interval for LSSCs is to group like components and stagger their testing equally over the interval identified for a specific component based on the probabilistic analysis and deterministic evaluation of each individual component. Initially, it would be desirable to test at least one component in each group every refueling outage. For example, component grouping should consider valve actuator type for power-operated valves and pump driver type, as applicable. With this method, generic age-related failures can be identified while allowing immediate implementation for some components. For component groups that are insufficient in size to test one component every refueling outage, the implementation of the interval should be accomplished in a more gradual stepwise manner. The RI-IST program should justify the selected test frequency for LSSCs that are to be tested on a staggered basis.

The licensee should perform the following:

a. For components that will be tested in accordance with the test frequency and method requirements specified in the OM Code of Record for the current 120-month IST program interval or OM Code Cases approved in RG 1.192 as incorporated by reference in 10 CFR 50.55a, no specific implementation schedule needs to be provided to the NRC as part of the 10 CFR 50.55a(z)(1) alternative request. The licensee's RI-IST program should document the test frequency and method.

b. For components that will tested by other methods, the licensee's 10 CFR 50.55a(z)(1) alternative request for the proposed RI-IST program should describe the test strategies (i.e., interval extension plan and test methodology).

The licensee should increase the test interval for components in a stepwise manner (i.e., equal or successively smaller steps, not to exceed one refueling cycle per step). If no significant time-dependent failures occur, the interval can be gradually extended until the component is tested at the maximum proposed extended test interval. An acceptable approach is to group similar components and test them on a staggered basis. Guidance on grouping components appears in Position 2 of NRC Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," dated April 3, 1989 (Ref. 24), for check valves, and in Supplement 6 to NRC Generic Letter 89-10 and Section 3.5 of ASME Code Case OMN-1 (Ref. 25) for MOVs, as well as in other NRC-endorsed documents.

3.3 Performance Monitoring

Performance monitoring in RI-IST programs refers to the monitoring of IST data for components within the scope of the RI-IST program (i.e., including both HSSCs and LSSCs). The purpose of performance monitoring in an RI-IST program is twofold. First, performance monitoring should help confirm that no insidious failure mechanisms that are related to the revised test strategies become important enough to alter the failure rates assumed in the justification of program changes. Second, performance monitoring should, to the extent practicable, ensure that adequate component capability (i.e., margin) exists, above that required during design basis conditions, so that component operating characteristics over time do not result in reaching a point of insufficient margin before the next scheduled test activity. RG 1.174 provides guidance on performance monitoring when testing under design basis conditions is impracticable. In most cases, component-level monitoring will be expected.

Two important aspects of performance monitoring are whether the test frequency is sufficient to provide meaningful data and whether the testing methods, procedures, and analysis are adequately developed to ensure that performance degradation is detected. Component failure rates cannot be allowed to rise to unacceptable levels (e.g., significantly higher than the failure rates used in the risk evaluation to support the RI-IST program) before detection and corrective action take place.

The NRC staff expects that licensees will integrate, or at least coordinate, their monitoring for the RI-IST program with existing programs for monitoring equipment performance and other operating experience at their sites and, when appropriate, throughout the industry. In particular, monitoring performed as part of Maintenance Rule (10 CFR 50.65) implementation can be credited in the RI-IST program when the monitoring performed under the Maintenance Rule is sufficient for the SSCs in the RI-IST program. As stated in RG 1.174, if an application requires monitoring of SSCs not covered by the Maintenance Rule, or involves SSCs that need a greater resolution of monitoring than specified in the Maintenance Rule (e.g., component-level versus train- or plant-level monitoring), it may be advantageous for a licensee to adjust the Maintenance Rule monitoring program rather than develop additional monitoring programs for RI-IST purposes. Therefore, it may be advantageous to adjust the performance criteria within the licensee's Maintenance Rule program to meet the guidance below.

The proposed monitoring programs should be capable of adequately tracking the performance of equipment that, when degraded, could alter the conclusions that were key to supporting the RI-IST program. Monitoring programs should be structured such that SSCs are monitored commensurate with their safety significance. This allows for a reduced level of monitoring of components categorized as having low safety significance, provided the guidance below is still met.

The licensee's performance monitoring process should have the following attributes:

a. The process includes enough tests to provide meaningful data.

b. These tests are devised such that incipient degradation can reasonably be expected to be detected.

c. The licensee trends appropriate parameters as necessary to provide reasonable assurance that the component will remain operable over the test interval.

Assurance should be established that degradation is not significant for components placed on an extended test interval, and that failure rate assumptions for these components are not contradicted by test data. It should be clearly established that those test procedures and evaluation methods are implemented that reasonably ensure that degradation will be detected, and corrective action will be taken.

3.4 Feedback and Corrective Action

The licensee's corrective action program for this application should contain a performance-based feedback mechanism to ensure that if a particular component's test strategy is adjusted in a way that is ineffective in detecting component degradation and failure, particularly potential CCF mechanisms, the RI-IST program weakness is promptly detected and corrected. Performance monitoring should be provided for SSCs with feedback to the RI-IST program for appropriate adjustments when needed.

If component failures or degradation occur at a higher rate than assumed in the basis for the RI-IST program, the following basic steps should be followed to implement corrective action:

a. The causes of the failures or degradation should be determined, and corrective action implemented.

b. The component's test effectiveness should be reevaluated, and the RI-IST program should be modified accordingly.

The licensee's corrective action program evaluates RI-IST components that either fail to meet the test acceptance criteria established by the licensee detailed in the RI-IST program plan or are otherwise determined to be in a nonconforming condition (e.g., a failure or degraded condition discovered during normal plant operation).

An acceptable evaluation does the following:

a. It complies with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action."

b. It promptly determines the impact of the failure or nonconforming condition on system or train operability and follows the appropriate technical specification when component capability cannot be demonstrated.

c. It determines and corrects the apparent or root cause of the failure or nonconforming condition (e.g., improves testing practices, repairs or replaces the component). The root cause of failure should be determined for all components categorized as having high safety significance, as well as for components categorized as having low safety significance when the apparent cause of failure may contribute to CCF.

d. It assesses the applicability of the failure or nonconforming condition to other components in the RI-IST program (including any test sample expansion that may be required for grouped components such as relief valves).

e. It corrects other susceptible RI-IST components, as necessary.

f. It considers the effectiveness of the component's test strategy in detecting the failure or nonconforming condition. The test interval, test methods, or both, are adjusted as appropriate when the component (or group of components) experiences repeated or age-related failures or nonconforming conditions.

The corrective action evaluations should periodically be provided to the licensee's PRA group so that any necessary model changes and re-grouping are done, as appropriate. The effect of the failures on overall plant risk should be evaluated, and it should be confirmed that the corrective actions taken will restore the plant risk to an acceptable level.

The RI-IST program documents should be revised to document any RI-IST program changes resulting from corrective actions taken. The licensee does not need to report the results of the RI-IST corrective actions to the NRC (except as required by specific NRC reporting requirements), but should retain them on-site for inspection.

3.5 Periodic Reassessment

RI-IST programs should contain provisions to periodically feed component performance data back into both the component categorization and component test strategy determination (i.e., test interval and methods) processes. Such assessments should also consider corrective actions that have been taken on past IST program components. (This periodic reassessment should not be confused with the 120-month program updates required by 10 CFR 50.55a(f)(5)(i), whereby the licensee's IST program must comply with later versions of the OM Code that have been approved by the NRC.)

The assessment should address the following:

a. Review and revise as necessary the models and data used to categorize components to determine whether component groupings have changed.

b. Reevaluate equipment performance to determine whether the RI-IST program should be adjusted (based on both plant-specific and generic information).

The licensee should have procedures in place to identify the need for more emergent RI-IST program updates (e.g., following a major plant modification or following a significant equipment performance problem).

Licensees may wish to coordinate these reviews with other related activities, such as periodic PRA updates, industry operating experience programs, the Maintenance Rule program, and other risk-informed program initiatives.

The test strategy for RI-IST components should be periodically assessed to reflect changes in plant configuration, component performance, test results, and industry experience.

4. Element 4: Submit Proposed Change

The NRC staff's review of the RI-IST program is to ensure that the analyses conducted by the licensee were sufficient to conclude that the key principles of risk-informed regulation are met. To facilitate the staff's review, documentation of the evaluation process and findings are should be maintained. Additionally, the information submitted should include a description of the process used by the licensee to ensure its adequacy and specific information to support the licensee's assertion that the RI-IST program would provide an acceptable level of quality and safety.

4.1 Licensee's Risk-Informed Inservice Testing Submittal Documentation

The licensee's RI-IST alternative request should contain the following documentation:

a. a request to implement an RI-IST program as an authorized alternative to the current NRC-approved OM Code pursuant to 10 CFR 50.55a(z)(1) for the current 120-month IST program interval;

b. a description of the proposed RI-IST program (Regulatory Position C.1.1);

c. a summary of key technical and administrative aspects of the overall RI-IST program that includes the following:

(1) a description of the process used to identify candidates for reduced and enhanced IST requirements, including a description of the categorization of components using the PRA and the associated sensitivity studies (Regulatory Position C.2.2.3.2);

(2) a description of 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 (Regulatory Position C.2.2.3.1);

(3) a description of how the impact of the change is modeled in the IST components (including a quantitative or qualitative treatment of component degradation) and a description of the impact of the change on plant risk in terms of CDF and LERF and how this impact compares with the decision guidelines (Regulatory Position C.2.2.3.3);

(4) a description of how the key principles of risk-informed decisionmaking were (and will continue to be) met (Regulatory Positions C.2.2 and C.2.3);

(5) a description of the integrated decisionmaking process used to help define the RI-IST program, including any decision criteria used (Regulatory Position C.2.3);

(6) a general implementation approach or plan (Regulatory Positions C.3.1 and C.3.2);

(7) a description of the testing and monitoring proposed for each component group (Regulatory Position C.3.3);

(8) a description of the RI-IST corrective action plan (Regulatory Position C.3.4); and

(9) a description of the RI-IST program periodic reassessment plan (Regulatory Position C.3.5).

d. a summary of any previously approved relief requests for components categorized as HSSC along with any exemption requests, technical specification changes, and relief requests needed to implement the proposed RI-IST program (Regulatory Position C.2.1.2);

e. an assessment of the appropriateness of any other relevant previously approved relief or alternative requests;

f. a summary of those components categorized as HSSC that are not in the licensee’s current IST program and their proposed testing (Regulatory Positions C.1.2, C.2.2.2, C.2.2.3.2, C.3.1, and C.3.2);

g. whether provisions of 10 CFR Part 50, Appendix B, apply to the RI-IST program and its updates—the licensee would be expected to control the RI-IST program and its updates in a manner commensurate with its impact on the facility’s design and licensing basis and in accordance with all applicable regulations and its quality assurance program description;

h. a discussion of measures used to ensure the PRA is acceptable for the application PRA, such as a report of a peer review augmented by a discussion of the appropriateness of the PRA model for supporting a risk assessment of the licensing-basis change being considered—the submittal should address any analysis limitations that are expected to affect the conclusion on the acceptability of the proposed change; and

i. the licensee's resolution of the findings of the peer review that have not been closed by an NRC-accepted process (see Regulatory Position C.4.2 of RG 1.200 for additional guidance).

4.2 Archival Documentation

The licensee should maintain archival documentation, as part of its quality assurance program, so that it is available for on-site inspection and examination. The following documentation of the analyses conducted to support changes to a plant's licensing basis should be maintained as lifetime quality records, in accordance with RG 1.33, "Quality Assurance Program Requirements (Operation)" (Ref. 26):

a. the overall IST program plan,

b. administrative procedures related to the RI-IST program,

c. component or system design-basis documentation,

d. piping and instrument diagrams for systems that contain components in the RI-IST program,

e. PRA and supporting documentation (Regulatory Position C.2.2.3),

f. categorization results, including the RI-IST process summary sheet for each component or group of components (Regulatory Position C.2.2.3.2),

g. integrated decisionmaking process procedures, including expert panel meeting minutes (if applicable) (Regulatory Position C.2.3),

h. detailed implementation plans and schedules (Regulatory Position C.3.2),

i. completed test procedures and any supplemental test data related to RI-IST (Regulatory Position 3.3),

j. corrective action procedures (Regulatory Position C.3.4),

k. plant-specific performance data (e.g., machinery history) for components in the RI-IST program (Regulatory Positions C.2.2.3.3 and C.3.1), and

l. a description of individual changes made to the RI-IST program after implementation (Regulatory Position C.1.3).

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, "Backfitting," and as described in NRC Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests" (Ref. 27), nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." 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 process in Management Directive 8.4.

REFERENCES

1. U.S. Nuclear Regulatory Commission (NRC), Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Washington DC. [2]

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

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

4. American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants, New York, NY. [3]

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

6. NRC, Standard Review Plan, Section 3.9.7, "Risk-Informed Inservice Testing," Revision 0, August 1998, Washington. DC.

7. NRC, NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants—Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants—Final Report," Rev. 2, October 2013, Washington, DC.

8. NRC, NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking—Final Report," Rev. 1, March 2017, Washington, DC.

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

10. ASME/American Nuclear Society, RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," New York, NY.

11. 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.

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[2] 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.

[3] Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Two Park Avenue, New York, New York 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, SECY-95-280, “Framework for Applying Probabilistic Risk Analysis in Reactor Regulation,” November 27, 1995.

13. NRC, “Nuclear Regulatory Commission International Policy Statement,” Federal Register, Vol. 79, No. 132, July 10, 2014, pp. 39415-39418.

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

15. International Atomic Energy Agency (IAEA), Safety Standard SSG-2, “Deterministic Safety Analysis for Nuclear Power Plants,” Rev. 1, 2019, Vienna, Austria. [4]

16. IAEA, Safety Guide SSG-3, “Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants Specific Safety Guide,” Vienna, Austria, 2010.

17. IAEA, Fundamental Safety Principle, SSR-2/1, “Safety of Nuclear Power Plants: Design,” Rev. 1, 2016, Vienna, Austria.

18. NRC, RG 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code.” Washington, DC.

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

20. NRC, NUREG/CR-6141, BNL-NUREG-52398, “Handbook of Methods for Risk-Based Analyses of Technical Specifications,” December 1994, Washington, DC. ASME, “Risk-Based Inservice Testing: Development of Guidelines,” Research Report, CRDT-Vol. 40-2, Volume 2, New York, NY, 1996.

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22. Electric Power Research Institute, TR-105396, “PSA Applications Guide,” Palo Alto, CA, August 1995. [5]

23. NEI, “Industry Guidelines for Risk-Based Inservice Testing, Draft (Revision B),” March 19, 1996.

24. NRC, Generic Letter 89-04, “Guidance on Developing Acceptable Inservice Testing Programs,” Washington, DC, April 3, 1989.

[4] 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-400 Vienna, Austria.

[5] Copies of Electric Power Research Institute (EPRI) standards and reports may be purchased from EPRI, 3420 Hillview Ave., Palo Alto, CA 94304; telephone (800) 313-3774; fax (925) 609-1310.

25. ASME, Code Case OMN-1, “Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor Operated Valve Assemblies in LWR Power Plants,” OM Code-1995, Subsection ISTC, New York, NY.

26. NRC, RG 1.33, “Quality Assurance Program Requirements (Operation),” Washington, DC.

27. NRC, Management Directive 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests,” Washington, DC.

APPENDIX A

DETAILED GUIDANCE FOR INTEGRATED DECISIONMAKING

A-1. Introduction

The increased use of probabilistic risk assessment (PRA) in nuclear plant activities, such as in risk-informed (RI)-inservice testing (IST) programs, will require a balanced use of the probabilistic information with the more deterministic engineering information. Some structured process for considering both types of information and making decisions will be needed that will allow improvements to be made in plant effectiveness while maintaining adequate safety levels. This will be particularly important during initial program implementation and also for the subsequent early phases of the program. In some instances, the physical data from the PRA and from the deterministic evaluations may be insufficient to make a clear-cut decision. At times, these two forms of information may even seem to conflict. In such cases, the licensee should assemble the appropriate skilled staff to consider all the available information in its various forms and to supplement this information with engineering judgment to determine the best course of action. The participants involved in this important role have generally been referred to in various industry documents as an “expert panel.”

A-2. Basic Categories of Information to be Considered

Risk-importance measures may be used together with other available information to determine the relative risk ranking (and thus categorization) of the components included in the evaluation. Results from all these sources are then reviewed before making final decisions about where to focus IST resources.

Although the risk ranking of components can be used primarily as the basis for prioritizing IST at a plant, additional considerations need to be addressed (e.g., defense-in-depth, common cause, and the single-failure criterion), which may be more constraining than the risk-based criteria in some cases. Consideration should be given to these issues and component performance experience before determining the IST requirements for the various components.

IST experience should contribute to an understanding of the important technical bases underlying the existing testing program before it is changed. The critical safety aspects of these bases should not be violated inadvertently in changing over to an RI-IST program, and important plant experience gained through the deterministic IST should be considered during the change.

The plant-specific PRA information should include important perspectives with respect to the limitations of PRA modeling and analysis of systems, some of which may not be explicitly addressed within the PRA analysis. An understanding should also be provided about how the proposed changes in pump and valve testing could affect PRA estimates of plant risk.

Plant safety experience should provide insights associated with the deterministic analyses (refer to Chapter 15 of the plant final safety analysis report) and any effect that proposed changes in testing might have on the deterministic perspective of overall plant safety.

Plant operational input should supplement the insights of plant safety with additional information on the operational importance of components under normal, abnormal, and emergency conditions. There should also be input on operating history, system interfaces, and industry operating experience to supplement information from the IST.

Maintenance considerations should provide perspectives on equipment operating history, work practices, and the implementation of the Maintenance Rule (Title 10 of the *Code of Federal Regulations* (10 CFR) 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants").

Systems design considerations should include the potential effect of different design configurations (e.g., piping, valves, and pumps) on planning for an RI-IST, particularly if contemplating future plant modifications or if systems are temporarily taken out of service for maintenance or replacement or repair.

A-3. Specific Areas to be Evaluated

This section addresses some technical and administrative issues that are particularly important for RI-IST programs:

a. The RI-IST application should confirm that proper attention was given to component classifications in systems identified in emergency operating procedures (and other systems) depended upon for operator recovery actions, primary fission product barriers excluded from the PRA because of their inherent reliability (such as the reactor pressure vessel), passive items not modeled in the PRA (such as piping, cable, supports, building or compartment structures such as the spent fuel pool), and systems relied upon to mitigate the effects of external events in cases where the PRA considered only internal events.

b. Failure modes modeled by the PRA may not be all-inclusive. Consideration should be given to the failure modes modeled and the potential for the introduction of new failure modes related to the IST application. For example, if valve mispositioning has been assumed to be a low-probability event because of independent verification, and therefore, is not included in the PRA assumptions, any changes to such independent verifications should be evaluated for potential impact on the PRA results.

c. The resource information base should include other qualitative or quantitative analyses that shed light on the relative safety importance of components, such as failure modes and effects analyses, shutdown risk, seismic risk, and fire protection.

d. Attention should be given to the fact that component performance can be degraded from the effects of aging or harsh environments; this issue will need to be addressed and documented.

e. The engineering evaluation should include the choice of new test frequencies, the identification of compensatory measures for potentially important components, and the choice of test strategies for both high safety-significant components and low safety-significant components.

f. IST tests should be evaluated before choosing the test methods to be used for the high safety-significant components and low safety-significant components, depending on the components' expected failure modes, service conditions, and other factors.

g. Because of the importance of maintaining defense-in-depth, particular attention should be given to identifying any containment systems involving IST components.

h. The RI-IST application should include stepwise program implementation, as discussed in Regulatory Position C.3.2, as part of the licensee's integrated decisionmaking process.

i. The RI-IST application should include the licensee's performance-monitoring approach, as discussed in Regulatory Position C.3.3, as part of the licensee's decisionmaking process.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 19, "Severe Accidents," provides additional issues of a more general nature that may arise in expert panel deliberations.