U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

October 1997
Division 1
Draft DG-1063

DRAFT REGULATORY GUIDE

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DRAFT REGULATORY GUIDE DG-1063

AN APPROACH FOR PLANT-SPECIFIC, RISK-INFORMED DECISIONMAKING: INSERVICE INSPECTION OF PIPING

This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received complete staff review and does not represent an official NRC staff position.

Public comments are being solicited on the draft guide (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by January 9, 1998.

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1. INTRODUCTION

The NRC's Commission Policy Statement on probabilistic risk analysis (PRA) encourages use of risk-informed analysis techniques to improve safety during decisionmaking and improve regulatory efficiency. A number of NRC staff and industry activities are presently under development in response to the Commission's policy statement. One activity now under way is the use of PRA insights to support modifications to a nuclear plant's current licensing basis (CLB). A number of specific CLB changes are now under staff review.

This regulatory guide is being developed to describe acceptable approaches for incorporating insights from probabilistic risk assessment techniques to inservice inspection (ISI) programs for pipes. Given the recent initiatives by the American Society of Mechanical Engineers, it is anticipated that licensees will request changes to their CLB for a nuclear power facility that incorporates risk insights in their ISI programs (known as risk-informed inservice inspection programs -- RI-ISI). As always, licensees can and should identify how the chosen approach, methods, data, and criteria are appropriate for the decisions they need to make.

1.1 Background

Traditionally, regulation of the design and operation of commercial nuclear power plants has been based on conventional engineering criteria (meaning criteria developed using traditional engineering analysis methods without applying probabilistic methods as in PRA). These engineering criteria continue to successfully assure that plants can be placed in a safe condition following a number of postulated design basis accidents. The traditional engineering criteria also provided the basis for identifying what plant structures, systems, components (SSCs), and activities are important to safety. Regulation of these "safety-related" SSCs and activities is controlled through regulatory requirements.

During recent years, both the NRC and the nuclear industry have recognized that PRA has evolved to the point that it can be used increasingly as a tool in regulatory decisionmaking. In August 1995, the NRC adopted a policy statement regarding the expanded NRC use of PRA (Ref. 1). In part, the policy statement states that:

- The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional philosophy of defense-in-depth.

- PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal of additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

- PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

- The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

In its approval of the policy statement, the Commission articulated its expectation that implementation of the policy statement will improve the regulatory process in three areas: foremost, through safety decisionmaking enhanced by the use of PRA insights; through more efficient use of agency resources; and through a reduction in unnecessary burdens on licensees.

In parallel with the publication of the policy statement, the staff developed a regulatory framework that incorporates risk insights. That framework was articulated in a November 27, 1995 paper (SECY-95-280) to the Commission (Ref. 2). This regulatory guide, which addresses ISI programs of welds in pipes at nuclear power plants, implements, in part, the Commission's policy statement and the staff's framework for incorporating risk insights into the regulation of nuclear power plants.

While the conventional regulatory framework, based on traditional engineering criteria, has and continues to serve its purpose in assuring the protection of public health and safety, the current information base contains insights gained from over 2000 reactor-years of plant operating experience and extensive research in the areas of material sciences, aging phenomena, and inspection techniques. This information, combined with modern risk assessment techniques and associated data can be used to develop a more effective approach to ISI programs of pipes.

The current ISI requirements for piping components are found in 10 CFR 50.55a and the General Design Criteria listed in Appendix A to 10 CFR Part 50 (Ref. 3). Requirements for piping are scattered throughout the General Design Criteria, such as in Section I, *Overall Requirements*, Section II, *Protection by Multiple Fission Product Barriers,* Section III, *Protection and Reactivity Control Systems,* Section IV, *Fluid Systems*, etc.

10 CFR 50.55a references Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) (Ref. 4). Section 50.55a addresses the codes and standards for design, fabrication, erection, construction, testing, and inspection of piping systems. The objective of the ISI program is to identify conditions, such as flaw indications, that are precursors to pipe leaks and ruptures, thereby meeting, in part, the requirements set in the General Design Criteria and 10 CFR 50.55a. ISI programs are intended to address all piping locations that are subject to degradations. Many of the inspections are focused on critical locations, such as welds, if such locations have the highest likelihood for failure. However, experience over many years has shown that while the location of examination using the current Section XI criteria have been effective for Category

B-J[1] welds - Class 1 piping, many of the actual reported problems (Ref. 5) were in other locations. The majority of flaws found in Category B-J piping welds have been caused by factors outside the scope of the current selection criteria. Some of the inspected locations that are not exposed to active degradation mechanisms have led to unnecessary radiation exposure to personnel implementing the inspections. Incorporating risk insights into the programs can have the potential to focus on the more important locations for inspections and reduce personnel exposure while at the same time maintaining or improving public health and safety.

As a result of the above insights, more efficient and technically sound means for selecting and scheduling ISIs of piping are under development by the ASME [(Ref. 6) and (Ref. 7)].

This regulatory guidance document builds upon the knowledge base documented in NUREG/CR-6181, Rev.1 (Ref. 8), and it reflects the experience gained from the ASME initiatives (pilot plant activities). When categorizing pipe segments in terms of their contribution to risk, it is the responsibility of a licensee to justify that the categorization of pipe segments and the resulting inspection programs provide a change in core damage frequency (CDF) that is consistent with the guidelines addressed in Draft Regulatory Guide DG-1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis" (Ref. 9). This draft regulatory guide is being developed to provide guidance on how to incorporate risk insights in an inservice inspection program, provide guidance on developing methods that identify locations where both increases and decreases in ISI inspections are needed to meet the requirements of 10 CFR 50.55a(a)(3)(i), and address performance objectives.

1.2 Purpose of the Guide

Changes to many of the activities and design characteristics in a nuclear power plant's CLB[2] require NRC review and approval. The current inservice inspection programs are performed in compliance with the requirements of 10 CFR 50.55a and with Section XI of the ASME Boiler and Pressure Vessel Code, which are part of the plant's CLB. This regulatory guide describes acceptable alternative approaches to the existing Section XI requirements for ISI programs. **Its use by licensees is voluntary**. This alternative approach provides an acceptable level of quality and safety {per 10 CFR 50.55a(a)(3)(i)} by incorporating insights from probabilistic risk analysis calculations. Licensees proposing to apply risk-informed inservice inspection programs will be required to amend their final safety analysis report (FSAR, Sections 5.3.4 and 6.6) accordingly.

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[1]Category B-J welds are pressure retaining welds in piping.

[2]This regulatory guide adopts the 10 CFR Part 54 definition of current licensing basis. That is, "Current Licensing Basis (CLB) is the set of NRC requirements applicable to a specific plant and a licensee's written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the licensee) that are docketed and in effect. The CLB includes the NRC regulations contained in 10 CFR Parts 2, 19, 20, 21, 26, 30, 40, 51, 54, 55, 70, 72, 73, 100 and appendices thereto; orders; license conditions; exemptions; and technical specifications. It also includes the plant-specific design-basis information defined in 10 CFR 50.2 as documented in the most recent final safety analysis report (updated FSAR) as required by 10 CFR 50.71 and the licensee's commitments remaining in effect that were made in docketed licensing correspondence such as licensee responses to NRC bulletins, generic letters, and enforcement actions, as well as licensee commitments documented in NRC safety evaluations or licensee event reports."

This regulatory guide addresses acceptable approaches that apply risked-informed (RI) methods to develop, monitor, and update more efficient ISI programs for pipes at a nuclear power facility. This guidance does not preclude other approaches for incorporating risk insights into the ISI programs. Licensees may propose alternative approaches for NRC consideration. It is intended that the approaches presented in this guide be regarded as examples of acceptable practices and that licensees should have some degree of flexibility in satisfying the regulatory needs on the basis of their accumulated plant experience and knowledge. This document addresses risk-informed approaches that are consistent with the basic elements identified in DG-1061 (Ref. 9) to inservice inspection programs. In addition, this document provides guidance on:

- acceptable methods for estimating leak, disabling leak, and rupture probabilities for pipe segments,
- identifying structural elements for which inservice inspection can be modified (reduced or increased) based on risk insights, defense in depth, as low as reasonably achievable (ALARA) principles for radiation exposure to personnel, etc.,
- determining the risk impact of changes to inservice inspection programs,
- capturing deterministic considerations in the revised inservice inspection program, and
- developing an inspection program that monitors the performance of the pipe elements that are consistent with the conclusions from the PRA.

The NRC staff will initiate rulemaking as necessary to permit licensees to implement RI-ISI programs, consistent with this regulatory guide and the accompanying Standard Review Plan (SRP) chapter, without having to get NRC approval of an alternative to the ASME Code requirements pursuant to 10 CFR 50.55a(a)(3). Until the completion of such rulemaking, the staff anticipates the need to review and approve each licensee's RI-ISI program as an alternative to the current Code-required ISI program, prior to implementation. As such, the licensee's RI-ISI program will be enforceable under 10 CFR 50.55a.

1.3 Scope of the RI-ISI Program

This regulatory guide only addresses changes to the ISI programs for inspection of pipes. In the majority of the cases, pipe welds are the point of interest in the inspection program, although within this regulatory guide, references to "welds" are intended to address inspections in general of critical structural locations including the base metal. On the average, pipe welds are anticipated to have approximately forty times the likelihood of experiencing a leak prior to the base pipe structure. Exceptions to this rule of thumb can occur when an active degradation mechanism is present, such as flow-assisted-corrosion (e.g., erosion-corrosion). The risk implication of each pipe segment is determined by the safety significance of a pressure boundary failure of the pipe at that location, augmented by the failure likelihood of the pipe segment. When the risk implications or degradation mechanisms along a pipe vary, the pipe is subdivided into segments, as discussed in Chapter 4.

To adequately reflect risk implications, the scope of systems, structures and components (SSCs) covered by this regulatory guide[3] includes:

- All Class 1, 2, and 3[4] pipes within the current ASME Section XI programs, and

- All pipes whose failure would compromise

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

  - Non-safety-related structures, systems, or components:

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

    - Whose failure could prevent safety-related structures, systems, or components from fulfilling their safety-related function; or

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

To ensure that the proposed RI-ISI program will provide an acceptable level of quality and safety, the licensee should use the PRA to identify the appropriate scope of pipe segments to be included in the program. This will include all pipes within the scope of the current ISI program. In addition, licensees implementing the risk-informed process may identify pipe segments categorized as high-safety-significant (HSS), which are not currently subject to the traditional Code requirements or to a level of regulation which is commensurate with their risk significance. PRA systematically takes credit for systems with non-Code piping that provide support, act as alternatives, and act as backups to those systems with piping that are within the scope of the current Section XI Code. To maintain the validity of the PRA as it is used to categorize pipe segments and to evaluate the effects of the proposed RI-ISI program on plant risk, all HSS pipe segments should be included in a licensee's RI-ISI proposal. Specifically, the licensee's RI-ISI program scope should include those ASME Code Class 1, 2, &3 and non-Code systems that the licensee's categorized as HSS.

The PRA should also be used to evaluate RI-ISI program inspection requirements as practicable. Consequently, the licensee should examine the inspection strategies for all welds in the final proposed ISI program, including those inspections in the current Section XI

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[3] It is anticipated that this regulatory guidance document will, at some future date, be consistent with the ASME's ongoing programs to incorporate risk-informed insights into the ASME Section XI programs.

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

program. The inspection strategy most capable of detecting the effects of the specific degradation mechanism to which each weld is exposed should be identified and selected.

1.4 Organization and Content

This regulatory guide is structured to follow the general four-element process for risk-informed applications discussed in Draft Regulatory Guide DG-1061. *Chapter 2* summarizes the four-element process developed by the NRC staff (referred to as, **staff**) to evaluate proposed CLB changes as it applies to the development of a risk-informed ISI program. *Chapter 3* discusses an acceptable approach for defining the proposed changes to an ISI program. *Chapter 4* addresses, in general, the traditional and probabilistic engineering evaluations performed to support risk-informed ISI programs and presents the risk acceptance goals for determining the acceptability of the proposed change.

Chapter 5 presents one acceptable approach for implementing, monitoring, and corrective actions for RI-ISI programs. The documentation the NRC will use to render its safety decision is discussed in *Chapter 6*. Detailed discussions of issues and/or acceptable approaches associated with the engineering evaluations needed to support an RI-ISI program are provided in *Appendices 1* through *5*. The existing ASME Section-XI traditional approach is highlighted in *Appendix 6*.

1.5 Relationship to Other Guidance Documents

As stated in Section 1.2, this regulatory guide discusses acceptable approaches to implement risk insights into an ISI program and directs the reader to Draft Regulatory Guide DG-1061 for general guidance, where appropriate.

Draft Regulatory Guide DG-1061 describes a general approach to risk-informed regulatory decisionmaking and includes discussions on specific topics common to all risk-informed regulatory applications. Topics addressed include:

- PRA quality[5] - data, assumptions, methods,
- Scope - internal and/or external event initiators, at-power and/or shutdown modes of operation, consideration of Level 1, 2, and 3[6] analyses requirements, etc.,
- Risk metrics - core damage frequency, LERF and importance measures,
- Sensitivity and uncertainty analyses, and
- Process for ensuring quality - relationship to 10 CFR Appendix B.

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[5]Draft NUREG-1602, "Use of PRA in Risk-Informed Applications," provides technical details that support Draft Regulatory Guide DG-1061 (Ref. 21).

[6]Level 1 - accident sequence analysis, Level 2 - accident progression and source term analysis, and Level 3 - consequence analysis

Regulatory guides that contain ASME Code Cases for inservice inspection programs and that are based on traditional engineering criteria include (Ref. 10), (Ref. 11), and (Ref. 12). For references to other risk-informed applications, the reader is directed to regulatory guides pertaining to inservice testing (IST) (Ref. 13), graded quality assurance (GQA) (Ref. 14), and technical specifications (Tech Specs) (Ref. 15). SRP sections associated with each of the risk-informed regulatory guides are addressed in (Ref. 16), (Ref. 17), (Ref. 18), (Ref. 19), and (Ref. 20).

Regulatory guides are issued to describe to the public methods that are acceptable to the NRC staff for implementing specific parts of the NRC's regulations, to explain techniques used by the staff in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations, and compliance with regulatory guides is not required. Regulatory guides are issued in draft form for public comment to involve the public in developing the regulatory positions. Draft regulatory guides have not received complete staff review; they therefore do not represent official NRC staff positions.

The information collections contained in this draft regulatory guide are covered by the requirements of 10 CFR Part 50, which were approved by the Office of Management and Budget, approval number 3150-0011. The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

1.6 Abbreviations/Definitions

AEC	Atomic Energy Commission
ALARA	As Low as Reasonable Achievable
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CDF	Core Damage Frequency
CLB	Current Licensing Basis
ECC/AM	Emergency Core Cooling and Accident Mitigation
ECCS	Emergency Core Cooling System(s)
FMEA	Failure Modes and Effects Analysis
FSAR	Final Safety Analysis Report
Expert Elicitation	This refers to experts in a specific field, normally outside the level of expertise found at the plant. The expert elicitation is used to estimate the failure probability and the associated uncertainties of the material in question under specified degradation mechanisms. For example, if a fracture mechanics Code is not qualified to calculate the failure probability of plastic pipes, then experts in plastic pipes and their failure may be used to estimate the failure probabilities.
Expert Panel	Normally refers to plant personnel experienced in inservice inspection programs and other related activities/disciplines that impact the decision under consideration.
FV	Fussell-Vesely Importance Measure
GQA	Graded Quality Assurance
HSS	High-Safety-Significance

HSSC	High-Safety-Significant Component
IGSCC	Intergranular Stress Corrosion Cracking
Importance Measures	Used in PRA to rank systems or components in terms of risk significance
IPE	Individual Plant Examination
ISI	Inservice Inspection
IST	Inservice Testing
LERF	Large Early Release Frequency
LOCAs	Loss-of-Coolant Accident
LSSC	Low-Safety-Significant Component
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NPAR	Nuclear Plant Aging Research
NRC	Nuclear Regulatory Commission
NUMARC	Nuclear Management and Resources Council
PDI	Performance Demonstration Initiative
POD	Probability of Detection
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
RAW	Risk Achievement Worth
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RI-ISI	Risk-Informed Inservice Inspection
RWST	Refueling Water Storage Tank
staff	Refers to NRC Employees
Sensitivity Studies	Varying parameters to assess impact due to uncertainties
SER	Safety Evaluation Report
SRP	Standard Review Plan
SRRA	Structural Reliability/Risk Assessment (refers to fracture mechanics analysis)
SSCs	Structures, Systems, Components
Tech Specs	Technical Specifications

2. PROCESS OVERVIEW

For the licensee who elects to incorporate risk insights into its inservice inspection programs, it is anticipated that the licensee will build upon its existing probabilistic risk analysis (PRA) activities, beginning with the individual plant examination programs (IPE). Figure 2.1 illustrates the five key principles involved in the integrated decisionmaking process which is described in detail in Draft Regulatory Guide DG-1061. In addition, Draft Regulatory Guide DG-1061 describes a four-element process for evaluating proposed risk-informed changes to the CLB as illustrated in Figure 2.2.

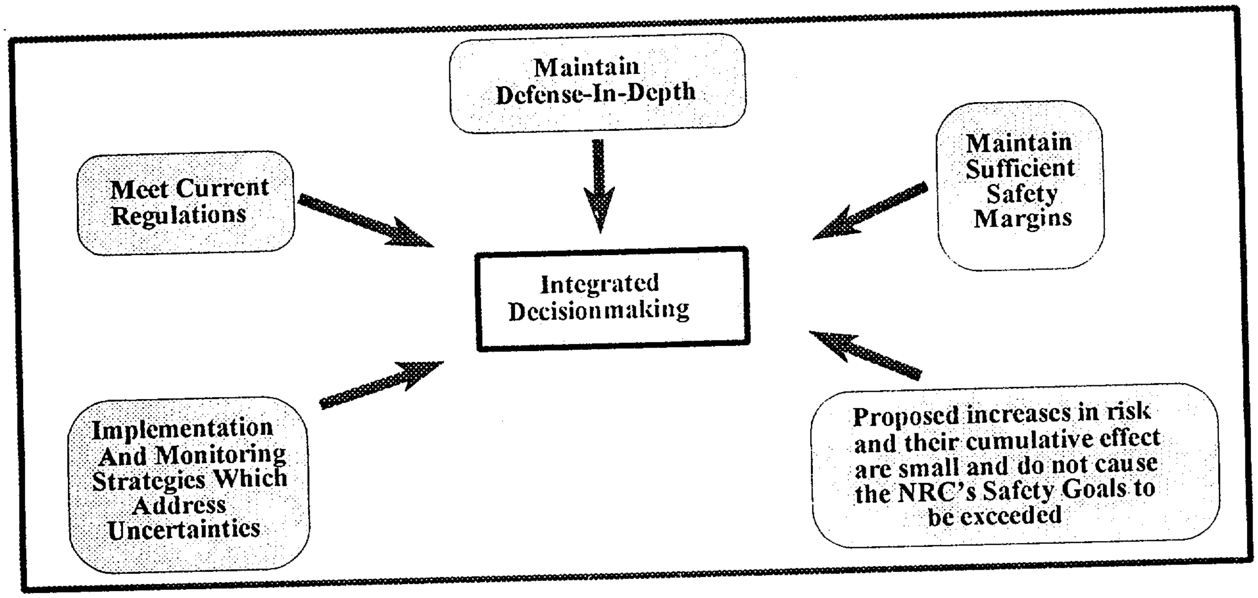


Figure 2.1 Principles of Risk-Informed Regulation.

The key principles and the location in this guide where each is addressed for RI-ISI programs are as follows:

1. *The proposed change meets the current regulations.* [This applies unless the proposed change is explicitly related to a requested exemption or rule change.] *(Section 3.1)*

2. *Defense-in-depth is maintained. (Section 4.1.1)*

3. *Sufficient safety margins are maintained. (Section 4.1.2)*

4. *Proposed increases in risk and their cumulative effect are small and do not cause the NRC's Safety Goals to be exceeded. (Sections 4.2 and 4.4)*

5. *Performance-based implementation and monitoring strategies are used that address uncertainties in analysis models and data and provide for timely feedback and corrective action. (Chapter 5)*

The individual principles are discussed in detail in Draft Regulatory Guide DG-1061, and are not repeated here. However, an overview of the four-element process is provided and specific issues that arise for risk-informed ISI are discussed.

The four-element process described below begins with a set of proposed changes to ISI. The process for developing the initial proposal for changes is left to the licensee, but can benefit from an examination of PRA information, including distinguishing the affected pipe segments through a categorization process based on various importance measures and engineering insights.

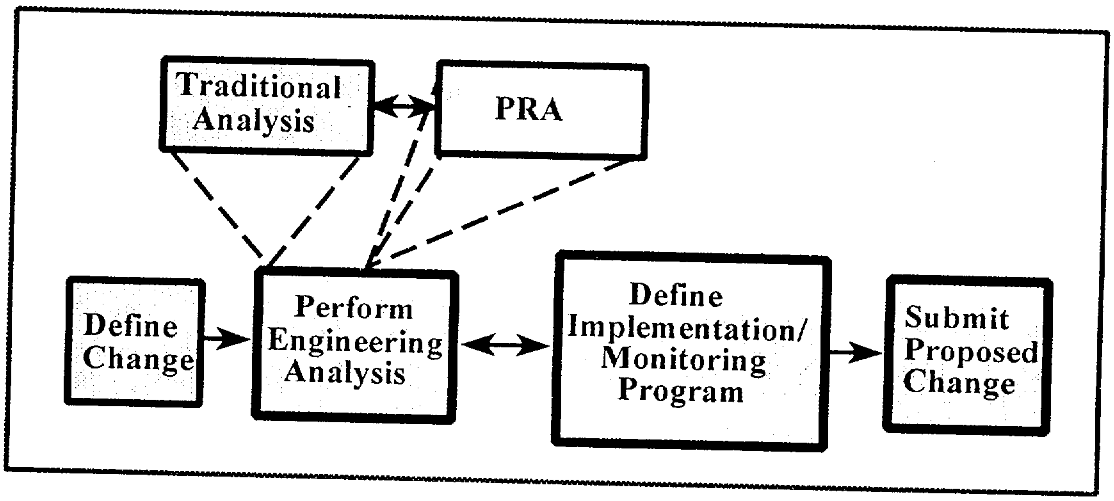


Figure 2.2 Principle elements of risk-informed, plant-specific decisionmaking.

Element 1: Define the proposed change

In this element the licensee identifies the pipes and welds that are affected by the change in inspection practices. This would include components currently in the ISI program and additional pipes categorized as high-safety-significant (HSS). Specific revisions to the inspection programs, schedules, and techniques should be documented. Plant systems and functions that rely on the affected pipes should be identified.

The licensee should assess whether an adequate PRA is available for risk-informed evaluations (see (Ref. 9) and (Ref. 21)) and how the existing regulations—the plant's current licensing basis—may be impacted by the proposed change. Finally, plant-specific experience with inspection program results should be examined and characterized relative to the effectiveness of past inspections and the types of flaws that have been observed. Chapter 3 provides a more detailed description of Element 1.

Element 2: Perform engineering analysis

In element 2, the proposed changes are evaluated with regard to:

- Maintaining adequate defense-in-depth,

- Maintaining adequate safety margins,

- The risk impact of the changes, including the treatment of uncertainties. The principle that the proposed increase in risk and their cumulative effect are small and do not cause the NRC Safety Goals to be exceeded is also addressed.

- Comparison of the PRA results with the acceptance guidelines in Draft Regulatory Guide DG-106I.

- An integrated decision-making process that considers insights from both the engineering and probabilistic risk analyses.

Traditional engineering and PRA methods are used in this evaluation. The results of the complementary traditional and PRA methods are considered together in an integrated decisionmaking process. During the integration of all of the available information, it is expected that many issues will need to be resolved through the use of a well-reasoned judgment process often involving a combination of different engineering skills. This activity has typically been referred to in industry documents as being performed by an "expert panel." As discussed in this document, this important process is the licensee's responsibility and may be accomplished by means other than a formal panel. It is the licensee's responsibility to ensure that any submittal to the NRC is accurate and complete. In carrying out this process, the licensee will need to make a number of decisions based on the best available information. Some of this information will be derived from traditional engineering practices and some will be probabilistic in nature, resulting from PRA studies. It may be that certain issues discussed in this guide are best evaluated through the use of traditional engineering approaches, but for other issues, PRA may have advantages. It is the licensee's responsibility to ensure that its RI-ISI program is developed using a well-reasoned and integrated decision process that considers both forms of input information (traditional engineering and probabilistic) including those cases in which the choice of direction is not obvious. Examples of this latter situation are when there is insufficient information to make a clear decision or if the PRA results appear to disagree with the traditional engineering data. Depending on the issues involved, technical or otherwise, this important decision-making process may at times require the participation of special combinations of licensee experts (staff) and/or outside consultants. This integrated decisionmaking process is discussed further in Section 4.3.

More details concerning Element 2 are contained in Chapter 4.

Element 3: Develop Implementation, Performance-Monitoring, and Corrective Action Strategies

In this element, plans are formulated to monitor factors that reflect (piping) reliability commensurate with the pipe's safety significance. For example, operating and environmental conditions should be monitored for consistency with the assumptions in the PRA analysis. In addition, the results of the individual ISIs should be monitored to ensure that piping degradation is not beyond the assumptions of the PRA. In the event that pipe failures or unanticipated degradations occur in an RI-ISI program, guidance for evaluating the need for and the implementation of corrective actions should be included in the plans. Specific guidance for Element 3 is given in Chapter 5.

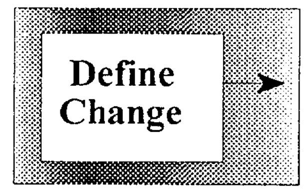
Element 4: Document evaluations and submit request for proposed change

The final element involves preparing the documentation to be included in the submittal and to be maintained by the licensee for later reference (i.e., archival) as needed. The submittal will be reviewed by the NRC following the guidelines set in the standard review plans (NUREG-0800) Chapter 19 and Section 3.9.8. Documentation requirements for RI-ISI programs are given in Chapter 6 of this regulatory guide.

3. ELEMENT 1: DEFINE THE PROPOSED CHANGES TO INSERVICE-INSPECTION PROGRAMS

3.1 Description of Proposed Changes

In this first element of the process, the proposed changes to the ISI program are defined. This involves describing the scope of ISI components that will be incorporated in the overall assessment and how their inspection would be changed. Also included in this element is an identification of supporting information, and a proposed plan for the licensee's interactions with the NRC throughout the implementation of the RI-ISI.



ELEMENT 1

A full description of the proposed change in the ISI program is prepared. This description would include:

(1) An identification of the aspects of the plant's CLB that would be affected by the proposed RI-ISI program.
(2) An identification of the specific revisions to existing inspection schedules, locations, and methods that would result from implementation of the proposed program.

(3) Any piping not presently covered in the plant's ISI program, but which are determined to be categorized as high-safety-significant (e.g., through PRA insights) should be identified and appropriately addressed. In addition, the particular systems that are affected by the proposed changes should be identified since this information is an aid in planning the supporting engineering analyses.

(4) An identification of the information that will be used to support the changes. This will include performance data, traditional engineering analyses and PRA information.

(5) A brief statement describing the way the proposed changes meet the objectives of the Commission's PRA Policy Statement.

3.2 Formal Interactions With The Nuclear Regulatory Commission

The licensee can make changes to its approved RI-ISI program under the following conditions:

1. Changes made to the NRC-approved RI-ISI program that could affect the process and results that were reviewed and approved by the NRC staff (including the change in plant risk associated with the implementation of the RI-ISI program) should be evaluated to ensure that the basis for the staff's prior approval has not been compromised. If there is a question regarding this issue, the licensee should seek NRC review and approval prior to implementation.

2. All changes should also be evaluated using the change mechanisms described in existing applicable regulations (e.g., 10 CFR 50.55a, 10 CFR 50.59) to determine if NRC review and approval is required prior to implementation.

For example:

- Changes to component groupings, inspection intervals, and inspection methods that do not involve a change to the overall RI-ISI approach where the overall RI-ISI approach was reviewed and approved by the NRC do not require specific (i.e., additional) review and approval prior to implementation provided that the effect of the changes on plant risk increase is insignificant.

- Component inspection method changes involving the implementation of an NRC endorsed ASME Code, NRC-endorsed Code Case, or published NRC guidance which were approved as part of the RI-ISI program do not require prior NRC approval.

- Inspection method changes that involve deviation from the NRC-endorsed Code requirements require NRC approval prior to implementation.

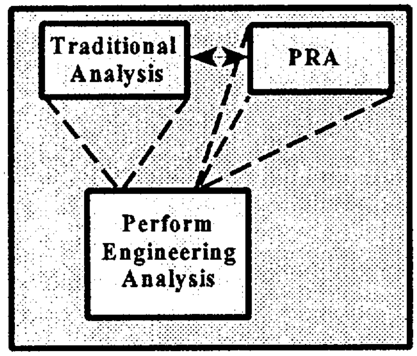
- Changes to the RI-ISI program that involve programmatic changes (e.g., changes to the plant probabilistic model assumptions, changes to the grouping criteria or figures of merit used to categorize components, and changes in the Acceptance Guidelines used for the licensee's integrated decision-making process) require NRC approval prior to implementation.

Piping inspection method changes will typically involve the implementation of an applicable ASME Code or Code case (as approved by the NRC) or published NRC guidance. Changes to the piping inspection methods, which nonetheless meet applicable Code requirements and/or NRC guidance do not require NRC approval. However, inspection method changes that involve deviation from the NRC approved Code requirements do require NRC approval prior to implementation.

The licensee will include in its submittal, a proposed process for determining when formal NRC review and approval are or are not necessary. As discussed, once this process is approved by the NRC, formal NRC review and approval are only needed when the process determines that such a review is necessary, or when changes to the process are requested.

4. ELEMENT 2: ENGINEERING ANALYSIS

This chapter summarizes the regulatory issues and engineering activities that a risk-informed inservice inspection program should consider. The discussions are divided into traditional and PRA analyses, as illustrated in Figure 2.2. Section 4.1 addresses the traditional engineering analysis, Section 4.2 addresses the PRA related analysis, Section 4.3 describes the integration of the traditional and PRA analyses, and Section 4.4 outlines the acceptance guidelines.



ELEMENT 2

The key principles of the engineering evaluations are to:

- Demonstrate that adequate defense-in-depth is maintained;
- Demonstrate that adequate safety margins are maintained;
- Demonstrate that the proposed ISI program changes do not result in unacceptable risk to the public and plant personnel, and are consistent with the decision metrics guidance identified in Draft Regulatory Guide DG-1061; and
- Support the integrated decisionmaking process.

The scope and quality of the engineering analyses[7] performed to justify the proposed changes to the ISI programs should be appropriate for the nature and scope of the change. The decision criteria associated with each key principle identified above are presented in the following subsections. Equivalent criteria can be proposed by the licensee if such criteria can be shown to meet the principles set forth in Section 2.1 of DG-1061. Germaine to the assessment of the impact of the proposed ISI change on plant risk, technical details on the use of risk importance measures are highlighted in draft NUREG-1602 (Reference 21) and in Appendix 2.

4.1 Traditional Analysis

This part of the evaluation is based on traditional engineering methods. Areas to be evaluated from this viewpoint include meeting the regulations, defense-in-depth attributes and safety margins. Probabilistic risk insights may be useful in the evaluation by providing information on relative importance of various SSCs.

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[7]Augmented inspection programs of pipes (e.g., NRC mandated programs) are also addressed in the engineering analysis performed by licensees when electing a risk-informed inspection program. The potential core damage contributions from failures of pipes that experience active degradation mechanisms may not be negligible. However, appropriate inspection programs with compensatory measures (e.g., replacement of pipes at appropriate intervals) could result in negligible contribution to core damage frequency.

4.1.1 Regulations

The engineering evaluation should assess whether the proposed changes in the ISI programs have compromised compliance with the regulations. The evaluation should consider the appropriate general design criteria, national standards, or other regulatory guidance. Specifically, the evaluation should consider:

- 10 CFR 50.55a,
- Appendix A to 10 CFR Part 50,
  - Section I - "Overall Requirements"
  - Section II - "Protection of Multiple Fission Product Barriers"
  - Section III - "Protection and Reactivity Control Systems"
  - Section IV - "Fluid Systems," etc.
- ASME Boiler and Pressure Vessel Code, Section XI,
- Regulatory Guide 1.147, and

4.1.2 Defense-in-Depth Evaluation

As stated in Draft Regulatory Guide DG-1061, the General Design Criteria and national standards are to be considered in the engineering evaluation. Defense in depth for ISI programs focuses on barriers (both preventive and mitigative) to core damage, containment failure, and population exposure.

The licensee should assess whether the proposed changes to the ISI program adversely impacts the CLB's conclusions on defense-in-depth. One acceptable set of guidelines for making that assessment are summarized below. Other equivalent decision criteria will also be considered.

Defense-in-depth is preserved when:

- a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved;

- over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided;

- system redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences of challenges to the system;

- defenses against potential common cause failures (**CCFs**) are preserved and the introduction of new CCF mechanisms are monitored for prevention;

- independence of barriers is not degraded; and

- defenses against human errors are preserved.

A PRA systematically assimilates all the above attributes of defense-in-depth into a coherent package. From this package, a detailed analysis can be performed to assess the impact of proposed modifications on those attributes. For example, the degradation of balance among prevention of core damage, prevention of containment failure, and consequence mitigation can lead to calculated CDF outliers.

4.1.3 Safety Margins

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

The engineering analyses should assess whether the impact of the proposed ISI changes are consistent with the principle that adequate safety margins are maintained. An acceptable set of guidelines for making that assessment are summarized below. Other equivalent decision criteria are acceptable.

Sufficient safety margins are maintained when:

- codes and standards (as given in Section 4.1.1) or alternatives approved for use by the NRC are met, and

- safety analysis acceptance criteria in the current licensing basis (e.g., updated FSAR, supporting analyses) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty.

Performance based inspection programs that monitor for degradations that can lead to leaks and are measured to acceptable target leak frequency goals, such as those identified in Section A2.7.3.3 of Appendix 2 (or alternative goals approved by the staff), can help provide confirmation of adequate safety margins. For example, if it can be demonstrated that reduction in inspections for a specific pipe segment will not lead to more leaks than the present ASME Section XI performance, then one can argue that the existing safety margins (due to inspections) are excessive and unnecessary. In the same sense, if the performance of a pipe segment exceeds the target leak frequency, then the safety margins are not sufficient and additional attention to that segment is needed.

4.1.4 Engineering Fracture Mechanics Evaluation

An important input to inservice inspection programs is the identification of structural mechanics parameters, possible degradation mechanisms, design limit considerations, operating practices and environment, and the development of a data base or analytic methods for predicting the reliability of piping systems. Design and operational stress/strain limits are assessed. This information is available to the licensee in its design information for its plant. The loading and resulting stresses/strains on the piping is needed as input to the fracture mechanics calculations that predict the failure probability of a pipe segment. Use of validated fracture mechanics computer programs, with appropriate input, is strongly recommended, because it facilitates the regulatory evaluation of a submittal. The method of applying computer simulation to calculate piping degradation has now achieved a level of maturity and validation that it can be applied in probabilistic risk applications. This topic is discussed in detail later in Appendix 1.

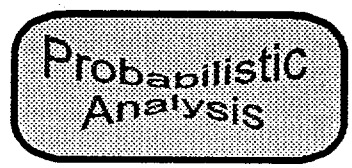
Where validated analytic computer programs are not available to predict the consequences for the degradation mechanisms or material in question, applicable data bases and expert elicitation programs can be applied to provide the necessary information.

4.1.5 Engineering Failure Modes & Effects Analysis

Sound engineering practices include validation of the parameters and consequences. An acceptable process that provides the risk insights to ISI programs includes detailed walkthrough of a nuclear power facility. Assessment of internal and external events, including resulting primary and secondary effects of pipe degradations (e.g., leaks and breaks), are important parameters for the risk-informed program. A detailed engineering failure modes and effects analysis (FMEA) provides an acceptable, disciplined, approach to the engineering analyses. Alternate methods should be submitted to the NRC for review and approval.

4.2 Probabilistic Risk Assessment

Using PRA to Assess the Change in Risk Associated with Changes to an ISI Program



The risk-informed application process is intended not only to support relaxation (number of inspections, inspection intervals and method), but also to identify areas where increased resources should be allocated to enhance safety. An acceptable RI-ISI process should, therefore, not focus exclusively on areas in which reduced inspection could be justified. This section addresses ISI-specific considerations in the PRA to support relaxation of inspections, enhancement of inspections, and validation of component operability.

The general methodology for using PRA in regulatory applications is discussed in Draft Regulatory Guide DG-1061, with reference to draft NUREG-1602, where technical details on scope, quality, and uncertainty issues are provided to support Draft Regulatory Guide DG-1061. General PRA issues specific to the development of a risk-informed ISI program are discussed below. Detailed discussions on an acceptable quantitative approach are provided in Appendix 2. Other approaches can be proposed and will be acceptable if they adequately address all issues discussed here and in Appendix 2.

For the results of the PRA to play a major and direct role in the ISI decision-making process, there is a need to ensure that the results are derived from quality analyses. Figure-4.1 identifies attributes of a quality ISI analysis.

Figure 4.1 Example Attributes for Risk-Informed ISI Programs

Attributes of a Quality ISI Risk-Informed Methodology

* Failure modes (e.g., small leak, disabling (large) leak, break) that can have either direct consequences (e.g., disable a system) or indirect consequences (water spray, pipe whip, etc.) are addressed.
* Full range of failure mechanisms (e.g., mechanical fatigue, thermal fatigue, stress corrosion cracking, flow assisted corrosion, etc.) that can contribute to component failure have been addressed.
* Evaluation of failure potential makes use of plant specific operating experience and industry data bases on failure occurrences.
* Categories for failure potential are related to well defined numerical ranges of failure frequencies or probabilities such that the assignment of the failure potentials to categories can be supported and/or benchmarked with failure experience data and with predictions based probabilistic structural mechanics models.
* Evaluation of failure potential include degradation mechanisms which are seldom or not yet experienced. Structural mechanics models may indicate that these mechanisms, although outside the scope of current operating experience and/or industry data bases, can contribute to the lower failure potential category(ies).
* Identification of pipe segments with particularly high failure consequences, so that inspection programs for these segments can be designed to detect degradation mechanisms which are either unexpected or more aggressive than expected.
* Identification of structural elements having the highest relative contributions to core damage risk, and include 100 percent of these top elements in the list of ISI locations.
* Identification of a population of structural elements which, as a group, contribute only a small fraction to the overall core damage risk associated with piping components. These stuctural elements can be subject to a reduced level of inspection.
* Evaluation of the impact or change in plant risk associated with implementation of the proposed RI-ISI programs.

The licensee is expected to use its judgment, drawing from the appropriate technical disciplines for the CLB change being considered, of the complexity and difficulty of the implications of the proposed CLB change to decide upon adequate engineering analyses to support the regulatory decisionmaking. Thus, 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. Application of qualitative simplification of risk-assessment may be found acceptable if benchmarked by quantitative methods. Any approach should develop performance objectives and means to achieve those objectives. That includes technically justified means to impose both increases and decreases in inspection requirements. The method needs to clearly illustrate the generality of the approach to strive for specified safety objectives.

4.2.1 Scope of Piping Segments

To adequately reflect risk implications, the scope of SSCs covered by this regulatory guide includes

- All Class 1, 2, and 3 pipes within the current ASME Section XI programs, and

- All pipes whose failure would compromise

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

    - Non-safety-related structures, systems, or components:

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

        - Whose failure could prevent safety-related structures, systems, or components from fulfilling their safety-related function; or

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

The final piping systems that are to be included in the scope of the RI-ISI program must be clearly defined. Similarly, piping systems not addressed by the RI-ISI programs must also be identified, documented and justified for exclusion. Table 4.1, adapted from (Ref. 7), provides an example list of systems included within the scope of an example risk-informed ISI program. Table 4.1 simply presents an example of information that should accompany a regulatory application. This regulatory guide recognizes that each plant choosing to submit changes to its ISI programs to incorporate risk insights will identify it own sets of systems that will differ from those listed in Table 4.1.

The basis for excluding a plant's piping systems from consideration for inspection should be clearly discussed in the context of the criteria outlined above. Any pipe in the plant can be selected for inservice inspection programs based on considerations outside the regulatory safety arena (e.g., pipes whose failure would have an inconsequential effect on safety, but could affect the economical operation of the plant). An example list of systems that may be considered for exclusion from consideration in a risk-informed ISI evaluation is provided in Table 4.2. Such a table should also accompany a regulatory submittal. All systems excluded from consideration must be justified.

Table 4.1 Example of Systems Identified as Falling under RI-ISI Programs for a Reference PWR (Adapted from Reference 7)

System Description	Basis
BDG - Steam Generator Blowdown	High Energy Line Break Concerns
CCE - Charging Pump Cooling	PRA[1]
CCI - Safety Injection Pump Cooling	PRA[1]
CCP - Reactor Plant Component Cooling	PRA & ASME Section XI
CHS - Chemical & Volume Control	PRA & ASME Section XI
CNM - Condensate	PRA[2]
DTM - Turbine Plant Miscellaneous Drains	ASME Section XI[3]
ECCS - Emergency Core Cooling	PRA & ASME Section XI
EGF - Emergency Diesel Fuel	PRA
FWA - Auxiliary Feedwater & Recirculation	PRA & ASME Section XI
FWS - Feedwater	PRA[2] & ASME Section XI
HVK - Control Bldg. Chilled Water	PRA
MSS - Main Steam	PRA & ASME Section XI
QSS - Quench Spray	PRA & ASME Section XI
RCS - Reactor Coolant	PRA & ASME Section XI
RHS - Residual Heat Removal	PRA & ASME Section XI
RSS - Containment Recirculation	PRA & ASME Section XI
SFC - Fuel Pool Cooling & Purification	PRA[5]
SIH - High Pressure Safety Injection	PRA & ASME Section XI
SIL - Low Pressure Safety Injection	PRA & ASME Section XI
SWP - Service Water	PRA & ASME Section XI

[1] Included in PRA boundary, but exempt by ASME Section XI pipe size.
[2] Modeled indirectly in PRA.
[3] Drain lines from MSS listed because of ASME Section XI.
[4] ECCS is a combination of piping segments which impact a number of systems - Charging, HPSI, LPSI, Quench Spray
[5] Not included in PRA internal events model, important to shutdown risk.

Table 4.2 Example of Risk-Informed Systems Excluded from Consideration in RI-ISI Programs for a Reference PWR

System ID	System Description	Resolution
DSM	Moisture Separator Drains & Vents	Determined to be non-risk significant*
DSR	Main Steam Separator Reheater Drains and Vents	Determined to be non-risk significant*
EGD	Emergency Diesel Fuel Exhaust & Comb. Air	Determined to be non-risk significant
ESS	Extraction Steam	Determined to be non-risk significant*
GMC	Stator Cooling Water	Determined to be non-risk significant
GMO	Generator Seal Oil	Determined to be non-risk significant
HDH	H.P. Feedwater Heater Drains	Determined to be non-risk significant*
HDL	L.P. Feedwater Heater Drains	Determined to be non-risk significant*
IAC	Containment Instrument Air	Determined to be non-risk significant
TMB	Turbine Control System	Determined to be non-risk significant
CCS	Turbine Plant Component Cooling	Determined to be non-risk significant*

* In addition, based on the outcome of the Feedwater, Condensate, SG Blowdown and Main Steam System piping segments evaluation, these other systems are considered bounded by these evaluations which determined all segments to be less safety significant.

4.2.2 Piping Segments

An acceptable method for modeling a run of a pipe in a PRA or to define its ISI requirements is to divide the pipe run into segments. Portions of pipes within the piping systems having the same consequences of failure should be systematically identified. Consequences of failure may be defined in terms of an initiating event, loss of a particular train, loss of a system, or combinations thereof. The location of the piping in the plant, and whether inside or outside the containment, should be taken into account in defining piping segments.

Piping sections subjected to the same degradation mechanism should be systematically identified. Most of the degradation mechanisms present in nuclear power plant piping are dependent on a combination of design characteristics, fabrication processes and practices, operating conditions, and service experience.

A piping segment should be defined as that run of piping for which the potential degradation mechanism is the same, and a failure at any point in the segment results in the same consequence. In addition, consideration should be given to identifying distinct segment boundaries at branching points such as flow splits or flow joining points, locations of size changes, isolation valve, MOV and AOV locations. Distinct segment boundaries should be defined if the break probability is expected to be significantly different for various portions of piping.

As can be noted from the previous discussions, the process of defining pipe segments is iterative. It generally requires an analyst to make several modifications to the pipe segment definitions before they are finalized.

See Section A2.3 of Appendix 2 for an acceptable approach on how to segment and display pipe segment information.

4.2.3 Modeling Pipe Failures in PRA

One acceptable approach to the incorporation of pipe failures into a PRA is to define logic model events to represent pipe segments, failures, and to incorporate them in the logic model in such a way that their consequence in terms of equipment failures (see Section 4.2.5) is captured. By estimating the probabilities of these pipe segment failures, their contribution to risk can be incorporated quantitatively in the PRA model.

An alternative acceptable approach is based on categorizing each segment's failure likelihood and the consequences of each segment's failures in terms of their impact on the plant. These two elements of risk, failure likelihood and consequences, are then systematically combined to determine the safety significance of each segment.

New initiators may need to be added to the PRA model if the greater resolution of the piping failures introduce different demands on mitigating systems than the generic pipe failures did in the baseline PRA. Correspondingly, when non initiating event pipe failure consequences cannot be captured by surrogate basic event failures, new basic events may need to be added to the models. For example, consider a system model that initially has only two basic events representing the failure of train A and train B of the system. Train A contains two parallel flow paths, one of which can be failed by the failure of a particular pipe segment. Since the model does not contain a surrogate basic event that represents the failure of the particular pipe segment, the model should be revised by adding basic events to represent the failure of each parallel flow path in train A. In addition, careful attention should be given to pipe failures which could cause initiating events and, at the same time, fail or degrade mitigating systems (common cause initiators).

See Section A2.4 of Appendix 2 for more details on an acceptable quantitative approach for modeling pipe failures in a PRA.

4.2.4 Piping Failure Potential

The determination of failure likelihood of piping segments, either as a quantitative estimate or a categorization into groups, should be based on appropriate values reflecting degradation mechanisms, operational characteristics, potential dynamic loads, flaw size and distributions, inspection parameters, experience data base, etc. The evaluation should include the appropriate quantitative definition of the failure potential (e.g., the failure rate or failure unavailability associated with the pipe and the basis for the quantitative definition. The failure probability or frequency used in the PRA should be appropriate for the specific environmental conditions, degradation mechanisms, and failure modes for each pipe location. When data analysis is used to develop a quantitative estimate, the data should be appropriate. When elicitation of expert opinion is used in conjunction with, or in lieu of probabilistic fracture mechanics analysis, a systematic procedure should be developed for conducting such elicitation. In such cases, a suitable team of experts should be selected and trained.

To understand the impact of specific assumptions or models used to characterize the pipe failure frequency or probability, appropriate sensitivity or uncertainty studies should be performed. These uncertainties include, but are not limited to, definition of limiting failure modes, such as loss of function as opposed to loss of structural integrity; design versus fabrication differences; variation in material properties and strength; effect of various degradation and aging mechanisms; variation in steady-state and transient loads; availability and accuracy of plant operating history; availability of inspection and maintenance program data; and capabilities of analytic methods and models to predict realistic results. Qualitative arguments may also be used to address these assumptions and models, but these arguments should be self supporting and self evident.

The methodology, process, and rationale used to determine the failure likelihood of piping segments should be independently reviewed during the final classification of the safety significance of each segment. This review should be documented and included in the submittal. When new computer codes are used to develop quantitative estimates, the techniques should be verified and validated against established industry codes.

See Section A2.5 of Appendix 2 for details on an acceptable approach for determining pipe failure likelihood for use in PRA.

4.2.5 Consequences of Failure

The impact on risk due to piping pressure boundary failure should consider both direct and indirect effects. Consideration of direct effects should include failures that cause initiating events, disable single or multiple components, trains or systems, or a combination of these effects. Indirect effects of pressure boundary failures affecting other systems, components and/or piping segments, also referred to as spatial effects such as pipe whip, jet impingement, consequential initiation of fire protection systems, or flooding should also be considered. Part of the analysis should incorporate insights obtained from the licensee's analysis of IPE, IPEEE, fire, flooding, etc.

The direct and indirect effects of pipe failures should be characterized to incorporate appropriate failure mechanisms and dependencies into the PRA model. An acceptable method of incorporating pipe failures is to classify pipe failures as leaks, disabling leaks, and breaks. Each of these failure modes has a specific failure probability and a corresponding potential for degrading system performance through direct and/or indirect effects. Leaks can result in moisture intrusion through jet impingement, flooding, and sprays. Disabling leaks (larger break area than for leaks) can result in initiating events and loss of system function in addition to indirect effects. Breaks can result in damage due to pipe whip in addition to all of the above mentioned damages. The corresponding failure probability or potential normally decreases as the break area increases.

To understand the impact of a specific assumption or model on the results of the PRA, appropriate sensitivity studies should be performed. Use of qualitative arguments should be self supporting and self evident.

The consequence evaluation should incorporate the contributions to risk from pipe failures as initiating events, mitigating system failures, and failures that cause both (common cause initiators). Risk assessment incorporates more than the contribution of pipe segment failing.

It includes operator actions, system interactions, common component (segment) interactions, etc. A qualitative assessment needs to consider the impact of these measures.

See Section A2.4 of Appendix 2 for more details on an acceptable approach for incorporating the consequences of pipe failures in a PRA.

4.2.6 Risk Impact of ISI Changes

A risk-informed ISI change request should demonstrate that principle four in DG-1061, highlighted in Section 4.4 of this regulatory guide, is met. Principle four states that proposed increases in risk, and their cumulative effect, are small and do not cause the NRC Safety Goals to be exceeded. Increase in risk caused by changes in the ISI program could arise from a decrease in the number of welds inspected, reduced efficiency from simplified weld inspections, or both. Decreases in risk could arise from inspecting welds not currently being inspected in the program, improved weld inspections, or both. The greater the potential risk increase in the proposed change in the ISI program (e.g., the larger the reduction in the number of welds to be inspected and of replacements of detailed inspections with simplified inspections) the more rigorous and detailed the risk analyses needed.

The licensee's risk assessment should be used to address the principle that proposed increases in risk, and their cumulative effect are small and do not cause the NRC Safety Goals to be exceeded. For purposes of implementation, the licensee should assess the expected change in CDF and LERF. The necessary sophistication of the evaluation is that needed to ensure that the potential risk impact of a change to the ISI program is acceptable. For changes that result in substantial impact, an in-depth and comprehensive PRA analysis of appropriate scope to derive a quantified estimate of the total impact of the proposed change will be necessary to provide adequate justification. In other applications, calculated risk importance measures or bounding estimates will be adequate. In still others, a qualitative assessment of the impact of the change on the plant's risk may be sufficient.

The fulfillment of principle four should be based on

- risk importance measures or bounding estimates capable of categorizing plant specific pipe element failure potential categories of high- and low- failure potential, and consequences categories of high- and low- safety significant piping (see Section A.2.7),

- a systematic process to combine failure potential and consequence to determine pipe element safety-significance,

- a weld inspection selection process which provides for changes in the ISI program based on the safety-significance of the pipe element,

- a discussion and evaluation of the aggregate risk impact of the set of changes requested in the ISI program, and

- an assessment and accounting of the sensitivities and uncertainties associated with the evaluations.

The process needs to demonstrate that the major part of the risk attributed to piping failure is covered by the RI-ISI program.

The general approach to risk impact evaluation for inspection of piping is illustrated in Figure 4.2. See Section A2.6 of Appendix 2 for details on an acceptable quantitative approach determining the risk impact of an ISI change.

FIGURE 4.2 General Approach to Risk-Impact Evaluation of Piping

- **Identify the Piping Segments and Associated Weld Population**
- **Determine the Risk Impact of thePiping Segment Failure**
  - **Determine the Initiating Event Frequency Increase Due to Piping Segment Failure**
  - **Determine the System Unavailability Increase Due to Piping Segment Failure**
- **Estimate the Piping Segment Failure Probabilities**
- **Determine the Piping Segment Risk Importance Combining the Risk Impact and Failure Probability**
- **Identify the Areas of the Piping Segment and the Sample Welds to be Inspected to Control the Risk Level Using the Risk Importance of the Piping Segments**

4.2.6.1 Initiating Events

For purposes of determining RI-ISI requirements, all initiating events (internal and external) and all operating modes should be evaluated to see whether initiating events and predicted plant response are affected by RI-ISI proposed changes. In addition, other initiators, including those that have been screened out (eliminated) from the base PRA, have to be considered by answering the following questions.

(1) Does the ISI issue involve a change that could lead to an increase in the frequency of a particular initiator already included in the PRA?

(2) Does the ISI issue involve a change that could lead to an increase in the frequency of a particular initiator initially screened out of the PRA?

(3) Does the ISI issue affect the quantification of previously identified accident scenarios for specific initiators that were screened out and eliminated from the PRA because of truncation?

(4) Does the ISI issue have the potential to introduce a new initiating event?

4.2.6.2 Dependencies and Common Cause Failures

The effects of dependencies and Common Cause Failures (CCFs) for ISI components need to be considered carefully because of the significance they can have on core damage frequency. Generally, data are insufficient to produce plant-specific estimates based solely on the data. For CCFs, data from generic sources may be required. (Ref. 21) and Appendix 2 to this document address CCFs in more detail.

4.2.6.3 Uncertainty and Sensitivity Analyses

Uncertainty and sensitivity analyses are expected to play an important (and complex) part in the support of risk-informed ISI program changes. These topics are discussed further in (Ref. 21 and Section A2.5.5 of Appendix 2 of this document. It is expected that certain application-specific guidance will be developed from the ongoing NRC reviews of the proposed RI-ISI pilot plant programs.

4.2.6.4 Human Reliability Analyses

See (Ref. 21) for further discussions on this topic. For ISI-specific analyses, the human reliability analysis methodology used in the PRA must account for the impact that the pipe segment break will have on the operator's ability to respond to the event. In addition, the reliability of the inspection program (both operator and equipment) that factor into the probability of detection should also be addressed (see Attachments 4 and 5).

4.2.7 Element Selection

This section discusses the establishment of the number of elements (e.g., welds) per segment requiring inspection and provides guidance on how to meet the requirements of 10 CFR 50.55a(a)(3)(i). 10CFR50.55a(a) states:

> (3) Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f) (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:
>
> (i) The proposed alternatives would provide an acceptable level of quality and safety....

Section 4.2.6 addressed the determination of the risk impact of ISI changes and the categorization of pipe segments as high- and low-safety significant. The segments categorized as low-safety significant will require less oversight and inspection than those categorized as high-safety significant.

One option for meeting the "acceptable level of quality and safety" criterion of 10 CFR 50.55a(a), is for a licensee to review existing industry experience with the ASME Section XI requirements and assess that performance on systems and piping components (e.g., developing target leak frequency goals based on existing ASME results). Meeting this performance standard with high assurance levels for the high-safety significant piping segments could be used as one element in the staff's determination of *acceptable level of quality and safety*.

Appendix 2 to this regulatory guide provides one example of determining leak target goals that conform to the 50.55a requirement, listed above. A licensee could propose its leak target goals and develop a program that meets those goals. Such a program would define the number of welds requiring inspection.

For the example "leak target goals" identified in Appendix 2, the target goals are applied to the ***system*** under consideration. For example, if a system is comprised of 36 segments, and 20 of which are categorized as high-safety significant, then the target goal for those 20 segments should meet the target leak goals at the 95% confidence level. The staff will find acceptable a 95% assurance level that the target leak goals will be met.

The target leak goals can be established on a system or element level. If established on an element level, the goals should ensure that the system's reliability is consistent with existing Section XI performance. For example, the leak target goals defined in Table A2.9 is based on global industry piping performance. These target goals would be applied only to the HSS segments in a system. Appendix 4 to this regulatory guide provides an examples of how one could calculate the number of welds to be inspected to meet the leak target goals.

Any analysis needs to consider the inspection method and the probability of detection. Appendix 4 provides an example of such a calculation.

4.3 Integrated Decisionmaking

This section discusses the integration of the technical considerations involved in reviewing submittals from licensees proposing to implement RI-ISI programs. General guidance for risk-informed applications is provided in Draft Regulatory Guide DG-1061. Specifically, the integrated decision process should assess whether or not:

- The comprehensive plant model, including the PRA and the integrated deterministic analysis, is technically sound and supports the rest of the findings regarding the proposed RI-ISI program.

- The analysis is based on the as-built and as-operated and maintained plant.

- All safety impacts of the proposed changes to the licensee's ISI program have been evaluated in an integrated manner as part of an overall risk management approach in which the licensee is using risk analysis to improve operational and engineering decisions broadly and not just to eliminate requirements seen as undesirable (i.e., the approach used to identify changes in requirements for ISI were used to identify areas where requirements in ISI should be increased as well as reduced).

- The proposed changes to the ISI program have been evaluated in an integrated fashion that ensures that all of the key safety principles are met.

- The cumulative risk evaluation accounting for all of the proposed ISI program changes confirms that changes to the plant core damage frequency (CDF) and large early release frequency (LERF) are small in conformance with the guidelines given in Section 2.4.2 of Draft Regulatory Guide DG-1061 and summarized below.

The risk acceptance guidelines discussed in DG-1061 are based on the principles and expectations for risk-informed regulations. As such, the licensee's risk assessment should:

- address the principle that increases in estimated CDF and LERF resulting from the proposed CLB changes will be limited to small increments,

- be sophisticated enough to support the determination of the expected change in the risk,

- be subjected to appropriate quality controls, and

- realistically reflect the actual design, construction, and operational practices of the plant requesting the proposed CLB change.

For the purpose of establishing objectives or guidelines for risk-informed decisionmaking, the CDF objective of 1E-04 per reactor year has been adopted. A large early release frequency (LERF) range of 1E-6 to 1E-5 per reactor year has been adopted as a containment performance guideline.

The acceptance guidelines have the following elements:

- For a plant with a mean core damage frequency at or above 1E-4 per reactor year (the Commission's subsidiary core damage frequency objective) or with a mean LERF at or above 1E-5 per reactor year, it is expected that applications will result in a net decrease in risk or be risk neutral.

- For a plant with a mean core damage frequency of less than 1E-4 per reactor year, applications will be considered which, combined with the LERF guidelines described below:

    - Result in a net decrease in CDF or are CDF neutral;
    - Result in increases of calculated CDF that are very small (i.e., CDF increases of less than 1E-6 per reactor year); or
    - Result in an increase in calculated CDF in the range of 1E-6 to 1E-5 per reactor year, subject to increased NRC technical and management review and considering the following factors:
        - The scope, quality, and robustness of the analysis (including but not limited to the PRA), including consideration and quantification of uncertainties,

- The base CDF and LERF of the plant,
- The cumulative impact of previous changes (the licensee's risk management approach),
- Consideration of the NRC's Safety Goals policy screening criteria in the staff's Regulatory Analysis Guidelines, which define what changes in CDF and containment performance would be needed to consider potential backfits,
- The impact of the proposed change on operational complexity, burden on the operating staff, and overall safety practices, and
- Plant-specific performance and other factors, including, for example, siting factors, inspection findings, performance indicators, and operational events.

AND

- For a plant with a mean LERF of between 1E-6 and 1E-5 per reactor year:

  - Result in a net decrease in LERF or are LERF-neutral;
  - Result in an increase in calculated LERF of up to 1E-6 per reactor year, subject to increased NRC technical and management review, as described above;

OR

- For a plant with a mean LERF of less than 1E-6 per reactor year:

  - Result in a net decrease in LERF or are LERF-neutral;
  - Result in increases in calculated LERF that are very small (i.e., LERF increases of less than 1E-7 per reactor year); or
  - Result in an increase in calculated LERF of up to 1E-6 per reactor year, subject to increased NRC technical and management review, as described above.

The rigor of analyses needed to support these different types of applications is addressed in Section 2 of DG-1061.

- Appropriate consideration was given to the uncertainties in the analyses and interpretation of the results.

- Plant-specific data was incorporated into the analyses, as appropriate.

- Defense-in-depth evaluations have been performed, and insights from these have been duly incorporated into the classification scheme, the performance goals, and the associated programmatic activities. These evaluations confirm that sufficient safety margins exist and the CLB's defense in depth evaluation is not compromised.

- The scope and models used were appropriate for the proposed change and the analysis was subjected to quality controls.

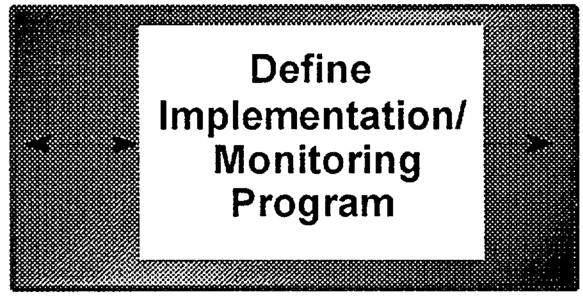
- Pipe segments have been identified and appropriately categorized for use in prioritizing and implementing the program. In particular, important components not modeled in the PRA have been identified and appropriately categorized using available deterministic supporting information.

- An appropriate monitoring program is proposed to assess plant performance and provide for feedback and corrective action if performance goals are not met.

- The data, analysis methods and assessment criteria used in the development of the RI-ISI programs are scrutable and available for public review.

In summary, acceptability of the proposed change should be determined using an integrated decision-making process that addresses three major areas: (1) an evaluation of the proposed change in light of the plant's current licensing basis, (2) an evaluation of the proposed change relative to the key principles and the acceptance criteria, 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-ISI program should be as realistic as possible, with proper 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-ISI program description including those areas that have been quantified through the use of PRA as well as qualitative arguments for those areas that cannot be readily quantified. Examples of qualitative subjects include ALARA for plant personnel, operator procedures that ease the burden on plant personnel, organizational human factors, etc. When making final programmatic decisions, choices must be made based on all of the available information. There may be cases where information is incomplete or where conflicts appear to exist between the traditional engineering data and the PRA-generated information. It is the responsibility of the licensee in such cases to resolve such issues.

5. ELEMENT 3: IMPLEMENTATION, PERFORMANCE MONITORING, AND CORRECTIVE ACTION STRATEGIES



ELEMENT 3

Using the information produced from Elements 1 and 2 of the RI-ISI process (as described in Chapters 3 and 4), the licensee develops a proposed RI-ISI program. The program should include implementation, performance monitoring, and corrective action strategies. The program should be self correcting as experience dictates. The programs should contain performance measures used to confirm the safety insights from the PRA. While the actual development of the RI-ISI program is left to the licensee's discretion, Appendix 5 provides a detailed discussion on an acceptable approach for developing an RI-ISI program.

Upon approval of the RI-ISI program, the licensee should have in place an implementation schedule for inspecting all HSSCs and LSSCs identified in its program. The number of required inspections should be a product of the systematic application of the risk-informed process.

5.1 Program Implementation

The implementation of a RI-ISI program for piping may begin at any point of the inspection interval as long as the examinations are scheduled and distributed to be consistent with the inspection interval requirements of the ASME Boiler and Pressure Vessel Code Section XI Edition and Addenda committed to by a licensee in accordance with 10 CFR 50.55a. The requirements for these intervals are contained in Section XI under Article IWA-2000 as they apply to Inspection Program B. **Initial RI-ISI programs should be submitted for NRC staff approval in accordance with 10 CFR 50.55a(a)(3) and documentation of program updates should be kept and maintained by the licensee on site for audit.** Updates to thee RI-ISI program should be performed at least on a periodic basis to coincide with the inspection program requirements contained in Section XI under Inspection Program B. These updates should be expedited as dictated by any plant established procedures to update their PRA which may be more restrictive than the Section XI period update. As plant design feature changes are implemented, changes to the input associated with the RI-ISI program segment definition and element selections should be reviewed and modified, as needed. Changes to equipment performance, the plant procedures that can affect system operating parameters, changes in component test intervals, valve lineups, operating modes of the equipment, or the ability of the plant personnel to perform actions associated with accident mitigation should be included in any RI-ISI program update. When scheduled RI-ISI program NDE examinations and pressure tests are completed with corresponding VT-2 visual examinations for leakage and flaws or indications of leakage are identified, the existence of these conditions should be evaluated as part of the RI-ISI update.

Each 10-year inspection interval is subdivided into inspection periods which end at 3, 7, and 10 years of plant service within each interval. Variations in these inspection program intervals and periods by plus or minus 1 year are allowed under Section XI based on refueling outage situations and may be employed by a licensee who implements an RI-ISI program. These same basic RI-ISI program interval and period requirements may also be used by a licensee who chooses to perform on-line Nondestructive Examination (NDE), but special considerations may have to be taken in regard to program updates during the performance of corrective actions that result from these examinations.

5.2 Performance Monitoring

RI-ISI programs are living programs and should be monitored continuously. Monitoring these programs encompasses many facets of feedback or corrective action that include periodic updates based on input and changes resulting from plant design features, plant procedures, equipment performance, examination results, and individual plant and industry failure information. Since the PRA used in the development of any RI-ISI program is a state of knowledge at the time of implementation, any significant change in parameters affecting the total plant's CDF or LERF needs to be considered upon identification. Plant administrative procedures should be in place to implement these changes into the PRA and incorporate any relevant results into the RI-ISI program outside of any periodic update.

The purpose of performance monitoring is to confirm:

(1) the assumptions in the PRA that could affect the probability or consequences of pipe failures,
(2) the target objectives or goals used in the integrated decisionmaking process are being met,
(3) the known degradation mechanisms are understood, and
(4) that any unknown degradation mechanisms are identified before they have a detrimental impact on risk.
(5) that the integrated decisionmaking process remains current with plant and industry experiences.

Performance monitoring of the risk-informed ISI plan is intended to confirm that:

- Piping reliabilities used in the calculation of risk contributions from passive piping components remain valid and thereby justify continuation of the ISI plan without modifications

- Appropriate modifications to the ISI plan are developed if new or unexpected degradation mechanisms occur.

The inspection procedures and analyses must provide assurances that performance degradation is detected with sufficient margin that there is no adverse effect on public health and safety (i.e., the failure rates cannot be allowed to rise to unacceptable levels before detection and corrective action take place). The basic elements of an acceptable performance monitoring program are illustrated in Figure 5.1 and summarized in the following subsections.

Periodic Updates.

Updates to an RI-ISI program should be performed at least on the basis of periods that coincide with the inspection program requirements contained in Section XI under Inspection Program B. These updates would be expedited as dictated by any plant established procedures to update their PRA which may be more restrictive than a Section XI period type update.

Plant Design Feature Changes.

As plant design changes are implemented, changes to the inputs associated with RI-ISI program segment definition and element selections may occur. It is important to address these changes to the inputs used in any engineering assessment or structural reliability risk assessment (SRRA) model that may affect resultant failure probabilities in terms of pipe leakage, disabling leakage, and full rupture events. Some examples of these inputs would include the following:

- Operating characteristics (e.g., changes in water chemistry control)
- Material and Configuration Changes;
- Welding Techniques/Procedures;
- Construction and Preservice Examination Results; and
- Stress Data (Operating Modes, Pressure, and Temperature Changes)

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

Plant Procedure Changes.

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

Equipment Performance Changes.

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

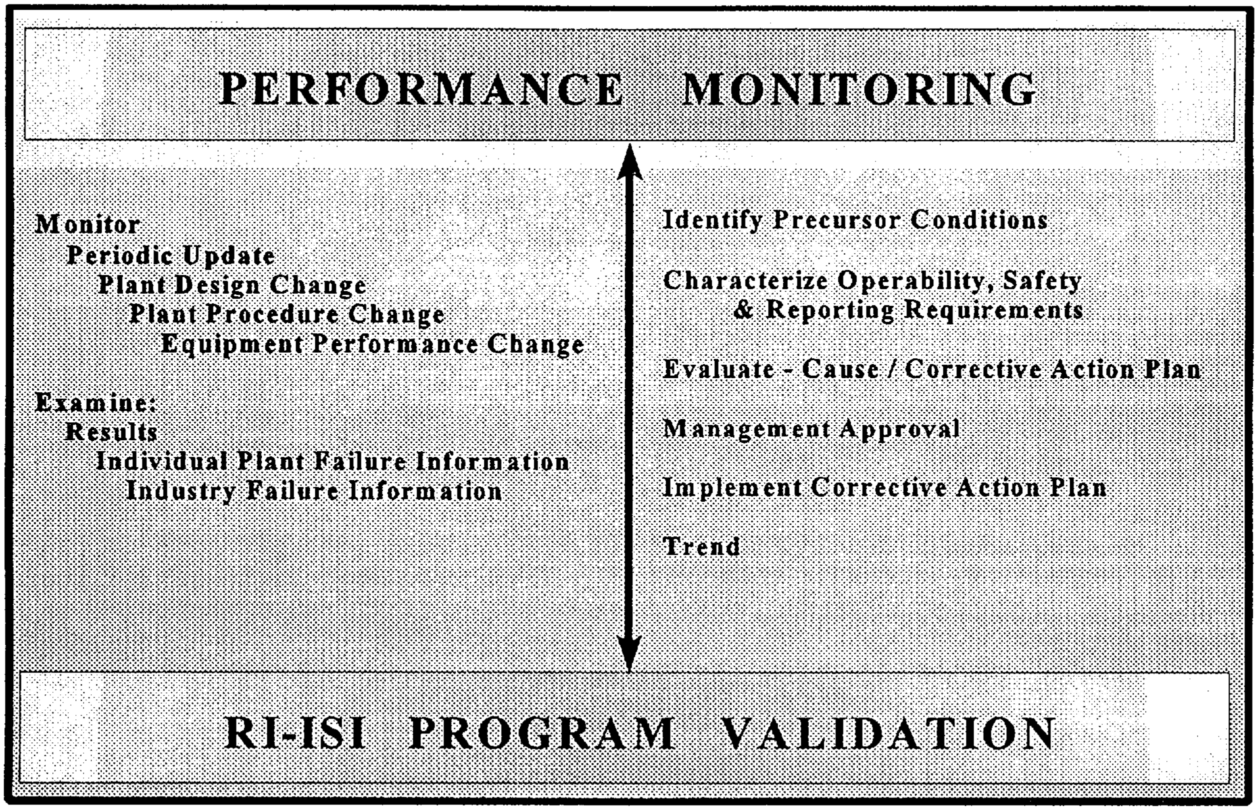


Figure 5.1 Elements of a Performance Monitoring Program

Examination Results.

When scheduled RI-ISI program NDE examinations and pressure tests are completed with corresponding VT-2 visual examinations for leakage, and flaws or indications of leakage are identified, the existence of these conditions should be evaluated as part of the RI-ISI program update.

Individual Plant and Industry Failure Information.

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

5.3 Corrective Action Programs

Each licensee of a nuclear power plant is responsible for having a corrective action program, consistent with Draft Regulatory Guide DG-1061. Measures are to 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. In the case of significant conditions adverse to quality, the measures must ensure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action are to be documented and reported to appropriate levels of management.

It is anticipated that a corrective action program will incorporate the following elements:

Identify.

Through the inspection location selection process established under an RI-ISI program, the structural element examinations performed should identify those conditions that would be adverse to quality in relation to identifying precursors to potential or actual leaks, disabling leaks, or pipe ruptures.

Characterize.

Depending on the timing of the condition identification and operational mode of the plant, (this may be a more critical situation when on-line NDE is performed) the initial issues to be addressed include:

- the effects on operability of safety related systems, structures, or components;
- if regulatory reporting is required; or
- the condition results in an immediate plant/personnel safety or operational impact.

If any of these three considerations exist, then the plant's management must be immediately notified through plant established procedures.

Evaluate

Evaluation has two parts: (1) determine the cause and extent of the condition identified, and (2) develop a corrective action plan or plans. Additional examinations should be considered an acceptable method in providing this cause and extent determination. Under an RI-ISI program, both quantitative and qualitative insights are used to identify postulated failure modes and elements to be examined. Performances of examinations on selected elements have been grouped into regions of "High" and "Low" failure potential and safety significance. These groupings provide the basis for additional examinations to be performed to determine the cause and extent of the condition identified. Acceptable sampling schemes such as those identified in ASME Section XI under IWB-2430 may be used with due consideration given to limit the additional examinations by piping segment, materials, service conditions, and failure modes already established in the RI-ISI program. Alternatively, due to the available information used in an RI-ISI program, an engineering evaluation may be used as a substitute for additional examinations to determine the cause and extent of the condition identified.

Once the true extent of the condition has been identified and documented by a licensee, then a corrective action plan should be developed. The plan could include repair, replacement, or monitoring of the condition identified depending on its safety significance. Several options of corrective action may be available to a licensee, but in all cases, needed success criteria must be defined and documented with the corrective action plan. These success criteria include the measurable attributes needed to evaluate the effectiveness of the corrective action in the prevention of a recurrence of the identified condition. The success criteria may be as simple as implementation of new element selections based on the new failure information during the next scheduled periodic update of the RI-ISI program and then performing the examinations to confirm that the issue has been corrected. Conversely, to prevent the condition from reoccurring, these criteria may require a plant design change, depending on the condition identified, and possible routine scheduled replacements.

Decide.

A decision should be made by appropriate levels of management on the owner's implementation of any corrective action plan. Agreement on the adequacy of the success criteria should be reached among the personnel involved and resources allocated to implement the plan. Cost will inevitably play a part in the decision process, but it is more important to fix the problem correctly the first time so as to avoid recurrence in the future.

Implement.

Complete the work necessary to both correct the problem and prevent its recurrence. In the case of an RI-ISI program, successive examinations may be one way to measure the effectiveness of the corrective action plan. A licensee could follow the requirements for successive examinations as described in Section XI, IWB-2420. These requirements could be used when flaws or conditions have been accepted by analytical evaluations and measurements of potential service related degradation. It is essential to avoid a future failure of a pipe element.

Monitor.

The first activity that must be monitored is whether or not the planned corrective action was implemented. Management should accomplish this as part of their oversight of daily work activities. In an RI-ISI program this may be as simple as having administrative procedures in place to verify that the program has been updated as a result of the corrective action plan and review the data to verify that the examinations are being performed as scheduled.

Once it has been determined that corrective actions have been implemented, the planned actions to verify that the desired results are obtained should be conducted. This is done by measuring the success criteria at regularly scheduled intervals in accordance with the corrective action plan. This measurement may indicate that the success criteria did not fix the problem or only partially fixed the problem. Additional corrective action plans may have to be developed and implemented if this situation occurs.

Trend.

The purpose of trending is to identify conditions that are significant based not only on individual issues, but on accumulation of similar issues. Even issues assigned low significance may be deemed of greater significance if there are an increasing number of similar issues. During the RI-ISI program, periodic updates of occurrences which required corrective actions should be reviewed by the ISI team and appropriate oversight groups/management to determine if these insights should result in additional or different locations for examination.

5.4 Acceptance Guidelines

The acceptance guidelines for the implementation, monitoring, and corrective action programs for the accepted RI-ISI program plan are presented below (*a.* through *g.).* In addition, acceptance guidelines for the initial development of the RI-ISI program plan, as described in Appendix 4, are provided (*h* through *t*). The acceptance guidelines include:

a. The implementation program will be evaluated based on the attributes presented in Section 5.1.

b. The monitoring strategy should evaluate that the RI-ISI components (i.e., pipe segments and elements) meet the guidelines addressed in Chapter 4 and are adequate to uncover components that fail to either meet the acceptance guidelines or are otherwise determined to be in a non-conforming condition.

c. The corrective action program should provide reasonable assurance that a nonconforming component will be brought back into conformance.

d. Evaluations within the corrective action program should:

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

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

(3) determine and correct the root cause of the failure or nonconforming condition

(4) assess the applicability of the failure or nonconforming condition to other components in the RI-ISI program

(5) correct other susceptible RI-ISI components as necessary

(6) incorporate the lessons in the data base and SRRA computer models, if appropriate

(7) assess the validity of the PRA failure rate and unavailability assumptions in light of the failure(s), and

(8) consider the effectiveness of the component's inspection strategy in detecting the failure or nonconforming condition. Adjust the inspection interval and/or inspection methods, as appropriate, when the component (or group of components) experiences repeated failures or nonconforming conditions.

e. The corrective action evaluation should be provided to the licensee's PRA group so that any necessary model changes and regrouping are done as might be appropriate.

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

g. A program is in place that monitors industry findings.

h. Piping Subject to Examination

The examination requirements include Class 1, 2, and 3 piping evaluated by the risk-informed process. Piping in systems evaluated as part of the plant PRA, but outside the current Section XI examination, and categorized as high-safety-significant, in accordance with Chapter 4 of this regulatory guide, are included.

i. Inspection Program

The examinations are to be completed during each inspection interval in accordance with the goals established for leak probability per weld per year (or other NRC approved performance monitoring criterion), with the following exceptions.

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

(2) If, during the interval, a reevaluation using the RI-ISI process is conducted and items are required to be added to the examination program, those items shall be added and the NRC informed.

j. Successive Inspections
Locations selected for inspections should be subjected to examinations consistent with Section XI requirements at appropriate intervals, such as given by items (1) through (3), below. Those locations with detected degradation (found to be at

acceptable levels) should be subject to more frequent examinations. An acceptable schedule for examinations is:

(1) The sequence of piping examinations established during the first inspection interval using the RI-ISI process shall be repeated during each successive inspection interval; however, the examination sequence may be revised to satisfy the requirements of Table IWB-2411-1 or Table IWB-2412-1, of Section XI.

(2) If piping structural elements are accepted for continued service by analytical evaluation in accordance with *m* (below), the areas containing the flaws or relevant conditions shall be reexamined during the next three inspection periods referenced in the schedule of the inspection program of *l* (above).

(3) If the reexaminations required by *j*.2 reveal that the flaws or relevant conditions remain essentially unchanged for three successive inspection periods, the piping examination schedule may revert to the original schedule of successive inspections.

k. Additional Examinations

Examinations performed in accordance with *l* (below) that reveal flaws or relevant conditions exceeding the acceptance standards are to be extended to include additional examinations. The additional examinations are to include piping structural elements described in Table 5.1 with the same postulated failure mode and the same or higher failure likelihood.

(1) The number of additional elements will be the number of piping structural elements with the same postulated failure mode originally scheduled for that fuel cycle.

(2) The scope of the additional examinations may be limited to those high-safety-significant piping structural elements within systems whose materials and service conditions are determined by an evaluation to have the same postulated failure mode as the piping structural element that contained the original flaw or relevant conditions.

If the additional examinations required above reveal flaws or relevant conditions exceeding the acceptance standards, the examination will be further extended to include additional examinations.

(3) These examinations are to include all remaining piping elements within Table 5.1 whose postulated failure modes are the same as the piping structural elements originally examined above.

(4) An evaluation will be performed to establish when those examinations are to be conducted. The evaluation must consider failure mode and likelihood.

For the inspection period following the period in which the examinations of above were completed, the examinations are to be performed as originally scheduled.

l. Examination and Pressure Test Requirements

Piping structural elements categorized as high-safety-significant are to be examined as required in Table 5.1.

Pressure testing and VT-2 visual examinations are to be performed on Class 1, 2, and 3 piping systems in accordance with Section XI specified in the licensee's ISI program.

Examination qualification and methods and personnel qualification are to be in accordance with the edition and addenda of Section XI specified in the licensee's ISI program.

m. Acceptance Standards for Identified Flaws

For component configurations or examination methods not addressed by Table 5.1, the licensee is to develop acceptance criteria consistent with the requirements of IWA-3000. The referenced paragraphs below and in Table 5.1 are to be applied in accordance with the edition and addenda of Section XI specified in the licensee's ISI program.

(1) Flaws that exceed the acceptance standards listed in Table 5.1 found during surface or volumetric examinations may be accepted by repair/replacement activities or approved analytical evaluation.

(2) Flaws or relevant conditions that exceed the acceptance standards listed in Table 5.1 found during visual examinations may be accepted by supplemental examination, corrective measures, repair/replacement activities, or approved analytical evaluation.

(3) Other unacceptable conditions not addressed above may be accepted by repair/replacement activities, or by approved analytical evaluation.

n. Repair/Replacement Procedures

Repair/replacement activities are to be performed in accordance with the Section XI requirements specified in the licensee's ISI program.

o. System Pressure Tests

System pressure tests should be performed in accordance with IWA-5000, IWB-5000, IWC-5000, IWD-5000 of the Section XI Edition and Addenda, as specified in the licensee's ISI program.

p. Records and Reports
Records and reports should be prepared and maintained in accordance with IWA-6000 of the Section XI Edition and Addenda as specified in the Licensee's ISI program.

q. The licensee's RI-ISI program submittal should be consistent with the acceptance guidelines contained throughout this regulatory guide, specifically with the findings listed in this section, or justify why an alternative approach is acceptable.

r. The licensee's proposed RI-ISI program should address the four principal elements of risk-informed decisionmaking (addressed in this document) by defining the proposed change, basing the new program on traditional analysis with insights from probabilistic risk assessments, and incorporating an implementation and monitoring program that enables the staff to conclude that the proposed RI-ISI program provides "an acceptable level of quality and safety" [10 CFR 50.55a (a)(3)(i)].

s. Administrative procedures should be in place to implement changes into the PRA and traditional analysis and incorporate any relevant results into the RI-ISI program during and outside any periodic update.

t. The RI-ISI program provides an acceptable level of quality and safety when compared to the existing Section-XI performance.

Table 5.1 Examination Category R-A, Risk-Informed Piping Examinations

Parts Examined	*Examination Requirement*[2]	*Examination Method*	*Acceptance Standard*	*Extent*[3] *and Frequency First Interval*	*Extent*[3] *and Frequency Successive Intervals*[5]	*Defer to End of Interval*
High-Safety Significant Piping Structural Elements[1]						
Elements Subject to Thermal Fatigue	IWB-2500-8(c)[1] IWB-2500-9,10,11 IWC-2500-7(a)[1]	Volumetric	IWB-3514	Inspect once per inspection inteval[2,4]	Same as 1st interval	Not Permissible
Elements Subject to High Cycle Mechanical Fatigue	IWB-2500-8(c)[1] IWB-2500-9,10,11 IWC-2500-7(a)[1]	Visual, VT-2[10]	IWB-3142	Inspect once per refueling outage	Same as 1st interval	Not Permissible
Elements Subject to Corrosive, Erosive, or Cavitation Wastage	Note 8	Volumetric[6] (for Internal Wastage) or Surface (for External Wastage)	IWB-3514 Note 8	Inspect once per inspection interval[2]	Same as 1st interval	Not Permissible
Elements Subject to Crevice Corrosion Cracking	Note 7	Volumetric	IWB-3514	Inspect once per inspection interval[2]	Same as 1st interval	Not Permissible
Elements Subject to Primary Water Stress Corrosion Cracking (PWSCC)[6]	Note 7	Visual, VT-2[10]	IWB-3142	Inspect once per refueling outage	Same as 1st interval	Not Permissible
Elements Subject to Intergranular Stress Corrosion Cracking (IGSCC)	IWB-2500-8 IWB-2500-9,10,11	Volumetric	IWB-3514	Inspect once per inspection interval[2]	Same as 1st interval	Not Permissible
Elements Subject to Microbiologically Influenced Corrosion (MIC)	IWB-2500-8 IWB-2500-9,10,11	Visual, VT-3 Internal Surfaces or Volumetric[6]	Note 8	Inspect once per inspection interval[2]	Same as 1st interval	Not Permissible
Elements Subject to Flow Accelerated Corrosion (FAC)	Note 9	Note 9	Note 9	Note 9	Note 9	Note 9

NOTES:

(1) The length for the examination volume shall be increased to include ½ in. beyond each side of the base metal thickness transition or counterbore.

(2) Includes all examination locations identified in accordance with the risk-informed selection process

(3) Includes 100% of the examination location. When the required examination volume or area cannot be examined due to interference by another component or part geometry, limited examinations shall be evaluated by the ISI Team for acceptability. Areas with acceptable limited examinations, and their bases, shall be documented.

(4) The examination shall include any longitudinal welds at the location selected for examination in Note 2. The longitudinal weld examination requirements shall be met for both transverse and parallel flaws examination volume defined in Note 2.

(5) Initially selected examination locations are to be examined in the same sequence during successive inspection intervals, to the extent practical.

(6) Applies to mill annealed Alloy 600 nozzle welds and heat affected zone (HAZ) without stress relief.

(7) The examination volume shall include the volume surrounding the weld, weld heat affected zone, and base metal, where applicable, in the crevice region. Examination should focus on detection of cracks initiating and propagating from the inner surface.

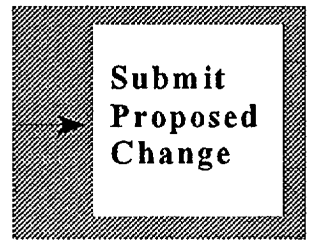
(8) The examination volume shall include base metal piping, welds, weld HAZ, including the piping near the weld, in the affected regions of carbon and low alloy steel, and the welds and weld HAZ of austenitic steel. Examinations shall verify the minimum wall thickness required. Acceptance criteria for localized thinning is in course of preparation. The examination method and examination region shall be sufficient to characterize the extent of the element degradation.

(9) In accordance with the Owner's existing FAC program.

(10) VT-2 examinations may be conducted during a system pressure test or a pressure test specific to that component/element.

6. ELEMENT 4: DOCUMENTATION

The recommended format and content for a plant-specific risk-informed ISI submittal are presented in this section. Use of this format by licensees will help ensure the completeness of the information provided, will assist the NRC staff in locating the information, and will aid in optimizing the time needed for the review process. Unless otherwise noted, all information should be contained in the main submittal report.



ELEMENT 4

This format follows the staff's guidance identified in the Standard Review Plan Chapter 3.9.8 (Ref. 17). Additional guidance on style, composition, and specifications of safety analysis reports is provided in the Introduction of Revision 3 to Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" (Ref. 22).

Table 6.1 provides an overall summary of the documentation information needed to support a risk-informed ISI submittal.

6.1 Risk-Informed Inservice Inspection Program Plan

The licensee's submittal should describe the proposed RI-ISI program with enough detail to be clearly understandable to the reviewers of the program. The description should cover the five items listed in Chapter 3 including sufficient detail such that reviewers of the program can understand how the program would be implemented. These items are: (1) changes to the plant's CLB, (2) changes to inspection schedules, locations, and methods, plus a description of the process used for determining these, (3) listing of affected components including an explicit description of any grouping of components, (4) identification of supporting information, and (5) brief statement regarding the way in which the proposed changes are consistent with the Commission's PRA Policy Statement.

The licensee's submittal should describe how its proposed RI-ISI program addresses the four principle elements of risk-informed decisionmaking (addressed in this document) by defining the proposed change, basing the new program on traditional analysis with insights from probabilistic risk assessments, and incorporating an implementation and monitoring program that enables the staff to conclude that the proposed RI-ISI program provides "an acceptable level of quality and safety" required by 10 CFR 50.55a(a)(3)(i).

The submittal should document the administrative procedures in place to implement changes into the PRA and traditional analysis and incorporate any relevant results into the RI-ISI program during and outside any periodic update. The submittal should be consistent with the guidelines contained throughout this regulatory guide, or provide justification for an alternative approach.

The submittal should also include a description of the process that was used for the categorization of components (further discussed in Section 6.2.2) and for the determination of **when formal interaction with the NRC is or is not needed when making changes to an approved RI-ISI program** (as described in Section 3.2). Exemptions from the regulations, technical specification amendments, and relief requests that are required to implement the licensee's proposed RI-ISI program should also be specified and included in the application.

6.2 Engineering Analysis Records and Supporting Data

The licensee's submittal should describe how the proposed RI-ISI program ensures that plant risk is maintained at acceptable levels. The description should cover the four items listed in Chapter 4 in sufficient detail such that reviewers can determine whether the proposed plan ensures risk is maintained at acceptable levels. Theses items are: (1) illustrate that defense-in-depth is maintained, (2) illustrate that adequate safety margins are maintained, (3) demonstrate that the proposed ISI program changes do not result in unacceptable risk to the public and plant personnel and are consistent with the guidelines identified in Draft Regulatory Guide DG-1061 (and presented in Chapter 4), and (4) support the integrated decisionmaking process. Items 1 and 2 are discussed in Section 6.2.1, and Item 3 is discussed in Section 6.2.2. Item 4 is discussed in Section 6.3.

6.2.1 Traditional Analysis Records and Supporting Data

This section should describe how the proposed RI-ISI program continues to ensure that defense-in-depth is maintained (Item 1) and how the ISI program ensures that adequate safety margins are maintained (Item 2). This description should include a presentation of the decision criteria used to determine whether defense-in-depth (see Section 4.1.1) and adequate safety margins (see Section 4.1.2) are maintained and a discussion of how the proposed ISI program meets these criteria.

6.2.2 Probabilistic Risk Assessment Records and Supporting Data

This section should describe the plant's probabilistic risk assessment in sufficient detail to allow a reviewer to ascertain whether the PRA accurately reflects the current plant configuration and operational practices, and whether the change in risk is acceptable by providing discussions on the topics identified below.

6.2.2.1 Scope

The application should clearly articulate the boundaries for the scope of piping systems, segments, and elements to be included in the RI-ISI program as follows.

Piping Systems

The licensee should document that the piping systems incorporated in the scope of the RI-ISI program include:

- All Class 1, 2, and 3 pipes within the current ASME Section XI programs, and

- All pipes whose failure would compromise

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

- Non-safety-related structures, systems, or components:

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

  - Whose failure could prevent safety-related structures, systems, or components from fulfilling their safety-related function; or

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

In addition, the details of the process used by the licensee to determine the final piping systems list for the RI-ISI program should be included.

Any systems excluded from the scope of the RI-ISI program should be justified by appropriate documentation.

Pipe Segments

Criteria or procedures used to establish pipe segments within the piping systems should be provided. Documentation should be sufficient to allow a reviewer to determine whether consequences of failure, degradation mechanisms, and segment boundaries are properly considered for defining pipe segments in accordance with the guidance in this document. Any deviations from the guidance in this document should be fully documented and justified.

Structural Elements

Piping structural elements included in the scope of the RI-ISI program should be documented to confirm those pressure retaining welds, base metal areas, weld counterbore areas, nozzle welds, valves and fittings are subject to ISI in accordance with the guidelines provided in this document. **Deviations from the guidelines should be documented in sufficient detail to allow NRC review.**

Lots

If structural elements from more than one pipe segment are subsumed within one *lot* for the purpose of statistical inspection sampling, as described in Appendix A4, **the criteria used and the justification for subsuming the welds (elements) should be documented in sufficient detail to allow NRC review and approval.** For a hypothetical example, assume that four segments in a piping system are identical, with respect to degradation mechanisms and failure likelihood. Each segment may contain four elements. It may be argued that a random sampling strategy be developed by which the sixteen (4 * 4) elements are subsumed in one calculation (e.g., one lot comprising of 16 elements), versus performing four separate calculations for each segment (e.g., four *lots,* with four elements in each *lot*).

6.2.2.2 Determination and Quantification of Accident Sequences

This section should present the methods and techniques used to identify and quantify the accident sequences. As part of the documentation, a table should be provided that summarizes how the PRA model used to develop the risk insights compares with the PRA technical issues identified in draft NUREG-1601. In addition, specific detailed information should be provided on identifying and quantifying initiating events; developing or modifying event trees; developing or modifying system models (i.e., fault trees); identifying, modeling, and quantifying passive component failures (i.e., pipe segment failures); identifying, modeling, and quantifying human actions; sequence quantification; and uncertainty/sensitivity calculations as described below.

Initiating Events

The process used to identify initiating events and the results from the process should be documented. For the process, describe how it will result in the identification of all, or the complete set of, initiating events important to the ISI analysis, including those initiating events that may result from the failure of ISI-affected passive component (i.e., pipe segments). For each initiating event identified by the process, present: (1) a description of the initiating event, (2) the rational for including or excluding the event, (3) the event's frequency, and (4) a discussion of how the frequency was estimated. If individual initiating events are collapsed into a group, describe the basis for such a grouping. All information should be provided in the main report.

Event Trees

The process used to develop the event trees should be documented. Provide example event trees that illustrate pictorially the logic structure. The description should include: (1) how the structure of the event tree was developed (i.e., what top events were included and why), (2) a description of each top event, including the success criteria for each top event, (3) and a description of each core damage sequence modeled in the event tree.

System Model Fault Trees and Passive Component Failures

The fault trees used to model the systems (top events) in the event trees should be documented. In addition, the method used to identify and incorporate passive component (pipe segment) failures into the analysis should be discussed, including the impact of each failure.

For each system model, provide: (1) a graphic representation of the logic structure (i.e., fault tree), (2) a simplified piping and instrumentation diagram or a one-line diagram with all pertinent (both active and passive) components identified including any dependencies, and

(3) a list or graphic representation of all dependencies associated with the system. The graphical representation of the fault trees should be provided in an appendix.

For the passive component (pipe segment) failures, provide a description of : (1) the method used to identify the passive failures, (2) how the passive component failures are incorporated into the analysis, (3) the direct (i.e., the system or train function lost) and indirect (e.g., pipe whip, spray impingement, and flood propagation) impacts associate with the loss of each component, (4) how the failure probability for the passive component was estimated using experience data sources, structural reliability methods, and/or expert judgment, and (5) uncertainties associated with each failure probability. *(NOTE: The NRC's preferred approach to estimating the failure probability of a pipe is the use of accepted fracture mechanics codes and operational data.* ***Use of expert elicitation should be fully documented and the results submitted to the NRC for information.*** *This enables the NRC to monitor new degradation mechanisms and to monitor consistency within the industry. The NRC recommends that the use of expert elicitation be performed by an industry group or professional society and the results incorporated into the fracture mechanics codes. This process ensures consistency in industry-wide application of RI-ISI programs.)* The following information should be provided to document the estimated failure probabilities for each component/pipe segment and structural element within the systems being addressed:

- Failure mechanism(s) that dominates the overall failure probability
- Flaw frequencies and size distribution used in the fracture mechanics calculations
- Assumptions used in calculating failure probability for every failure degradation mechanism, including the qualification of the method of analysis.
- Failure mode(s) for the component (rupture, large leak, etc.) that was identified as having safety consequences
- Method used to estimate each failure probability
- Estimated numerical mean value of failure probabilities for the identified failure mode(s) and mechanism(s) *(NOTE: Table 6.2 provides an example summary of possible methods for obtaining failure probabilities based on specified degradation mechanisms. The staff recommends that licensees provide such a table with supporting discussions.)*
- Estimated numerical mean value of each segment's CDF used in the categorization and in the ΔCDF and ΔLERF calculations
- Overall failure probabilities for each system and for each pipe segment corresponding to the total contribution from all sub elements making up the system or pipe segment being addressed
- Detailed discussion (for each system) of the major contributors to the structural failure probabilities.

Human Actions

The technique(s) used to identify and quantify human actions should be described. For each action, describe: (1) how the action was identified (e.g.,explicit identification of the immediate response action for a specific initiating event), (2) what method was used to estimate the failure probability associated with the action (e.g., THERP), (3) which performance-shaping (or performance-influencing) factors were or were not considered, (4) how the factors identified in (3) were estimated, and (5) how the effects of the factors identified in (3) were incorporated into the estimate of the action's failure probability.

Sequence Quantification

The method used to quantify the accident sequences should be described. This description should: (1) identify what software package was used to quantify the accident sequences, (2) identify the truncation limit used to eliminate sequences from the analysis, and show that the truncation limit conforms to the criteria as described in DG-1061, (3) list the failure probability or unavailability value used for each basic event in the analysis, including any uncertainty associated with the event, (4) list the core damage frequency for each sequence analyzed, and (5) present the results and discuss the implications of the uncertainty and sensitivity studies.

Uncertainty/Sensitivity Calculations

The data used in any uncertainty calculations (i.e., uncertainty distributions for basic events or input parameters) and any sensitivity calculations (e.g., giving additional or less credit for operator actions than that considered in the base case) should be provided consistent with the guidance provided in Draft Regulatory Guide DG-1061. How uncertainty was accounted for in the segment categorization, and what sensitivity studies were performed to ensure the robustness of the categorization should be described.

6.2.2.3 Contribution to Risk and Risk Importance Measures for Pipe Segments

Total CDF and LERF, prior to and following implementation of a RI-ISI program, should be documented and compared against the acceptance guidelines in Draft Regulatory Guide DG-1061 and Chapter 4 to provide assurances that pressure boundary failures associated with plant piping systems do not impose an undue increase in risk. Appropriate assumptions (i.e., no credit for ISI when categorizing, credit for ISI for total CDF/LERF considerations) used to obtain risk-categorization values for ISI should be documented.

Importance measures should be documented and shown to be in accordance with the threshold values specified in Appendix A2 of this document. *The role of the QA program and the procedure used by an appropriate panel to further review pipe segments and piping structural elements that may be inappropriately categorized as low-safety-significant should be documented to provide assurances that the PRA strengths and limitations, deterministic insights, operational insights, industry pipe failure data, and Maintenance Rule insights are taken into consideration.*

6.3 Integrated Decisionmaking Process Records

In addition to the general documentation requirements identified in Draft Regulatory Guide DG-1061, provide a description of each issue considered in the integrated decision-making process and a discussion of how the resolution of each issue impacts the original probabilistic categorization. Information should be provided in the main report.

6.4 Development of ISI Program

Describe the ISI program. The licensee's program for monitoring the performance of both HSS and LSS segments should be described. The description if the RI-ISI program should include: (1) the inspection frequency or frequencies associated with each category, (2) the method or methods of inspection associated with each element within a passive component, and (3) comparisons between existing ASME Section XI inspections and RI-ISI inspections.

The applicant should provide adequate documentation that verifies that the degradation mechanisms, postulated failure modes, and configuration of piping structural elements are incorporated in the definition of the inspection scope and inspection locations. Selected inspection locations are reviewed to confirm that stress concentration, geometric discontinuities, and terminal ends are included in the inspection program. In addition, the documentation should verify that plant-specific pipe cracking experience has been considered in selecting inspection locations. Sampling methods (e.g., the Assurance Level Sampling Method recommended by the Perdue-Abramson method are identified in Chapter 4 and Appendix 4) used to identify elements to be inspected should be documented, justified and compared to existing Section XI licensing basis requirements. The licensee needs to document if alternate methods are specified to ensure structural integrity in cases where examination methods cannot be applied due to limitations, such as inaccessibility or radiation exposure hazard. The licensee should document that its RI-ISI program continues to perform pressure tests and visual examinations of piping structural elements on all Class 1, 2, and 3 systems in accordance with ASME BVPC Section XI programs regardless of whether the segments contain locations that have been classified as high- or low-safety-significant and high- and low-failure-potential.

The licensee should document that its proposed RI-ISI inspection program and examination methods and acceptance guidelines currently included in the ASME BVPC Section XI program are used as guidance. Examination methods and acceptance guidelines should be documented to ensure compliance with the acceptance guidelines.

The procedure to evaluate pipes containing flaws that exceed the acceptable flaw standard should be documented to ensure that the techniques employed are in accordance with the acceptance guidelines.

As required by the ASME Code, a record of each inspection should be maintained in which component degradation and failures occurred and corrective action was required. Procedures should be in place which are initiated by piping failures that are detected by the RI-ISI program as well as by other mechanisms (e.g., normal plant operations, inspections, industry experience, etc.). Procedures should also exist to determine their impact on the plant PRA. Piping-specific performance data should be used to support periodic PRA and RI-ISI program updates.

The submittal should also include a proposed schedule for initiating the RI-ISI program pending NRC approval.

6.5 Implementation Plans and Schedule

The licensee's implementation plans should be provided including a proposed schedule for initiating the program pending NRC approval. Describe the process for determining when formal NRC review and approval are or are not necessary (Section 3.2). As discussed, once this process is approved by the NRC, **formal NRC review and approval are only needed when the process determines that such a review is necessary, or when changes to the process are requested.**

In addition, document the types of information that will be submitted to the NRC for information only, to enable the NRC to monitor operational experience by the industry, including new degradation mechanisms.

6.6 Quality Assurance

The NRC expects that the quality of the engineering analyses conducted to justify proposed CLB changes will be appropriate for the nature of the change. In this regard, it is expected that for traditional engineering analyses (e.g., deterministic engineering calculations) existing provisions for quality assurance (e.g., Appendix B to 10 CFR Part 50 for safety-related SSCs) will apply and provide the appropriate quality needed. Similarly, when a risk assessment of the plant is used to provide insights into the decisionmaking process, the NRC expects that the PRA will have been subject to quality control.

To the extent that a licensee elects to use PRA information to enhance or modify activities affecting the safety-related functions of SSCs, the following, in conjunction with other guidance contained in this guide, describe an acceptable way to ensure that the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 are met and that the PRA is of sufficient quality to be used for regulatory decisions:

- utilize personnel qualified for the analysis
- utilize procedures that ensure control of documentation, including revisions, and provide for independent review, validation or checking of calculations and information used in the analyses (an independent peer review can be used as an important element in this process)
- provide documentation and maintain records in accordance with the guidelines in Draft Regulatory Guide DG-1061.
- provide for an independent audit function to verify quality (an independent peer review can be used for this purpose)
- utilize procedures that ensure appropriate attention and corrective actions are taken if analyses or information used in previous decision making is determined to be in error.

Where performance monitoring programs are used in the implementation of proposed change to the CLB, it is expected that those programs will be implemented utilizing quality provisions commensurate with the safety significance of affected SSCs. An existing PRA or analyses can be utilized to support a proposed CLB change, provided it can be shown that the appropriate quality provisions have been met.

Table 6.1 Documentation Summary Table

PRA Certification	Address the adequacy of the PRA model used in the calculations. Address the acceptance guidelines in Chapter 4 of this document and in Draft Regulatory Guide DG-1061.
Failure Probability Calculations	Address the method(s) used to calculate the failure probability/frequency of a pipe element. Any use of expert elicitation should be fully documented.
Changes in CDF and LERF	Address the change in total CDF and LERF resulting from the CLB program versus the RI-ISI program.
ISI Systems	Identify all the systems inspected based on the CLB programs and compare the systems for the RI-ISI programs. For the Class 1 and 2 pipes, provide a schematic diagram identifying the CLB and RI-ISI inspection locations and frequency of inspections.
Segmentation	Identify methods used to segment piping systems.
Categorization	Identify methods used to categorize pipe segments and elements as HSS, LSS, HFP, and LFP. Identify all the HSS and HFP elements. Document additional piping elements that will undergo ISI, but are outside the scope of this document. This will eliminate future regulatory misinterpretation.
Sampling Method	Identify the method used to calculate the number of welds to be inspected. Document the method used to establish elements within a lot (e.g., use of the Assurance Level or Global statistical sampling method as described in Appendix 4, or alternative method).
Locations of Inspections	Provide a system/piping diagram that overlays the existing CLB locations of inspection and overlay the RI-ISI location of inspection. Discuss the differences.
Failure Probabilities	Identify the methods used to arrive at the failure probabilities for pipe segments.
Performance Monitoring	Discuss the performance goals and corrective action programs.
Periodic Reviews	Identify the frequency of performance monitoring and activities in support of the RI-ISI program. Address consistency with other RI programs (e.g., Maintenance Rule, IST, Tech Specs, etc.).
QA Program	Describe the QA program used to assure proper implementation of RI-ISI process and categorization and consistency with other RI programs.
Expert Elicitation	Identify any use of expert elicitation used to estimate a failure probability. Address the reasons why an expert elicitation was required, provide all supporting information used to by the experts, document the conclusions, and address how the results will be incorporated in an industry data base or computer code.
Each weld to be inspected	Identify: 1. the NDE method to be used 2. the applicable degradation mechanism to be inspected, and 3. the frequency of inspection
Compliance with Regulations	Verify compliance with applicable regulations.
Defense in Depth	Address any impact on defense in depth
Safety Margins	Confirm adequate safety margins exist.
Implementation and Monitoring Program	Address the Acceptance Guidelines outlined in Chapter 5 of this Reg Guide.

Table 6.2 Example Summary of Methods Used to Estimate Pipe Failure Probabilities for Risk Categorization

Failure Mechanism			Methods for Estimating Probability		
Name of Mechanism	Contributing Factors	Failure Mode	Stainless Steel	Carbon Steels	Other Materials
High Cycle Fatigue	Thermal Striping Flow Induced Vibration Mechanical Vibration	Crack Initiation	Code Name	Code Name	Failure Database
		Crack Growth	Code Name	Code Name	
Low Cycle Fatigue	Thermal Stratification Heat-up and Cool-down Thermal Cycling	Crack Initiation	Code Name	Code Name	Failure Database
		Crack Growth	Code Name	Code Name	
Corrosion Cracking	Coolant Chemistry Crevice Corrosion Susceptible Material High Stresses (Residual, Springing)	Crack Initiation	Code Name	Not Applicable	Failure Database
		Crack Growth	Code Name	Not Applicable	
Wastage	Flow Accelerated. Corrosion Microbiologically Ind. Corr. Pitting and/or Wear	Wall Thinning	Name of Code	Name of Code	Failure Database
Other Mechanisms	Creep Damage Thermal Aging Irrad. Embrittlement	Miscellaneous Modes	Failure Database	Failure Database	Failure Database

REFERENCES[1]

1. USNRC, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," Federal Register, Volume 60, p. 42622, August 16, 1995.

2. USNRC, "Framework for Applying Probabilistic Risk Analysis in Reactor Regulation," SECY-95-280, November 27, 1995.

3. Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," 10 CFR Part 50, current version.

4. American Society of Mechanical Engineers, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code, 1989 Edition, New York.

5. American Society of Mechanical Engineers, "Evaluation of Inservice Inspection Requirements For Class 1, Category B-J Pressure Retaining Welds in Piping," ASME Section XI Task Group on ISI Optimization, Report No. 92-01-01, Revision 0, December 1994.

6. S. R. Gosselin et al., "Risk-Informed Inservice Inspection Evaluation Procedure," Electric Power Research Institute (EPRI), TR-106706, June 1996.

7. WCAP-14572, "Westinghouse Owners Group Application of Risk-Based Methods to Piping Inservice Inspection," March 1996.

8. T.V. Vo, H.K. Phan, B.F. Gore, F.A. Simonen, S.R. Doctor, "A Pilot Application of Risk-Informed Methods To Establish Inservice Inspection Priorities for Nuclear components at Surry Unit 1 Nuclear Power Station," USNRC, NUREG/CR-6181, Revision 1, February 1997.

9. USNRC, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," Draft Regulatory Guide DG-1061, June 1997.

10. USNRC, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division I," Regulatory Guide 1.147, Revision 11, October 1994.

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[1] Copies of Commission policy statements, EPRI and WCAP reports referenced herein are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

Copies of NUREGs are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-2249); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

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11. USNRC, "Design and Fabrication Code Case Acceptability, ASME Section III, Division I," Regulatory Guide 1.84, Revision 30, October 1994.

12. USNRC, "Materials Code Case Acceptability, ASME Section III, Division I," Regulatory Guide 1.85, Revision 30, October 1994.

13. USNRC, "An Approach for Plant-Specific Risk-Informed Decisionmaking: Inservice Testing," Draft Regulatory Guide DG-1062, June 1997.

14. USNRC, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance," Draft Regulatory Guide DG-1064, June 1997.

15. USNRC, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Draft Regulatory Guide DG-1065, June 1997.

16. USNRC, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," Standard Review Plan, NUREG-0800, Draft Chapter 19, March 1997.

17. U. S. Nuclear Regulatory Commission, "Standard Review Plan for Risk-Informed Decisionmaking: Inservice Inspection of Piping," Standard Review Plan, NUREG-0800, Draft Section 3.9.8, September 1997.

18. USNRC, "Risk-Informed Inservice Testing Applications," Standard Review Plan, NUREG-0800, Draft Section 3.9.7, March 1997.

19. USNRC, "Risk-Informed Decisionmaking: Graded Quality Assurance," Standard Review Plan, NUREG Draft Section 16.1, March 1997.

20. USNRC, "Risk-Informed Decisionmaking: Technical Specifications," Standard Review Plan, NUREG-0800, Draft Chapter 16.1, March 1997.

21. USNRC, "The Use of PRA in Risk-Informed Applications," Draft NUREG-1602, June 1997.

22. USNRC, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Regulatory Guide 1.70, Revision 3, November 1978.

Appendix 1: PROBABILISTIC STRUCTURAL MECHANICS COMPUTER CODES FOR ESTIMATING FAILURE PROBABILITIES

A1.1 Introduction

This regulatory guide does not require or endorse particular computer codes or preclude the use of alternative codes to those cited here as examples. Nevertheless, the use of validated computer codes is recommended for estimating failure probabilities. It is anticipated that use of validated and controlled computer codes will lead to a more efficient and timely regulatory review.

In all applications the computer codes and associated structural reliability and risk assessment (SRRA) models and methodology should be documented and/or referenced. Such documents should identify the failure mechanisms modeled, describe the underlying analytic/engineering models, identify the parameters that are simulated as random variables, describe the input for these variables, and describe the numerical methods (e.g., Monte Carlo simulation) used to calculate failure probabilities. New computer codes should be validated by comparison with results from other generally accepted and documented codes, including applicable data.

Structural mechanics computer codes are valuable tools for estimating failure probabilities of piping components. Such codes can evaluate the impacts of parameters related to component design, stresses, operating conditions, material characteristics, and fabrication practices on failure probabilities. Predictions of these models can be useful in estimating both absolute and relative values of structural failure probabilities. Structural mechanics computer codes also predict the progress of degradation (e.g., crack growth) with time, and thereby provide a basis for selecting appropriate inspection intervals. Figure A1.1 illustrates

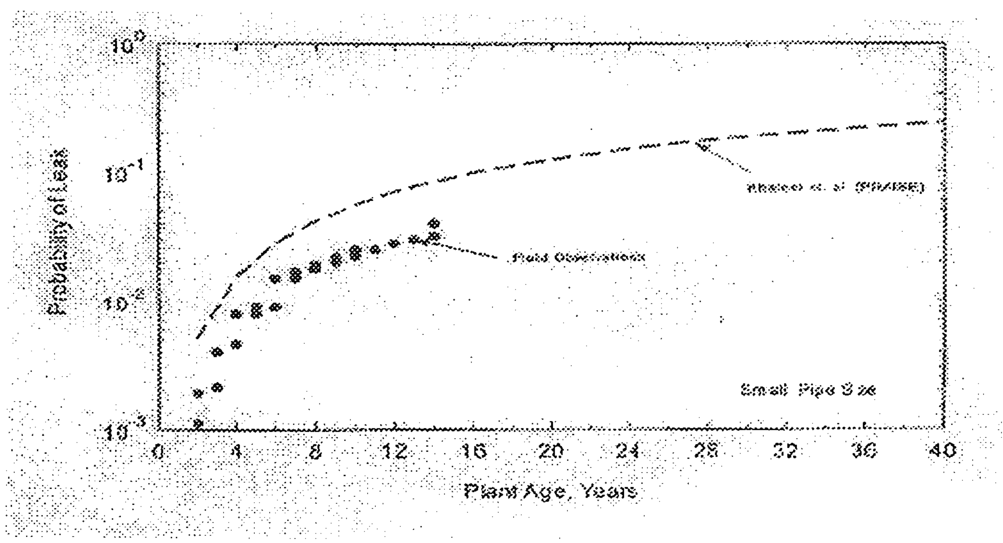

Figure A 1.1 Stress Corrosion Cracking PRAISE vs FieldData

the capability of a structural mechanics computer Code. This Appendix provides the present criteria by which the NRC will judge acceptability of codes for use in estimating failure probabilities of piping components and a detailed discussion of selected structural reliability Code issues.

A1.2 Areas of Structural Reliability Code Review

The areas of review of the structural mechanics computer codes include the following:

- Addressing the failure mechanisms under consideration.
- Addressing the structural materials and component geometries under consideration.
- Assuring that the structural mechanics models are based on pertinent engineering principals and approximations used in the models are appropriate.
- Assuring that the probabilistic aspects of the structural mechanics models address those parameters with the greatest variability and uncertainty.
- Assuring that the model calculates failure probabilities using realistic considerations, without conservative or non-conservative assumptions that would inappropriately bias risk-based categorizations towards particular systems, failure mechanisms or operating conditions.
- The numerical methods, including Monte Carlo (or appropriate) simulations and importance sampling techniques.
- The inputs to the codes are within the knowledge base of the experts applying the Code.
- Internally assigned (hardwired) parameters and probability distributions are documented and supported by available data and knowledge base.
- Documentation of technical bases of the model is available for peer review.
- Limitations of the Code are identified and cautions provided for cases when alternative structural mechanics models and/or other estimation methods should be used.
- Benchmarking with structural mechanics codes considered acceptable by the NRC such as pc-PRAISE.
- Calculated failure probabilities are consistent with historical failure rate data from plant operating experience.
- The development of the computer Code, documentation, and application are consistent with quality assurance requirements in Appendix B to 10 CFR Part 50.

The evaluation should identify limitations of the codes, and should establish the appropriate role (absolute or relative probabilities) for the calculated failure probabilities obtained from the codes.

A1.3 Selected Structural Reliability Code Issues

A1.3.1 Loads and Stresses

Inputs for loads and stresses to SRRA models should address both conditions anticipated during the design of the systems, and unanticipated loads that have become known only through operating experience at the plant of concern or at other similar plants. SRRA evaluations should use realistic input for loads and stresses and for occurrence rates of plant transients.

It should be noted that calculated stress levels in piping stress reports are generally based on conservative analysis assumptions. It is appropriate in the evaluations to treat such calculated stresses as upper bounds on uncertainty bands for the actual operating stresses, with expected values being lower than those cited in stress reports. The exception may be stresses due to internal pressures which are subject to less uncertainty in calculations than other stresses such as the stresses from restraint of thermal expansions.

Loads and transients should be based as much as possible on actual operating experience rather than on design or bounding conditions. Loadings having low estimated probabilities of occurrence should not be neglected but should be addressed explicitly in a probabilistic manner in the evaluations. Given the computation effort of probabilistic calculations, the loading cases should be limited to those that have the largest potential contributions to component failure probabilities. Insights from engineering calculations along with bounding estimates of loading frequencies and conditional failure probabilities should be used to eliminate from consideration those load cases and/or transients with little potential contribution.

A1.3.2 Vibrational Stresses

Uncertainties associated with high cycle fatigue stresses, such as from mechanical vibration and thermal fatigue, should be given special consideration in calculating failure probabilities. High cycle fatigue applies whenever the number of stress cycles is sufficiently large such that cracks grow through the pipe wall thickness within a small portion of the design life, given that the cyclic stress levels exceed the threshold ΔK for fatigue crack growth.

The following factors govern the growth of such cracks:

Threshold ΔK - In applications of the pc-PRAISE Code, published data have been used to estimate appropriate inputs for ΔK_{th} for stainless steels (4.6 ksi/in for an R-Ratio = 0.0) whereas ΔK_{th} = 0.0 has been assumed for ferretic steels in accordance with the ASME Section XI.

R-Ratio - The structural mechanics models and inputs to these models should account for the impact of mean stresses on reducing the governing values of ΔK_{th}.

Vibrational Stress Levels - Because vibrational stresses are random in nature, the levels of these stresses are difficult to estimate in practice. Such stresses tend to be greatest for smaller pipe sizes. The guidelines developed on the pilot application of risk-informed inservice inspection to the Surry-1 plant provide an acceptable basis for estimating vibrational stresses, as follows, where the cyclic stresses are given in terms of a stress amplitude (i.e., ½ (σ_{max} - σ_{min}):

Pipe Diameter inch	Upper Bound Cyclic Stress, ksi	Median Cyclic Stress, ksi
1.0	6.0	3.0
5.0	2.5	1.25
> 10.0	1.0	0.5

Occurrence Rate - In most cases the probability that the vibration stress will occur is relatively low, and also the duration of these stresses may be limited to the time periods of intermittent operation of vibrational sources such as pumps. It is acceptable to adjust calculated failure probabilities to account for these uncertainties.

A1.3.3 Residual Stresses

Residual stresses can be the major factor in the growth of cracks by the mechanism of stress corrosion cracking, and can also enhance crack growth by fatigue by increasing the level of mean stress as characterized by the calculated R-Ratio. Guidelines developed on the pilot application of risk-informed inservice inspection to the Surry-1 plant provide an acceptable basis for estimating residual stress levels. *These guidelines recommend a lognormal distribution with maximum stresses distributed by two standard deviations, corresponding to 90 percent of the material flow stress.*

These guidelines quantified the uncertainties in welding residual stresses, and addressed the possibilities that residual stresses can attain yield strength levels or can be essentially zero in other cases. Statistical distributions to describe uncertainties in residual stresses should be truncated at the material flow strength (average of yield and ultimate strengths). Levels as high as 90% of the flow strength should have relatively low probabilities corresponding, for example, to a 90th percentile of a lognormal distribution.

A1.3.4 Preservice Inspection

The effects of preservice inspections by such methods as ultrasonics and radiography should be included either explicitly or implicitly in the calculation of failure probabilities. In most cases, such inspections are addressed implicitly through their effects on the estimated number and sizes of initial fabrication flaws. In such cases, the simulation of preservice inspection in the structural mechanics model is inappropriate since such a simulation would result in double counting of the effects of preservice inspections.

A1.3.5 Proof Test

It is recommended that the effects of proof tests performed after fabrication but before plant operation be included in the probabilistic structural mechanics calculations. Simulated failures that occur during such proof tests should not be included in the failure probabilities addressed by the inservice inspection program.

A1.3.6 Leak Detection

In calculating pipe degradation (leaks to ruptures) probabilities, the effects of leak detection from through wall flaws should be addressed, and pipe failures that would be detected by observations of leakage should not be included in the calculation of leak/rupture probabilities. Leak detection can be due to explicit leak monitoring measures, or to detection of leaks by plant staff in the course of plant walkdowns or system testing. Leak rate calculations and leak detection thresholds used in the calculations of pipe failure probabilities should be documented and justified. The leak rate model in (Ref. 1) is an acceptable basis for predicting leak rates from through wall cracks.

A1.3.7 Failure Modes (Leak Versus Break)

Failure probability calculations should address the failure modes of concern to the risk-categorization process, and should include the categories of small leaks (through-wall cracks), large leaks that disable a system (labeled as a ***disabling leak***), and pipe breaks. The leak rate for the disabling leak category should be based on the consequences considerations identified in the plant PRA and safety analyses reports.

The methodology identified in Reference 1 is an acceptable basis for predicting leakage through cracks for use in calculations of large leak probabilities and for simulating the impact of leak detection of pipe failure probabilities. An example of an acceptable implementation of this leak prediction methodology is currently part of the pc-PRAISE Code.

A1.3.8 Service Environment

The service environments that affect both corrosion rates and crack growth rates should be addressed in the SRRA models. Such environments are often described in the SRRA models in terms of discrete categories such as air versus water or high versus low oxygen environments. The selected environments used in each SRRA calculation should be documented along with the rationale for the selections. Data bases used to develop distributions of crack initiation and crack growth rates should represent the range of operating conditions expected for the structural component being addressed by the SRRA models. In those cases for which the service environment is subject to large uncertainties and variations, the SRRA models can be structured to simulate these variations, and to use the models to simulate the effects of these variations on the resulting failure probabilities.

A1.3.9 Initial Flaw Size Distributions

Stresses at most pipe locations are sufficiently low such that the calculated failure probabilities are essentially zero, unless there is an initial fabrication flaw present at the structural location of concern (e.g., weld). Therefore, SRRA models should simulate the number, size, and location of such fabrication flaws. These characteristics should be estimated and should be described statistically with distributions that are appropriate for the material, wall thicknesses, welding practices, and inspection procedures for the specific location of interest.

The documentation of the SRRA calculations should describe and justify the number and sizes of defects that were assumed. The model developed in (Ref. 2) for simulating fabrication defects is an acceptable method for estimating initial flaw densities and size (depth and length) distributions. Applications of this model to pipe welds and data from detailed examinations of actual welds, suggest flaw densities of one or more defects per weld, but with less than ten percent of these flaws being inner surface connected. The flaw depth distributions from this model can be approximated by a lognormal distribution with the mean flaw depth being on the order of the thickness of one weld bead.

A collection of flaw distribution calculations has been performed with the model in Reference 2 to support a pilot application of risk-informed inservice inspection for the pilot plant. These calculations addressed a wide range of welds, and the results provide an acceptable basis for estimating the numbers and sizes of flaws in most cases of piping welds. A future report will describe details of these calculations along with trend curves that describe flaw densities and flaw depth distributions as a function of pipe wall thickness, material (stainless versus ferritic steel), and post weld inspection (i.e., with or without radiographic examination). The results indicated the following trends:

- Flaw densities are best characterized in terms of flaws per unit length of weld rather than in terms of flaws per unit volume of weld material. This measure of flaw density can be conveniently described by curves giving flaw density as a function of pipe wall thickness.

- Most fracture mechanics models conservatively assume that all flaws are surface breaking flaws at the pipe inner surface. Therefore, only a small fraction of the total flaw density should be included in the flaw density used in fracture mechanics calculations, in order to account for the fact that buried defects are less likely to cause failures than surface breaking defects.

- Radiographic inspection has a significant impact on the number (density) of flaws, but relatively little impact on the size distributions of the flaws.

- The number (density) of flaws is similar for stainless and ferritic steels, but the probability of a very deep flaw being present is greater for welds in ferritic steel piping.

- For the cases of manual metal arc and tungsten inert gas welding processes, the number of flaws and the sizes of these flaws are insensitive to the particular process used to make the weld.

- The model in Reference 2 addresses generalized basis for estimating the number and sizes of flaws in pipes, and is a method that covers a wide range of pipe sizes and fabrication practices. **The final selection of the number and sizes of flaws has to be documented and submitted for NRC review.**

A1.3.10 Flaw Initiation

Operating experience shows many cases whereby flaws have initiated during service due to such mechanisms as stress corrosion cracking or fatigue associated with cyclic stresses (e.g., thermal fatigue). Unless service induced cracks can be justified to be negligible contributors to failure probabilities, the SRRA models for components should account for the potential contributions of initiated cracks to failure probabilities. These contributions should be added to the contributions from initial fabrication cracks. Documentation of SRRA calculations should describe and justify the explicit or implicit approaches taken to address crack initiation.

Various direct and indirect approaches can be used to account for crack initiation. The pc-PRAISE Code provides an approach for simulating the initiation of IGSCC cracks. SRRA models for the mechanism of fatigue, including pc-PRAISE, do not yet simulate the contributions of fatigue crack initiation, although such effects may be approximated through inputs regarding the number and sizes of very small inner surface defects. For example, (Ref. 3) assumed each weld had one small inner surface flaw with the depth described by a uniform distribution ranging from 0.002 to 0.010 inch.

A1.3.11 Crack Growth Rates

The prediction of crack growth rates by fatigue and by stress corrosion cracking is a critical step in the calculation of piping failure probabilities. Large experimental efforts are required to perform crack growth tests, and to develop predictive equations that correlate data bases from laboratory tests. It is recommended that probabilistic structural mechanics codes make use of recognized and accepted correlations.

The correlations described in the documentation for the pc-PRAISE Code provide an acceptable basis for predicting crack growth rates for stainless and ferritic steels. These equations should be applied only for the relevant materials and service conditions. Other crack growth relationships should be used to address materials and service conditions outside the scope of the equations developed for pc-PRAISE. Such equations should be justified on the basis of measured crack growth rate data, the effects of mean stresses or R-Ratio (i.e., K_{min}/K_{max}), and should address threshold ΔK levels.

A1.3.12 Material Property Variability

Variability and uncertainties in material properties can be simulated by the SRRA models. Only those properties that have significant variability and/or for which the failure probabilities are particularly sensitive need to be simulated. Other properties can be treated as deterministic inputs. Typical variables that should be simulated in the probabilistic model include material strength levels, fracture toughness, and crack growth rates due to fatigue and/or stress corrosion cracking. Documentation for SRRA calculations should state which material property inputs were treated as deterministic parameters, and which parameters were simulated in the probabilistic model. The bases for assigning mean values, standard deviations, and distribution functions should be documented.

A1.3.13 Comparison with Service Experience

The numerical estimates of failure probabilities from SRRA models should be compared with the service experience for the structural components being addressed. In most cases the predictions will give very low leak and rupture probabilities. Calculations should be compared for consistency with the plant specific experience regarding leaks and detected degradation. Since the failure probabilities for specific structural locations are almost always too small to permit meaningful comparisons, it is recommended that comparisons of calculations with service experience also be made for the total failure probability for all components for each system. Data on pipe rupture occurrences will seldom, if ever, be available. Therefore it is more likely that data on leaks and detected material degradation will provide evidence that the component designs and/or operational conditions are sufficiently severe to enhance the probability for pipe ruptures. Industry wide experience for similar materials, designs, and operating conditions should also be used as an additional basis to check the credibility of SRRA calculations.

A1.3.14 Effects of Inservice Inspection (CDF vs Importance Measure Calculations)

As documented in the body of this report, one acceptable approach to RI-ISI programs consists of two components. The first component is the quantification of the total CDF (or ΔCDF) that results from the proposed change in the ISI program. The second component is to categorize a pipe segment as high- or low-safety significant.

For calculating the total CDF (or ΔCDF) from changes to the programs, the calculated pipe failure probabilities should be consistent with the operations and procedures of the plant. That *includes effects of the inservice inspection programs.*

However, when calculating failure probabilities for use in establishing risk importance measures to be used in component categorization scheme, the analyses should assess both the effects of *implementing inservice inspection programs* (ISI) and the effects of *no inservice inspection programs.*

To support the development of effective ISI programs, SRRA modeling should also be applied with the simulations of inspections to evaluate alternative inspection strategies. Two critical inputs to such SRRA calculations are the inspection method (as characterized by a probability of detection curve), and the time interval between the inservice inspections. Inputs for detection probabilities should be relevant to the materials, component geometries, and degradation mechanisms for the structural location being addressed. Inputs for detection probabilities should be documented and justified.

A1.3.15 Cumulative Effects of Repeated/Periodic Inspections

Failure of an inspection to detect a particular flaw is often due to physical factors such as crack tightness, crack orientation, etc. Such factors can prevent detection regardless of how many inspections are performed. Calculations of the benefit of inservice inspection should assume that nondetection of a particular flaw in one trial will be correlated with the outcome (nondetection) during a subsequent inspection. Overly optimistic estimates of ISI effectiveness can be predicted if the alternative assumption of independent outcomes is assumed.

A1.3.16 Review and Treatment of Uncertainties

Uncertainties in modeling assumptions and inputs to calculation should be identified and quantified. Figure A1.2 identifies parameters that should be reviewed for their impact on the calculated uncertainties. The use of conservative assumptions and inputs to address uncertainties should be avoided since inflated values of failure probabilities can give unwarranted inspection priority to components at the expense of other components that may actually have greater safety significance. The uncertainty distributions for the calculated failure probabilities should be addressed in the PRA analysis.

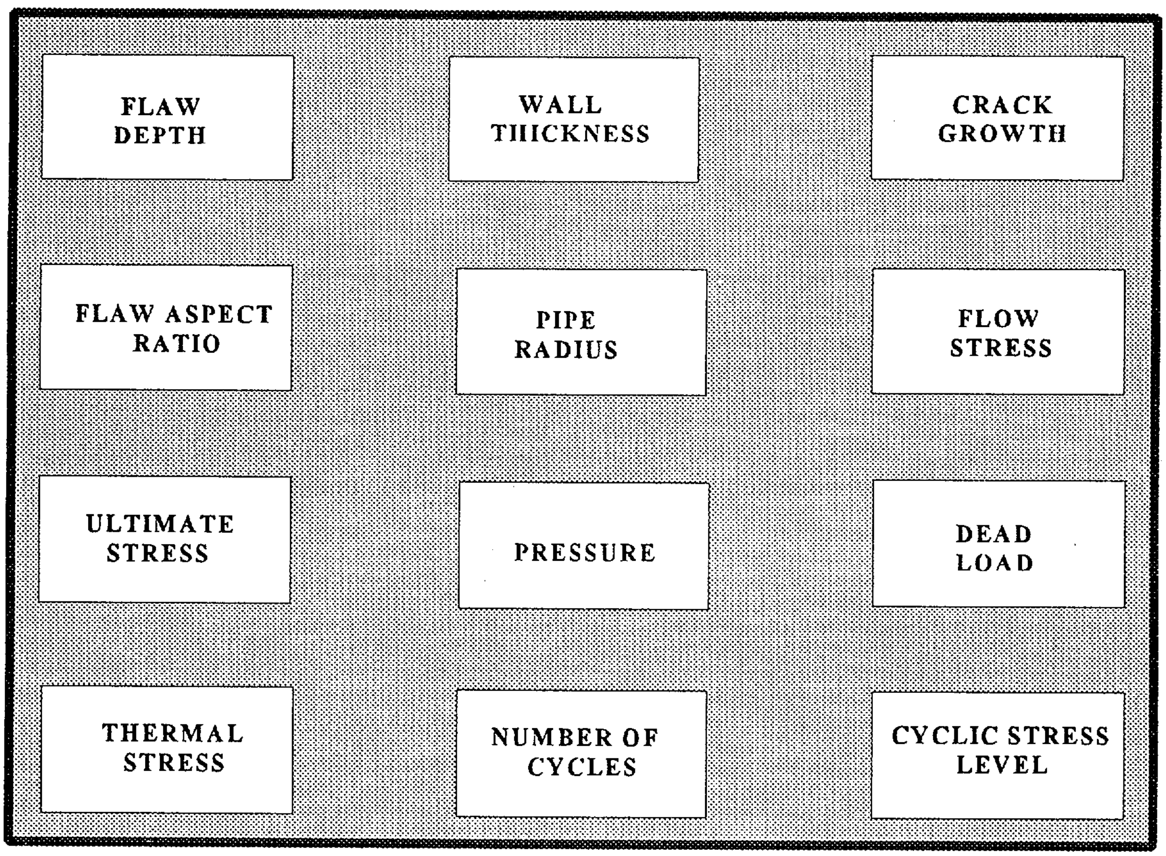

Figure A1.2 Example Of Major Parameters That Can Influence Calculated Pipe Failure Probability

A1.3.17 Realistic Versus Conservative Calculations

Structural reliability calculations should be based on realistic considerations rather than assumptions and inputs that ensure conservative estimates. The introduction of conservatisms on a selective and/or nonuniform basis for particular components or particular failure mechanisms will have the undesired effect of biasing the importance categorizations. The result can be inappropriately low categorizations for some pipes that are truly more risk significant. The use of conservative assumptions (to address uncertainties) should be part of the sensitivity studies. Results of such sensitivity studies should go through a rigorous quality assurance (QA) process or an expert panel as a potential basis for adding locations to the category of high-safety-significant ISI locations.

Although there may be large uncertainties in the estimated failure probabilities, the relative values (e.g., from location-to- location in a given system) are generally calculated with a higher level of confidence. However, even relative values can become increasingly uncertain, when comparisons are made from one system to another (due to different failure mechanisms, pipe sizes, materials/fabrication practices, and operating environments), and for comparisons of different failure mechanisms within a given system. Sensitivity studies can be useful in evaluating potential impacts on risk categorizations due to systematic biasing of estimated failure probabilities from one system to another.

A1.3.18 Consideration of Failure Mechanisms

The failure mechanisms of most concern for reactor piping are the initiation and growth of fatigue and/or stress corrosion cracks, and wall thinning by erosion corrosion. Each of these mechanisms will be addressed by separate structural mechanics models, either within a single computer Code or by separate computer codes. The mechanism of fatigue is a concern for both ferritic and stainless steel piping. Stress corrosion cracking is limited to stainless steel piping, whereas erosion corrosion needs to be addressed only for ferritic steels having susceptible material compositions and operating under specific flow conditions.

Calculations of failure probabilities are contingent on the availability of a computer Code that addresses the dominant failure mechanism for the piping segment of concern. The first decision, before any calculations are performed, is that of the adequacy of the selected Code to model the identified failure mechanism(s). The model must not only address the relevant failure mechanisms, but the scope of the model must cover the specific material type and grade, and the relevant operational conditions (temperature, chemical environment, flow velocities, material heat treatment, etc.).

Inappropriate applications of structural mechanics models will result in calculated failure probabilities of no value for risk informed purposes. Submittals should provide justification that the scope of selected computer codes addresses the components, operating conditions, and failure mechanisms of concern. Alternative methods should be used to estimate failure probabilities when there are no applicable computer codes.

A1.3.19 Materials Considerations

The governing failure mechanisms and associated failure probabilities are impacted by the particular types and grades of materials used to fabricate the pipe of concern. Some material considerations, such as yield and ultimate strength levels, are addressed by user provided inputs to the probabilistic calculations. Because materials related inputs are seldom known with precision, computer codes must simulate the uncertainties in these input parameters which are associated with the scatter in material properties.

Probabilistic structural mechanics codes must address material parameters that are beyond the knowledge base of the expected Code users. For example, predictions of growth rates for fatigue and stress corrosion cracks are a challenge even to researchers working in this specialized area of fracture mechanics. Therefore, the users of SRRA codes must usually rely on the validity of default or hardwired values for crack growth parameters, or use the guidance and/or examples given in documentation for the computer codes.

Acceptable SRRA computer codes should provide technically sound and documented approaches to predict crack growth rates. Applications of crack growth relationships should not require specialized knowledge of fracture mechanics, but should permit sufficient flexibility to permit more knowledgeable users to refine predictions of fracture mechanics models.

A1.3.20 Consideration of Component Geometries

Probabilistic structural mechanics codes are generally based on Monte Carlo simulations, which involve repeated deterministic calculations to calculate failure probabilities. The large number of calculations dictates that the models be limited to relatively simple geometries, such as straight lengths of pipes with circumferential or axial cracks. Applications of the simplified models to more complex geometries involves assumptions and approximations. For example, inputs can specify stresses for simplified models to numerically approximate the level and distribution of stress from a more detailed stress calculation performed with a finite element Code outside the framework of the probabilistic model.

Acceptable SRRA codes should address appropriate geometric considerations for the failure mechanisms of concern. For fatigue and stress corrosion mechanisms, the models should address internal surface circumferential cracks, with the ability to approximate the axial crack case. Erosion corrosion models should address piping failures associated with enhanced levels of hoop stress due to wall thinning.

A1.3.21 Deterministic Structural Mechanics Models

Since probabilistic models are based on the repeated application (e.g., Monte Carlo simulations) of deterministic models, the validity of predicted failure probabilities depends on the correctness of the underlying deterministic model. As indicated above, deterministic models in probabilistic structural mechanics codes are generally limited to relatively simple structural geometries, with effects of more complex geometries addressed through suitable manipulations of the inputs that prescribe the levels and distributions of the stresses.

The critical features of the deterministic fracture mechanics models are as follows:

- calculation of crack tip stress intensity factors as function of crack depth, crack length, crack orientation, applied stress level, through wall variation in stress, and residual stresses

- models for predicting subcritical crack growth (or wall thinning) as a function of stress intensity factors, material properties, and operating conditions (temperature and chemical environment)

- models for predicting critical crack sizes and critical depths of wall thinning that correspond to piping failure by leaks or breaks

A1.3.22 Selection of Probabilistic Variables

Once the deterministic structural mechanics model has been defined, it is then necessary to select those variables that will be simulated in the probabilistic calculations as opposed to those variables that will be treated as single valued deterministic parameters.

Variables selected for simulation should be limited to those with the most significant uncertainty due both to lack of knowledge and/or limited base of data or due to known variability (as indicated by scatter in data). In probabilistic structural mechanics calculations a typical division between deterministic and probabilistic variables is shown in Table A1.1.

Table A1.1 Determination vs Probabilistic Variables

Deterministic Parameters	Probabilistic Parameters
Pipe Diameter Initial Pipe Wall Thickness Location of Fabrication Flaws (Surface or Buried) Chemical Environment (Air, Water, Oxygen Content, etc.) Operating Temperature	Stress Level Material Strength Fracture Toughness Crack Growth Rates Number of Fabrication Flaws Sizes of Fabrication Flaws (Depth and Length)

In many cases it will be necessary and appropriate to address certain probabilistic variables outside the framework of the structural mechanics Code. For example, the probabilities or frequencies of loading cases (e.g. pressure temperature transients for pressurized thermal shock accidents) may be the subject of ongoing detailed evaluations. Such decomposition of the failure probability calculations into a set of conditional failure probability cases can also facilitate sensitively calculations and the independent reviews of failure probability estimates.

Documentation for probabilistic structural mechanics codes should clearly state which variables are treated as deterministic parameters, and which variables are simulated in the probabilistic calculations. The documentation should also state the distribution function(s) used to describe each simulated variable, along with user defined parameters (e.g. mean, standard deviation, truncation of distribution tails, etc.) and any distribution function that has been "hardwired" as part of the probabilistic model.

A1.3.23 Numerical Methods

The accuracy and computational efficiency of computer codes are impacted by the numerical approaches used to implement the probabilistic structural mechanics model. The most commonly used approach is that of a Monte Carlo simulation, since it has general applicability to complex physical phenomena involving interactions between variables and discontinuous behaviors. A Monte Carlo approach is also relatively straight forward to program and does not require advanced mathematical knowledge of probabilistic and statistical methods. Resulting computer codes will be relatively robust, but may lack in the numerical efficiency desired for the calculations where very low values of failure probabilities are of interest.

There are a number of acceptable numerical techniques to enhance the speed of failure probability calculations. For example, the pc-PRAISE Code (Ref. 4) uses stratified sampling, and the Westinghouse structural mechanics Code (Ref. 5) uses importance sampling. In both cases the more sophisticated sampling procedures are used as an enhancement to the underlying Monte Carlo simulation.

Care must be exercised in applications of enhanced sampling methods to ensure that the methods are correctly implemented and are not applied to model situations with complex probabilistic structures. For example, stratified sampling is precluded in the pc-PRAISE Code for stress corrosion cracking because pc-PRAISE models multiple crack initiation sites and treats crack interactions and coalescence. In all cases, the validity of enhanced sampling methods and their implementation should be verified by comparisons of numerical results with those from conventional Monte Carlo simulations.

The documentation for the computer codes should include guidance on selecting the user inputs that control sampling procedures. Complex sampling procedures should be avoided if an unreasonable level of statistical insight is required on the part of new, occasional, and inexperienced users of the Code.

A1.3.24 Assignment of Input Parameters

The user of a probabilistic structural mechanics code has the responsibility of assigning the inputs for the calculations that address particular pipe segments. This task has as large an impact on the credibility of the failure probability estimates as the development of the computer Code itself. Much of the discussion in this appendix bears directly or indirectly on issues related to the inputs for the calculations.

It is not the intent here to repeat or summarize guidance provided elsewhere in this Regulatory Guide. However, the following steps will further the objective of consistent values for the input parameters.

- Documentation for the Code should provide detailed guidance for assigning input parameters.

- Example calculations should be presented along with a narrative describing considerations used to assign input parameters and the sources of data that support the assigned numerical values.

- Developers of the codes should provide training sessions for new users of the Code, should be available for consultation, and should organize workshops to permit interactions among the Code users.

- The Code documentation should provide guidance of a more prescriptive nature for those input parameters (e.g. flaw size distributions, crack growth equations, fracture toughness correlations, etc.) that are either outside the expected knowledge base of the Code users or where the expected variations in judgments made by several users could result in differing/inconsistent inputs.

- To further the objectives of the above bullet, a consensus process should be followed to develop the guidelines on suitable numerical values for the more difficult-to-define input parameters used in the structural reliability calculations. The objective would be to enhance the level of uniformity and consistency in the calculated failure probabilities that are used to support risk-informed inspections.

A1.3.25 Supporting Data Bases

Certain inputs for probabilistic calculations are outside the knowledge base of expected users of the SRRA codes. Examples of such inputs are flaw density and size distributions and material characteristics related to crack growth rates and erosion corrosion rates. An essential part of developing a Code is to make a selection of suitable inputs available to the Code user, either as a menu of "hardwired" options or in the user documentation as recommended default values for consideration by the user.

A major part of developing a probabilistic structural mechanics Code should be the compilation of data bases for use in quantifying parameters of the model. An equally important task is the development of statistical correlations of the data into such a form that it is suitable for the computation models. Documentation of computer codes should describe the data and statistical correlations used to support the model along with the approaches used to derive the statistical correlations.

A1.3.26 Documentation and Peer Review

Probabilistic structural mechanics computer codes should be documented and subject to peer review prior to widespread dissemination and application to risk-informed inspection. The scope of the recommended Code documentation is addressed throughout this appendix.

Documentation is essential to permit peer reviews of the technical basis for the structural mechanics codes, and is also essential to permit correct and appropriate applications of the Code by the user community. Part of the peer review process should be trial calculations by independent outside users of the codes. Such applications will result in improved insights regarding the strengths and limitations of the computer codes and their associated documentation.

A1.3.27 Identification of Code Limitations

It is essential to identify limitations of structural mechanics models to avoid inappropriate applications or false levels of confidence in calculated failure probabilities. Guidance should identify situations for which the codes are expected to give the most accurate absolute values for failure probabilities, as well as other situations for which the calculations should be used as indication of relative failure probabilities.

The Code documentation should state the assumptions made in the structural mechanics models, and the expected impacts of these assumptions on the calculated failure probabilities. Limitations should be specifically stated regarding failure mechanisms addressed along with the applicable operating conditions in terms of temperatures, operating environments, material types.

A1.3.28 Benchmarking with Other Computer Codes

The predictions of probabilistic structural mechanics codes should, whenever possible, be benchmarked against results from other computer codes that have gone through peer review and validation, such as the pc-PRAISE Code. Differences in calculated failure probabilities should be identified, and the reasons for any significant differences in the numerical results should be reconciled. Acceptance of a particular Code in the light of numerical differences should be technically justified, if these differences are due to improved modeling approaches or improved sources of supporting data.

Advances continue to be made in the field of probabilistic structural mechanics. Therefore codes will often not be available to support benchmarking of new and improved computer codes. In these cases, other approaches can accomplish the benchmarking objectives as follows:

- A matrix of demonstration calculations to cover a wide range of input parameters which result in predicted failure probabilities covering the range from very high (i.e. approaching unity) to very low (e.g less than 10^{-8} over the design life of the component),

- Sensitivity calculations covering all input parameters to demonstrate that changes to input values result in consistent changes in calculated failure probabilities,

- Selected benchmarking calculations that address consistency with operating experience in accordance with the discussion of Section A1.3.29 below. These calculations should cover both normal or design conditions, and also cases of actual (but unanticipated) operating conditions that have resulted in component failures or service related degradation.

A1.3.29 Consistency with Operating Experience

Failure probabilities for most structural components are very low, such that failures are not expected to occur over the intended operating life of individual components. Few (if any) failures are expected to occur even if a large population of similar components is considered. This sparsity of data on actual failures, provides the incentive to use probabilistic structural mechanics models as a method to estimate failure probabilities. In this regard, probabilistic models predict component failure probabilities making use of the better known data on the individual variables (e.g. flaw occurrence rates, flaw sizes, crack growth rates, material strengths and fracture toughness properties) that govern the component failure probabilities. However, there are large uncertainties regarding the assumptions and input data. Therefore, predictions from probabilistic structural mechanics models should be compared for consistency with trends from operating experience.

The following approaches are recommended for establishing the consistency of model predictions with the limited amount of data regarding failures available from operating experience:

- In many cases there will be no reported failures corresponding to the conditions addressed by the structural reliability calculations. The calculations can be validated in the sense that the predicted failure probabilities are indeed very low, and are shown not to be inconsistent when no failures have occurred for a known population of components over a defined span of operating years.

- While operating experience may show no failures by the mode of pipe rupture, the data may indicate other more common occurrences of pipe leaks and/or of detected cracks. Such data should be used for consistency checks of calculated probabilities for pipe leaks and for crack growth to detectable depths. The occurrences of stress corrosion cracking and erosion corrosion at nuclear power plants have been relatively frequent, and can provide a basis for validating predictions of structural mechanics codes.

- There are documented cases where unanticipated operating conditions (e.g. thermal fatigue and erosion corrosion) have caused reactor pipes to become severely degraded (cracking and wall thinning) over relatively short periods of operation. Such reports of service experience can be used to test the ability of a probabilistic structural mechanics models to predict component performance under limiting situations of severe operating conditions.

- The literature documents studies in which piping specimens have been tested under conditions of fatigue and stress corrosion cracking. Such data can be used to evaluate the capability of the structural mechanics models to predict the conditions that result in relatively high probabilities of failure.

A1.4 Formal Process for Validating and Updating SRRA Codes

As previously stated, this regulatory guide does not require or endorse any particular SRRA computer code. However, if such codes are used, a formal process for validating and updating the codes should be in place to ensure they represent, and continue to represent, the best engineering fracture mechanics knowledge available at the time of their use. The process will also contribute to the uniformity and consistency of estimated failure probabilities for identical or similar components as calculated by different codes and/or by different organizations, and thereby enhance the credibility of the ranking and selection methodology. While the specifics detailing the formalized process for validating and updating an SRRA code are the responsibility of those owning the code, the formalized process should contain the following general attributes.

- The primary means of code validation should be by direct comparison of the code's results with applicable historical and experimental data (both generic and plan-specific) for each failure mechanism modeled in the code. Implicit in this is that such a source of historical data exists, collected, periodically updated as new information becomes available, and that mechanism-specific failure probabilities have been determined or can be determined from the data.

- A secondary means of code validation is to compare a code's results with other codes that have already been successfully validated.

- As new information becomes available, either additional failures for known failure mechanisms, failures attributable to here-to-fore unknown failure mechanisms, or new calculational techniques, this information should be incorporated into the code in a timely fashion such that results from the updated code once again reflects the best current knowledge basis in the areas of fracture mechanics and numerical quantification.

- The code's documentation identifying the failure mechanisms modeled, describing the underlying analytic/engineering models, identifying the parameters that are simulated as random variables, describing the input for these variables, and describing the numerical methods (e.g., Monte Carlo simulation) used to calculate failure probabilities should be updated as new information, models, or techniques are incorporated into the code.

A1.5 References for Appendix 1[1]

1. "Evaluation and Refinement of Leak-Rate Estimation Models," USNRC, NUREG/CR-5128, Revision 1, June 1994

2. O.J.V. Chapman, "Simulation of Defects in Weld Construction," PVP-Vol. 251, "Reliability And Risk in Pressure Vessels and Piping," The 1993 Pressure Vessels And Piping Conference, Denver, Colorado, July 25-29, 1993, American Society of Mechanical Engineers, 1993.

3. M.A. Khaleel and F.A. Simonen, "A Parametric Approach to Predicting the Effects of Fatigue on Piping Reliability," *Service Experience and Reliability Improvement: Nuclear, Fossil and Petrochemical Plants*, ASME PVP Vol. 288, pp. 117-125, 1994.

4. D.O. Harris and D.D. Dedhia, "Theoretical and User's Manual for pc-PRAISE, A Probabilistic Fracture Mechanics Computer Code for Piping Reliability Analysis," USNRC, NUREG/CR-5864, July 1992.

5. B.A. Bishop and J.H. Phillips, "Prioritizing Aged Piping for Inspection Using a Simplified Probabilistic Structural Analysis Mode," ASME PVP-Vo. 25, *Reliability and Risk in Pressure Vessels and Piping,* pp. 141-152, American Society of Mechanical Engineers, 1993.

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[1] Copies of Commission policy statements, EPRI and WCAP reports referenced herein are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

Copies of NUREGs are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-2249); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

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Appendix 2: USING PRA TO EVALUATE THE CHANGE IN RISK ASSOCIATED WITH CHANGES TO AN ISI PROGRAM

This section discusses the characteristics of a PRA that are acceptable for use in developing risk-informed ISI programs. The PRA provides the basis for calculating the impact of structural failures on the CDF and other risk measures, and thereby provides a risk basis for establishing appropriate ISI programs. Traditional PRA approaches are generally suitable for this evaluation, with some added refinement to address the passive failures of pipes.

The general methodology for using PRA in regulatory applications is discussed in Draft Regulatory Guide DG-1061 with reference to draft NUREG-1602 (Ref. 1), which provides guidance on the minimum requirements that a PRA must satisfy to be suitable for risk-informed regulatory applications. General PRA issues specific to the development of a risk-informed ISI program are discussed in Section 4.2. Detailed discussions on an acceptable quantitative approach are provided below.

The development of risk-informed ISI programs consists of two major elements. The first element quantifies the total risk impact that result from the proposed changes to the existing design basis ISI programs. Once the total change to public risk is evaluated, compared with the acceptance guidelines (decision metrics), and found acceptable, the second element will then incorporate risk insights (e.g., by use of importance measures) in the selection of pipe locations for inspection. Since the selection of pipe locations to be inspected is required to calculate the change in total risk impact, the process, by its nature, is iterative. One acceptable approach for performing the PRA analyses to assess the impact of the risk-informed ISI programs is shown in Figure A2.1. The procedural steps to accomplish this include:

- Determine Scope -- This defines the scope of piping to include in the plant PRA model. *(See Section 4.2.1 for guidance on this step.)*

- Develop PRA Model -- This defines acceptable approaches for modifying PRA models to include models for passive components and their associated leak and break probabilities. *(See Section 4.2.2 through 4.2.5 for general guidance.)* More detailed discussions are provided below.

- Develop Risk Impact of ISI Changes -- This determines the collective impact on risk from changes to inspection intervals, locations and methods for the plant piping. The risk calculated using the revised inspection programs is evaluated according to the decision guidelines discussed in Section 4.4 to determine if the revised inspection programs are acceptable. *(See Section 4.2.6 for general guidance.)* More detailed discussions are provided below.

As noted in Draft Regulatory Guide DG-1061, one principle that must be met to demonstrate the acceptability of a risk-informed submittal is a comparison of the plant's risk with the acceptance guidelines (decision metrics) contained in Draft Regulatory Guide DG-1061. Thus, at a minimum, the licensee must perform an analysis that is capable of showing that any increase in the calculated risk is consistent with those guidelines. The licensee also has an option of performing a Level 2 and a Level 3 PRA to demonstrate compliance with the decision matrix if such an analysis would prove useful to the risk-informed ISI program.

Changes to the ISI program are not expected to have an impact on the accident progression analysis or the performance of the containment structure. However, ISI program changes could affect failure probabilities for piping in containment systems (containment sprays, etc.) and containment bypass probabilities (failure of inter-facing piping). However, these can be modeled simply by assigning new failure probabilities to the affected piping. Thus, changes to the ISI program do not impact the performance of the Level 2 analysis except to address the failure probabilities assigned to pipes. Furthermore, the methods for performing a Level 3 analysis are not affected by changes to ISI since the objective of a Level 3 analysis is to estimate the consequences of events modeled during a Level 1 and Level 2 analysis. Thus, Level 2 and Level 3 methodologies are not further discussed in this document. Those ISI-related changes that impact the Level 1 PRA are discussed below.

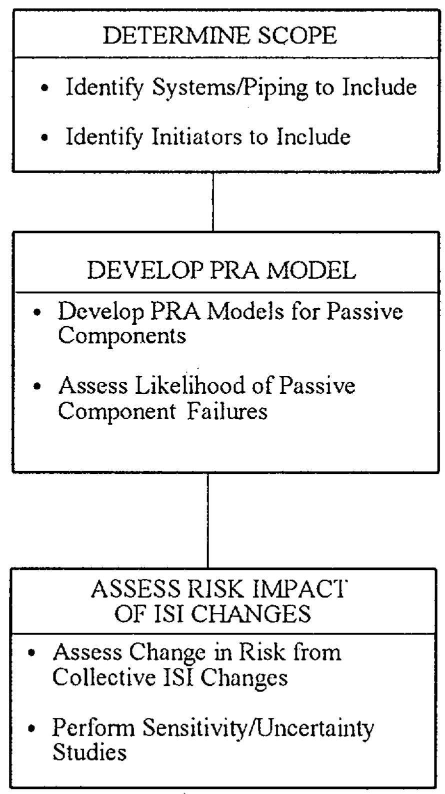

Figure A2.1 Process for probabilistic analysis for risk-informed ISI.

For the PRA to provide proper insights to the decisionmaking, there should be a good functional mapping between the piping associated with ISI and the PRA basic event probability quantification. Part of the basis for the acceptability of any RI-ISI program is a demonstration by use of a qualified PRA that established risk measures are not significantly increased by the proposed extension in inspection intervals or reduction in the number of inspections for selected pipes. To establish this demonstration, it is necessary that the PRA includes models that appropriately account for the change in reliability of the components as a function of inspection interval (or frequency), the number of elements inspected, and degradation mechanisms. When feasible, it is also desirable to model the effects of an enhanced inspection method. For example, enhanced inspections might be shown to improve or maintain component reliability, even if the interval is extended or the number of inspections reduced. That is, a better inspection method might compensate for a fewer number of inspections and/or longer interval between inspections. Licensees who apply for increases in inspection interval and/or decreases in the number of inspected elements are expected to address this area, i.e., to proactively seek improvements in inspections that would compensate for the increased intervals under consideration and/or decreased number of elements inspected. Licensees are encouraged to employ enhanced inspection techniques to improve detection of degraded components. This includes both conscious efforts to improve inspections according to state-of-the-art guidance, and, for licensees who wish to invoke credit for detecting degraded components, improvements in reliability modeling of a basic event probability as a function of the inspection programs.

As part of developing the risk impact of an ISI change, the following steps should be performed:

(1) Identify all RI-ISI systems, and components.
(2) Identify all affected cut sets and RI-ISI-related basic events.
(3) Review the method used to assess each affected basic event. Most fundamentally, the process should consider the effect of inspection strategy (interval and inspection method) on unavailability.
(4) Assess the effects that the changes have on the base case CDF and LERF.
(5) Address degradation mechanisms.
(6) Address uncertainties.
(7) Address NRC's defense-in-depth considerations.

A2.1 Modeling Passive Systems in PRA

Pipe leaks and breaks are traditionally modeled as initiators in PRAs (e.g., loss-of-coolant accidents (LOCAs), feedwater line breaks, floods), but the failures are not normally modeled in detail. The PRAs focus on the system responses necessary to prevent core damage, rather than a detailed treatment of the probability of the initiator occurring. That is, they do not usually model individual pipe segments or the structural elements within the pipe segments. However, since the goal of risk-informed ISI is to detect flaws so that failures are averted in those structural elements that have a significant impact on plant risk, it will be necessary to use models that are more detailed than traditional PRA models. The PRA will need to be modified so that a more detailed treatment of the probability of pipe failures and the influence of such failures on other systems are incorporated into the model. Acceptable approaches for addressing pipe failures in a PRA are summarized in this section and illustrated in the flow chart shown in Figure A2.2.

A2.2 Determine Consequences of Pipe Failures

The direct and indirect effects of pipe failures need to be characterized so that the appropriate failure mechanisms and dependencies can be incorporated into the PRA model. One acceptable means for incorporating pipe failures in a PRA is to consider three types of postulated pipe failures:

(1) leak,
(2) disabling leak, and
(3) break.

Each failure mode has a likelihood for degrading system performance through direct and/or indirect effects. For example, leaks can result in moisture intrusion through jet impingement, flood, and sprays. Disabling leaks (larger break area than for leaks) can result in similar damages as described for leaks, in addition to an initiating event and loss of system function. Breaks can result in all of the above-mentioned damages, including damages resulting from pipe whips. For each break size, the analyst calculates a failure probability and consequences resulting from the postulated failure. A failure modes and effects analysis (FMEA) with system walkdowns identify the failures required for the PRA calculations. The failure probability changes (decreases) as the break area increases (in most cases). Fracture mechanics computer models can be used to calculate failure probabilities. Acceptable methods for calculating failure probabilities of pipes are addressed later.

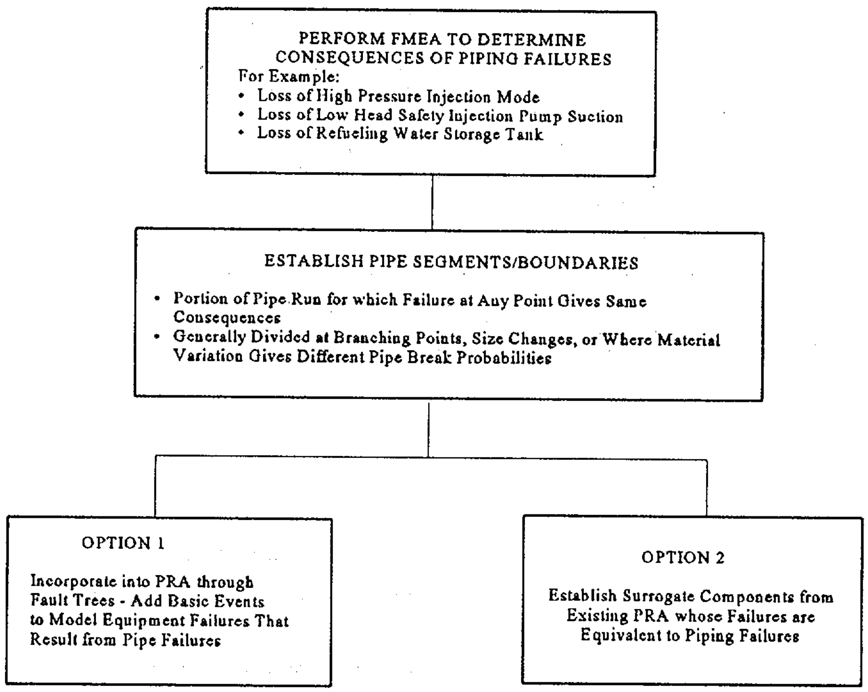

Figure A2.2 Process for Modifying PRA to Include Passive Components

Examples of direct effects that can result from pipe failures include:

- failure that causes initiating events such as a LOCA or a reactor trip,
- failure that disables a single train or system,
- failure that disables multiple trains or systems, and
- failures that cause a combination of the above situations.

Table A2.1 illustrates direct consequences postulated for several pipe segments, considering possible operator actions and their impact on the consequences for the plant examined in Reference 2.

Indirect effects include failures to additional equipment (including equipment in other systems) as a result of pipe whip, jet impingement, or flooding. An examination of indirect effects must also include a determination of how operator actions can be affected by improper instrument indications that could result from equipment failures/malfunctions caused by a pipe failure. A FMEA is an acceptable structured approach that can be used to catalogue these possibilities. The evaluation should also consider the potential actions that plant personnel can take to recover from a pipe break event. An example FMEA, adapted from (Ref. 2), is summarized in Table A2.2. Additional sources of information regarding the effects of pipe breaks that should be considered include the plant hazard evaluations performed to meet requirements of NRC's Standard Review Plan (Ref. 3), and any internal flooding analysis that has been performed at the plant[1].

Table A2.1 Examples of direct consequences from pipe segment failures.

Segment ID	Segment Description	Postulated Consequence (without operator action)	Postulated Consequence (with operator action)
ECCS-0	RWST to flow split to LPSI, HPSI, and Charging - MOVs 8812A, 8812B, LCVs 112D, 112E, V8884 and MOV 8806	Loss of refueling water storage tank (RWST)	Loss of RWST
ECCS-1*	From CV8819C and CV8818C to CV8847C	Loss of RWST**	Loss of all RHR and HPSI
ECCS-5*	Flow from SI CV 8847A and ACC CV 8956A to join to CV 8948A	Loss of RWST**	Loss of all RHR, HPSI and one accumulator
RCS-7	LPSI connection from Loop A cold leg tee to CV 8948A	Large LOCA with loss of HPSI, LPSI, and ACC injection to one cold leg	Large LOCA with loss of HPSI, LPSI, and ACC injection to one cold leg
FWS-1	Main feedwater flow from MOV35A to gate valve FCV510	Feedline break initiator	Feedline break initiator

* The only operator action that could be taken would result in closure of MV8835 (no HPSI to any paths) and closure of MV8809A or B (loss of two LPSI paths). However, given the short time available to take operator actions following a LOCA where LPSI is required, no operator action could be credited with closing MV8809A or B to save two injection paths. However, closure of MV8809A (or B) does result in preventing a loss of RWST.

** During the ISI team of expert meetings, the postulated consequence (without operator action) was changed to a loss of RWST inside containment resulting in an earlier transfer to recirculation and the loss of one injection path. An operator recovery action could not be taken due to limited time and the difficulty in diagnosing the actual location of the break during a LOCA.

[1]Section 2.2 of draft NUREG-1602 describes the attributes of a traditional flooding analysis. The major difference between an ISI analysis and a traditional flooding analysis is that in the ISI analysis the direct and indirect effects of ***each*** pipe segment must be considered and incorporated into the PRA model—no screening of flooding sources or propagation paths takes place.

Table A2.2 Example FMEA (Adapted from (Ref. 4))

Pipe Segment (location and pipe size)	Failure Mechanism	Failure Consequence	Recovery Action	Remarks
RWST to Valve 1-CS-25 • 6 Welds • 2 Elbows (16" diameter)	• Concern with chloride SCC (all locations) • Movement of tank during seismic event (at elbow nearest to tank)	• Loss of HPI Mode • Loss of Low Head SI Pump Suction • Loss of RWST	• Cross-Tie to Unit 2 RWST • Follow EOPs	• RWST is the primary source for the LPI and HPI systems during injection mode

A plant walkdown is required to assess the potential for indirect effects. Prior to a plant walkdown, existing documents (e.g., flooding analyses, etc.) that can provide insights into possible indirect effects should be examined. Possible sources of indirect effects can be obtained from the plant's equipment qualification program, hazards review program, and other documents that examine local effects of pipe breaks for the systems in the ISI program. Systems and trains affected by a break in the area should be identified. The plant layout drawings, for areas not covered by the documentation review, should be examined. Plant areas for which documentation was not clear or specific equipment not listed should be identified and resolved.

One good practice for pre-walkdown preparation is to develop summary sheets that examine the effects of spray wetting, flooding, temperature, pipe whip, jet impingement, rotating machinery, and pressure boundary ejected missiles. Development of such summary sheets should take advantage of the experiences gained from the ASME's Validation and Verification pilot programs (e.g., Code Case N577 Virginia Power's Surry plant). The hazards evaluation should include the examination of the emergency safety features building, the auxiliary building, the diesel generator building, the fuel building, the recirculating and service water pump house, the turbine building, the containment building, and the hydrogen recombiner building.

The personnel performing a walkdown should include representatives from the following organizations or groups:

- PRA
- Piping
- ISI
- Operations
- Engineering

The following is an example of the results from a walkdown performed for the reference plant Reference 2:

The walkdown of the turbine building resulted in several areas needing further consideration for the PSA modeling. The turbine building component cooling water has a small surge tank and virtually any pipe break/leak will eventually fail the system which will lead to reactor trip. The three plant air compressors are located side by side near the condensate pump discharge header. A postulated break in the header could potentially fail all three compressors which would cause a reactor trip. The location of the motor driven and 2 turbine driven pumps makes the system susceptible to losing all pumps due to a pipe break.

Hazards evaluation concludes pipe break will not target cable trays, but should further investigate effects of losing cable tray. No additional interactions found. Train B valves located away from postulated break locations. Pipe break will only effect FWA Train A. Need to consider the CCP interaction for inclusion in the segments analyzed.

An example of a walkdown worksheet documenting the information gathered is presented in Table A2.3.

A2.3 Pipe Segments

One acceptable method for modeling a run of pipe in a PRA is to divide (*segment*) the pipe-run such that a failure at any point in the pipe segment results in the same consequences. Distinct segment boundaries are identified at branching points or size changes where a significant difference in consequence (e.g., where pipe materials change), or the break probability is expected to be markedly different due to environment or other factors.

An example of a system and some of its defined pipe segments is shown in Figure A2.3. In this example, ECCS pipe segment #1 is defined as a pipe-run between check valves 1-SI-241, 1-SI-235, and 1-SI-79. Failure/break of this pipe segment is postulated to result in the loss of the inventory of the refueling water storage tank (RWST) inside containment. Similarly, ECCS segments 2, and 3 are defined for the other injection points into the RCS cold legs.

Another example of a pipe segment shown on Figure A2.3 is LHI pipe segment #1. This segment is defined as a pipe-run between check valves 1-SI-46B, 1-SI-47, and 1-SI-50. Failure of this pipe segment is postulated to result in loss of RWST outside containment, resulting in the loss of all injection and recirculation.

The number of pipe segments defined for an ISI analysis will be plant-specific. For the application described in Reference 2, the total number of segments defined and the systems are shown in Table A2.4.

Given that system boundaries involve system functions and may also involve interactions between different systems, the definition of these boundaries requires a careful, logical approach. All interfaces must be identified to ensure that there is consistency between the defined boundaries, when viewed from the systems on either side of each boundary, and that no safety functions are overlooked.

Table A2.3 Example of walkdown worksheet. *Adapted from Table 3.4-2 of Reference 2*

INDIRECT EFFECTS WALKDOWN WORKSHEET
Item #: 5 Building: ESF
Cubicle/Area: 011 Elevation: 21" - 6"
Indirect Effect of Concern: Loss of Train A equipment due to any pipe break in area (aux. feedwater suction or discharge piping), including a CCP pipe.

Components/Equipment in Cubicle/Area					
System	Comp. Type	Tag. No.	Train	Needed for Safe Shutdown?	Support System?
FWA	Pump	3FWA*PA	A	Y	N
FWA	Valve	3FWA*HV31D[1]	A	Y	N
FWA	Valve	3FWA*HV31A[1]	A	Y	N
FWA	Valve	3FWA*V4[2]	A	Y	N
FWA	Valve	3FWA*AV61A[3]	A	Y	N
FWA	Valve	3FWA*AV23A[3]	A	Y	N
FWA	Valve	3FWA*HV31CB[4]	B	Y	N
FWA	Valve	3FWA*HV31C[4]	B	Y	N
FWA	Valve	3FWA*AV62B[4]	B	Y	N

Comments

Cable tray numbers listed in Hazards Evaluation did not match those marked on the overhead trays in the room. Additional checks needed.

Conclusions

Apparent discrepancy with cable tray identifiers noted. Hazard Eval. concludes pipe break will not target cable trays, but should further investigate effects of losing cable tray. No additional interactions found. Train B valves located away from postulated break locations. Pipe break will only affect FWA Train A. Need to consider the CCP interaction for inclusion in the segments analyzed.

1. Located at far side of room from unisolable break
2. Near pump
3. Located at postulated break location
4. Located at far end of room away pump and postulated break

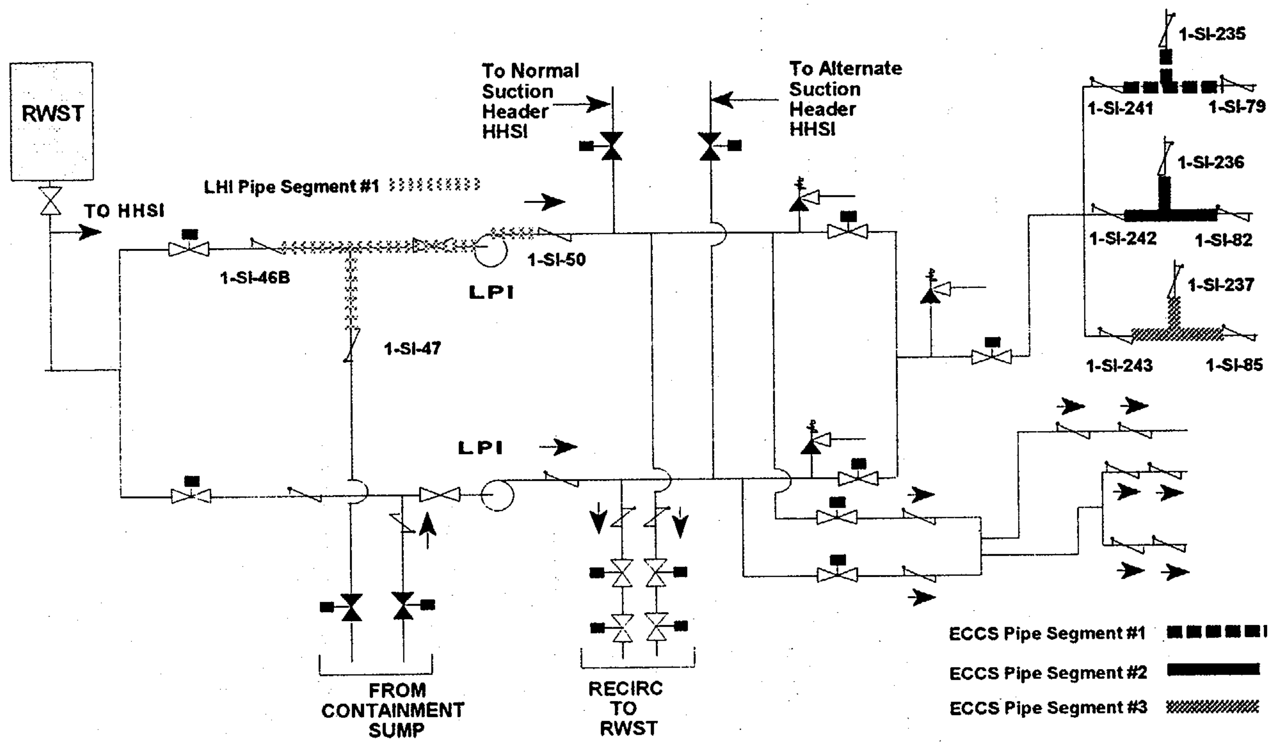

Figure A2.3 System pipe segment examples.

Table A2.4 Example list of piping segments

System	Number of Segments
BDG (SG Blowdown)	4
CCE (CHS Cool)	2
CCI (SI Cool)	with SIH
CCP (CCW)	14
CHS (CVCS)	23
CNM (Condensate)	with FWS
DTM (Turbine Plant Drains)	with MSS
ECCS*	9
EGF (DG Fuel)	4
FWA (Aux Feed)	15
FWS (Feedwater)	19
HVK (Control Bldg Chilled Water)	1
MSS (Main Steam)	30
QSS (Quench)	5
RCS	66
RHS (RHR)	with SIL
RSS (Recirc)	11
SFC (Fuel Pool)	4
SIH (HPI)	10
SIL (LPI)	13
SWP (SW)	29
TOTAL	259

* ECCS system was created to capture piping common to several systems including SIH, QSS, and SIL.

A2.4 Incorporate Pipe Segments Into PRA Model

To adapt the PRA model for risk-informed ISI, the initiators will need to be refined to reflect the direct and indirect effects of pipe breaks, if such breaks introduce new initiating events. Similarly, events for pipe breaks that occur subsequent to an initiator should also be analyzed. The effects of inservice testing of standby systems should also be addressed. These refinements can be made through various approaches. One acceptable approach involves direct modeling of the pipes in the PRA fault trees (Option 1 in Figure A2.2). An acceptable alternative (used in (Ref. 4)) involves using "surrogate components" that capture the effects of the pipe failures (Option 2 in Figure A2.2).

If Option 1 is used, new initiators may need to be added to the PRA model to reflect failures of the piping segments if such failures introduce new initiating events. If the pipe segment failure yields the same consequences as some other initiator already included in the PRA (e.g., a large LOCA), it could be accounted for by increasing the frequency of the initiator that is already included or by directly incorporating the pipe segment into a model (i.e. fault tree) of the initiating event. The importance of the pipe segments can be separated out at the end by considering the fraction of the initiator frequency due to that particular pipe segment failure or by grouping all cutsets with a particular pipe segment basic event. If the FMEA for the pipe segment identifies effects not included in any other initiator (e.g., spray effects that fail additional systems), then a new initiating event should be incorporated into the PRA. Event trees will need to be constructed for any new initiators that are added. Guidance for identifying initiating events and developing appropriate event trees is provided in draft NUREG-1602.

When selecting Option 1, the PRA fault trees should be modified to model events corresponding to pipe segment failures. The segment failure events can be included[2] as basic events in the fault trees, i.e., incorporated as additional failure mechanisms for the event(s) impacted by the pipe segment failure.

When using the second option to address pipe segment failures in a PRA, the PRA is not actually modified, but instead the impact of pipe segment failures is calculated by modifying the results of an existing PRA. For this approach, surrogate components are identified whose failures capture the effects of pipe segment failures. The risk corresponding to a revised ISI plan is then calculated by adjusting the frequencies of sequences or cut sets containing these surrogate components. Section A2.6 discusses the calculations that are performed to obtain these results.

Pipe failure frequencies will need to be determined for each pipe break initiator included in the PRA. Similarly, pipe segment failure probabilities will be needed for events included in the system models. These failures can reflect either failure probabilities (on demand) and failure rates (per hour or per year), and care must be taken to ensure that the correct units are applied. Acceptable methods for calculating failure probabilities for piping are discussed in Section A2.5.

Pipe segment **failure rates** for normally operating systems are analogous to active component failure rates used in PRAs, where the rate is the number of observed failures divided by the number of years of operation. A failure rate is used for events such as initiating events (e.g., LOCAs and steam line breaks) and for systems that are continuously operating (i.e., not demand-based, such as a pump failure to run for a desired mission time).

The **demand-based** piping failure probability is analogous to the active component failure probabilities that are used in PRAs, where the probability is the number of observed failures over the number of demands (such as a pump failure to start on demand). The demand-based piping failure probability is used for events in which a piping segment/system is in standby and is called upon to function given an event.

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[2]Some PRA codes allow the user to transform an existing fault tree basic event into the original event plus some combination of other basic events (e.g., pipe segment failures). Use of such a Code feature is an acceptable alternative to actual fault tree modification.

A2.5 Piping Failure Potential

The process of estimating component failure probabilities is at the heart of a quantitative risk-informed ISI program. Failure probabilities and failure rates of pressure boundary components are required as inputs to the calculation of CDF and risk. It should be noted that quantification of failure probabilities (i.e., estimating the impact of ISI on reducing failure probabilities) is also part of developing a risk-informed ISI program.

A2.5.1 Overview of Estimation Procedure

Figure A2.4 shows the process for estimating failure probabilities. The steps are described as follows:

- Identify locations of high failure probability and their associated failure modes/mechanisms. The failure probability should be for a break size that can degrade a system from fulfilling its mission. It may be a leak that results in secondary failures, such as an electrical bus, a disabling leak, and/or a break.

- Review and revise the initial selections for high failure probability locations as well as the failure modes/mechanisms for these locations. This review may make use of a technical group (i.e., a panel) of individuals with specific areas of expertise in plant operations and maintenance, fracture mechanics, and PRA.

- Assemble the detailed data needed to estimate failure probabilities, including piping design data, loadings, materials, and operating experience.

- Estimate failure probabilities of critical location(s) for each pipe segment using historical failure rate data, structural reliability computer codes, or expert judgment elicitation. If expert judgment elicitation is required, then it should be performed generically through the ASME or industry group and incorporated into the structural reliability computer Code. (NRC should be informed of such activities.)

- Estimate relative failure probabilities for other less critical locations within the piping segments using the probability estimated for the critical location(s) as the reference value.

- Calculate the overall failure probability for each system and the combined probability for all plant systems.

- Review calculated failure probability estimates. This review could be performed by the ISI team or by an independent panel.

- Tabulate final estimates of failure probabilities for use in PRA calculations to estimate the CDF and/or risk associated with each pipe segment and/or structural element.

- Perform sensitivity studies to evaluate potential impacts of modeling and input data uncertainties in failure probability estimates on estimated failure probabilities.

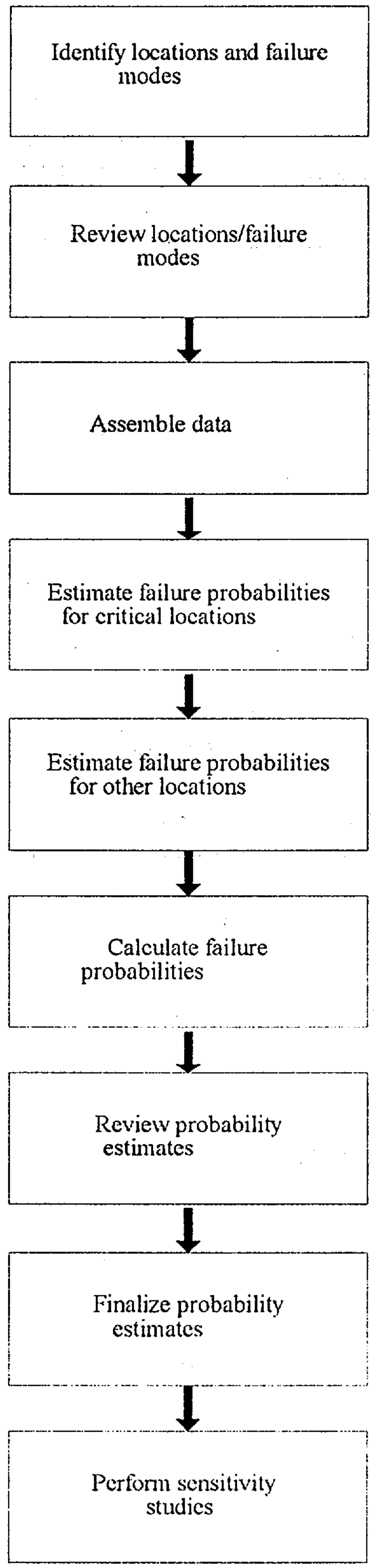

Figure A2.4 General process for estimating failure probabilities.

Detailed considerations that should guide the failure probability estimation process are provided below.

A2.5.2 General Guidance on Issues

Realistic Versus Conservative Estimates - The objective of risk-informed calculations is to make realistic estimates of failure probabilities rather than conservative or non-conservative estimates. The introduction of conservatism on a selective and/or nonuniform basis for particular components or particular failure mechanisms will have the undesired effect of biasing the CDF or risk estimates and the inspection locations.

Effects of ISI - For CDF and/or risk calculations (LERF, ΔCDF), pipe segment failure probabilities should be estimated assuming that ISI is performed during the plant's licensed period (e.g., 40 or 60 years). Structural mechanics calculations should include effects of inservice inspections. For segment categorization (discussed in this Chapter), ***no*** credit for ISI should be taken. In the application of historical data from operating reactor experience, it can be assumed that past ISI programs for most components have had only modest impacts (if any) because the selection criteria focused on locations of high-stress/high fatigue usage (among other criterion) on component failure probabilities, while at the same time leading to unnecessary personnel exposure to radiation. One exception would be a situation where augmented ISI programs have been implemented (e.g., inspections for stress corrosion cracking of BWR piping, and inspections of piping for erosion corrosion for both PWRs and BWRs).

Aging Effects - The effects of aging mechanisms on failure rates should be included in estimating failure probabilities. Specific aging mechanisms known to be of concern to nuclear pressure boundary components are irradiation induced embrittlement for reactor pressure vessels and for vessel internal components, and thermal aging for cast stainless steel.

It should be noted that statistical analyses have not identified increasing failure rate trends, based on component failure data, as a function of component age (Ref. 5) and (Ref. 6). Such trends are consistent with results of computer calculations. Structural Reliability/Risk Assessment (SRRA) models of fatigue and stress corrosion cracking for typical operating conditions have indicated that failure rates should be very low, and that aging will not increase failure rates until well beyond the design life of the components. However, aging effects should be considered for those locations at which service induced structural degradation (cracking or wall thinning) is present.

Credit for Leak Detection - Leak detection can provide advance warning of pipe degradation prior to break. For calculating the change in core damage frequency that results from changes to the inspection program, leak detection should be credited. However, when calculating the relative risk importance of a segment, leak detection should not be credited. The present defense in depth process includes ISI programs, operator walkarounds, leak detection systems, system tests, and pressure tests. These should not be credited in the importance measure calculations used for classifying a pipe as high- or low-safety significant.

Failure Probability Calculation - Applying the guidelines outlined above (including additional criteria addressed later in this report), the failure frequency is normally calculated as a *cumulative* failure probability over the 40 or 60-year license of the plant (as justified) and divided by the number of years (40 or 60) to obtain the average rate of failure in any one year. This process addresses aging effects calculated by the computer Code and results in an average failure rate on a per-year basis. (See Section A2.5 for additional implementation details.)

Failures on Demand Versus Failure Frequencies - The term "failure probability" refers to both demand-related and time-related probabilities. Section A2.6 of this regulatory guide addresses the use of these measures of failure probability in the calculation of CDF and/or risk, and recommends methods for relating demand-related probabilities to failure frequencies.

Failures of components in standby systems will have safety consequences only if the piping fails or is in a failed state during the limited time periods when the system is required to mitigate an accident or to otherwise maintain the plant in a safe condition. Failure probability estimates should be apportioned to exclude pipe failures that occur and are detected during other periods, such as standby and testing modes, and are subsequently repaired. However, structural integrity evaluations should account for structural degradation (e.g., corrosion) that can develop during these non-demand periods, because such degradation can subsequently lead to failures when maximum loads are applied to the degraded components for a demand situation.

Evaluations of standby systems should establish the likelihood of piping failures during periods of demand as opposed to failures during standby periods or during periods of operability testing for which failures will not impact plant safety. It can be assumed that structural failures during standby periods or during testing will be detected, such as by visual observation of gross leakage, and that the failed components are promptly repaired. The failure mechanisms and frequency should be compared with the calculated results.

Identification of Failure Mode and Mechanism - As stated above, it is important to identify the appropriate failure mode (leak, disabling leak, or full break) for each individual component, so that the failure mode corresponds to the consequence addressed by the probabilistic risk assessment. In most cases a pipe break is the failure mode of concern, although in some cases a pipe leak (for jet impingement) or a disabling leak (for loss of system function) can also have safety consequences. While failure modes corresponding to a pipe leak may not be of concern from the standpoint of safety consequences, such modes would be of concern from the standpoint of plant availability, economic impacts (which are outside the scope of this regulatory guide), or public perception (safety concern).

Operating experience on leaks and cracking, as well as other detectable modes of degradation, are significant to the risk-informed ISI process. Such observations are often associated with conditions (i.e., design and material deficiencies, fabrication errors, unanticipated stresses, aggressive environments, etc.) that could cause a pipe break at another location in the system and/or during future periods of operation. This information should be used for estimating pipe failure probabilities.

Information on observed degradation mechanisms should also influence inputs to structural reliability calculations and used for benchmarking, such as done for the computer Code, pc-PRAISE (Figure A2.5). For example, structural reliability models predict (in addition the pipe break probability) probabilities of leaks and significant crack growth and/or wall thinning. Uncertainties regarding inputs and modeling assumptions can be addressed by calibrating the structural reliability codes to the trends of service experience, for example, as for modeling of stress corrosion cracking with the pc-PRAISE Code (Ref. 7), and (Ref. 8).

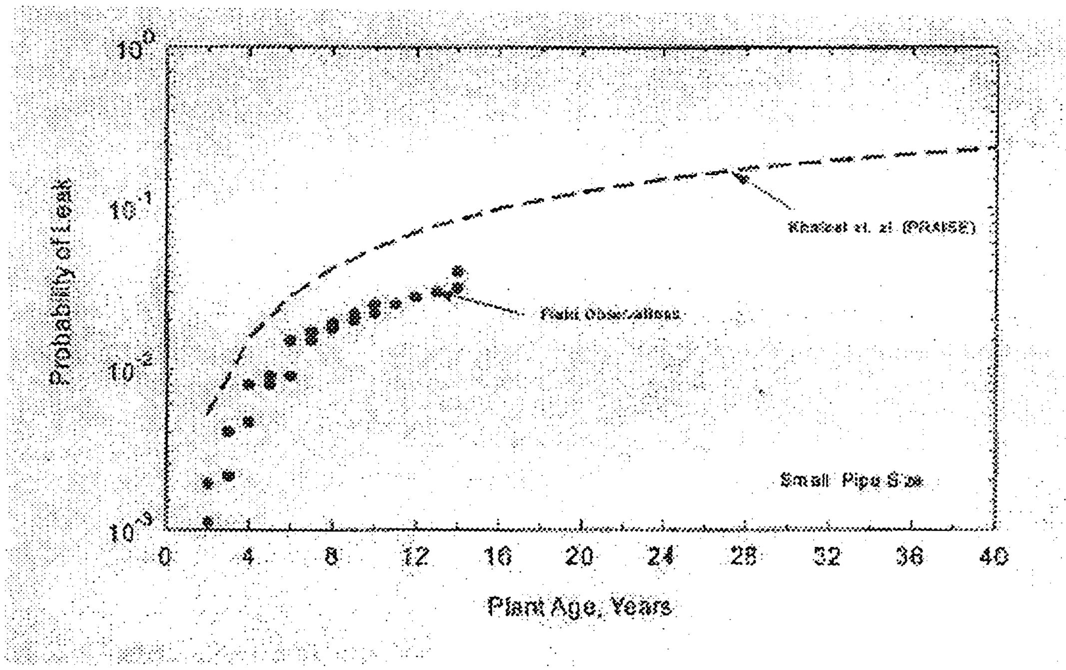

Figure A2.5 Example Code -vs- Service Experience.

The estimation procedure should address each component (e.g., pipe segment) and structural element (e.g., weld), and should assign:

- a dominant failure mechanism (e.g., fatigue cracking at the inside surface), and
- a numerical value for the failure probability.

Identification of failure mechanisms is a significant step. This information is an important input to the subsequent step of developing inspection strategies, since different failure mechanisms will dictate different inspection methods to detect the presence of structural degradation and damage.

Common Cause Failures - Special situations that can result in CCFs should be identified as part of the failure probability estimation process. For example, extending the inspection intervals could make CCFs more important. CCFs are of concern only if the failures occur within the same time period, as for example, during the course of a given accident scenario. The method of segmentation of pipes and requiring one element be inspected in each segment that is categorized as high-safety-significant can reduce the likelihood for CCF by detection.

Situations that could result in CCFs that occur within the same time period include:

- Piping that is not subject to routine pressure testing to verify its integrity. Such piping could experience long term degradation (corrosion/wall thinning), resulting in multiple failures when it is suddenly pressurized during a critical demand period of an accident scenario.

- Degraded piping that is subject to routine pressure testing to verify its integrity, but is subject to over pressure conditions (e.g., interfacing system LOCA or waterhammer loads) during a critical demand period of an accident scenario.

- Degraded piping subject to severe loads from external events such as a seismic event.

- Multiple pipe failures caused by indirect effects from pipe breaks (e.g., a broken pipe swings and impacts an adjacent pipe causing the impacted pipe to break).

Undefined Failure Mechanisms - In some pipe run locations it can be difficult to identify any failure mechanism (either from plant service experience or from SRRA calculations) that can result in other than very small failure probabilities. The arbitrary assignment of a zero failure frequency is unrealistic and could bias the ISI process by eliminating from consideration locations that have relatively high consequences of failure. The technical approach should include a procedure for estimating zero frequencies (i.e., approximately 10^{-8} - 10^{-9} failures per year) for such locations to account for modeling limitations associated with very low values of calculated probabilities, and/or to account for uncertainties regarding unidentified failure mechanisms. The assignment of such low failure frequencies is consistent with an expectation that plant operation is unlikely to experience significant material degradation. A potential failure mechanism should also be assigned to these locations to provide a basis for developing inspection strategies.

Failure Probabilities for Other Locations - Given the number of structural elements within each pipe segment, it is not practical to perform detailed evaluations for each location (e.g., element or weld). The recommended approach is to identify the critical location(s) within each pipe segment which has the highest expected failure probability, and to focus the detailed evaluations on these locations. It may not always be clear without detailed evaluations which of the structural locations within a segment has the greatest failure probability. In these cases, detailed structural mechanics evaluations should be performed for each location. Additional evaluations can also establish relative differences in failure probabilities within the segment, and thereby provide an improved technical basis to assign probabilities.

Having estimated the range of expected failure probabilities for critical structural elements within a segment, the failure probabilities can be estimated for the other less critical locations. Typical estimates in pilot applications (Ref. 9) have assigned at least 50% (and typically 90% or more) of the overall segment failure probability to a critical location. It is important to make failure probability estimates for the other structural locations to determine if a large number of small contributions from such locations contribute significantly to the overall failure probability of the segment.

Total Failure Probabilities for Systems - A total failure probability is calculated for each system based on the probabilities estimated for the individual segments that make up the systems. The total failure probability for pipe segments within a system is the sum of the individual pipe segment failure probabilities. These totals should be reviewed by the licensee to facilitate the review of the failure probability estimates. Such system level information is more readily benchmarked with the limited data regarding pipe failures from plant specific and industry wide experience. Unreasonably large or small system level probabilities, when compared to data, should be cause to modify the inputs and/or assumptions used to estimate the segment level failure probabilities. Total system level failure probabilities should also be reviewed to look for reasonable and consistent trends regarding relative contributions of particular systems and failure mechanisms to overall plant wide failure probabilities. All assumptions in the calculations should also be reviewed and revisions made as appropriate.

A2.5.3 Methods for Estimating Failure Probabilities

This regulatory guide describes three acceptable methods for estimating failure probabilities for piping. It is recommended that these methods be used in combination. In typical applications, some aspects of all three methods will usually be used although one method may be the primary method. For example, while the primary method may be application of structural reliability computer codes, some inputs to the computer model will be based on experts where data is lacking. Similarly, experts make use of available data, including results from computer models. Furthermore failure probability estimates, from both experts and computer models, are always subject to "reality checks" by comparisons of the estimated probabilities with plant-specific failure experience and industry wide historical data on failure rates. The degree to which one relies on one method or another is predicated on the availability of experts and applicable structural reliability models.

Approaches for estimating failure probabilities include:

Historical Data - Studies by Bush (Ref. 10)(Ref. 11)(Ref. 12), Jamali (Reference 5), Thomas (Ref. 13), and Wright et al. (Ref. 14) have estimated break probabilities for systems and components based on data from the few documented occurrences of pipe breaks along with additional knowledge of the relevant number of years of plant operation. While such data bases will not fully reflect plant specific factors (e.g., operational conditions, service experience, materials selection, design features, etc.) needed for an individual plant evaluation, the information can serve as useful baseline data to guide estimates. Table A2.5 lists a number of sources of failure data that can be used to guide the estimation of piping failure probabilities.

SRRA Calculations - Structural reliability risk analysis (SRRA) computer codes use models based on probabilistic structural mechanics methods and can be applied to estimate failure probabilities for important components. SRRA estimates can take into account a higher level of component specific information than methods based on historical data or expert elicitation. SRRA models can be particularly useful for estimating relative values of failure probabilities to permit locations within a system with higher values of failure probabilities to be identified.

Table A2.5 Sources of Failure Data That Can Be Used To Guide Estimation of Failure Probabilities.

Database	Narrative description	Comment	Reference
NPRDS	Computerized database maintained on behalf of electric utility industry by INPO	Contains component hardware reliability. Covers experience on maintenance, inspection and repair of nuclear plant components	--
LERs	Computerized database maintained by NRC	Contains information submitted by operating plants. Small fraction of reports deal with component/structural degradation and failure. Extensive screening required to locate information relevant to maintenance and inspection	--
Plant records	Maintained by individual plant and vendors of plant operating experience	Useful information. Contains inspection, maintenance, and repair information. Accessing this information involves commitments of time and money for visits to plants	--
Expert elicitation	Developed by NRC and national laboratories. Provides useful information on undocumented field experience	Contains failure probabilities and rates of pressure boundary components and structures. Contains estimates of important safety parameters useful for performing PRAs	(Ref. 15) (Ref. 16) (Ref. 17)
NPAR	Summary conclusion of NRC research on age-degradation of pressure boundary components	Describes service failures and degradation at operating plants	(Ref. 18)
Assessment of plant life extension	Utility industry prepared through NUMARC. Each report addresses issues for a particular type of component (e.g., primary coolant system components)	Identifies degradation potentially important to plant safety	--
ASME Task Group on Fatigue	Special ASME, Section XI Task Group report. Reviews fatigue of nuclear power plant components and makes recommendations to ASME, Section XI.	A comprehensive review of operating experience, and describes occurrences of cracking	(Ref. 19)
NRC Pipe Crack	Formed by the NRC to evaluate the causes of unexpected cracking of reactor piping systems	Identifies potential solutions for eliminating or mitigating reactor piping systems cracking	(Ref. 20) (Ref. 21)
EPRI	EPRI-sponsored study on material degradation and environmental effects on components for plant life extension	Contains information relating to fabrication processes that contribute to degradation. Identifies flaws in LWR components	(Ref. 22)
EPRI	Computer software developed by EPRI to predict piping locations subject to erosion/corrosion	Widely used by utilities	(Ref. 23)
EPRI	A compilation of data on nuclear piping failures.	Includes estimates of generic failure probabilities for particular systems.	(Ref. 5)
INEL	Summary of pipe break accidents.	Information intended to use in probabilistic risk assessments	(Ref. 14) (Ref. 24)
Bush	Review and interpretation of data on piping failures and service related degradation	Author brings perspective of Code and regulatory issues.	(Ref. 10) (Ref. 11) (Ref. 12) (Ref. 25) (Ref. 6)

SRRA models also predict the progress of degradation and/or crack growth as a function of time while quantitatively accounting for the impact of random loadings, such as earthquakes. These results can be useful for selecting appropriate intervals over the service life of the components for periodic ISI examination. Section A2.5.4 of this regulatory guide provides example guidance on the application of SRRA models to the development of risk informed inspection programs.

The following steps should be applied in application of SRRA models for estimating failure probabilities.

- Select a structural reliability model(s) that addresses the materials, operating conditions, and failure mechanism(s) that apply to the structural location of concern

- Gather detailed data needed as input to the SRRA model including pipe dimensions, materials and welding parameters, operating temperatures, operating pressures, cyclic loadings, chemistry/flow rates for fluids, and operating stresses for normal and upset conditions.

- Use design basis stress analysis as a source of stress data.

- Use plant operating staff knowledge to address input parameters such as susceptibility to IGSCC, wall thinning, and thermal high cycle fatigue.

- Neglect effects of inservice inspections when defining inputs to the SRRA calculations which will estimate the failure probabilities to be used in the PRA.

- Review final input data for an appropriate SRRA, and follow guidance provided in Section A2.5.1 of this regulatory guide.

- Calculate the failure probabilities.

- Assess values for calculated failure probabilities for consistency with operating experience and expert judgment. Identify inconsistencies in predicted probabilities for detectable degradation, leak probabilities, and break probabilities. Benchmark SRRA calculations with operating experience and expert input.

- Document SRRA calculations by providing details of input data, modeling assumptions, and resulting values of calculated failure probabilities.

<u>Expert Input</u> - Elicitation of experts has gained acceptance as a means to quantify input to PRAs and risk-based studies. A systematic procedure, as described by References 9, 16, and (Ref. 15), has been developed for conducting such elicitations to address major industry safety issues. Application of the procedure has been demonstrated in a research program that estimated failure probabilities for use in a pilot application of PRA methods to inservice inspection (Ref. 26) and Reference 6. (See Appendix 3.)

The expert elicitation process is (as a generic methodology) applicable to any issue where there are large uncertainties, data are lacking, or predictive models are not well validated. As such, the methodology need not be applied directly to make estimates of structural failure probabilities, but can be applied to address needed inputs to structural mechanics models. A full scale expert judgment process as described in References 16 and 25 can be laborious and normally requires staff and expertise outside utility capabilities. Therefore, if expert judgment elicitation is required, it should be performed generally through the ASME or an industry group and incorporated into the structural reliability computer code. ***(Reliance and conclusions from an expert elicitation program should be documented in a licensee's submittal to the NRC.)*** For example, inputs for crack growth rates, loading conditions, and flaw distributions could be addressed through an elicitation process, with full documentation of the process and results.

A2.5.4 Structural Reliability Computer Codes

Structural reliability computer codes are useful tools for estimating failure probabilities of piping components. These codes make use of probabilistic structural mechanics methods to model the uncertainties and variability in such parameters as material properties, mechanical loadings, operating environment variables, and flaw distributions. Some of the benefits of using such codes are:

- The subjective nature of estimating failure probabilities is decreased. The judgmental aspect of the estimation process is reduced to a series of smaller decisions regarding some specific inputs to a structural mechanics model rather than being combined into a single judgment needed to assign a failure probability.

- A greater level of consistency and uniformity to the process of estimating failure probabilities is achieved. This adds credibility to the risk ranking process. Regulatory reviews of failure probability estimates are facilitated since the methodology and associated computer codes need only be reviewed once thereby limiting reviews of plant specific evaluations to address only the inputs used for calculations.

- Structural mechanics models, by simulating the range of uncertainties in governing input parameters, provide an improved technical basis to conclude that particular failure mechanisms can only make relatively small contributions to failure probabilities.

- Structural mechanics codes model the physical interactions of the various factors that impact failure probabilities. As such, the calculations can give good predictions for relative numerical differences in failure probabilities from segment to segment within a given system, and thereby enhance the credibility of the categorization process (see Section A2.7).

- It has been demonstrated that structural reliability calculations can be performed relatively efficiently. Therefore, the time and costs to estimate failure probabilities can be significantly reduced compared to alternate approaches such as with the formal conduct of an expert judgment elicitation.

- Structural reliability computer codes provide reproducible results by independent parties.

- Knowledge gained from plant operating experience regarding observed degradation mechanisms and failures by leak or break (or lack of such observations) should be incorporated into the structural reliability computer codes on a continuous basis.

There are limitations to structural mechanics computer codes, which must be recognized by the user:

- A structural mechanics code may not be available to address the particular failure mechanisms or materials of interest. Inappropriate application of existing codes could give misleading predictions of failure probabilities. Code users must be fully aware of the code's limitations and resort to other estimation methods, as needed. **The new estimates should be validated, incorporated into the code, and reported to the NRC.**

- There can be a lack of information to assign inputs to the computer codes. This means that expert judgment will be used in assigning input parameters to calculations rather than in a direct manner to estimate failure probabilities. **The results of the experts should be fully documented, incorporated into the code, and reported to the NRC.**

- As with any technical computer program, a false sense of confidence can be attached to calculated failure probabilities, since many of the physical assumptions and numerical parameters used in the calculations are not evident to most users. For example, most users will have little basis to evaluate the applicability and reasonableness of parameters associated with crack growth rate correlations and density and size distributions for flaws. A quality assurance program should be in place to ensure the proper selection of input parameters.

- There are uncertainties in the modeling of structural failure mechanisms and the quantification of the inputs to the models. Therefore, a review of the estimated failure probabilities should be performed to determine if the probabilities are consistent with plant specific and industry experience regarding expected contributions from specific systems and failure mechanisms.

The applicability of the available structural reliability code models should be evaluated along with the feasibility of adequately defining the needed inputs to the model on the basis of available plant specific data. **In those cases where none of the proposed modeling approaches are capable of calculating credible results, other methods for estimating failure probabilities, based on historical experience and/or expert judgment, should be performed, incorporated into the code and reported to the NRC.**

It is recommended that calculations of failure probabilities with structural reliability models be performed when suitable models are available to address the component of concern. A number of suitable codes based on probabilistic fracture mechanics codes have been applied in the past, such as the pc-PRAISE Code (Ref. 27) and (Ref. 28). Simplified models (e.g., *SRRA* computer Code) have also been developed (Ref. 29) and (Ref. 30). In general, these simplified models have been built from more detailed models, such as the pc-PRAISE Code.

Appendix 1 provides a detailed discussion of structural reliability codes. The following summarizes the criteria for evaluating the acceptability of computerized structural reliability codes for estimating failure probabilities of piping components:

- Addresses the failure mechanisms under consideration.
- Addresses the structural materials under consideration.
- Structural mechanics model based on suitable engineering principles and the approximations used in the model are appropriate.
- Probabilistic part of the structural mechanics model addresses those parameters with the greatest variability and uncertainty.
- The inputs to the codes must be within the knowledge base of the experts applying the code.
- Internally assigned parameters and probability distributions are documented and supported by available data and knowledge base.
- Documentation of technical basis of model is available for peer review.
- Limitations of code are identified and cautions provided for cases when alternative structural mechanics models and/or estimation methods should be utilized.
- Benchmarked with codes considered acceptable by the NRC such as pc-PRAISE.
- Benchmarked with applicable data and operating reactor experience.
- The development of the computer code, documentation and application was conducted in accordance with approved quality assurance procedures.

A2.5.5 Screening and Sensitivity Studies for the Purpose of Categorizing Pipe Segments

Screening and sensitivity studies should be performed to eliminate pipe segments from further consideration and evaluate the change in the calculated failure probability estimates and their potential impacts on the pipe segment prioritization or categorization process. Uncertainty in the calculated piping segment failure probabilities will contribute to uncertainties in the calculated CDF and LERF. Thus, while performing the sensitivity calculations identified in (Ref. 2) are necessary, additional calculations should focus on those aspects of estimating pipe segment failure probabilities and other PRA related activities which could significantly affect the categorization of pipe segments, thereby impacting the estimate of a plant's CDF and risk. Particular emphasis should be placed on identifying and understanding the screening and sensitivity studies that would move a segment from a lower risk category to a higher risk category and vice versa.

The objective of the screening and sensitivity calculations is to remove from consideration pipe segments to be included in the list of high safety significant segments. The following calculations can provide useful insights on how pipe segment categorization can be affected by changes in a pipe segment's estimated failure probability. Other screening calculations should be considered as appropriate.

- In some cases the calculated failure probabilities for pipe segments will be, for all practical purposes, zero (incredible events). In such cases, a small probability will be assigned to account for unknown/undefined failure mechanisms. This is done to ensure that a pipe segment is accounted for in the PRA and categorization calculations. In the Westinghouse Owners Group study (Ref. 2), this minimum (bounding in the sense that no degradation mechanism can be identified) probability for pipe segments was taken to be a cumulative probability of 10^{-8} for a pipe rupture over the 40-year design life of the plant. Screening calculation should be performed to address the issue of truncation limit affects on categorization by assigning an even lower failure probability. This lower probability can be based either on the actual calculated probability (assuming that the numerical approximations gave a non zero number), or on an assigned probability with a lower value (say by factor of 100 lower). If the results confirm a low risk from these segments, categorize as low.

- Because estimated probabilities for certain failure mechanisms could be systematically high or low, a number of screening calculations addressing systematic biases in estimating pipe segment failure probabilities should be performed. To perform these calculations, the failure probabilities should be increased by a factor of 100 for all affected pipe segments, and the probabilistic model could be recalculated to eliminate the problem of truncation owing to the increased failure probability. The following areas of concern have been identified:

    - Segments for which erosion corrosion is the failure mechanism of concern,
    - Segments consisting of small pipe sizes,
    - Segments containing ferritic steel, and
    - Segments exposed to a common set of environmental conditions.

If still low risk, categorize low.

- Estimates of leak probabilities (through wall cracks) can be made with higher level of confidence than the corresponding estimates of pipe break probabilities. Probabilistic structural mechanics codes calculate both leak and break probabilities. Therefore a sensitivity calculation should be performed that replaces all disabling leak and break probabilities for each pipe segment with the leak probability for the segment. In addition, leak probabilities could be selectively used for segments governed by degradation mechanisms that tend to promote the development of leaks (e.g., intergranular stress corrosion cracking). If still low, categorize low.

- Because operator actions can be important in mitigating the effects of a pipe break, sensitivity calculations should be performed that remove credit for all operator actions incorporated into the PRA in response to specific pipe segment failures (e.g., operator terminates inventory loss from the reactor water storage tank). Additionally, a sensitivity study should be performed that increases the failure probability by a factor of 10 of all operator actions to account for the possible additional stress associated with responding to a pipe failure. If still low, categorize low.

- If the initial pipe segment failure probabilities were calculated assuming no credit for ISI programs, then a sensitivity study should be preformed where credit is taken for these programs. If low risk, categorize low.

The effects of the above screening and sensitivity studies should be integrated in the decision-making process for categorizing segments as high- or low-safety significant.

A2.6 Risk Impact from Proposed Changes to the ISI Program

Applying a PRA model developed in accordance with the guidelines outlined in this chapter, the risk impact of the proposed changes in the ISI program can be evaluated. The acceptability of the change in risk due to the change in the ISI program is addressed in Section 4.4. To aid in that assessment, uncertainty and sensitivity analyses will be needed. General guidelines for these analyses are provided in Draft Regulatory Guide DG-1061 and draft NUREG-1602. ISI-specific uncertainty and sensitivity analysis guidelines are addressed in the following sections of this document.

When using the first approach (Option 1, Figure A2.2) for incorporating pipe segment failures into the PRA (i.e., incorporating basic events representing pipe segment failures into the fault trees), the risk corresponding to a revised ISI plan is calculated by simply requantifying the PRA using pipe segment failure probabilities/frequencies appropriate to the revised ISI plan. For the second approach (Option 2), using surrogate components, the risk is calculated by adjusting the base PRA results to reflect the new initiators and events that simulate the consequences of a postulated pipe failure. This Option 2 process is outlined in Figure A2.6. The calculations and the equations required by this approach are described later in the section (Ref. 2).

In order to evaluate the risk impact from proposed changes to the ISI program, one first uses the Option 2 approach with the pipe failure rates calculated with credit for the current ISI program, and then with the proposed changes to the ISI program. The risk impact is the difference between the sum of the piping failure contributions to the core damage frequencies as calculated with the two different ISI programs. Realistically one should consider, when considering the risk impact from proposed changes to the ISI program, the appropriate operator recovery actions for isolating the pipe breaks, with the appropriate human error probabilities.

The equations used in the Option 2 approach can also be used to find the contribution to the core damage frequency from each piping segment for prioritization or risk categorization purposes. For this case, the pipe failure rates without the inspections one is considering eliminating should be used, but other inspections can be included. The following discussion provides additional clarification on this subject.

<u>Estimation of Failure Probabilities for Risk Categorizations</u>

When using fracture mechanics codes to estimate failure probabilities, the following conditions are used for risk categorization calculations:

- For piping segments that are included in augmented programs (such as erosion-corrosion and stress corrosion cracking programs), the calculated failure probabilities with ISI but without leak detection are used.

- For other piping segments, the failure probability without ISI and without leak detection are used.

<u>Basis for Not Crediting Leak Detection and Operator Walkarounds in Risk Categorization</u>

Most fracture mechanics codes can calculate a failure probability which credits leak detection at the defined leak rate entered as an input to the model. This leak detection assumes immediate detection of the leak and subsequent repair/shutdown. In addition, operator walkaround can also be credited to identify leakage.

However, the purpose of RI-ISI programs is to identify degradation prior to leakage and/or rupture. Therefore, taking credit for these factors would mask important piping segments that should require non-destructive examination (NDE) inspection to identify the degradation prior to failure. Leak detection systems and operator walkarounds are recognized as additional mechanisms that ensure defense-in-depth in maintaining the pressure boundary prior to piping failures that lead to initiating events or mitigating system failures.

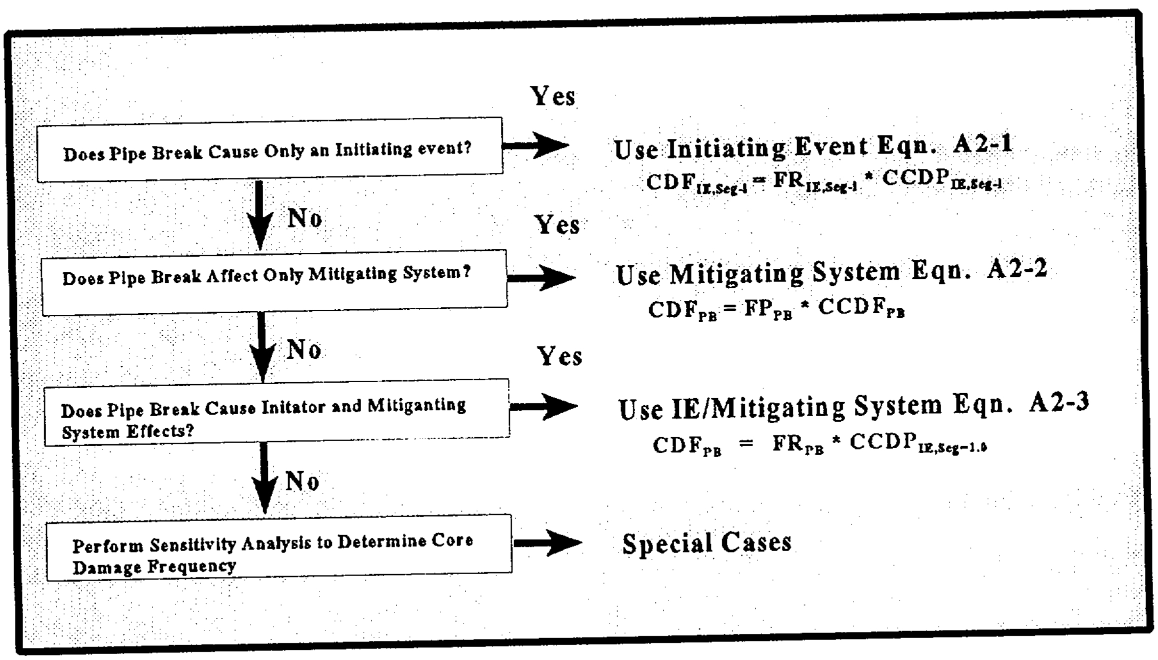

Figure A2.6 Core Damage Frequency Calculation Process (Adapted from Figure 3.6-2 of Reference 7, 2).

CDF Calculations for Surrogate Component Approach

Initiating Events:

For a pipe whose failure is an initiating event, the portion of the PRA model that is impacted is the initiating event and its frequency:

$$CDF_{IE,Seg\text{-}I} = FR_{IE,Seg\text{-}I} * CCDP_{IE,Seg\text{-}I} \qquad \text{(EQN. A2-1)}$$

where

$CDF_{IE,seg\text{-}I}$ = CDF from initiating event associated with failures of pipe segment I (events per year)

$FR_{IE,seg\text{-}I}$ = pipe segment I failure rate (events per year) that results in the initiating event, assuming the appropriate ISI for the given case: for risk prioritization, as discussed above in the paragraph entitled "Estimation of Failure Probability for Risk Categorizations"; for the current ISI program, the failure rates with the current ISI; and for the revised ISI case, the failure rates with the revised ISI program.

$CCDP_{IE,Seg\text{-}I}$ = Conditional core damage probability for the initiator for pipe segment I (determined from the accident sequences and associated minimum cut sets given the pipe failure as the initiating event.)

$FR_{IE,Seg\text{-}I}$, the pipe segment failure frequency (in events per year), is normally calculated using an appropriate SRRA computer Code, such as PRAISE, or other applicable codes, for the appropriate ISI case. However, the SRRA codes typically provide cumulative failure probabilities over a specified time interval (40 or 60 years for this application). To obtain the failure rate, one only needs to divide the cumulative failure probability by the number of years the plant is licensed:

$$FR_{IE,seg\text{-}I} = FP_{IE,Seg\text{-}I} / EOL$$

where

$FP_{IE,Seg\text{-}I}$ = pipe segment failure probability that results in the initiating event for the appropriate ISI case

EOL = number of years the plant is licensed (e.g., 40 years. If remaining years of plant license is <40 years, such as 20 years, then 20 years may be used as long as it accounts for aging/degradation effects over the 40 years of plant operation)

The conditional core damage probability is determined from existing PRA results or from solving the PRA model—if necessary to minimize truncation problems (see draft NUREG-1602).

Mitigating System(s) Consequence:

For pipe failures that cause only mitigating system(s) degradation or loss, the core damage frequency for the pipe segment is determined by the following equation:

$$CDF_{PB} = FP_{PB} * CCDF_{PB} \qquad \text{(EQN. A2-2)}$$

where

CDF_{PB}	=	CDF from a pipe failure (events/year)
FP_{PB}	=	Pipe break failure probability (dimensionless), for the appropriate ISI case
$CCDF_{PB}$	=	Core damage frequency (CDF), in events/year, given that the segment is failed (PB = 1), minus the CDF, given that the segment is not failed (PB = 0):

$$CCDF_{PB} = CDF_{PB=1} - CDF_{PB=0}$$

When calculating the pipe failure probability, FP_{PB}, the contribution of inservice testing of pumps should be addressed. An exposure time should be evaluated for a pipe segment and incorporated into the analysis. Exposure time is defined as the down time for the failed systems/trains, or the time the systems/trains would be unavailable before the plant is shutdown. It is a function of the test interval, the detection time, and allowed outage time (AOT). Two types of pipe failures may be distinguished, for which tests of active components may be useful in their detection. In the first type of failure, the pipe fails while the system is in standby, but the pipe failure is not detected until the next test. In the second type of pipe failure, the pipe degrades to the point where, on the next demand, either true demand or test demand, the pipe fails. In this second case one can call the pipe degradation occurring between tests a "latent" failure. The pipe does not fail until the stresses caused by the test or true demand occurs. In addition, pipe failures may be detected immediately in certain cases, and not require a test to reveal the failure. Examples are normally operating systems, such as the charging pump system.

The key attributes in determining the exposure time are the system states when the pipe failure is expected to occur (standby, test, or real demand), and the time required for the break detection (means available to detect diversion of the flow) (Ref. 31). The failure may be detected by different types of tests, and this should be taken into consideration. For example, some piping failures will be detected by monthly or quarterly pump surveillance tests; others will be detected only by full flow system tests occurring during refueling. The exposure time, when multiplied by the pipe failure rate, gives the probability that an accident sequence initiating event, occurring at a random time, will occur with the pipe failed, or in a latent failure condition, so that it will fail on demand. There is a second contribution to the increase in core damage frequency caused by a pipe break. Here, an initiating event occurs, and then the pipe break occurs during the mission time for the mitigating system (say, 24 hours). The probability of the pipe break here is the product of an operating system pipe failure rate times the mission time. The pipe break failure rate when the system is operating may be different than the pipe break failure rate when the system is in standby.

Two cases are distinguished. In the first case the system is normally in standby, and detection occurs during a test of an active component in the system. In the second case, the system is normally operating; detection is assumed to be immediate.

Piping Failures in Standby Systems

Here, one obtains

$$CDF_{PB} = (FP_{PB} / EOL) * T_{EXPOSURE} * CCDF_{PB}$$

where

FP_{PB} is the cumulative failure probability over the number of years the plant is licensed to operate for the appropriate ISI case.

EOL is the number of years the plant is licensed to operate (e.g., 40 years)

$$T_{EXPOSURE} = 0.5 * T_{Test\ Interval} + OT$$

The term OT (outage time) here may refer to the Allowed Outage Time (AOT), if the plant would, for the particular piping failure be maintained at power for the allowed outage time, but this term may also be the mean repair time for the piping segment, if the pipe is repaired in less than the AOT, or it may be only the time necessary for a controlled shutdown, if this is what would be done for the particular piping failure. The contribution of the pipe failure during the system mission time, after an initiating event has occurred, is here omitted. Because the mission time is short compared to the test interval, this term will have a small contribution.

To calculate the $CCDF_{PB}$, a surrogate component (basic event or set of basic events, such as a pump or valve) that is already modeled in the plant PRA is identified in which the consequence or impact on the CDF matches the postulated consequence for the piping failure. The surrogate component is assumed to fail with a failure probability of 1.0 and the PRA model is solved to obtain a new total plant core damage frequency. This is the conditional plant core damage frequency, given that the pipe is failed, denoted by $CDF_{PB=1}$. One also needs the plant core damage frequency, given the pipe is not failed, denoted by $CDF_{PB=0}$. However, since the piping component was not modeled in the PRA (very likely), this is just the base case PRA, so that $CDF_{PB=0} = CDF_{base}$. In any event, even if the pipe failure probability is in the base case PRA, its contribution is likely very small, and the CDF obtained is little different than the CDF with the pipe assumed not to fail. Therefore,

$$CCDF_{PB} = CDF_{PB=1} - CDF_{base}$$

Alternatively, one can calculate $CCDF_{PB}$ by isolating the cutsets associated with the pipe segment, and quantifying them (with the condition that the pipe segment failure probability equals unity).

The second method, the method of isolating the cutsets, permits one to perform an uncertainty analysis directly. If, instead, one calculates $CCDF_{PB}$ as $CDF_{PB=1}$ - $CDF_{PB=0}$, then, in performing an uncertainty analysis, one must take into account the correlations between $CDF_{PB=1}$ and $CDF_{PB=0,}$ arising from fact that the same basic events occur in both calculations, and, although there may be uncertainty in the values of the failure probabilities for these events, the uncertainty distributions are completely correlated: for example, even though the failure probability of a high pressure injection pump may be uncertain, it has exactly the same failure probability in both cases. The correlations can be taken into account by performing correlated Monte Carlo calculations.

Systems Continuously Operating:

For systems that are continuously operating before an initiating event occurs and are required to respond to the initiating event, the unavailability calculation may be calculated as:

$$FP_{PB} = FR_{PB} * (T_m + OT)$$

Where:

FP_{PB} is the failure probability for the appropriate ISI case
FR_{PB} is the failure rate (in events per unit time)
T_m is the total defined mission time (e.g., 24 hours)
OT, the outage time, is defined as for standby systems

From the fracture mechanics computer calculations, the failure rate (in hours) is estimated by:

$$FR_{PB} = FP_{EOL} / (\text{EOL years} * 8760 \text{ hrs/year})$$

This equation can also be applied to piping segments that are continuously under constant static pressure and are attached to storage tanks. Thus, the failure is identified by alarms and the segment unavailability is immediately recognized, thereby eliminating the need to consider detection time; the exposure time consists only of OT, the time between detection and repair or shutdown.

The distinguishing characteristic of continuously operating systems is the immediate detection of the pipe break. Also, since the system is continuously operating, it is legitimate to identify the operating failure rate as the pipe failure probability at the end of lifetime, divided by the plant lifetime. For standby systems, which spend most their time in standby, and not operation, it may not be possible to do this. For such systems, as was done above, the standby failure rate is the failure probability at the end of life, divided by the plant lifetime, and it would be more difficult to estimate the operating failure rate. However, as mentioned above, the term involving the contribution of the pipe failure during the system mission time is, for a standby system, small, and the difficulty in estimating the operating failure rate for such a system does not introduce any real difficulty in estimating the contribution to the core damage frequency of a pipe break in a standby system.

Initiating Event and Mitigating System Degradation Consequence:

For piping failures that cause an initiating event and mitigating system degradation or loss, core damage sequences involving both events simultaneously must be evaluated. To evaluate this case, the event tree for the initiator which is impacted by the piping segment failure is requantified with the surrogate component for the mitigating system assumed to be failed (that is, with a failure probability of unity). For piping failures that cause an initiating event and system degradation, the following equation is used:

$$CDF_{PB} = FR_{PB} * CCDP_{IE,Seg=1.0} \qquad \text{(EQN. A2-3)}$$

where

CDF_{PB}	=	Core damage frequency from a pipe failure (events per year)
FR_{PB}	=	Pipe failure rate (events per year)
$CCDP_{IE,Seg=1.0}$	=	Conditional core damage probability for the initiator with mitigating system component assumed to be failed

The conditional core damage probability for the initiator is determined by the following equation:

$$CCDP_{IE,Seg=1.0} = CDF_{IE,seg=1.0} / FREQ_{IE}$$

where

$CDF_{IE,seg=1.0}$	=	CDF from the initiating event with segment failed
$FREQ_{IE}$	=	Initiating event frequency

Recall that the failure probability calculated with an approved fracture mechanics Code is cumulative for the licensed period of the plan, and that the failure rate, for the appropriate ISI case, is therefore calculated as:

$$FR_{PB} = FP_{PB} / EOL$$

Special Cases:

When applying surrogate component methodology, cases may arise where not all of the pipe break locations fit into the three categories described above and on Figure A2.7. Each pipe segment is analyzed separately to determine the best calculational method. Some pipe locations may fall into several of these categories depending on the circumstances. For example, a failure in the piping segment in the charging system is postulated to result in a reactor trip and subsequent loss of RWST. This segment has two separate cases considered that are then added together to obtain the total core damage frequency for the segment. First, the segment is modeled as a reactor trip and loss of RWST using equation A2-3; then the segment is modeled as a loss of RWST for the remaining initiating events using equation A2-2.

Total Pressure Boundary CDF

Each piping segment within the scope of the program is evaluated to determine its CDF due to piping failure. Once this is computed, the total pressure boundary CDF is calculated by summing across each individual segment. This provides the baseline from which to determine the risk importance measures of the segments that can then be used to categorize the segments within ISI-issue. The total pressure boundary CDF provides a measure of the risk associated with the ISI program. The difference between the CDF calculated using the existing licensing basis program and the RI-ISI program describes a measure of the change in risk. For consistency, the base PRA should include the realistic pipe failure rates established for the RI-ISI pipe segments.

A2.7 Selection of Locations To Be Inspected

This section provides guidelines and describes an acceptable approach for selecting pipe segments and structural elements (e.g., welds) for inspection in accordance with the risk-informed inservice inspection programs. The selection of locations should be based on the following considerations:

- The selected group of pipe segments and structural elements identified in the ISI programs should continue to meet the intent of all existing deterministic requirements for structural integrity, such as defined by 10 CFR 50.55a, Appendix A to 10 CFR Part 50, and the ASME Pressure & Vessel Code, Section XI.

	LOW-SAFETY SIGNIFICANT SEGMENT	HIGH-SAFETY SIGNIFICANT SEGMENT
HIGH-FAILURE POTENTIAL SEGMENT	OWNER DEFINED ELEMENT INSPECTION PROGRAM	SUSCEPTIBLE ELEMENT LOCATION
LOW-FAILURE POTENTIAL SEGMENT	ONLY SYSTEM PRESSURE TEST & VISUAL ELEMENT EXAMINATION	ELEMENT INSPECTION LOCATION SELECTION PROCESS

ELEMENT 2 (Continued)

- The proposed inspection program, including the set of selected ISI locations, should meet the probabilistic criteria as described in Section 4.4.

To meet the intent of existing requirements, the program should identify a set of inspection locations for which:

- Failures will have greatest potential impact on safety, and

- There is a greater likelihood of detectable degradation and consequently a greater potential for identifying piping degradation prior to failure.

Section A2.7.1 describes one acceptable method for classifying pipe segments. Section A2.7.2 describes guidelines for categorizing structural elements within pipe segments based on the likelihood for failure and safety classification associated with each pipe segment. Section A2.7.3 discusses one acceptable strategy for inspections based on performance measures.

A2.7.1 Methods of Selecting Pipe Segments for Inspection

To maintain consistency with the intent of current regulatory criteria for inservice inspection programs, the selected pipe locations for inspection should address:

- Locations where failures would have greatest potential impacts on safety, and

- Locations where detectable degradation and consequently potential piping failures are more likely to occur first.

For some segments, welds in certain locations are known to be more vulnerable to developing flaws or increasing the flaw size than other welds. If some welds are known to be more vulnerable than others, the welds with the higher conditional probability or frequency for a flaw growing to a leak should be sampled first [e.g., in a straight run of pipe, the degradation and fluid conditions may be similar for all elements, or welds. However, structural mechanics analyses may be able to identify a subset of the elements as exhibiting relatively greater stresses and potentially greater (though still minimal) likelihood for identifying degradation when compared to all the elements in the segment]. While this procedure is biased compared with random sampling, it is biased in a conservative direction, provided only that the average flaw probability of the welds in the sample is larger than the average flaw probability of all the welds in the segment. If there are some welds which are never sampled because they are inaccessible, the bias that is introduced by this constraint can still be conservative, provided that the average flaw probability condition stated above still holds.

Risk Importance and Categorization - This section identifies an acceptable approach to incorporate risk insights for selecting inspection locations. Quantitative calculations of risk contributions, as identified in previous sections, are used to demonstrate that the risk contribution criterion is satisfied.

To augment engineering calculations and engineering judgment traditionally used by licensees to select pipes for inspection, this section identifies one acceptable approach to incorporate quantitative risk insights in categorizing pipes in terms of failure potential and safety significance. These guidelines are based on quantitative information from PRAs and calculated failure probabilities for pipes. The calculations provide the input needed to apply risk importance measures, which provide a means for categorizing pipe segments and structural elements in terms of their associated risk to the public. This regulatory guide does not imply that licensees can only base the selection process on calculated risk importance, although the use of such measures can facilitate the selection of an optimum set of ISI locations commensurate with risk.

For ISI prioritization or categorizing, a modified Fussell-Vesely (FV) importance measure (FV_{ISI}) can be used to categorize components (i.e., pipe segments) selected for ISI examination. Use of importance measures generally requires the determination of the total CDF or LERF. For ISI importance measures, the total CDF or LERF used in calculating the modified FV importance measure should be determined by summing the contributions of all pressure boundary failures in the plant piping systems. This ensures that the categorization of the pipe segments for ISI consideration is focused, such that the ISI programs developed from this categorization will ensure that important pressure boundary failures in plant piping systems do not become major contributors to total plant risk (i.e., CDF or LERF) as a result of unexpected or age degradation mechanisms.

Failure Potential Estimation - In this method, historical or service data, deterministic insights (e.g., material, fluid chemistry, loadings, and inservice experience from the pant and industry), expert judgment, and/or structural reliability/risk assessment calculations are used to estimate pipe segment and structural element failure probabilities. The preferred approach is to use structural reliability codes validated with applicable data. As highlighted in Section A2.5.3, if an expert judgment elicitation process is required, then it should be performed generically through the ASME or an industry group and incorporated into the structural reliability computer Code. ***Use of expert elicitation should be reported to the NRC for information.*** The guidance of Section A2.5.3 applies to the estimation of failure probabilities. The use of conservative assumptions in estimating failure probabilities (to address uncertainties) should only be used as part of sensitivity studies to assess the impact on categorizing components. Results of such sensitivity studies should be addressed in the decisionmaking process.

Importance Measures - General guidelines for risk categorization of components using importance measures and other information are provided in Appendix A to draft Regulatory Guide DG-1061. These general guidelines address acceptable methods for carrying out categorization and some of the limitations of this process. The basic elements to be considered when implementing importance measures include:

a. Truncation Limits
b. Different Risk Measures
c. Completeness of Risk Model
d. Consideration of all Allowable Plant Configurations and Maintenance States
e. Sensitivity Analysis for Component Data Uncertainties
f. Sensitivity Analysis for Common Cause Failures
g. Sensitivity Analysis for Recovery Actions
h. Multiple Component Considerations
I. Relationship of Importance Measures to Risk Changes
j. SSCs not included in the Final Quantified Cut Set Solution

In calculating risk importance measures for the categorization process, the failure probabilities used for each pipe segment should ***not*** credit ISI inspections or leak detection, except for those in an augmented inspection program. (Note, this is not the case when evaluating the change in the CDF and LERF, as addressed in Section 4.2.) Guidelines that are specific to the ISI application are given in this section. As applied here, risk categorization refers to the process for grouping ISI components into LSS and HSS categories.

Risk importance measures from the PRA may be used as one of the inputs to the categorization process. Some components of interest to RI-ISI may not be addressed in the existing PRA, and so there is no quantified risk importance information for these components. When feasible, adding these components to the PRA should be considered by the licensee. In cases where this is not feasible, detailed discussions should accompany the application request that addresses how traditional engineering analyses and judgment (e.g., integrated decisionmaking process) were applied to determine if a component should be categorized as LSS or HSS.

In addition to component categorization efforts, the determination of safety significance of components by the use of PRA-determined importance measures is important for several other reasons:

- When performed with a series of sensitivity evaluations, it can identify potential risk outliers by identifying ISI components which could dominate risk for various plant configurations and operational modes, PRA model assumptions, and data and model uncertainties.

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

- System level importance results can provide a high level validation of component level results and can provide guidance for categorizing ISI piping not modeled in the PRA.

While categorization is an essential step in defining how the RI-ISI program will be implemented, it is not an essential part of ensuring the maintenance of an acceptable level of plant risk. The sensitivity of risk importance measures to changes in ISI strategy (i.e., proposed for RI-ISI) can be used as one input to the overall understanding of the effect of this strategy on plant risk. However, the traditional engineering evaluation, augmented with the calculation of change in the overall plant risk, provide the major input to the determination of whether the risk change is acceptable or not.

Criterion for Selection - Table A2.6 summarizes the guidelines used in the identification of high-safety-significant pipe segments to be used in making the final selection of inspection locations. **The total CDF or LERF in the risk significance evaluation should *only* account for those contributors associated with pressure boundary failures in piping systems.** Pipe segments that exceed the FV_{ISI} importance measure guideline range[3] in Table A2.6 are classified as having a high safety significance. (Note: for this application, the denominator in the FV importance is limited only to the cumulative contribution of all pipe segments. It does not include contributions from other system components.) Those segments with a value less than the range given in Table A2.6 are classified as having a low safety significance (LSS). The risk measures are then supplemented by sensitivity studies to provide estimates in the variability of these measures. The final categorization into HSS and LSS is performed using additional deterministic and qualitative insights and information. Plant design and operating features and their relationship to component categorization should be explored and understood; in some cases, this will result in changing a component's ranking or category from what it might otherwise have been if based solely on the PRA results, and allow categorization of components not analyzed in a PRA.

The use of the Risk Achievement Worth (RAW) is an important measure that provides the risk impact from a pipe segment failure. It is the conditional core damage probability or conditional core damage frequency calculated for the pipe segment depending upon whether the pipe failure causes system unavailability degradation or an initiating event. The RAW identifies pipe segments whose failure has high risk impact and high safety impact and which needs consideration.

The categorization, where the judgment of the ISI team of experts is needed, is reached by consensus. Although most decisions of the ISI experts will be reached by 100% consensus, there will be times when differing professional opinions will exist. These differences must be documented.

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[3]The criterion in Table A2.6 is provided as guidance. Other criteria can be proposed by a licensee. If such criteria are proposed, the licensee must provide sufficient justification to ensure that important pressure boundary failures in plant piping systems do not become major contributors to total plant risk as a result of unexpected or age degradation mechanisms.

Table A2.6 Approach to Overall Risk Significance Determination for Alternative Risk-Informed Selection Process for Inservice Inspection

Risk Importance Measure	Criteria[a] Pipe Segment
Quantitative Measures:	
Modified FV Importance Measure (FV_{ISI})	$>$ 0.001 – 0.005
Risk Achievement Worth (RAW)	The utility's submittal should identify a RAW value such that if RAW ≥ a utility defined value for either CDF and LERF, the pipe segment could be considered as important If RAW < the defined value, then the pipe segment could be considered as less important
Sensitivity Studies	Uncertainties
Qualitative Input	**Items to be considered in the establishment of qualitative criteria** * Level of Redundancy * System Trains * Groupings of Components into Supercomponents for modeling purposes * Truncation limits during quantification * Operational Histories * Others

[a] These example criteria apply to the use of a total CDF_{PIPING} or $LERF_{PIPING}$, which is the total CDF or LERF attributed to pressure boundary failure in plant piping systems. A range of values is provided. The basis for the final selection criterion used in the submittal should be justified.

The process used to categorize the segments should be documented in a licensee's submittal to the NRC.

Cumulative Risk Contribution - In addition to the criterion of Table A2.6, the approach identified in this regulatory guide requires a supplemental calculation at the pipe segment level. This calculation should demonstrate that the risk-informed analysis of piping identified the piping that contributed 95% of the plant risk CDF_{PIPING} **AND** $LERF_{PIPING}$. Figure A2.7 is provided as an illustration of a cumulative CDF risk diagram for a plant's piping.

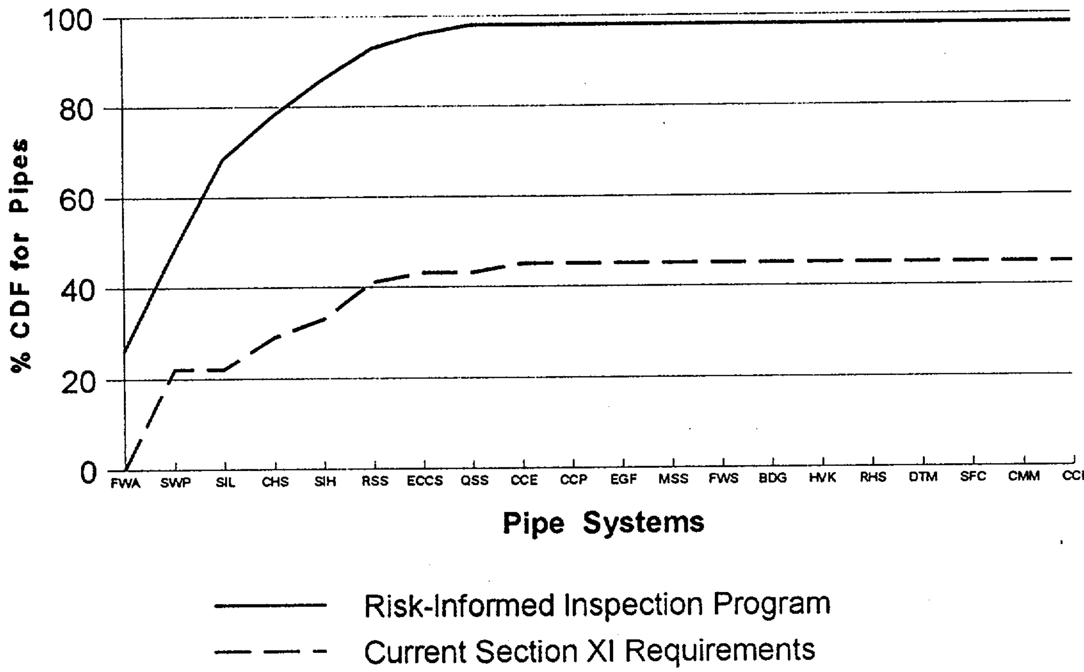

Figure A2.7 Cumulative Risk Contribution of a Plant's Piping

Engineering Considerations - While risk importance can guide the selection process, there are other deterministic considerations that should be integrated into the decision making process to ensure that the results of the selection process continue to meet the existing criteria, such as 10 CFR 50.55a and the ASME Section XI. Such engineering considerations include:

- **Early Detection of Degradation Mechanisms** - A goal of inservice inspection is the early detection of new and unexpected degradation mechanisms. Accordingly, the selection of ISI locations should include locations where degradation is first expected to develop. These locations may or may not be the same locations with the greatest risk contributions as identified by calculations of risk importance based on estimated consequences of failures and break probabilities.

  The risk selection process should include a sample of representative locations within each piping system identified as contributing to risk, thereby enabling the detection of degradation mechanisms that may be active within the system. These locations should, in part, correspond to locations for which the probability of degradation is considered greatest, independent of the calculated risk importance parameters.

- **Leak Versus Break Probabilities** - The selection process, including the calculations of risk importance, should use leak probabilities. Consideration of leaks is appropriate since the risk importance is only intended for use in the categorization and selection process to indicate priorities based on the relative benefits to be gained from inspecting a particular location.

Use of leak probabilities to augment the selection of inspection locations is also consistent with the stated objective of early detection of degradation mechanisms. In this context a pipe leak criterion serves to establish a definition for a significant level of degradation.

It is also acceptable to use leak probabilities because the uncertainty for calculated leak probabilities is less than the uncertainties for calculated break probabilities. Similarly there is less uncertainty in the estimation of leak probabilities versus break probabilities.

Structural mechanics models often calculate very low probabilities that through wall defects will result in breaks rather than pipe leaks. However, the associated fracture mechanics calculations are based on many uncertain modeling assumptions and inputs (i.e., inputs for the defect sizes, defect growth characteristics, and leak detection capabilities) which can significantly impact the likelihood for pipes to break rather than to leak. The use of leak probabilities in the categorization and selection process minimizes the effects of this issue.

If the calculated leak and break probabilities are similar for a pipe segment and the pipe segment is not found to be important from a risk viewpoint, but is on the borderline, then consideration should be given to add the pipe segment to the list for inspection due to non probabilistic considerations.

- **Operational Insights** - Reviews of the selected pipe segments should ensure that the proposed inspection program includes insights from operational and maintenance experience, using both information from the plant and relevant information from other plants.

- **Defense-in-Depth** - Reviews of the selected pipe segments should identify any proposed relaxations of inspection requirements from prior practices and assess that effect on plant safety.

Relationship to Augmented Inspection Programs - Mandated programs for augmented piping inspections (e.g., boiling water reactor piping for stress corrosion cracking and balance of plant piping for wall thinning by erosion/corrosion) should be taken into consideration when selecting locations for inservice inspection. It is acceptable to coordinate otherwise independent inspection programs by selecting common locations to the extent possible. This regulatory guide does not eliminate the need to comply with the requirements of existing augmented inspection programs in effect at the plant.

A2.7.2 Structural Element Selection Within Pipe Segments

The plant ISI engineering team reviews all pertinent information and determines the final safety classification for each pipe segment included within the scope of the risk-informed ISI program. The team uses qualitative and quantitative information associated with PRA and failure probability calculations in combination with classic engineering insights and design basis information to develop the final classification categories of high-safety-significant and low-safety-significant pipe segments. This information is then used to develop a matrix to assist in the selection of structural elements for examination, as shown in Figure A2.8, for all pipes included in the risk-informed ISI program.

The criteria for determining how many structural elements should be selected for examination are based on the safety significance of the segment and the failure likelihood within that segment.

The risk calculations used to support the safety significance determination involve combining consequences with pipe failures that are initiating events and/or with pipe failures that occur on demand as a result of a plant event. Engineering insights and design basis information also provide input to the classification of a segment as high-safety-significant. In addition, the process is well established for the plant's engineering team, or expert panel (if used), to confirm that the segments were properly classified as either high- or low-safety-significant.

The probability for pipe failure directly drives the need for an effective examination method(s). This attribute is categorized by a demarcation of "high-failure-potential" (HFP) versus "low-failure-potential" (LFP) (see Figure A2.8) using the following definitions:

High Failure Potential - As determined by the engineering team,[4] a segment is of high failure potential if it has either an active failure mechanism that is known to exist, which may be currently monitored as part of an existing augmented inspection program, or alternatively is analyzed as highly susceptible to a failure mechanism, which could, in the future, lead to a leak or break. The ISI team applies engineering insights such as material, fluid chemistry, loadings, and inservice experience from the plant and industry experience to make this determination. Examples of failure mechanisms that would typically result in this classification are excessive thermal fatigue, corrosion cracking, primary water stress corrosion cracking, intergranular stress corrosion cracking, microbiologically influenced corrosion, erosion-cavitation, high vibratory loadings on small diameter pipes, and flow-accelerated corrosion.

Low Failure Potential - As determined by the engineering team, a segment meeting this description would not meet the above criteria for a high failure potential segment. Examples that would typically result in this classification would have no known failure mechanisms other than fatigue based upon normal and design basis loadings.

Probabilistic insights from SRRA results are used to confirm the engineering team's determinations. A segment should be considered to have a "high failure potential" if at any element in that segment exceeds any one of the two following criteria:

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[4]The engineering team, which is sometimes called the "engineering subpanel," the "component ISI team," or "focused structural element expert panel," consists of the following expertise:

- Inservice inspection program
- Non-destructive examination methods
- Piping stress & materials
- Plant/industry failure, repair & maintenance experience

	LOW-SAFETY SIGNIFICANT SEGMENT	HIGH-SAFETY SIGNIFICANT SEGMENT
HIGH FAILURE POTENTIAL SEGMENT *(Degradation mechanism usually present)* $P_{LEAK} > 10^{-5}$ -- 10^{-4} or $P_{BREAK} > 10^{-8}$ -- 10^{-7} per 40 year operating life	OWNER DEFINED *ELEMENT* INSPECTION PROGRAM *(Incorporates Augmented Inspection Programs)* 3	SUSCEPTIBLE *ELEMENT* LOCATION *(100% Inspection or NRC Approved Owner Inspection Program)* 1
LOW FAILURE POTENTIAL SEGMENT *(Degradation mechanism not usually present)*	ONLY SYSTEM PRESSURE TEST & VISUAL *ELEMENT* EXAMINATION 4	*ELEMENT* INSPECTION LOCATION SELECTION PROCESS *(or NRC Approved Owner Inspection Program)* 2

Figure A2.8 Structural element selection matrix

(1) $P_{LEAK} > 10^{-5} - 10^{-4}$ per 40 year operating life

(2) $P_{BREAK} > 10^{-8} - 10^{-7}$ per 40 year operating life

SRRA sensitivity studies have been performed which have shown that pipe locations with failure probabilities below these values are essentially benign. Piping systems that do not exhibit a leak-before-break attribute could exceed the above break probability criteria even if the leak probability is determined to be less than the leak probability criterion. *In such cases, the break criterion would dictate that the segment be classified as "**high failure potential.**"*

Figure A2.8, illustrates a four-region matrix for identifying locations for periodic examinations. The *safety significance* matrix is based on the probabilistic categorization of the pipe segments[5]. The failure potential matrix applies SRRA tools, as appropriate. Each of the four regions has an examination rule base as follows:

Region 1 All susceptible locations in the segment identified by the engineering team as likely to be affected by a known or postulated failure mechanism, must be inspected. Exceptions include existing augmented programs[6] or other inspection programs approved by the NRC.

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[5]While the initial categorization of pipe segments is based on probabilistic considerations, the utility is free to increase the safety significance of any pipe segment for reasons of their own choosing.

[6]Segments with failure modes that have established augmented programs (e.g., flow-assisted corrosion, intergranular stress-corrosion cracking) would be inspected in accordance with that existing program.

Region 2 The engineering team selects locations for examination in these segments based on the guidance provided in Section A2.7.3.2. In this region, a low failure potential was identified. In most cases, fatigue is anticipated to be the failure mechanism. Based on the guidance provided, portions of the pipe segment that would experience the highest loads or highest degradation potential, would generally be selected for inspection. If the degradation potential is equally dispersed among the elements in a lot, then a random element(s) may be selected. At a minimum, one element will be examined to account for uncertainty and unknown degradation mechanisms in the segment or lot, and to guard against CCF. *The NRC will consider other owner inspection programs, as justified.*

Region 3 All susceptible locations in the segment identified by the engineering team as likely to be affected by a known or postulated failure mechanism, and that are not already in an augmented program, will be examined in accordance with an Owner Defined Program and reported to the NRC (if not already reported). While failure of these segments would have a minimal safety impact, the impact on plant operations may be significant in terms of unplanned outage time, repair costs, and other consequential impacts.

Region 4 Only system pressure tests and visual examinations are required for segments of low failure potential and low-safety-significance.

System pressure tests and visual examinations are performed for pipes in Regions 1, 2, and 3, as well.

<u>Guidelines for Selection of Locations in Regions 1 and 2</u>

The risk-informed selection process includes assessments and evaluations of the pipe structural elements in each of the high safety-significant pipe segment. These structural elements include the following examination items:

- all pipe welds, including those to nozzles, valves and fittings such as elbows, tees, reducers, branch connections, and safe ends

- areas and volumes of base material and examination zones such as weld counterbore areas and fitting material, as appropriate.

Welded attachments and pipe supports are not included in the assessment and evaluations.

For the high-safety-significant pipe segment exhibiting low-failure-potentials, at a minimum, one location in each pipe segment must be inspected. The number of inspection locations is based on a statistical sampling technique outlined in Section A2.7.3.2 and Appendix 4.

Should a pipe segment (categorized as high-safety-significant ***and*** high-failure-potential-Region 1) consist of several elements (e.g., welds), of which the majority of the elements exhibit low-failure-potential, then the licensee may consider separating the elements into two lots. One lot requiring 100% inspection (HSS and HFP lot) and the other lot (HSS and LFP lot) requiring an inspection program similar to that required for Region 2. **Such separations should be justified and documented in the RI-ISI submittal to the NRC.** Simplified P&IDs showing the segment boundaries are reviewed along with piping isometrics, plant and industry operating experience, the previous pipe segment evaluations performed to determine the high-safety-significant pipe segments and system design, fabrication, and operating conditions. Based on the postulated failure mechanism and the loading conditions for the pipe segment, the areas in which this failure mechanism is most likely to occur are identified considering the following factors:

Configuration Dependent. This factor considers the effect of piping layout and support arrangement. For example, piping with low flexibility for thermal expansion will experience high bending moments which, in turn, can drive crack growth.

Component Dependent. For example, socket welds have low resistance to sustained vibration. Elbows or piping immediately downstream of valves, which add turbulence to the flow, are locations susceptible to erosion-corrosion-wear.

Materials/Chemistry Dependent. Intergranular stress corrosion cracking (IGSCC) and dissimilar metal welds are examples of how materials and chemistry can play a role.

Loads Dependent. An example of this is the number of cycles seen by the piping segment. Another example is piping where inadvertent operation may lead to water hammer events. Seismic events are also included in this category.

Determination of the inspection location(s) within a pipe segment is dependent on the above factors. In general;

Component dependent failure modes are usually localized to a single or small number of locations.

Materials dependent or operations dependent mechanisms are often present throughout the segment. In such cases, interactions with other effects must be considered for determining the location(s).

Load dependent failure modes typically involve undetected preexisting flaws or degradation that could fail under high loads. The high loads could arise from dynamic (seismic, water hammer) events, large thermal expansion loads (configuration dependent), or external loading. Locations where such loads could have the greatest impact can often be determined.

Table A2.7 provides some additional insights based on postulated failure mechanisms that assist in identifying the susceptible areas of pipes.

A2.7.3 Inspection Strategy - Reliability and Assurance Program

The previous sections focused on: assessing the changes to public risk by modifying existing ASME Section-XI ISI programs with risk-informed ISI programs; and categorizing pipe segments as high- and low- safety significant with high- and low- failure potential. An acceptable structural segment selection matrix guideline is illustrated in Figure A2.8. Once a pipe segment is categorized via the selection matrix guideline, different inspection programs are applied based on their safety significance. As illustrated in Figure A2.8, the order from most to least safety significance is: Region 1, 2, 3, and 4. This section addresses Region 2, a segment categorized as high-safety significant with low failure potential.

Any proposed inspection strategy should:

1. Define a reliability goal for piping systems;
2. Define a method that strives to meet the reliability goal;
3. Quantify the existence of a flaw or probability for a leak in a weld;
4. Consider that the inspection technique is not perfect;
5. Consider that not every weld will be inspected;
6. Consider the implication for calculating confidence or assurance that the inspected sample contains non of the defective welds in the lot; and
7. Demonstrate that the final results provide reliability and assurance that the reliability goal will be achieve.

The target reliability goals are addressed later in Section A2.7.3.3.

One acceptable method for addressing the above seven elements is discussed in Appendix 4. This method is consistent with the ASME Code Case N577, Case A. The method integrates statistical techniques with input from fracture mechanics calculations and/or data for flaws.

A2.7.3.1 Risk-Informed Lot Selection and Element Selection for Inspection

In the previous sections, seven elements were identified for consideration when developing a statistical inspection sampling program. One acceptable application of a weld sampling technique was identified in Appendix 4. This sampling process is used in Region 2 of Figure A2.8 matrix, where the ISI engineers are unable to differentiate the elements (welds) within a pipe segment as having significantly different probability for degrading. How does one inspect a pipe segment categorized high-safety-significant and high-failure-potential where only one element (weld) in the segment experiences an active degradation mechanism, and the balance of the welds have similar low failure potential? One acceptable method is to place the outlier element in one lot, requiring 100% inspection, and subsume the balance of the elements in a separate *lot* for statistical sampling, as described in the previous sections.

The concept of a *lot* can be broadened into more than one pipe segment. That is, several pipe segments with similar elements (e.g., same low failure potential, no known degradation mechanisms, same environmental conditions, etc.) may be subsumed within one *lot* for purpose of statistical inspections. An example may be all welds attaching the cold legs to the reactor vessel inlet nozzles. **Any collapsing of segments or elements within a segment into one lot will require NRC review and approval.**

A2.7.3.2 Sequential Sampling

This section addresses the guidance for additional examinations should an inspection identify unacceptable degradation in a pipe. The Assurance Level Sampling or Global method, addressed in Appendix 4, identifies the number of welds that should be inspected. The RI-ISI engineers select the limiting weld or the weld most likely to degrade first, as the

Table A2.7 Insights for Identifying Inspection Locations

Failure Mechanism	General Criteria	Susceptible Areas
Thermal Fatigue	Areas where hot and cold fluid mix, areas of rapid cold or hot water injection, areas of potential leakage past valves separating hot and cold water	Nozzles, branch pipe connections, safe ends, welds, heat-affected zones, base metal, areas of concentrated stress
Corrosion Cracking	Areas exposed to contamination and areas with crevices; high stresses (residual, steady-state, pressure), sensitized material (304 SS) and high coolant conductivity are all required; lack of stress relief or cold springing could also lead to residual stresses	Base metal, welds, and heat-affected zones
Microbiologically influenced corrosion	Areas exposed to organic material or untreated water	Fittings, welds, heat-affected zones, crevices
Vibratory Fatigue	Configurations susceptible to flow induced vibration and flow striping or for vibratory resonance with rotating equipment (pump) frequencies	Welds, branch pipe connections
Stress Corrosion Cracking	Areas of high oxygen and stagnant flow	Austenitic steel welds and heat-affected zones
Flow accelerated corrosion	Areas of low chromium material content, high moisture content, and high pH, high pressure drop or turning losses	
Low cycle fatigue	Areas with high loads due to thermal expansion for heat-up and cool-down thermal cycling.	Equipment nozzles and other anchor points, near snubbers, dissimilar metal joints
Others?		

first weld to be inspected. Presumably, this would be the weld that was used in the classification of the segment as a low failure potential. Next, the failure frequency attributed to this limiting weld is then conservatively assumed to apply to all the other welds in the lot, so that a conservative estimate of assurance (by use of the binomial distribution) is generated. The only time a random selection of a weld would occur is when engineering analysis can offer no guidance as to which element is most likely to degrade.

If the inspection uncovers a flaw, then the "Additional Examinations" requirement of Section XI (IWB-2430, page 82) would still be applied (paraphrased):

- If no flaws are found in the first sample(s), then stop (note that this implies a "zero defect acceptance criterion" as discussed in Appendix 4).
- If one or more flaws are found, then take another sample equal in size to the first sample.

- If one or more new flaws are found, inspect the rest of the lot.

The risk-informed process is consistent with the experience gained from the ASME Code. One acceptable performance guideline is striving for a 95 percent probability that the occurrence of a leak would not exceed a frequency of 1E-06/yr/weld. If that performance guideline is not met, then a root cause analysis is performed and the inspection period and number of locations will be recalculated based on the new information.

Implementing this approach in the eight element example in Appendix 4, if only one of the eight elements has a high failure potential, then that one element is allocated its distinctive *lot* and the balance is combined into a separate *lot* for inspection purposes. Thus, the one-element lot will be inspected (Section 1 of Figure A2.8 - 100% inspection or NRC approved owner program) and the remaining seven elements (if not combined with elements from other segments) will be sampled based on the guideline of a 95% probability that the development of a leak will not exceed a frequency of 1E-06/yr/weld.

A2.7.3.3 Historical Failure Data and Target Reliability Matrix Guideline Criteria

Studies performed by Dr. Spencer Bush indicate that the frequency of leaks from pipes at nuclear plants has shown some decreasing trends over the years of plant operations. For the exiting population of plants in the U.S. (approximately 110), the industry observes a total of about 100 leaks per year. These leaks are primarily from the balance of plant systems, such as corrosion type failures due to poor quality water in copper nickel tubing. In safety related systems (including the RCS) the small number of failures appeared to be focused at small diameter branch piping, such as a vent line near an RCS pump whose failure mechanism is vibration fatigue. The ratio of leaks to breaks is a function of the failure mechanism involved, among other factors, and can be as large as 1:1 for erosion/corrosion to 1000:1 or less for intergranular stress corrosion cracking.

On the average, there are about 10,000 welds (or structural elements) in pipes at a typical plant. From this estimate, a leak frequency of (100 leaks per year)/(110 plants)/(10,000 welds/plant) or ~ 1E-04 leaks per weld per year can be calculated. This would include pipes of all sizes, all systems and all failure mechanisms.

The RCS pipes (Class 1), however, have experienced lower leak rates than the overall leak rate for all of the plant's pipes. Estimates for pipe failures for a PWR RCS are less than 1E-06 per weld per year. This performance standard is a conservative representation of the operating experience for Class 1 pipes under the existing ASME requirements and is one acceptable target goal for RI-ISI application for high-safety-significant pipe segments.

Applying the above data, the following trends for pipe leak frequencies have been observed:

All pipes	--	1E-04 per weld per year
RCS pipes	--	< 1E-06 per weld per year

Further analysis of nuclear power plant operating experience has led to categorizing detectable piping leak rates, as identified in Table A2.8.

Table A2.8 Operating Experience Insights to Leak Frequencies

LEAKS - 1965 - 1996			
MATERIAL	**PIPE SIZE**	**# OF FAILURES**	**LEAK FREQUENCY (leak/yr- weld)**
Stainless Steel	≤ 1 -inch	546	18 E-06
Ferric Steel	≤ 1 -inch	414	13 E-06
Stainless Steel	>1 ≤ 4	290	10 E-06
Ferric Steel	> 1 ≤ 4	136	4 E-06
Stainless Steel	> 4	170	5 E-06
Ferric Steel	> 4	253	8 E-06

Referring back to the statistical sampling technique described in Appendix 4, Table A2.9 provides an example of a potential matrix guideline for implementing RI-ISI programs on *high-safety-significant pipes*. It is anticipated that these goals such as these would be achieved with a 95% assurance level for only that part of the system categorized as high-safety significant. For example, if a system consists of 20 segments, 10 of which are categorized as high-safety significant, the leak target goal would only apply to the 10 segments as a system. A licensee should identify and justify the leak target goals it intends to monitor.

Table A2.9 Target Detectable Leak Frequency Goals

LEAK TARGET GOALS		
MATERIAL	**PIPE SIZE**	**TARGET LEAK FREQUENCY (leak/yr- weld)**
Stailess Steel	< 1 -inch	<1 E-05
Ferric Steel	≤ 1 -inch	<1 E-05
Stainless Steel	>1 ≤ 4	<1 E-05
Ferric Steel	> 1 ≤ 4	<1 E-06
Stainless Steel	> 4	< 1 E-06
Ferric Steel	> 4	< 5 E-06

As addressed in Appendix 4, an input for the binomial distribution in the Assurance Level Sampling Method is the probability of a flaw. One acceptable definition of a flaw is to apply the ASME definition [e.g., a flaw whose depth exceeds about 10% of the wall thickness (a/t ~ 0.1)]. *This does not imply that the flaw is unstable and will lead to a through the wall crack.* It is a flaw that requires additional analyses. For example, a typical probability for an unacceptable flaw for a large pipe in a PWR RCS may be on the order of 3E-03/year/weld. For a weld containing such a flaw, the probability of a detectable leak is on the order of 4.3E-08 per year per weld, for a disabling leak it is 5.1E-10 per year per weld, and for a break it is 3.0E-13 per year per weld. The probability of a flaw is calculated with the structural mechanics model, discussed in Appendix 1. Application of the sampling model should account for the uncertainties in the calculated probability of a flaw per weld per year, and account for that part of the system categorized as HSS, using appropriate goals for each segment to achieve the system performance target goal.

The above matrix guideline is conservative in that a detectable leak is used as the figure merit. Meeting these guidelines maintain, as a minimum, the current level of safety provided by the existing ASME Section XI Code, and would likely result in increased safety as the RI process expands the regulatory scope of inservice-inspection to other systems not currently addressed by Section XI, and potentially a decrease in radiation exposure to plant personnel..

A2.7.3.4 Inspection Location Summary

This section addressed one acceptable method for pipe segment classification (high versus low safety significant classification; high versus low failure potential; etc.), it discussed the use of a statistical procedure for selecting the number of welds to be inspected (e.g., the Assurance Level or Global method for high-safety significant with low failure potential), it addressed (through reference to Appendix 4) one acceptable method for incorporating uncertainties in the inspection technique (probability of detection), and it addressed sequential sampling where the initial testing identified potential flaws.

Focusing our attention on the high-safety significant with low failure potential elements, once the number of locations to be inspected (among the total number in a given log) has been established, the next step in the procedure is to select the actual inspection locations. It should again be noted that all locations for the lots of interest will have low failure potentials, and that the number of sample locations on a percentage basis will be small. The objective for the sample inspections is to detect degradation using a strategy that inspects those locations where degradation is first most likely to occur, along with inspections of different types of structural elements (welds, fittings, etc.), thereby providing diversity to the sample set.

The estimated failure probabilities for the low-failure potential elements will typically have been assigned to a common small value (e.g., the limiting element in a lot) for purposes of risk-categorization calculations. Nevertheless, the selection of sample locations for inspections should be based on a location where degradation is most likely to occur. These evaluations can be based on consideration of factors such as identified in the previous sections, as well as the failure mechanisms and susceptible areas listed in Table A2.5. Results of probabilistic structural mechanics calculations and data from operating experience can also guide the selection. When fatigue is the failure mechanism of concern, the criteria from ASME Section XI can also provide useful guidance by directing attention to terminal ends, locations of high calculated stress and fatigue usage factors, and dissimilar metal welds.

For high-safety-significant elements with high failure potential, 100% inspection or an NRC approved owner's program is required. An example of an NRC approved owner's program is the *erosion-corrosion program.*

Final Selection Process - It is the responsibility of a licensee to ensure that the categorization of elements and the location of inspections are performed in accordance with sound engineering practices and licensing requirements. This regulatory guide does not endorse one method over another. In other risk-informed programs (i.e., maintenance, inservice testing, technical specifications, graded quality assurance, etc.), the industry incorporated the use of an expert panel for providing plant management the information it requires to render its decision. Whether an expert panel is used or not, the issues that an expert panel addresses need to be addressed in the process. These issues include:

- Concurrences that the systems included in the scope of the program are correct and that no other systems should be included/excluded

- Verification that the system boundaries are adequate

- Verification that the consequences assumed for each piping segment are accurate (both direct and indirect effects)

- Concurrence that shutdown risk, containment performance, operational history, etc. have been appropriately considered in the analysis

- Verification that appropriate operator recovery has been considered (i.e., consideration of available indications, timing, and alternate actions)

- Upgrading the safety significance of a pipe segment based on economic or other considerations that are outside the regulatory program through a consensus process and documenting the basis for such an upgrade

- Concurrence that the structural elements selected for examination and the type of examination method selected meets the requirements of the program

- Integrate the insights from other risk-informed programs for consistency and proper coverage.

A review group or panel cannot downgrade a high-safety-significant pipe to a low-safety-significant (LSS) pipe if it comports with the guidelines in this report.

In rendering the final decision, the licensee ensures that the program solicits experts in the areas of PRA and engineering disciplines to develop a final list of high-safety-significant pipe segments. As indicated above, the licensee can select to inspect pipes for factors other than the decision criteria identified in this chapter. Such factors might include economic considerations that have no safety impact or other non-safety considerations as deemed appropriate by the utility. Inspections based on non-safety considerations (upgrading pipes no ranked high-safety-significant), *are not considered under this regulatory guide.*

For consistent application of risk-informed programs, it is recommended that the licensee incorporate the insights gained from the Maintenance Rule and other risk-informed programs at the plant. The licensee should solicit its experts in the areas of:

- plant engineering, operations, maintenance, and maintenance rule coordination;
- plant work, planning, and control;
- piping design and stress analysis;
- inservice inspection;
- NDE;
- structural design and support engineering;
- welding and materials test engineering;
- industry failure, repair and maintenance history;
- safety analysis; and
- probabilistic safety assessments.

The licensee should build upon the industry's documentation format developed for the RI-ISI pilot demonstration plants. These documents help lead the ISI teams to consider the major issues for each step of the program.

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[7] Copies of Commission policy statements, EPRI and WCAP reports referenced herein are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

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Appendix 3: ESTIMATION OF FAILURE PROBABILITIES USING EXPERT JUDGMENT ELICITATION (Ref. 1)

A3.1 Introduction

In pilot applications of risk-informed ISI methods (Ref. 2) and (Ref. 3), expert judgment was selected as a method for estimating failure probabilities of piping system components. This appendix describes the elements of the formalized process for conducting an expert judgment elicitation. For plant-specific applications there are time and cost limitations that will usually preclude application of this process in its entirety. Nevertheless, much of the guidance provided in this appendix can be applied to making the many judgmental decisions involved in estimating failure probabilities, whether by application of data bases or by application of probabilistic structural mechanics computer codes. In other cases it may be appropriate to systematically apply the expert judgment elicitation process to address generic issues related to structural reliability. Industry is encouraged to make such generic applications to estimate baseline failure probabilities for particular systems, materials, and operational conditions and incorporate that knowledge in the structural mechanics computer codes to increase the consistency and uniformity of plant-specific failure probability estimates. However, in practice it will be necessary and appropriate to modify any such generic estimates to address plant-specific conditions.

A3.2 Background

As in any scientific endeavor, expert engineering and scientific judgment (often referred to as expert opinion) is an essential aspect of any method (including application of historic data and structural mechanics computer codes) selected for estimating failure probabilities. In identifying the systems and components to be studied, expert judgment can be used to

- precisely define what is meant by a failure
- formulate a mathematical failure mode
- identify and assess relevant data
- combine all of these elements to obtain the desired results in a useful format

For such tasks, expert judgment is usually applied, but in an informal and unstructured manner.

For many such problems, this approach yields satisfactory results in an efficient manner. However, an informal and unstructured approach may be unsatisfactory when relevant data are sparse or nonexistent, or when the issue studied is complex or likely to receive extensive review and criticism. A formal expert judgment process has a predetermined structure for the collection, processing, and documentation of expert knowledge. The advantages and drawbacks of using such a process, as opposed to an informal process, are outlined in Bonano et al., 1990 (Ref. 4). The advantages include:

- improved accuracy and reliability of the expert judgments

- a reduced potential for critical mistakes leading to suspect or biased judgments
- enhanced consistency and comparability of procedures
- improved scrutability and documentation for communication and external review

The drawbacks include:

- an increase in the resources and time required to carry out the process
- a reduction in the flexibility to make changes in the ongoing process
- an increased vulnerability to criticism due to the relative transparency provided by a formal documentation of the procedures and findings, including differences expressed by the various experts.

Reference 4 cautions that, while a formal process often requires more resources and time than an informal process initially requires, a faulty process that fails to withstand criticism or must be redone because of inappropriate design or improper execution may end up failing to satisfy the project objectives and cost more in both time and resources. The potential for further costs in an informal study should be considered when evaluating the need for an formal process.

The formal use of expert judgment has been extensively applied to a number of recent major studies in the nuclear probabilistic risk assessment area [(Ref. 5), (Ref. 6), and (Ref. 7)]. Although scientific inquiry and decisionmaking have always relied on expert judgment, the formal use of expert judgment as a well-documented systematic process is a relatively new development. However, because of the many potential pitfalls in using expert judgment, it is essential that analysts be familiar with the state of the art and utilize the services of experienced practitioners in order to avoid wasting time and resources. Useful discussions of potential pitfalls and approaches to overcoming them may be found in (Ref. 8), (Ref. 9), and (Ref. 10).

The expert judgment process used in NUREG-1150 (Reference 5) is presented in (Ref. 11) and outlined in (Ref. 12). This methodology was developed in response to criticisms of the previous Reactor Safety Study (Ref. 13) and an earlier draft of NUREG-1150. The history of this development underscores the importance of basing the expert judgment process on state-of-the-art techniques and of making use of experienced practitioners in this difficult area.

A3.3 Expert Judgment Elicitation Process

A flowchart of the expert judgment process is given by Figure A3.1, taken (with minor changes) from Reference 12. The expert judgment process has 10 steps, outlined below. This process, with some modifications, was used to estimate break probabilities for selected components at Surry-1, as discussed in References 2 and 3. Specific techniques for the elicitation, use, and communication of expert judgment may be found in References [(Ref. 3), (Ref. 7), (Ref. 8), (Ref. 13), and (Ref. 14)].

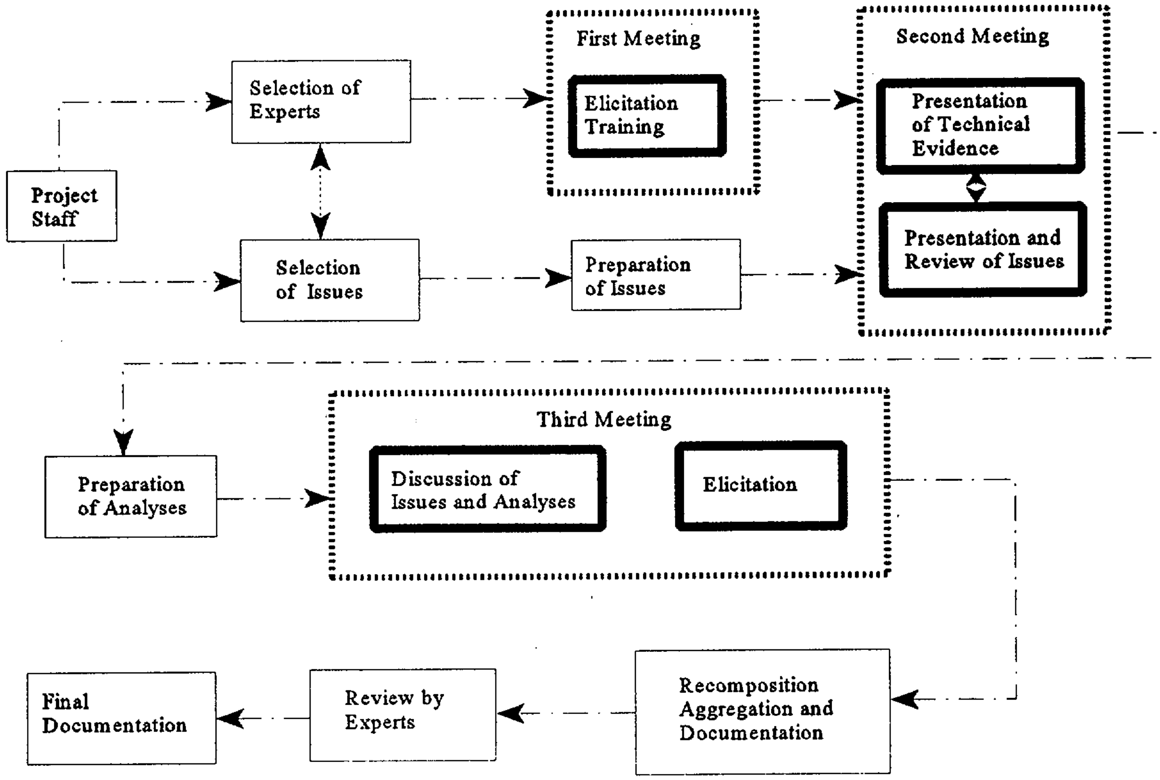


Figure A3.1 Expert judgment process.

A3.3.1 Selection of Issues

The initial selection of issues should be made by the project staff and is used to guide the selection of the experts. Two primary criteria for the issue selection are as follows:

(1) *The issue has significant impact on the risk and/or uncertainty.*

(2) *Alternative sources of information such as experimental and observational data, or validated computer models are not available.*

A3.3.2 Selection of Experts

Experts are selected on the basis of their recognized expertise in the areas of interest and chosen to ensure a balance of viewpoints. To address the issues of concern to the nuclear power industry, experts from reactor vendors, utilities, the federal government, national laboratories, consultant, and academia should be included. The goal is to obtain multiple and diverse input so that the issues can be thoroughly examined from many viewpoints.

There are two ways to organize the experts - by panels or by teams. The panel approach was used in NUREG-1150 (one panel for each of six groups of related issues) and the Lawrence Livermore seismic hazard study (one seismicity panel and one ground motion panel) described in Reference 6. The team approach was use by the Electric Power

Research Institute seismic hazard study (six balanced teams, each containing seismicity and ground motion experts) described in Reference 7.

In addition to the experts to be elicited, substantive and normative experts are needed to facilitate discussions, make presentations, and train the experts. The substantive expert(s) must be knowledgeable about decision theory and the practice of probability elicitation.

A3.3.3 Elicitation Training

The purpose of elicitation training is to help the experts learn how to encode their knowledge and beliefs into probabilistic or other quantitative forms. Elicitation training can significantly improve the quality of the experts' assessments by avoiding psychological pitfalls that can lead to biased and/or overconfident assessments. Training should include information about the methods used to process and propagate subjective beliefs, introduction to the assessment tools and practice with these tools, calibration training using almanac questions, and an introduction to the psychological aspects of probability elicitation. The training should be conducted by a normative expert with assistance by a substantive expert.

For NUREG-1150, the elicitation training took place at the first meeting and required a half day. Depending on their familiarity with elicitation techniques, some experts may require less or more than a half day of training. It is recommended that training occur at the beginning of the process so that the experts can familiarize themselves with the types of assessment they will be making before they decide on the specific issues to be addressed. However, when the training session takes place, it is important that it not be abbreviated due to time pressure.

A3.3.4 Presentation and Review of Issues

The initial list of issues selected by the project staff should be sent to the experts before the first meeting for review. Relevant data sources, models, and reports should also be included. The experts would be invited to propose additions, deletions, or modifications to the list. When the experts meet, substantive experts present the issues to the expert panel. The purposes of the presentation and review are:

- to ensure that a common understanding of the issues is addressed
- to ensure that the experts respond to the same elicitation questions
- to permit unimportant issues to be excluded and important issues to be included
- to allow modification or decomposition of the issues
- to provide a forum for the discussion of alternative data sources, models, and forms of analysis

An essential aspect of issue presentation is issue decomposition, which allows the experts to make a series of simpler assessments rather than one overall assessment of a complex issue. This step should be executed with great care, as the decomposition of an issue can vary by expert, thereby significantly affecting its assessment. Care should also be taken to present the issues so as to minimize potential biases in their assessment.

A3.3.5 Preparation of Analyses

The experts should be given sufficient time and resources to analyze the issues before the elicitation session. This step may entail support by the project staff, e.g., by performing computer calculations or other requested analyses. Some experts may choose to alter the proposed decompositions or create new ones. While many calculations necessarily bear on the probability assessments that the experts make in the elicitation sessions, the experts should be cautioned to avoid making any subjective probability assessments until the elicitation. This is necessary to avoid making any subjective probability assessments until the elicitation and to avoid the psychological bias of anchoring (Reference 8).

A3.3.6 Discussion of Issues and Analyses

Prior to the elicitation session, the experts should present the results of their analyses and research. The goal of this step is to ensure a common understanding of the issues and the database. It is not to reach agreement on the issue decompositions and the elicitation variables. To take advantage of the diversity of approaches, it is essential that each expert analyze each issue according to his/her own interpretation, and use the decomposition and elicitation variables with which he/she is most comfortable.

A3.3.7 Elicitation

The elicitation sessions should be held immediately follow the discussion of issue analyses. An elicitation team should meet separately with each expert. This avoids the pressure to conform and elides the other group interactive dynamics that may arise if the expert judgments are elicited in a group setting. The elicitation team should consist of a substantive expert, a normative expert, and a recorder. It is also useful to add as a fourth member the person who will prepare the final documentation.

The elicitation sessions serve two purposes. The first is to obtain the decompositions and quantitative assessments for each issue from each of the experts. Insofar as possible, the uncertainty of each quantitative assessment should also be elicited. The second purpose is to obtain the rationales for the decompositions and assessments. The experts should be questioned about their stated beliefs and asked to reflect on and explain the reasoning behind the decompositions and quantitative assessments they have provided.

Much of the documentation of the experts assumptions and reasoning can be completed during the elicitations. However, some follow-up work is usually necessary to fill voids in the logic provided by the experts or to obtain missing assessments.

A3.3.8 Recomposition and Aggregation

Each expert's assessments must be recomposed by the normative and substantive experts to organize them into a common form for each issue. Recomposition is necessary because the assessments for the elicitation variables in the decomposition for each issue must be combined into an assessment for the issue as a whole. Since each expert may have employed a unique decomposition, the end result for each expert must be in a common form suitable for aggregation. This will typically be a subjective probability distribution for a parameter of interest.

After the recomposition of each expert's elicitation, the results should be aggregated to yield a final assessment for each issue. It is essential that the aggregation reflect the uncertainties as expressed by the experts. There are two general classes of aggregation methods: methods that tend to consensus and methods that tend to preserve the variability among the experts. Genest and Zidek (Ref. 15) provide informative reviews on the many proposed aggregation methods.

When variability among the experts is greater than the uncertainty for each expert, a simple aggregation method is sometimes used. Each expert's assessment is replaced by a central value (the realistic estimate) and the central values are plotted. Converting the plot of central values to a box-and-whisker plot (Ref. 16) is a convenient way to summarize the assessments that reflects the uncertainties. This method was used in Reference 2 to estimate component break probabilities.

While consensus methods are often easy to implement (e.g., averaging over the experts), they should not be automatically applied without careful consideration. Because one of the primary goals of the expert judgment process is to reflect the state-of-the-art uncertainty as expressed by the diversity of expert judgments, an aggregation method should not be used if it tends to mask the diversity of expert judgment. For example, consider a case where half the experts judge the probability P of a phenomenon to be close to zero while the other half judge P to be close to one. Averaging over the experts is equivalent to the case where all experts judge P to be approximately ½. These two cases, however, are quite different since there is no disagreement among the experts in the second case, while there is a great deal of disagreement among the experts in the first case. In the second case, a decision maker would have high confidence that P is approximately ½, while in the second case, he/she does not know what value to assign to P. If he/she would make one decision when $P = 0$ and another decision when $P = 1$, premature averaging in the first case might deprive the decision maker of essential information. In general, an aggregation method should be used only if a sensitivity study indicates that it does not destroy information that might significantly affect the options of a decision maker.

A3.3.9 Review by Experts

Following the initial recomposition, aggregation, and documentation, written analyses of each issue should be distributed to each panel expert, substantive expert, and normative expert for review. A substantive (and nonvoting) expert might be an individual from the plant technical staff with detailed knowledge of plant design and/or operations, whereas a normative expert might be an individual with knowledge of probability and statistics who could assist the other experts in translating their engineering knowledge into numerical estimates of failure probabilities. The purpose of this review is to provide the experts with the opportunity to revise their earlier assessments, and ensure that potential misunderstandings are identified and resolved before final documentation. The revised assessments are then recomposed and reaggregated. To prevent an expert from arbitrarily changing his/her assessment so as to influence the aggregated assessment in a preferred direction, the experts should be required to provide a rationale for any significant reassessment.

A3.3.10 Documentation

Documentation has a number of important purposes. First, clear comprehensive documentation is essential to ensure that the expert judgment process is accepted as credible. Second, documentation can be used by the experts involved to provide assurance that their judgments are correctly reflected. Third, it can be used by potential users of the process to enhance their understanding. Fourth, it can be used by peer reviewers of the process to provide an informed basis for their review. Finally, documentation can be extremely useful to update the analyses when future research provides additional information.

A3.4 Example Application to Nuclear Piping Systems

Since elicitation of expert opinion was recognized as an acceptable means to quantify input to PRAs and risk-based studies, this method was selected for estimating pressure boundary failure probabilities for use in a pilot application of risk-based ISI methods performed by Pacific Northwest National Laboratory (PNNL). The systematic procedure, as described in References 11 and 5, guided the elicitation process. The following paragraphs summarize the procedures as indicated by Figure A3.2 and describe sample results obtained. Detailed discussions of the procedures as well as the complete results can be found in References 2 and 3.

PNNL conducted two expert judgment elicitation meetings. The meetings addressed only structural failures that were perceived as important to plant risk, or that could significantly affect core damage frequencies. The specific objective was to develop numerical estimates for the probabilities of catastrophic or disruptive failures for the selected pressure boundary systems and components at a PWR plant.

Experts at these meetings included specialists in the areas of materials science, structural mechanics, inservice inspection, data bases on service experience, plant operational practices, and plant specific knowledge of the plant. The first meeting on May 8-10, 1990, at Rockville, Maryland, addressed failure probabilities for the reactor pressure vessel, reactor coolant system, low pressure injection system, auxiliary feedwater system and accumulators (Ref. 2).

The second meeting occurred on February 3-6, 1992, in Washington, DC. This meeting addressed the high pressure injection system, residual heat removal system, service water system, component cooling system, and power conversion system (Reference 3).

The panel of experts brought to bear a large base of experience with structural integrity issues at operating plants as well as an understanding of the response of structural materials to service environments. The experts consisted of knowledgeable representatives from utilities, vendors, federal government agencies, and consultants. Prior to the workshop, reference materials were sent to the experts, including data sources, reports, and recent PRA results. Panel members were asked to study these materials and formulate initial estimates of failure probabilities.

To resolve issues thoroughly from many viewpoints, the elicitation was designed as a face-to-face meeting. A formal presentation was provided for each system of interest. The presentations discussed technical descriptions, historical component failure mechanisms, elicitation statements, suggested approaches, questionnaire forms, and any supporting materials. The issues were presented in a manner to avoid preconditioning or biasing responses.

All experts were encouraged to get involved in subsequent discussions. Knowledge from experts regarding plant design and operation, failure history, and material degradation mechanisms was brought to the discussions. Since the process was designed to take advantage of the diversity of the knowledge, each expert provided an independent estimate. No effort was made to seek a consensus among the experts on estimated break probabilities. Each expert completed questionnaires addressing location-specific break probabilities for the systems of interest. This data covered realistic estimates of probabilities, uncertainty estimates, and the rationale for these estimates.

Following the elicitation meeting, information provided by the expert panel was recomposed and aggregated. The written analyses of each system, including the recomposition and additional plant specific data, were then returned to each expert for review. This review provided the experts with an opportunity to revise their earlier assessments, and ensured that potential misunderstandings were identified and resolved and that the documentation correctly reflected the experts' judgment. The revised analyses were then again recomposed and aggregated to provide a single composite judgment for each break probability.

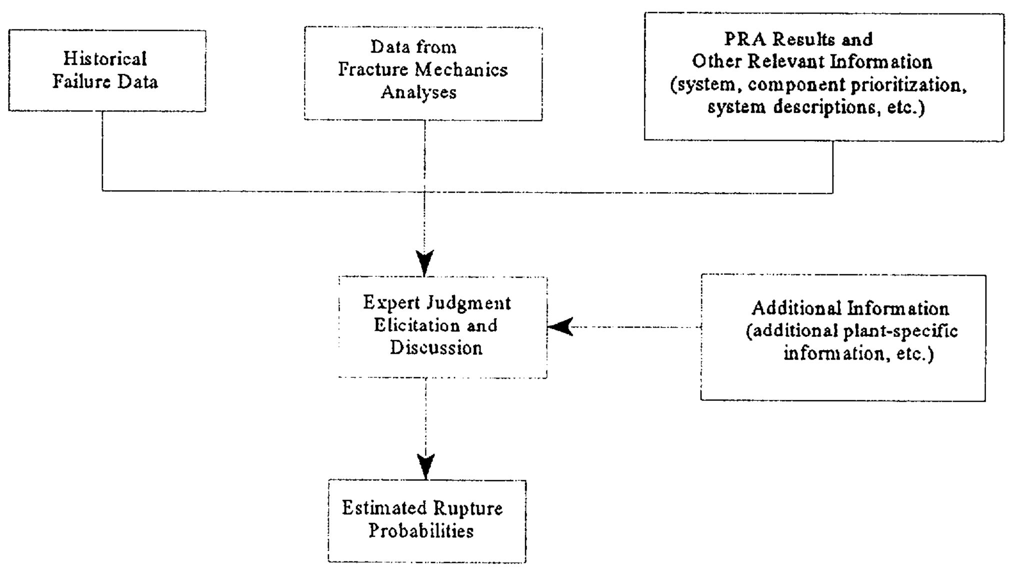


Figure A3.2 Process for estimating failure probability using expert judgment.

Figures A3.3 and A3.4 are samples of estimated failure probabilities obtained from the expert judgment approach. The probabilities are expressed as failures per year. Because, as in most expert judgment applications, the data set was not symmetric about a single peak, the median was used. Unlike the mean, the median is not influenced by extreme values. The interquartile range (75th percentile minus the 25th percentile) is used to describe variability in the data set.

As shown in the figures, the realistic estimates obtained from the population of experts are summarized in a series of box and whisker plots. These plots of the distribution associated with the expert population display the following features:

(1) the "whiskers" identify the extreme upper and lower bound values;
(2) the box is determined by the 25 and 75 percentiles (i.e., the lower and upper quartiles). Its length is the interquartile range (IQR).
(3) the middle 50% of the data points lie within the box;
(4) the circles indicate the median of the distributions.

The experts provide a wide range of responses regarding failure probabilities. This range is entirely consistent with the large uncertainties associated with the performance of the components being addressed. Since no attempt was made to seek a consensus from the expert panel, the median of the experts' estimates was suggested as a realistic probability for use in the risk-based studies. The evaluation should incorporate an uncertainty analysis, as illustrated in Figures A3.3 and A3.4.

For the systems selected for study, the extreme values of the failure estimates varied between 1.0E-09 and 1.0E-03 failures per year. For a given component within a particular system, the inter quartile range generally represented variations between a factor of 10 to 100. The component medians within a given system generally vary within a factor of 10, with the notable exception of the control rod drive mechanisms and the instrument lines of the reactor pressure vessel.

In summary, the data appeared to be reasonable and generally agree with the PWR plant operating experience. Typical areas of high-break probabilities correspond to such factors as high-cycle thermal stresses (e.g., places where mixing of fluids with large temperature differences occur) and places where erosion or corrosion effects are active. A tremendous amount of technical information was gathered from the exchange of information between the experts and the observers, and the elicitation greatly enhanced the realism and credibility of the plant analyses.

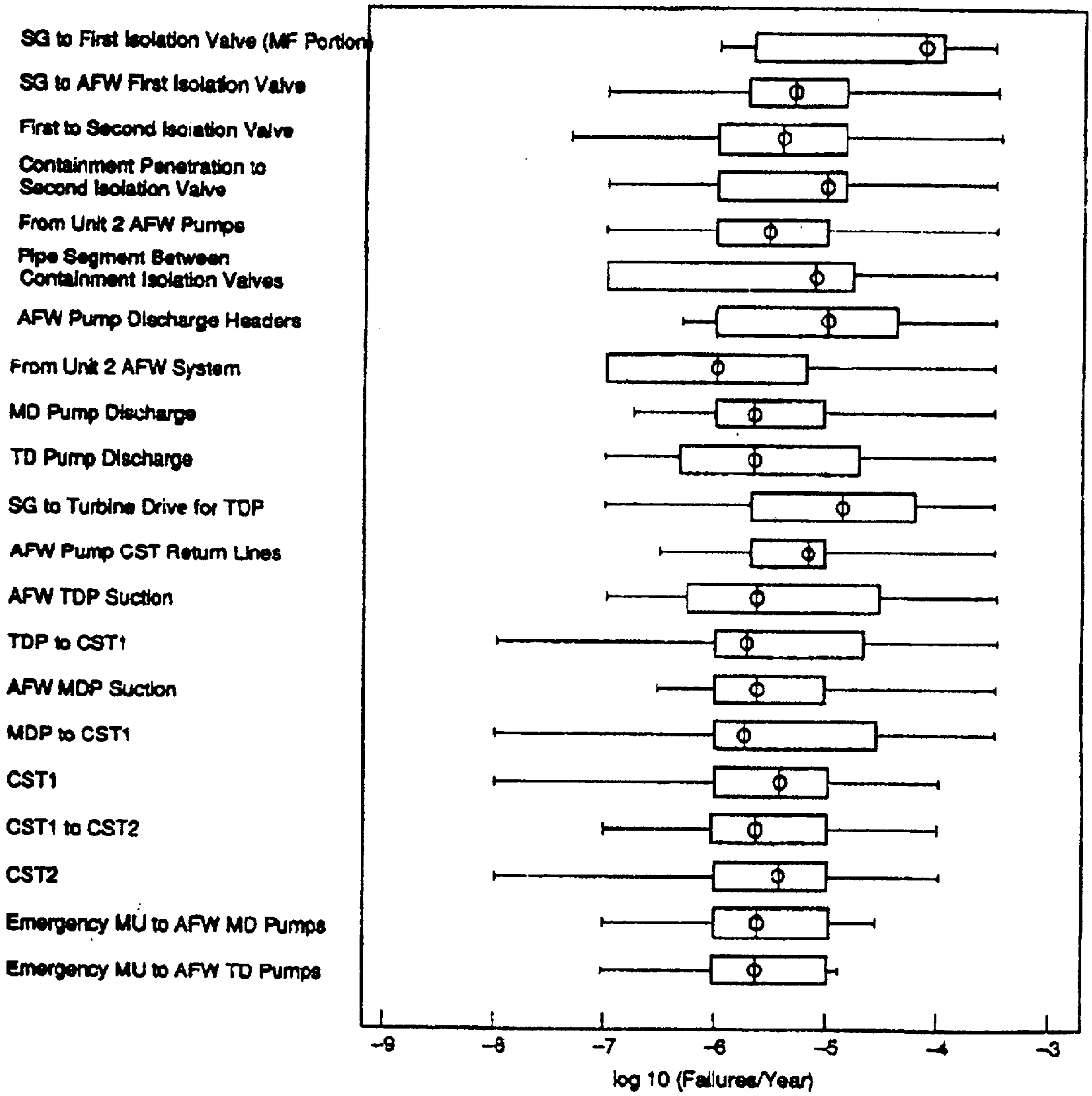
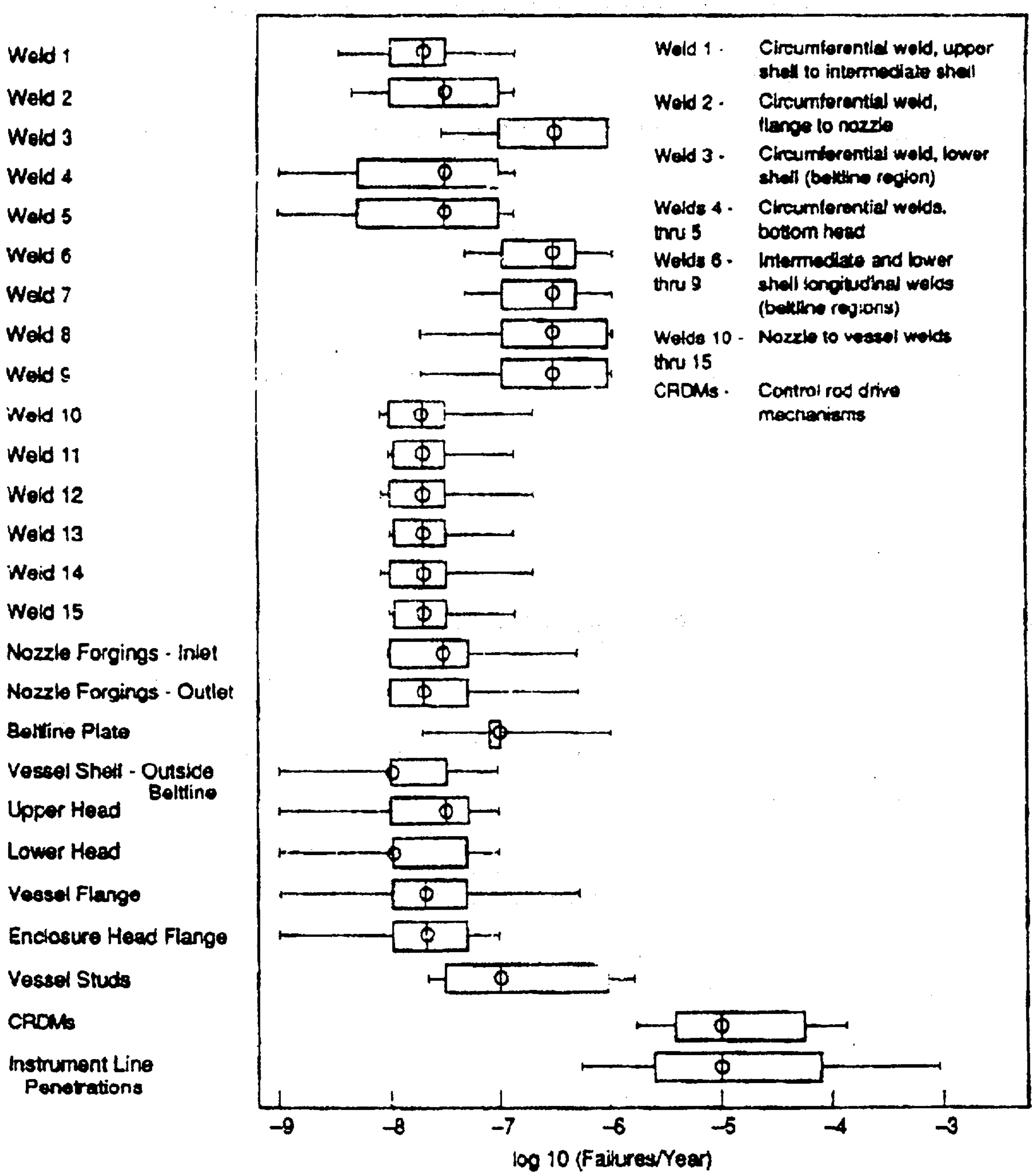


Figure A3.3 Failure Frequency Estimates for the Auxiliary Feedwater (AFW) System Components

Figure A3.4 Failure frequency estimates for the reactor pressure vessel



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Appendix 4: INSPECTION STRATEGY—RELIABILITY AND ASSURANCE PROGRAM

The purpose of this appendix is to illustrate one acceptable method for identifying the number of welds (as well as other structural locations) to be inspected in a risk-informed inservice inspection program. This appendix relies on statistical sampling techniques. As such, certain terms typically used by statisticians should not be confused with those used elsewhere in this regulatory guide. For example, the term *"consumer risk,"* or risk, as used in this Appendix, is not to be confused with the plant risk (CDF or LERF) used elsewhere. The plant risk used in the previous sections focused on: assessing the changes to public risk resulting from replacing existing ISI programs with the risk-informed ISI programs; and assessing high and low safety significant pipe segments. This appendix uses the term risk as used by statisticians when applying statistical sampling techniques. Here, risk refers to the probability of experiencing a detectable *leak* in a pipe (versus a break). Keeping this distinction in mind, the following provides one acceptable process for identifying the number of pipe elements to be inspected in a RI-ISI program. This process incorporates reliability, confidence, and the probability of detection (POD) of the inspection procedures to identify degradation prior to leak. This method is extracted from a paper by Perdue (Ref. 1) and augmented by Dr. Lee Abramson (from the NRC), through the ASME-Research program on RI-ISI. For reference, we will refer to this method as the Perdue-Abramson method. The Perdue-Abramson method focuses on two analyses. The first analysis focuses on flaws and the potential that a flaw exists and develops into a leak. The second analysis focuses on the global operating experience that directly compares observed leak frequencies with the target leak frequency. Combined, the process provides a check and balance.

The following sections will:

- Introduce the concept of statistical risk for quantifying the adequacy of an inspection plan.
- Illustrate a general method that can be applied to calculate risk for any reliability demonstration under the implicit assumption of perfect ability to detect a flaw given that the flaw is in the sample drawn.
- Incorporate how to address less-than-perfect ability of detecting a flaw given that the flaw is in the sample.
- Assess the implications for calculating the confidence/assurance that the sampling plan achieves the desired level of risk.

A4.1 The Concept of Statistical Risk

Consider a hypothetical pipe segment that consists of eight potentially inspectable elements (welds) that have not been previously inspected. Assume further that no risk-informed or other information is available so that, from the plant ISI team's perspective, the eight elements are clones of each another. Assuming that we stay within the current Section XI rules, one-quarter (25%) or 2 of the eight elements in this segment can be randomly selected for inspection in an upcoming outage.

If we inspect the 2 elements, what confidence can we place that the other elements within the segment are of similar condition? The question is similar to asking what "risk" is attached to this particular sampling plan? Risk is a concept from the field of statistical acceptance (or inspection) sampling that can be defined as follows. Assume that one specifies that a *minimum reliability level* for a lot is *X* defects. If a sample drawn from that lot is inspected and the whole *lot* is judged to be "acceptable" if the sample contains no defects, then risk is the probability that the lot will have more than the *X* permissible defects, whenever the sample contains no defects. Equivalently, risk is the probability that the inspection plan will let a *lot* (consisting of the elements of interest) be accepted with an unacceptable level of defects. Acceptance sampling or reliability demonstration is concerned with developing plans that "demonstrate" specified levels of risk or, equivalently, "confidence" (= 1 minus risk). To calculate risk, one needs to define:

- *Lot* size
- Sample size
- Flaw or defect
- Acceptance number (i.e., number of flaws found that will lead to rejection of the lot)
- A priori probability that a lot contains *X* defects
- Minimum allowable reliability level to be demonstrated.

The ASME Section XI can be said to provide definitions or guidance for all but the last item, the minimum reliability level to be demonstrated. In particular, the current code implies acceptance number of zero (more about this later). As for the minimum reliability to be demonstrated, it is useful to show the confidence associated with various postulated minimum reliability levels.

A4.2 Calculation of Risk

The measure of the minimum acceptable reliability level is the failure rate, where 'failure' is typically defined as a pipe break. Inspection, however, is concerned with finding "flaws" before they turn into leaks and breaks and, hence, there is a need to translate the failure rate measure into an equivalent number of (code-defined unacceptable) flaws. Information for a representative system may indicate, for example, that only four out of every 100 repairable flaws can be expected to propagate to a leak and only 1 in 1000 of the latter to a rupture over a 40 year interval. Such information, which is potentially obtainable from the combined exercise of structural reliability and risk assessment models, and probabilistic encoding of engineering judgment, can be used to translate a specified failure rate into an equivalent number of flaws or vice versa. For illustrative purposes only, returning to the simple hypothetical example of 8 elements in a pipe segment, the following assumptions are made:

- Probability of a flaw exceeding 10% of the pipe wall thickness in any one of the eight welds = 0.0065

- Conditional frequency that a flaw will grow to a leak /yr/weld is 3E-5

Given a probability of .0065 that any element will turn up flawed, the *binomial distribution* for N = 8 and p = .0065 can be used to calculate the probability that 0, 1, 2, et cetera flaws will exist in the lot of 8 *prior* to inspection. This is illustrated in column 3 of Table

A4.1 rom spreasheet model (Ref. 1, where, for example, the probability of precisely zero defects in the lot is 95%, 1 defect = 5% and so on.

Column 2 of Table A4.1 contains the failure rate for each number of flaws as calculated by fracture mechanics methods. Thus, given that one flaw has 3E-5/yr chance of becoming a leak, then 2 such flaws have about 6E-5/yr chance of producing a leak and so on.

Column 4 contains the cumulative counterpart of the binomial distribution in column 3. Thus, for example, the value of .999 in column 4 =0.949 + 0.0497 from column 3 and can be interpreted as "the probability of observing 1 or less flaws-or, equivalently, ***the probability of a leak frequency of 3E-05 or lower is 99.9%***." This cumulative distribution is dubbed the *Pre-ISI Probability Curve*. It indicates, for example, that there is a 99.9% chance of finding one or fewer flaws or, equivalently, that there is a 99.9% probability that the failure rate would be no more than 3E-5/year in the absence of inspection. This, of course, implies a 0.1% chance that the probability of a leak would be more than 3E-5/year. This 0.1% is the risk in the absence of inspection.

Interpreted within the context of Bayes' theorem, the distribution in column 3 of Table A4.1 can be called the "prior to inspection" distribution. Column 5 is called an "operating characteristic" or OC curve in acceptance sampling. For purposes of Bayesian reliability demonstration, however, it can be interpreted as a "likelihood" function because it shows the likelihood or probability of accepting the lot-given that said lot has the number of flaws indicated in Column 1. Like any OC curve, this one is calculated by using the *hypergeometric distribution,* which is tabulated in (Ref. 2) and is also built into a number of software packages (e.g., EXCEL). Keep in mind that the specified acceptance number for this example is zero; that is, the lot will pass only if zero flaws are found in the sample of 2 elements. Thus, referring to the second row in column 5, for a lot size N = 8, sample size n = 2, number of defects in lot k = 1, the hypergeometric distribution can be used to calculate that the probability of finding x = 0 flaws is 0.75. The analogous probability for k = 2 and x = 0 is 0.536 and so on. If the acceptance number had been, say, 1 flaw, then the calculations would use x=1 and proceed to find the probability of (8,2, k, 1) for different values of k. If a different sample size, say 3, had been used then the probability to look up would have been (8, 3, k, x).

Given the prior and the likelihood function, the next step in the application of Bayes Theorem is to simply multiply the two columns (i.e., column 3 times column 5) to get column 6. The latter column is not itself a proper probability distribution because it does not sum to unity. This is fixed by summing column 6 and then dividing each of its elements by the column sum to get the "post-Inspection" probability distribution in column 7. The cumulative counterpart of the latter distribution, called here the *"Post-ISI Assurance"* distribution is column 8.

To examine the effect of the target leak frequency goal, assume that the minimum allowable reliability is associated with a failure rate of 1E-6 per year for the lot (**not** per element but rather for all 8 elements that make up the lot). Assume further (for the moment) that if a flaw appears in the sample, the inspectors will see it (POD=1). Column 8 of Table A4.1 indicates that if the sample passes the inspection (i.e., if no defects are found), then ***we have 96.2% confident that the reliability is no worse than the maximum allowable failure rate of 1E-6/year***. Equivalently, the "risk" probability associated with this inspection plan is 1 - .962 = .038 or 3.8 percent. Once a specified level of "risk" is defined, different inspection strategies can be evaluated by the above method until one is found that meets the goal.

Table A4.1 Evaluation of Risk for N = 8, n = 2, and Zero Defect Acceptance Criterion

1	2	3	4	5	6	7	8
No. of Flaws (k) in N Elements	Conditional Leak Frequency Leak/yr/Lot Given a Flaw >0.1 Wall Thickness	Binomial Probability of k Flaws in the lot (Prob. of a Flaw > 0.1 Thickness of The Pipe Wall = 6.5E-3/weld)	Pre-ISI (i.e., No ISI) Probability of k or Fewer Flaws in The Lot	OC Curve Hypergeometric Distribution Probability That 0 Flaws are in the Sample of 2, Conditional on k Flaws in the Lot (0,k,2,8)	Col. 3 x Col. 5	Post-Inspection Probability That k Flaws are in the Lot and Given None are In the Sample (Col. 6 / Sum Col.6)	Post ISI Assurance — Probability of k or Fewer Flaws (Cumulative Sum of Col. 7)
0	0	0.949	0.949	1	0.949	0.962	0.962
1	0.00003	0.0497	0.999	0.750	0.0373	0.0377	0.999
2	0.00006	0.00114	1.000	0.536	0.00061	0.00062	1.000
3	0.00009	0.00001	1.00000	0.357	5.3E-06	5.4E-06	1.000
4	0.00012	0.00000	1.00000	0.214	2.6E-08	2.6E-08	1.000
5	0.00015	0.00000	1.00000	0.107	6.8E-11	6.9E-11	1.000
Col. Total		1.00000			0.987		

Key: N = Lot (population) size (8)
n = Sample size (2)
k = Number of defects in lot (1-8)
x = Number of defects in sample (0)

A4.3 Correction For Imperfect Detection

Table A4.1's OC curve in column 5 implicitly assumes that the nondestructive evaluation (NDE) techniques used to find flaws are perfectly accurate - i.e., if a flaw ends up in the sample, then it will be detected and properly sized. The OC curve can be corrected to reflect any hypothesized or real NDE level of accuracy (usually expressed as the probability of detection or POD).

Figure A4.1 illustrates the logic for an imperfect detection process where it is assumed that one flaw exists in a lot. The outcome of a sampling process could:

1. Detect the flaw if it is in the sample, and reject the lot,

2. Not detect the flaw even if it is in the sample (due to the inaccuracy of the detection process), and accept the lot, or

3. Accept the lot because the flaw was not in the sample selected for inspection.

It is assumed that there are no false detections, i.e., NDE never calls an item defective when it is not.

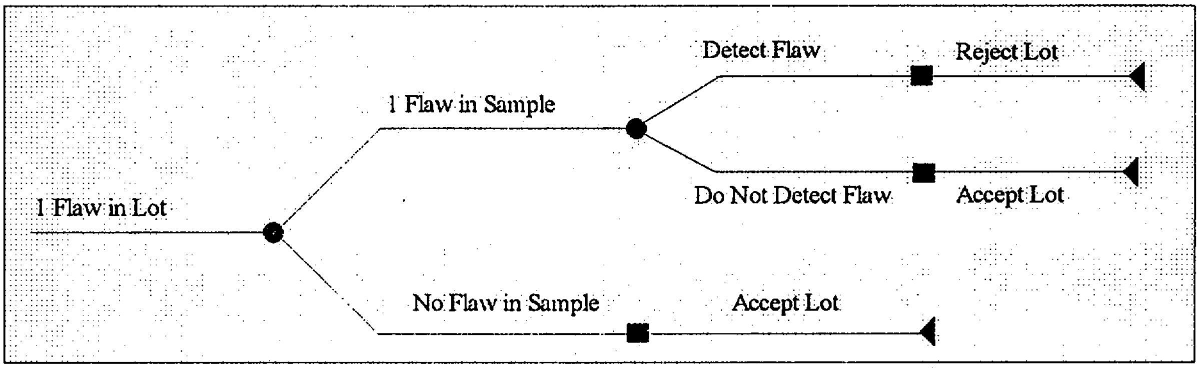

Figure A4.1 Single Sample Plan Logic

Only for the case where the flaw was within the sample *and* detected, would the lot be rejected. The lot would be accepted if the flaw was in the sample and not detected or if the flaw was not in the inspection sample. Thus, the probability of detection can have an important role in the analysis and needs to be addressed in the analysis.

The probability of accepting a lot, given that one flaw exists in the lot, is the sum of the probability of all the paths identified in Figure A4.1. Applying the hypergeometric distribution function to this process, the probability of accepting the lot is:

HYPGEOMDIST(0,2,1,8) + HYPGEOMDIST(1,2,1,8)*(1-0.65)

Where: HYPGEOMDIST(0,2,1,8) signifies the probability of getting zero flaws in a sample of 2, given that the lot has one flaw in a lot consisting of eight elements, and HYPGEOMDIST(1,2,1,8) is the probability of getting one flaw in a sample of 2, given that the lot has one flaw. The term (1-0.65) is the probability that the flaw will *not* be detected by the detection technique (1-probability of detection). Note that in practice, the probability of detection depends on both the mechanical detection technique as well as on the capability of the inspector performing the inspection.

The existing Section XI of the ASME Code calls for a double sampling plan. As an example, a double sampling plan can be summarized as follows: Take a sample of 1 and accept the lot if no flaw is found in that sample. Otherwise, take another sample of 1 and reject the lot if a flaw is found and accept the lot if no flaw is found. In general, if a flaw is detected in the sample, then take another sample of equal size. If a flaw is found in the second sample, then reject the entire lot. The logic for this is more complex, as illustrated in Figure A4.2.

Let us follow an example for the case leading to accepting the lot. The application of the hypergeometric distribution function takes on the following representation for 2 flaws in a lot with an initial sample of 1:

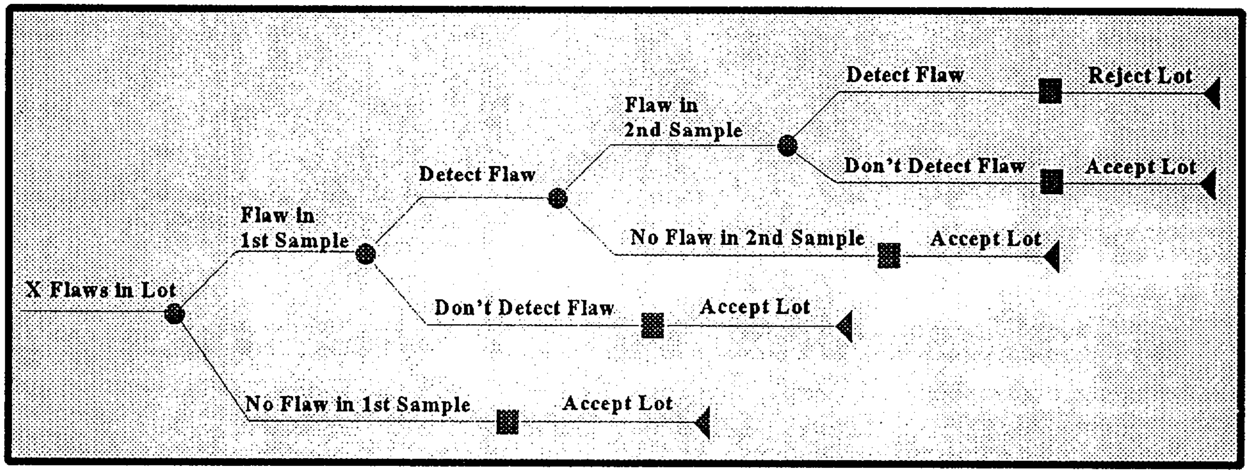

Figure A4.2 Double Sample Plan Logic

HYPGEOMDIST(0,1,2,8) + (HYPGEOMDIST(1,1,2,8))*(1-0.65) +
(1-HYPGEOMDIST(0,1,2,8))*0.65*HYPGEOMDIST(0,1,2-1,8-1) +
(1-HYPGEOMDIST(0,1,2,8))*0.65(1-0.65)*(1-HYPGEOMDIST(0,1,2-1,8-1))

The results of the above hypothetical example are listed in Table A4.2.

A4.4 System Assurance - Example Calculation

The methodology described above (applied to select a sampling plan for a single lot or segment) can be applied to provide a suitable level of confidence that a target leak frequency would not be exceeded. The NRC finds 95 percent confidence or assurance that the target leak frequency goal will be met as an acceptable objective for the system in question (e.g., summation of all the HSS segments in a system). However, achieving a 95 percent confidence for each segment of a system does not insure 95 percent confidence that the system itself will meet the target leak frequency. It is important to remember that selecting a single segment sampling plan on the basis of achieving a confidence of *at least* 95 percent will often necessitate choosing a sampling plan which yields considerably more than 95 percent confidence. This is demonstrated in Table A4-2, which adapts results from the Surry pilot plant, presented at a public meeting of February 12, 1997 (Ref. 3). Two segments, RC-41 and RC-42,43, could achieve the 95 percent confidence level only with 100 percent of the elements were inspected; thus producing (to 5 decimal places) 100 percent confidence in each segment that the leak frequency will be greater than or equal to the target rate. Suppose that none of the remaining nine segments are inspected because, in each case, their "Pre-ISI Confidence meets or exceeds 95 percent (this is "Plan A" in the second column of Table A4-2). The resulting RC SYSTEM confidence level actually

Table A4.2 Evaluation of Risk Using Bayes Theorem for Perfect (POD=1) and Imperfect (POD=0.65) Probability of Detection Cases

1	2	3	4	5	6	7	8	9	10	11	12	13	14
No. of Flaws (k)	Implied Leak/yr/Lot	Binomial Probability of k Flaws in a Lot of 8	Pre-ISI (i.e., No ISI) Probability of k or Fewer Flaws	OC Curve	Col. 3 x Col. 5	Post-Inspection Probability (Col. 6 / Sum Col.6)	Double Sample (each sample=1) Prob. of k or Fewer Flaws	OC Curve (double, 1st sample=2)	Col.3 x Col.9	(Post-ISI Inspection Probability) Col.10/ Sum Col.10	Double Sample (each sample=2, POD=0.65) Prob. of k or Fewer Flaws	Single Sample (POD=1) Prob. of k or Fewer Flaws	Single Sample (POD = 0.65) Prob. of k or Fewer Flaws
0	0	0.949	0.949	1	0.949	0.949	0.949	1	0.949	0.949	0.949	0.962	0.957
1	0.00003	0.0497	0.999	1	0.0497	0.0497	0.998	1	0.0497	0.0497	0.999	0.999	0.999
2	0.00006	0.00114	1.000	0.985	0.00112	0.00112	1.000	0.925	0.00105	0.00105	1.000	1.000	1.000
3	0.00009	0.00001	1.000	0.955	0.00001	0.00001	1.000	0.850	1.3E-05	1.3E-05	1.000	1.000	1.000
4	0.00012	0.00000	1.000	0.909	0.000	0.00000	1.000	0.718	8.7E-08	8.7E-08	1.000	1.000	1.000
5	0.00015	0.00000	1.000	0.849	0.000	0.00000	1.000	0.566	3.6E-10	3.6E-10	1.000	1.000	1.000
Col. Total		1.00000			1.000	1.00000			1.000				

demonstrated is the probability that no segment will exceed its target leak frequency, and this is equal to the *product* of the individual segment confidence probabilities in the third column:

$$\text{SYSTEM Confidence} = \prod (\text{Segment Confidence Probabilities}) \qquad \text{(Equation A4-1)}$$

which comes to 98.3 percent. Thus, even though no single segment is required to demonstrate more than 95 percent confidence, the resulting RC system confidence exceeds 95 percent (for the "safety significant" segments of interest). This will not always be the case, but the system result can always be checked by taking the product of the segment confidences associated with the sampling plans actually chosen.

Table A4-2: Surry RCS Segment Results

Segment (with # of elements in the Segment)	Plan A: Number of Elements Inspected	Plan A: Confidence (probability leak frequency below target)	Plan B: Number of Elements Inspected	Plan B: Confidence (probability leak frequency below target)
RC-44,45,51 (42)	0	1.000	1 per segment	1.000
RC-41 (3)	3	1.000	3	1.000
RC-42,43 (6)	3 per segment (= 6)	1.000	3 per segment	1.000
RC-18 (7)	0	1.000	1	1.000
RC-16,17 (14)	0	1.000	1 per segment	1.000
RC-37,38,39 (17)	0	0.999	1 per segment	1.000
RC-19 (7)	0	1.000	1	1.000
RC-27,28,29 (51)	0	0.984	1 per segment	1.000
RC-10,11,12 (6)	0	1.000	1 per segment	1.000
RC-13,14,15 (12)	0	1.000	1 per segment	1.000
RC-07,08,09 (30)	0	1.000	1 per segment	1.000
RC SYSTEM	9	0.983	31	1.000

Assuring an Acceptable System Confidence

Assume that the system product falls short of the 95 percent threshold (or assume that the product of all relevant systems falls short of the required *Plant-wide* confidence). "Plan B" in the Table could represent a second iteration in which the licensee would return to augment Plan A by selecting a minimum of one element to inspect from each segment. This produces an increase in RC system confidence to essentially 100 percent. Plan B is in fact the approach actually recommended for the Surry RC. Based on this analysis, the following process can be used to assure acceptable system confidence:

1. Select a sampling plan for each segment that achieves at least 95 percent confidence (no more than 5% risk of exceeding target leak frequency), *subject to the constraint that at least one element will be inspected in each high-safety significant segment..*

2. Calculate system confidence as the product of the segment confidences associated with the sampling plans initially chosen. If system confidence is below 95 percent, then rank-order the segments and proceed to augment inspection plans in the worst segments until the requisite system confidence (that no lot will exceed its target leak frequency) is achieved.

A4.5 The Global Analysis

The following presents the global analysis of the Perdue-Abramson method for calculating the number of inspections and for monitoring adherence to the leak frequency targets or goals. The global analysis assures that a specified target leak frequency is not exceeded for a given system of high-safety significant/low failure potential pipe segment. The target leak frequency is specified in terms of the frequency of leaks per year per weld. The analysis consists of the following steps, as shown in the flow chart in Figure A4.3.

1. Calculate the leak frequency for the given system without inspection.

2. If the calculated leak frequency does not exceed the target leak frequency, then no inspection is necessary, except for one weld to satisfy the defense in depth consideration.

3. If the calculated leak frequency exceeds the target leak frequency, then some inspection is necessary. Specify an inspection plan and recalculate the leak frequency.

4. If the recalculated leak frequency is less than the target leak frequency, then implement the inspection plan.

5. If the recalculated leak frequency exceeds the target leak frequency, then a more stringent inspection plan is necessary. Modify the inspection plan in step 3 and recalculate the leak frequency.

6. Iterate through steps 4 and 5 until the inspection plan results in a leak frequency which is less than the target leak frequency.

Assuring the Target/Goal by the Global Analysis

The target is to assure a maximum acceptable leak frequency per weld in a system consisting of N welds. For most cases of interest, this leak frequency is sufficiently small so that the chance of more than one leak in the system in a year is negligible. Therefore, it is assumed that at most one leak will occur. (The methodology can be extended if this assumption is not valid.) Accordingly, the leak frequency for the system of N welds is simply the probability that one of the welds will develop a leak. Denote the maximum acceptable leak frequency per weld by r_0. Then the maximum

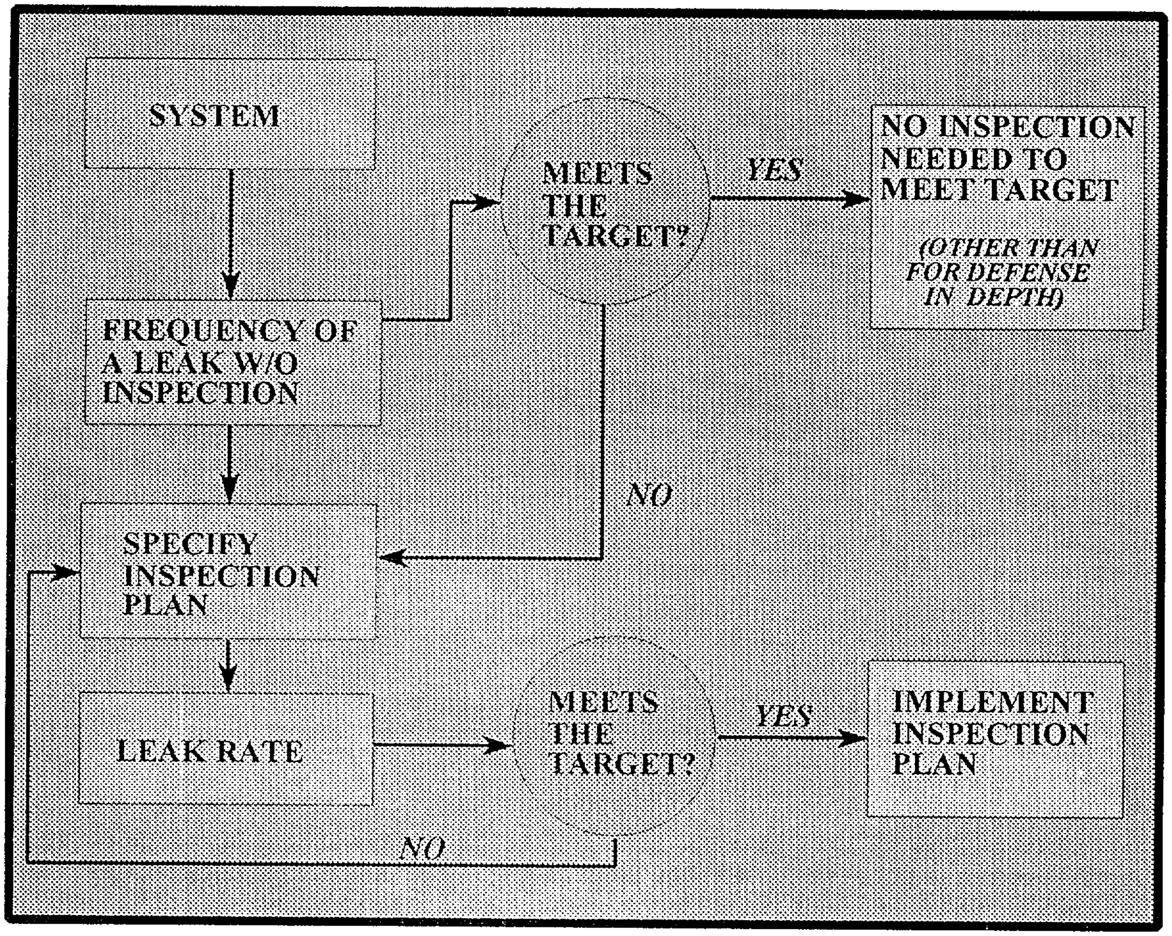

Figure A4.3

acceptable leak frequency for the system is Nr_0. Equivalently, the maximum acceptable probability of a leak in the system is Nr_0.

The purpose of inspection is to assure that the maximum acceptable leak frequency r_0 or Nr_0 is not exceeded. The inspections considered here attempt to identify flaws which, if not repaired, have the potential to develop into leaks. For any given system, its leak frequency depends on the number of flaws remaining after inspection and the probability that a flaw will develop a leak.

In the discussion below, it is assumed that all welds in the system have the same probability of (i) having a flaw, and (ii) having the flaw result in a leak. We will then show how the analysis can be generalized to the case where these probabilities are not constant.

First, consider the case where no inspection is performed. Let

p = Prob{ a weld has a flaw}

q = Prob{ a flaw will develop a leak}.

Then the probability that any given weld will develop a leak is pq. The leak frequency for the system is the probability that one of the welds in the system will leak and is equal to Npq. Comparing this with the target of Nr_0, we conclude that:

If $pq \le r_0$, no inspection is necessary to meet the target goal.

If $pq > r_0$, inspection is necessary to meet the target goal.
If inspection is performed, then the leak frequency will depend on the initial distribution of flaws and on the probability that one or more flaws will escape detection. Let k be the number of flaws in the system. Then k has a binomial distribution with parameters N and p. Conditional on k and on an inspection strategy S, let

$G_S(k)$ = Prob{ no flaws are detected | k , S }.

Denote the leak frequency for the system by R. Then

$$R = \text{Prob }\{\text{one leak in the system}\}$$

$$= \sum_{k=0}^{N} Prob(k)\ G_s(k)\ Prob(leak|k)$$

$$= \sum_{k=0}^{N} \binom{N}{k} p^k (1-p)^{N-k}\ G_s(k)\ kq$$

$$= q[Np(1-p)^{N-1} G_S(1) + N(N-1)p^2(1-p)^{N-2} G_S(2) + \ldots] \qquad (1)$$

As an example, consider a system with N = 8 welds and p = 0.0065. Let S = {inspect 2 out of 8 welds and accept the lot if no flaws are found in the sample of 2}. Then Prob{k} and $G_S(k)$ are given by the binomial and hypergeometric distributions, respectively.

k	Prob {k}	G_S {k}
0	0.949	1.
1	0.0497	0.750
2	0.00114	0.536
3	0.00001	0.357

Substitution into Equation 1 yields:

$$R = 0.0385\, q \tag{2}$$

This must be compared with the maximum acceptable leak frequency $Nr_0 = 8r_0$. Accordingly, the inspection scheme S meets the target provided

$$q \leq 207.8\, r_0 \tag{3}$$

For example, if $r_0 = 10^{-6}$, then any $q < 2.1 \times 10^{-4}$ meets the target.

Generalization of the Global Analysis

In many cases, the probability of a flaw and the probability that a flaw will result in a leak may differ from weld to weld. If there are N welds in a system, let

p_i = Prob{weld i has an unacceptable flaw}
q_i = Prob{weld i will result in a leak, given that weld i has an unacceptable flaw}

for i = 1., 2, ... , N.

If no inspection is performed, the leak frequency for the system is:

$$R_0 = \sum_{i=1}^{N} p_i q_i.$$

Comparing this with the target of Nr_0, we conclude that:

If $R_0 \leq Nr_0$, no inspection is necessary to meet the leak frequency target.

If $R_0 > Nr_0$, inspection is necessary to meet the leak frequency target.

If inspection is necessary, set $p^* = \max\{p_1, p_2, \ldots, p_N\}$ and $q^* = \{\max q_1, q_2, \ldots, q_N\}$. A conservative approach is to assume that all welds have the same probability, p^*, of having an unacceptable flaw and the same probability, q^*, that the flaw will result in a leak. Equation 1 can then be used to calculate an upper bound, R^*, on the leak frequency by replacing p by p^* and q by q^*. R^* can then be compared with Nr_0. If $R^* \leq Nr_0$, then the inspection strategy S meets the target. Otherwise, a more stringent inspection strategy is needed to meet the target.

A4.6 References for Appendix 4

1. R.K. Perdue, "A Spreadsheet Model for the Evaluation of Consumer Risk Associated with Inservice Inspection Plans," attached to USNRC, "Meeting Summary - ASME Research Meeting on Risk-Informed ISI: Statistical Sample Method of WELDS," January 8, 1997.

2. G.J. Lieberman and D.B. Owen, *Tables of the Hypergeometric Probability Distribution*, Stanford University Press, Stanford, CA, 1961.

3. A. McNeill et al., "Example Applications of WOG RI-ISI Process to Surry RCS," USNRC, Attachment to Meeting Summary, "Meeting Summary - ASME Research Meeting on Risk-Informed ISI: Pilot Plant Preliminary Results," February 25, 1997.

Appendix 5: RISK-INFORMED INSPECTION PROGRAM DEVELOPMENT

The methods discussed in Chapters 3 and 4 can be applied to support the development of improved inservice inspection plans (e.g., what to inspect, where to inspect, when to inspect, and by what method) by integrating risk insights into the program. In this regard, the development of a risk-informed inspection plan can be viewed as a three-step process:

- Step 1 - Selects the particular structural elements or locations that will be inspected; this selection should be made to ensure that the selected piping locations are those with higher failure probabilities, with greater impacts on plant safety, and those locations not expected or anticipated 30 years ago during the original design of a plant but identified through operating experience.

- Step 2 - Define inspection strategies for the selected locations, such that the NDE methods and inspection frequencies provide desired levels for detection of degradation and reductions of failure probabilities.

- Step 3 - Augment steps 1 and 2 to accommodate defense-in-depth review for unexpected degradation mechanisms.

The risk categorization study and the element selection process, described in Chapter 4, focuses on the first step. These methods can be applied to evaluate various inspection strategies to identify combinations of inspection methods (e.g., POD, sizing accuracy) and frequencies at selected locations that can be effective in maintaining or reducing the failure probabilities of passive reactor components. To accomplish this target, the inspection strategies must address the failure mechanisms of concern, and have sufficiently high probabilities of detection and sizing accuracy so that the expected damage can be detected (given various frequencies of inspection) and the components repaired before structural integrity is impacted. When analyzing the piping networks for failure degradation mechanisms, it is useful for the analyst to have a checklist table of degradation mechanisms, identification of materials susceptible to those degradation mechanisms, potential locations that are susceptible to the degradation mechanisms, and the contributing causes. The checklist, such as illustrated in Table A5.1, provides added confidence that the analysis takes into consideration the various degradation mechanisms and potential locations. The analyst also needs to consider acceptable approaches for determining the number of locations to be inspected (size of inspection sample) and the desired reliability and frequency of the inspections to be performed at these locations. Since several potential inspection strategies may provide the desired maintenance or reductions in failure probabilities, the final selection can be based on other important considerations including man-rem exposures to inspection personnel and cost effectiveness.

Table A5.1: Check List of Degradation Mechanisms for Inspection of Piping Systems

Degradation Mechanism	Susceptible Materials	Susceptible Locations	Contributing Causes
Low Cycle Fatigue	All materials	Terminal ends Dissimilar metal welds Near snubbers Near component nozzles Fittings	Operating transients High thermal expansion stresses Stress concentrations Construction defects
Thermal Fatigue	All materials	Mixing of hot and cold fluids Hot or cold water injection Valves (downstream from leakage) Feedwater nozzles Counterbores Horizontal lines	Valve leakage Thermal stratification
Vibratory Fatigue	All materials	Small diameter piping (ϕ < 2-inch) Terminal ends Socket welds	Rotating equipment
Intergranular Stress Corrosion Cracking	Stainless steels	Welds Heat affected zones Sensitized base metal materials BWR piping PWR piping (CVCS systems)	Elevated temperatures High coolant conductivity High carbon grades of SS Elevated oxygen levels Residual stresses Cold springing stresses Stagnant fluids

Degradation Mechanism	Susceptible Materials	Susceptible Locations	Contributing Causes
Transgranular Stress Corrosion Cracking	Stainless steels Iron-nickel-chromium alloys	Bolting	High carbon or low carbon materials High oxygen High welding stresses Severe cold working Presence of chlorides or sulfates Brackish environment Insulation materials with chlorides
Crevice Corrosion Cracking	Stainless steels Iron-nickel-chromium alloys		
Primary Water Stress Corrosion Cracking	Iron-nickel-chromium alloys		
Intergranular Attack	Iron-nickel-chromium alloys		
Flow Accelerated Corrosion (erosion/corrosion)	Ferritic steels	Elbows Reducers Tee fittings	Wet steam Single phase (water) flow Low alloy content Low oxygen High Ph High flow velocities
Slurry Erosion	All materials	Raw water systems	Sand or solids in raw water
Cavitation Wastage	All materials	Pumps and valves Highly localized areas	Phase change Droplets

Degradation Mechanism	Susceptible Materials	Susceptible Locations	Contributing Causes
Microbiologically Influenced Corrosion (MIC)	All materials	Buried piping (external surfaces) Other piping (internal Surfaces) Welds Fittings Heat affected zones Crevices	Exposure to organic materials Exposure to raw water Lack of coatings Lack of cathodic protection
General Corrosion	Ferritic steels Austenitic steels (occasionally)	Secondary systems Service water systems Dissimilar materials (galvanic effects)	Galvanic/electrolytic corrosion Crevice corrosion Acid attack Raw water Salt water corrosion Brackish water corrosion
Boric Acid Corrosion	Ferritic materials	Primary systems	Leak of boric acid solutions
Pitting	Ferritic materials	PWR feedwater nozzles	Leakage at thermal sleeves/joints
Structural Damage	All Materials	Small diameter piping Compression fittings	Water Hammer Impact Crushing Over Pressure Maintenance errors

An inservice inspection strategy can be defined by the following elements:

Element 1: Sampling Strategy

The sampling strategy is defined by the selection of structural elements that are proposed for inclusion in the inspection program. The selection of structural elements should be guided by the calculations of risk categorization and should include additional elements to address defense-in-depth for lower risk components, and to address unanticipated generic failure mechanisms that have not been detected or that have not yet occurred. The strategy should include immediate expansion of the sample when flaws are detected during an ISI through sequential sampling based on feedback from ISI findings and operating experience.

The structural elements (or locations to be inspected) should be appropriately defined, *such that the defined volume of metal for the elements includes the critical locations where degradation is most likely to occur.* Structural elements will be the basis for the "examination volumes" to be addressed by the detailed NDE procedures. In many cases, the structural elements should include base metal locations well removed from the weld heat affected zones to ensure that the NDE covers locations of stress concentrations, such as weld counter bores.

Element 2: Inspection Method

Inspection methods are selected to address the degradation mechanisms, pipe sizes and materials of concern. The inspection method includes the basic technique itself (e.g., ultrasonics) along with the particular equipment and the procedures to be applied for detecting and sizing flaws. Candidate inspection techniques for piping include ultrasonic testing, surface examinations with dye penetrants (or magnetic particles), visual examinations, and radiography. In a larger context, monitoring methods such as leak detection, thermal transient monitoring, and acoustic emission monitoring can be used to supplement or replace nondestructive testing methods. Detailed aspects of equipment, procedures, and personnel qualifications are significant factors that govern the reliability of the inspections. The risk-informed inspection concept requires that the reliability of the inspection method be established in order to justify the selection of a particular inspection strategy. Based on materials, environments, loads, and degradation mechanisms, probabilistic fracture mechanics calculations can establish the probability of detection, the sizing accuracy, and the frequency of inspection needed to meet targets for passive reactor component failure probabilities (see Chapter 4).

Element 3: NDE Reliability and Performance Demonstration

Qualification of the NDE system (personnel, procedure and equipment) is an important element of an inspection program. Inspection systems with known reliability are needed to achieve the desired levels in failure probabilities consistent with the goals of the risk-informed inspection process. A risk-informed inspection program should justify the inspection reliability using data from performance demonstration programs.

Element 4: **Time of Inspection**

The inservice inspection strategy must define when the inspections are to be performed. In most cases inspections are performed periodically at regular intervals such as with the 10 year interval of the existing ASME Section XI. A risk-informed inspection program will identify the appropriate inspection intervals, such that the inspection program provides the desired maintenance or reductions in component failure probabilities. Inspection intervals must be sufficiently short so that degradation too small to be detected during one inspection does not grow to an unacceptable size before the next inspection is performed.

This chapter discusses one approach for determining the appropriate examination methods, frequency, and level of qualification for the structural elements selected for examination in Regions 1 and 2 of Figure A2.9. As mentioned previously, SRRA tools have been and can be exercised to evaluate the effectiveness of a given examination method, frequency, and level of performance.

*Whereas Chapter 4, Section 4.3 of this regulatory guide focused on the selection of pipe segments and the number of structural elements to be inspected, **this chapter addresses the selection of inspection strategies**.* Guidance is provided to ensure that inspections are performed in a manner that ensures that the failure probabilities of passive piping components remain acceptably low. To accomplish this objective, the inspection strategies must address the failure mechanisms of concern and have a sufficiently high probability of detecting the expected damage before structural integrity is impacted.

Section A5.2 discusses acceptable approaches for determining the reliability of the inspections to be performed at these locations, and the frequencies of the inspections. Since several potential inspection strategies could provide a desired reduction in failure probabilities, the final selection by licensees can be based on other important considerations such as cost effectiveness and man-rem exposures to inspection personnel. As mentioned previously SRRA tools have been and can be exercised to evaluate the effectiveness of candidate inspection strategies.

A5.1 Elements of Inspection Strategies

An inservice inspection strategy may be comprised by use of the inspection strategy table in Figure A5.2. This is accomplished by selecting one option within each category identified in Figure A5.1 (Ref. 1). The following address some of the major categories identified in Figure A5.1.

Inspection Method - Inspection methods are selected to address the degradation mechanisms, pipe sizes, and materials of concern. The inspection technique includes the basic technique itself (e.g., ultrasonics) along with the particular equipment and the procedures to be applied for detecting and sizing of flaws. Appropriate inspection techniques for piping include ultrasonic testing, surface examinations with dye penetrants (or magnetic particles), visual examinations, and radiography. In a larger context, monitoring methods such as leak detection, thermal transient monitoring, and acoustic emission monitoring can be used to supplement or replace nondestructive testing methods.

Detailed aspects of equipment, procedures, and personnel qualifications are significant factors that govern the reliability of the inspections. The risk-informed inspection concept requires that the reliability of the inspection method be established in order to justify the selection of a particular inspection strategy.

Time of Inspection - The inservice inspection strategy must define when the inspections are to be performed. In most cases, inspections are performed periodically at regular intervals such as with the 10-year interval of ASME Section XI. The risk-informed inspection program will identify appropriate inspection intervals, such that the program provides the desired component failure probabilities (consistent with the PRA assumptions). Inspection intervals must be sufficiently short so that degradation too small to be detected during one inspection does not grow to an unacceptable size before the next inspection is performed.

Some techniques (e.g., acoustic emission monitoring) perform the inspections on a continuous rather than periodic basis. In other cases, the strategy may require inspections only after an unanticipated or a significant loading event has occurred, such as a severe thermal shock or a water hammer. Some inspections may be performed on a one-time-basis, as for example, to verify that a degradation mechanism experienced at a similar plant is not occurring at the plant of concern, or to otherwise support continued plant operation, such as part of a license renewal process.

NDE Qualification - Qualification of NDE (method, procedure, and personnel) is an important element of an inspection program, particularly for those components having high failure probabilities or safety significance. For such components, highly reliable inspections may be needed to achieve the desired failure probability goal.

A risk-informed inspection program should have a technical basis for the inspection reliability inputs that are used in structural reliability calculations of estimated failure probabilities for proposed inspection strategies. Such a basis can be provided by NDE performance demonstration programs. Generic data from studies of NDE reliability can also be useful. Such generic data are available from NDE round robin exercises. The reliability of any inspection is dependent on the specific qualifications and skill level of the inspection personnel. In addition, the reliability can be enhanced by the use of inspection teams having qualifications that meet industry codes and standards, and by the use of methods and procedures with accepted capabilities.

Figure A5.1 Inspection strategy table.

Inspection Methods	Time of Inspection	NDE Qualification	Locations	Development of NDE Methods	Sampling Strategy	Delivery Method
VT-1 Cracking/ Surfaces	10 Years	Personnel Visual Qualification	Welds	Customization	100%	TV Cameras
VT-3 Gross Damage/ Surfaces	Refueling 12 - 24 months	Performance Demonstration	Bolts	New Technique From Scratch	Various Options of 1, 2, & 3 Below	Divers
Remote/ Enhanced Visual	Continuous	Historical Data	Flexures	None	No Sequential Strategy	Submarines
Ultrasonic Testing	License Renewal		Surfaces/ Interfaces		Whatever is Accessible	Remote Tools
No Inspection	After Significant Event		Support Systems		1) Initial size	In-Situ
Monitoring Neutron Noise Parts	Based on Performance Goal Objectives		Dowel Pins		2) Sequential Rule	Removed from Vessel
Remote Replication			Unspecified		3) Choice of Sampling Location/ Period	
Mechanical Measurements			Known Degradation			
Metallurgical Examination						

Development of NDE Methods - This element of the inspection strategy, as indicated in Figure A5.1, addresses the possible development of new and improved NDE methods to achieve levels of NDE reliability which are consistent with the goals of the risk-informed inspection program identified in Section 4.3 (e.g., *frequency of a leak < 1E-06 per weld-year*). In most cases, special development effort will not be needed, since existing NDE methods can be utilized or adapted. As indicated by Figure A5.1, such activities as the industry-funded Performance Demonstration Initiative (PDI) can be considered an appropriate development effort, since it serves to enhance NDE reliability.

Sampling Strategy - In the context of this regulatory guide the sampling strategy is defined by the selection of an appropriate number of structural elements as described in Section A2.7 and Appendix 4. Expansion of the sample size (i.e., through sequential sampling) is addressed in the implementation of risk-informed inspection through feedback of ISI findings and other information on structural degradation gained from operating experience. Such information should impact the estimates of component failure probabilities, and will result in appropriate changes to the inservice inspection programs.

Delivery Method - The effectiveness of an ISI strategy can be enhanced by the use of improved methods that provide better access to the selected locations. Improved access and the use of remote systems can also provide benefits in terms of reduced radiation exposures to the employees.

A5.2 Failure Probability Considerations

An inservice inspection program should ensure appropriate failure probabilities for the inspected structural elements, thereby minimizing their contributions to the risk as measured by core damage frequency or by other risk measures. The licensee should justify the basis for the selected sample size, of locations to be inspected and justify the effectiveness of the inspections at these selected locations.

Inservice inspection programs for piping, in accordance with ASME Section XI and/or other requirements, are performed to maintain confidence in the structural reliability of pipes. In terms of risk-informed inspection one objective of inservice inspections is to maintain the failure probabilities to an acceptable low value.

This section describes how considerations of quantitative goals can guide the development of risk-informed inservice inspection programs. *For example, it is proposed below that a factor of ten reduction in calculated failure probability (over the probability of no inspection) can be used as a guideline to identify effective inspection strategies.* Such a goal also helps to eliminate ineffective inspection strategies for which the sampling plans, NDE methods, and inspection frequencies are inadequate to deal with the components and degradation mechanisms of concern. In other cases candidate strategies may be marginal in achieving the goal, in which case modifications to the NDE methods or to the inspection frequencies can be identified.

Service experience provides specific examples to demonstrate that inspections can reduce failure probabilities. There are cases of large and growing cracks, and of areas of wall thinning whereby inspection programs have provided timely detection of the damage such that repairs were performed before the defect sizes became critical. On the other hand there are other examples whereby ineffective inspection programs have failed to detect large defects which have eventually resulted in pipe leaks or pipe breaks. Such ineffective inspections, performed at considerable expense and often exposing personnel to radiation exposures, have not contributed to piping reliability.

While service experience identifies many examples of direct benefits from inspections which have provided examples of the timely detection and repair of piping, inservice inspections also provide other important indirect benefits which are more difficult to quantify. For example, the detection of ongoing degradation at a specific location not only impacts the failure probability for the inspected location, but also provides valuable information to the plant technical staff (and the industry in general) regarding materials performance issues and the structural integrity of similar piping locations.

Therefore, the finding of degradation during a particular inspection can have a significant impact toward reducing failure probabilities for a population of similar pipe locations. In such cases, the findings of a single inspection can be a key factor that leads to important corrective actions (e.g., additional inspections in accordance with requirements for expanded or sequential sampling, improved operational practices to reduce stress levels, replacement of pipes using improved materials and designs, etc.).

At a minimum, an adequate sample size includes sufficient representative locations within each piping systems to permit the detection of degradation mechanisms that may be operating within the system. *These locations should, in part, correspond to locations for which the probability of degradation is consider greatest, independent of the calculated risk importance parameters.*

For the selected locations, the ISI strategy should be based on an appropriately specified level of effectiveness for detecting structural degradation. An effective inspection strategy is a one that detects degradation before it grows through the depth of the wall. Licensees should identify the level of inspection effectiveness adopted as a criterion for the development of its proposed inspection programs.

As an example, the following is an acceptable rationale for adopting a criterion of a factor-of-ten reduction in calculated failure probabilities for the goal of the candidate inspection strategies.

SRRA calculations indicate that if pipe failure probabilities are estimated assuming no impact from ISI (e.g., no inspection) and then calculated assuming ISI has an impact (i.e., accounting for the probability of detection of defects and the subsequent repair or replacement of the affected pipe), reductions in the failure probabilities (i.e., ratio of failure probability without ISI over failure probability with ISI) are about a factor of 10 (Ref. 2), (Ref. 3), (Ref. 4), (Ref. 5), (Ref. 6), and Reference 8. Calculated reductions of failure probabilities, greater than a factor of 10, can often be difficult to justify, due to the limitations and uncertainties in NDE flaw detection probabilities, and the need for relatively frequent inspections for cases where cracks can grow relatively quick between inspections.

Inservice inspection locations for piping in ASME Section XI, are defined for individual structural elements. However, it is recommended that the desired reductions in failure probabilities be established in terms of total contributions from groups of the structural elements being addressed. This approach minimizes the impacts of uncertainties in the estimated probabilities for individual structural elements. A graded approach for reducing component failure probabilities is considered appropriate, such that the most aggressive inspection strategies focus on the top contributors from the risk categorization, with reductions short of the factor of 10 being acceptable for the less critical structural elements.

A5.3 Integration of Probabilistic Structural Mechanics Calculations

The selection of an inspection strategy for a structural element requires that the effectiveness of the candidate strategies in detecting structural degradation and reducing the failure probability of the structural elements be estimated. The effectiveness is governed by several factors including the NDE reliability (e.g., probability of detection), inspection frequency, and crack growth rates. In this regard, limited historical data on piping failures provides little information on the impacts of inspections on these probabilities, and it is therefore necessary to apply structural mechanics models to quantify the expected benefits of proposed strategies. Furthermore, the inspection strategies of interest are usually ones that will be newly implemented, and therefore an extended period of future operating experience would be needed before the failure rate data could indicate the effectiveness of a proposed strategy. *Even then, data on structural failures will be very limited, because actual failures (with or without inspections) are expected to occur only very infrequently.*

Efforts to calculate inspection related reductions in failure probabilities should compliment and build on the knowledge gained in recent years from ongoing work within the nuclear power industry by specialists in the area of NDE technology. This work has quantified the ability of NDE methods to detect and size defects in piping, and has resulted in new and improved requirements for performance-based demonstrations of the NDE methods, procedures, and personnel which are used to qualify the pipe inspections performed at nuclear power plants. Applications of probabilistic structural mechanics calculations, as described below, are an extension on the current industry studies of NDE reliability. The calculations integrate considerations of NDE reliability (i.e., as measured by probabilities of flaw detection and sizing errors) with considerations of degradation mechanisms and inspection intervals. The calculations model the degradation mechanisms of concern to pipe reliability, and use probabilistic fracture mechanics methods to simulate the effects of periodic inservice inspections. Results of these calculations provide a basis for screening candidate inspection strategies and identify the strategies that are the most effective in detecting growing flaws before such flaws become through wall cracks and/or cause pipe breaks or large leaks.

Structural reliability models can be used to address the various factors that govern the ability of ISI to detect degradation and reduce failure probabilities. For some situations, knowledge of only the probability of flaw detection for the proposed inspection method may be sufficient to estimate the effectiveness of a proposed strategy. However, this is seldom the case because the following additional factors govern the effectiveness of ISI:

- Detection probabilities are a function of flaw size. If small flaws are important to structural integrity, many NDE methods will lack the needed sensitivity. Therefore the expected sizes of fabrication and service induced flaws must be addressed by the structural reliability models.

- Flaws can grow in size over time when active degradation mechanisms are present. A structural reliability model must simulate the flaw growth rates, predict the sizes of growing flaws, and simulate the detection probabilities for the flaw sizes that are likely to exist when the periodic inspections are performed.

- Small detected flaws need not be repaired if they are less than the acceptable sizes as defined by the ASME codes. Some of these unrepaired flaws will contribute to pipe failures.

- In some cases there can be errors in measurements of flaw sizes, such that oversized flaws which should have been repaired are allowed to remain in service.

Structural reliability models should simulate the above factors to evaluate the benefits of inservice inspections. The models should simulate initial distributions of fabrication flaws in terms of their numbers and sizes, and also consider the possibility that degradation mechanisms can initiate new flaws during the service life of components at locations that were originally free of defects. For example, the initiation of new flaws should be addressed for those cases that calculations indicate that failures can occur for even the smallest sizes of the fabrication flaws. The structural reliability model should simulate the population of flaws of various sizes over the service life of the component, and predict the flaw sizes that could be present at the times when inservice inspections are performed. If particular flaws are detected and repaired, the model should then assume that these detected flaws no longer contribute to the failure probability.

Probabilistic models of inservice inspection should address the following:

- The primary consideration is a representation of a probability of detection curve that corresponds to the specific NDE method/procedure/personnel, degradation mechanism, material, pipe size, and component geometry of concern. Section A5.6 provides guidance on estimating the parameters for the curves for probability of detection (POD) as a function of flaw size.

- Consistent with the realistic approach used by the structural mechanics codes to simulate other parameters, the POD curve used to simulate ISI should be based on realistic curves without consideration of confidence levels in POD values. Separate uncertainty analyses can deal with concerns regarding confidence levels.

- The combined effects of a sequence of periodic or repeated inspections should be appropriately simulated. The detection (or nondetection) of a given flaw by successive inspections, or by inspections using different NDE methods are not usually independent events. Those random factors (excluding flaw size) which prevent detection for one inspection will also tend to preclude detection for the next inspection. For conservative calculations, the combined effects of repeated inspections can be bounded by taking credit only for the inspection having the greatest likelihood of detecting the flaw (e.g., the periodic inspection corresponding to the maximum size of a growing flaw, or the NDE method with the maximum POD capability).

- The simulations can address the effects of pre-service inspections on failure probabilities by treating this inspection as an inservice inspection performed at time equals zero within the service life of the component. However, the simulation of preservice inspections should be consistent with the assumptions made in estimating the distributions of initial fabrication flaws in the component, because pre-service inspection is a consideration in estimating distributions of initial flaws. Double counting of pre-service inspection effects can result if the simulated pre-service inspection was already addressed in estimating the initial flaw distribution. Pre-service inspections should be included in the calculations only if the inspections are in addition to those used as part of the fabrication process, and then only if the NDE method provides an enhanced level of NDE reliability.

- The simulations of inservice inspections should address the fact that detected flaws (more specifically small flaws) are not repaired if these flaws are smaller than ASME code flaw acceptance criteria.

The structural reliability calculations can be performed using the same computer code as used to estimate failure probabilities for the PRA calculations and risk importance measures. In many cases the benefits of proposed inspection strategies can be estimated by reference to prior generic calculations (e.g., from the literature) for the failure mechanisms, component designs, operating conditions and inspection strategies of concern (References 2, 3, 4, 5, and 6).

One potential benefit from risk-informed inservice inspection programs is the possible reduction of radiation exposure to personnel from reduction in the number of locations of inspections of radioactive pipes. Applying the NRC's ALARA and defense-in-depth principles, the NDE method used in locations where the number of inspections was significantly reduced should be optimized in terms of its probability of detection capabilities. Part of the steps to identify optimum detection methods include:

- Select a structural mechanics model that addresses the component, failure mechanisms, and inspection strategies of concern;
- Define the reliability of the candidate inspection methods;
- Calculate the failure probability of a component assuming no inservice inspections are performed;
- Calculate the failure probability of a component for each of the candidate inspection strategies;
- Calculate effectiveness of candidate inservice strategies as the ratio of failure probabilities, with the baseline being either no inspection or the current inspection strategy.

The calculations described above should make use of leak probabilities where the leak probabilities are used as a measure of inspection effectiveness, and as a surrogate for estimating the effects of ISI on reducing the probabilities of pipe breaks. The application of leak probabilities have significantly less uncertainties and is consistent with the regulatory philosophy of preventing breaks. It also avoids a large number of assumptions and uncertainties associated with calculations of pipe break probabilities. The numerical difficulties of calculating very small values of probabilities for pipe breaks can also impose excessive computational demands, which are largely avoided if the focus is directed to calculating leak probabilities.

One acceptable approach is to quantify the benefits of inspection strategies in terms of a relative failure probability, which can be expressed by various terms such as "factor of improvement" and "inspection efficiency" as follows:

$$\text{Factor of Improvement} = P_o/P$$

$$\text{Inspection Efficiency} = 1 - P/P_o$$

P_o = Failure Probability with baseline inspection strategy (e.g., no inspection)

P = Failure Probability with Inspection Strategy of Interest

These calculations of relative failure probabilities, that compare alterative inspection strategies, have been found to be relatively insensitive to such factors as uncertainties in the operating stress levels that govern the absolute values of failure probabilities.

If the baseline strategy is no inspection, the values of inspection efficiency can range from between 0.0 and 1.0 with a value of 0.0 corresponding to no ISI or a totally ineffective ISI strategy (i.e., the same as no ISI). A value of 1.0 corresponds to perfect inspection. Inspection efficiency is roughly correlated to the POD of flaws, and becomes the same as POD for the limiting case for which:

- the POD that is independent of flaw size, and
- all flaws are repaired without regard to their measured size.

The values for a factor of improvement can range from between 1.0 and infinity with a value of 1.0 corresponding to a totally ineffective ISI strategy (same as no ISI), and a value of infinity corresponding to a perfect inspection.

A5.4 Example Probabilistic Structural Mechanics Calculations

The selection of inspection program requirements for key locations in piping systems can be supported by SRRA evaluations. The literature provides many examples of such calculations, including the work of Khaleel and Simonen in Reference 5. This particular study was performed with the pc-PRAISE computer code as a series of sensitivity calculations for piping systems impacted by fatigue crack growth degradation. In this section we will present the results of both the individual SRRA calculations, and trend curves derived from the overall series of calculations. We will also describe how a selected inspection strategy (method and frequency) can be supported by the example trend curves.

Table A5.2 provides input parameters for the baseline case (no inservice inspection) of a 6-inch diameter pipe subject to fatigue cycling, which results in a calculated leak probability of only 6.0E-08 (cumulative probability per weld at 40 years). This modest level of fatigue cycling corresponded to a "**Q-Factor**" of 1.0, where the Q-Factor is a measure of the magnitude and number of stress cycles for the piping location being addressed. The series of failure probability calculations of Reference 5 covered a wide range of Q-Factors corresponding to more severe conditions of stress cycling giving results as follows:

Loading Condition	Q-Factor	Leak Probability
Low	$1.0 - 10^1$	$\approx 1.0\times10^{-7}$
Medium	$10^2 - 10^3$	$\approx 1.0\times10^{-4}$
High	$10^4 - 10^5$	$\approx 1.0\times10^{-1}$

Table A5.2 **PRAISE model of LPI system: baseline case**

Flaw Depth Distribution	Exponential (Mean Depth = 0.06 inch)
Flaw Aspect Ratio	Lognormal (Parameter = 0.689)
Stress Through Wall Thickness	Uniform Tension
Cyclic Stress Amplitude	15 Ksi / 5 cycles per year
da/dN Curves	As given in pc-PRAISE Documentation
Threshold ΔK for da/dN	0.00
Flow Stress	Normal (Mean = 43 ksi, C.O.V.* = 0.0977)
Pipe Inner Radius	2.75 inches
Pipe Wall Thickness	0.562 inch
Pressure	2.250 ksi
Dead Weight Stress	3 ksi
Thermal Expansion Stress	10 ksi
Inspection	No PSI and No ISI

* C.O.V. = Coefficient of variation = standard deviation / mean

The low Q-Factor should relate to all piping segments in Region 2 of Figure A2.9. The medium and high Q-Factors should relate to susceptible locations in pipe segments Region 1. The remaining locations in those Region 1 segments should have low Q-Factors, as for segments in Region 2.

Required inspection frequencies can be established using the trend curves such of Figure A5.2 which were developed from a set of probabilistic structural mechanics calculations as described in Reference 5. For example, let us assume that a licensee wants to reduce the probability of a leak by a factor of 10. The curves of Figure A5.2 are for an ultrasonic inspection method designated "very good," with a probability of detection curve (POD) having a 50% probability of detecting a crack with depth 10% of the wall thickness and a probability of 90% in detecting flaws greater than 50% of the wall thickness. The objective is to determine the time interval between inspections that will detect 90% of the growing cracks which could become through wall depth before the end of the 40-year design life.

The curves of Figure A5.2 indicate that an inspection frequency of 10 years with the first inspection at 5 years (5/10) can achieve the factor of ten reduction in failure probability. This reduction applies to a wide range of cyclic stress conditions (Q-factor from 1.0E+0 to about 1.0E+3 corresponding to 40 year leak probabilities of 1.0E-7 to 1.0E-1). The inspection efficiency decreases for higher values of failure probabilities, because the rates of crack growth are so high that the 10-year interval between inspections is inadequate. For very low values of the Q-Factor, the failure probabilities are also very low, because those failures that do occur are very early in life and are due to large fabrication defects which are not detected with the normal post weld inspections. These defects are best addressed by a high quality preservice inspection.

The results of Figure A5.2 show that improved NDE methods (that is, methods having the ability to detect smaller defects) can justify the use of longer time intervals between periodic inspections. Application of such improved NDE methods, even with longer inspection intervals, can decrease failure probabilities compared to less sensitive NDE method. The reduced number of inspections can also reduce radiation exposures to the workers performing the inspections.

The relationships and/or tradeoffs between detection capabilities and inspection frequencies can be explained in terms of the sequence of events that lead to structural failures. This sequence consists of the initiation of small cracks, an extended time period of slow crack growth, and a final period of rapid crack growth.

An effective inspection program detects small cracks before the crack growth rates increase to unacceptably high levels. The maximum allowable time interval between inspections is dictated by consideration of the crack growth rates. This time interval is governed by the difference between the smallest crack size that can be detected and the larger critical crack size that can result in a structural failure, with the optimum inspection interval corresponding to the time period needed to grow from the undetectable size to the critical size.

For many cycle stress levels the small cracks at the detection threshold will not grow to critical size over the plant operating life. In such cases one high quality inspection early in life is the most effective strategy. In cases of high cyclic stresses, the growth rates for these small cracks will be much greater. Therefore, depending on the crack growth rates, several inspections before during the plant life are required to ensure that cracks do not grow to critical size.

In summary, these results can give an indication of what type of program may be necessary to achieve an Improvement Factor that maintains the failure probability of a given pipe segment below an acceptable level. For those elements that have estimated leak probabilities above acceptable threshold values (e.g., 1×10^{-5} per weld lifetime for small leaks and/or 1×10^{-8} per weld lifetime for disabling leaks), inspection programs can be defined that will yield the necessary Improvement Factors.

In terms of defining an appropriate examination method(s) for various geometries and postulated failure modes, Table 4.1-1 in (Ref. 7) provides a comprehensive place to start in selecting appropriate examination methods.

A5.5 Additional Considerations for Selecting Strategies

Additional factors should be addressed by licensees during the selection of inspection strategies beyond those related to effectiveness of the inspection methods to achieve goals for failure probabilities. Considerations related to safety and structural reliability are as follows:

- Exposure of inspection personnel to hazardous environments, including man-rem exposure from radiation (reactor coolant system piping and fittings), hazardous materials, dangerous heights or climbing of scaffolds and unsteady platforms, rotating equipment or machinery, and falling objects. Man-rem exposure has the potential to not only impacts personnel health and safety, but also impacts on the overall costs of performing the inspections. ALARA considerations should be followed to develop strategies that reduce man-rem levels. In some cases, it may be

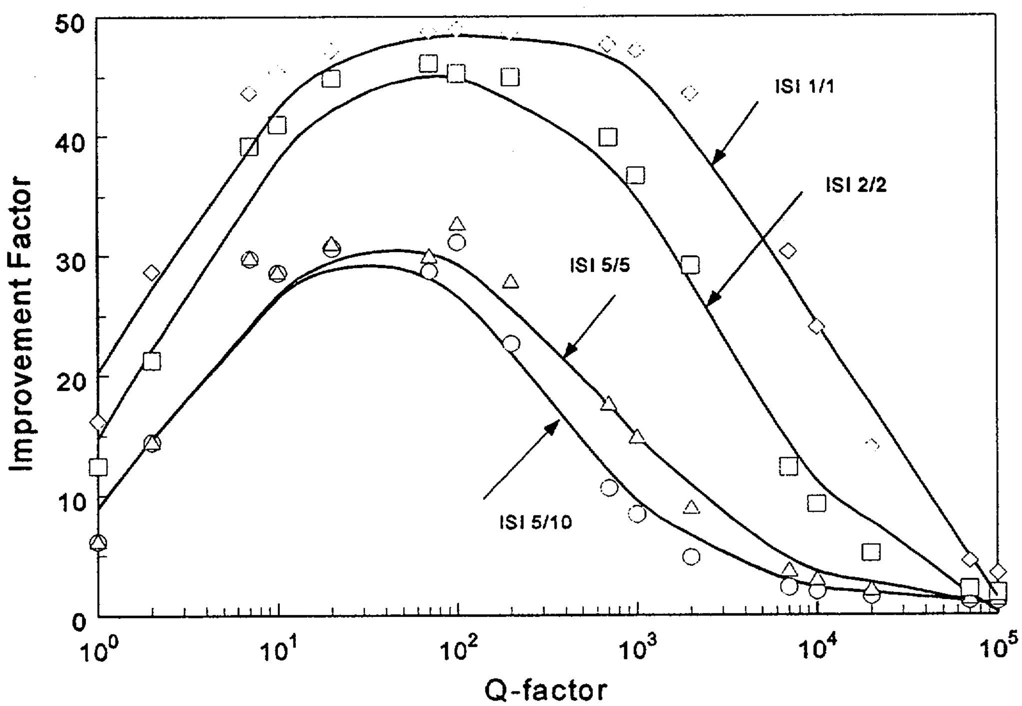

Figure A5.2 **Improvement factors for four inspection interval (NDE performance level for POD = "Very Good").**

justified to reduce the number of inspections that have marginal impacts on risk but with large contributions to manrem exposure.

- Damage to components can occur as a result of the inspection itself. In some cases the inspection requires that equipment be taken apart to gain adequate access to permit inspections of the structural locations of concern. The degree of success in reassembling systems and components that have to be taken apart or taken off-line to do the inspection (e.g., reactor vessel closure studs, steam generator manway covers, piping supports and attachments, pumps, valves, turbine generator-casings) should be a consideration.

- Movement of large equipment or structures (e.g., reactor vessel internals, reactor closure heads, large pipe supports, and restraints) can damage adjacent equipment and structures.

Concerns with disassembly or movement of components will not be a factor for most piping inspections. However, when such situations do occur, inspections should be coordinated with other maintenance needs that require the needed disassembly or movement operations. In other cases, it may be prudent to minimize such inspections unless the ISI locations are from the highest categories of the risk categorization scheme.

A5.6 Quantification of NDE Reliability

Evaluations of inservice inspection strategies require quantitative inputs to describe the reliability of NDE methods to be used. A primary input is a POD curve for the piping locations that are to be inspected. Other considerations include flaw sizing accuracies and the flaw acceptance criteria governing which sizes of flaws must be repaired versus flaws that are permitted to remain in service.

Factors Governing NDE Reliability - The POD curves and flaw sizing accuracies are related to the particular NDE method/procedure/personnel, degradation mechanisms, materials, pipe sizes, and component geometries being addressed. This section describes acceptable approaches for estimating POD curves and other parameters of the inspection process. Additional information on these topics is available in the literature, and has been summarized in Section 11 of the Probabilistic Structural Mechanics Handbook (Ref. 7).

In estimating the reliability of a candidate inspection strategy the following factors should be addressed:

- **NDE Method** - Visual examination, liquid penetrant testing, magnetic particle testing, radiographic testing, eddy current testing, ultrasonic testing, acoustic emission monitoring

- **Flaw Dimensions** - Depth, length, opening/crack tightness

- **Flaw Orientation** - Normal or parallel to surface

- **Material Type** - Stainless steel, ferritic steel, cast or wrought, fine grained or coarse grained

- **Access to Inspection Location** - Inside or outside surface, near or far side access to welds, presence of physical obstructions, need for disassembly

- **Surface Conditions** - Surface roughness, contamination/deposits, weld deposits, cladding

- **Extraneous Signals** - Large grained materials, geometric reflectors, weld roots, counter-bore geometries

- **Human Factors** - Inspector experience and training, motivational factors, low tolerance for false calls, time restraints, hostile environments (heat, humidity, poor lighting, confined spaces, protective clothing)

- **Qualification/Performance Demonstration** - Equipment, procedures and personnel, ASME Appendix VIII, detection and sizing capabilities

NDE Reliability Studies - There have been ongoing research efforts on a national and international level to develop data to better characterize the reliability of NDE methods for detecting representative service-type defects (cracks). Such efforts have included a number of round robin studies to determine the reliability of NDE as practiced in the nuclear power industry, and to take actions to improve NDE reliability.

Table A5.3 lists studies of NDE reliability which have provided information on the probabilities of detection for flaws in nuclear piping and other components (Ref. 7). These studies cover a range of components, inspection methods, and damage mechanisms. Early findings showed a relatively low level of NDE reliability, even though the inspection methods were often consistent with the minimum standards of existing codes as published by such organizations as the American Society of Nondestructive Testing (ASNT) and the ASME. Subsequent efforts have produced changes in codes and standards which were directed to improving the reliability of NDE as applied at nuclear power plants.

The usual approach in NDE reliability studies has been to use specimens with representative service-type defects (i.e., cracks) for training and for demonstrations of capability. The round robin data have shown large team-to- team variations in the detection and sizing of flaws. As shortcomings have been noted, the nuclear industry has responded with steps to strengthen minimum requirements such as in the ASME Code to improve the inspection of reactor pressure vessels and piping systems.

Performance Demonstrations - ASME Section XI has adopted Appendix VIII (Ref. 8) which follows a performance demonstration approach, through which inspection organizations must qualify the performance of equipment, procedures and personnel. In the new approach, inspection teams must achieve passing scores in tests of their capabilities to detect simulated service-type flaws in a matrix of samples that simulate conditions in reactor pressure vessels and piping. A passing score requires detection of a statistically significant fraction of the flaws in the sample set, while maintaining an acceptably low frequency of false calls. The performance demonstrations also require that a team attain passing scores on flaw-sizing capability.

Performance demonstrations provide a basis to identify those NDE methods that are most reliable, and those whose reliability is unacceptable. However, current performance demonstrations in the ASME Section XI Code require only a specified POD level for the collection of flaws in the sample set. The sample sets have a range of flaw sizes, beginning with the smallest size that is considered to be structurally significant. As now practiced, performance demonstrations are not designed to generate a full POD curve as a function of flaw depth as is needed for purposes of probabilistic structural mechanics calculations. To obtain a statistically based POD curve, additional detection data beyond the minimum demanded by current performance demonstration tests are required. Lacking such a complete set of data, POD curves for field inspections must be estimated based on engineering judgment and by making use of the currently available base of detection data as generated from inspection round robins and performance demonstration efforts.

Modeling of NDE Uncertainties - Consistent with the practice for simulating other parameters in probabilistic structural mechanics calculations, the POD curves used in simulations of ISI should be selected to represent mean values of POD without consideration of confidence levels. Arbitrary conservatism should not be applied in estimating the POD curves to be used in the probabilistic structural mechanics calculations, because such conservatism, if not applied uniformly, could improperly bias the selection of inspection strategies. While realistic mean values of PODs should be used as input to the structural

Table A5.3 Reliability studies of NDE for inspection of nuclear piping and other components.

Component	Inspection Method	Damage Mechanism	Responsible Organization	Reliability of Method	Reference
Beltline of Reactor Pressure Vessel	Ultrasonics using manual procedures and past ASME Section XI practices	Cracking embedded within thickness of plate	PISC-I (Plate Inspection Steering Committee). European Organization for Economic Cooperation and Development	Detection capability was marginal. Much less reliable than for the procedures used in the subsequent PISC-II trials	(Ref. 9)
Beltline and nozzles of PWR Reactor Pressure Vessels	Ultrasonics	Near surface cracking caused by disposition of weld cladding, volumetric weld defects, voids, and porosity	PISC-II (Program for Inspection of Steel Components) European Organization for Economic Cooperation and Development with participation of other organizations including U.S. Nuclear Regulatory Commission	Detection reliability relatively good; sizing capability relatively poor	(Ref. 10)
Beltline/Plates of Reactor Pressure Vessel	Ultrasonics, using methods proposed as sufficiently reliable for British regulatory requirements	Cracks near vessel inner surface, and within cladding	Risley Nuclear Laboratories, Risley, United Kingdom - Defect Detection Trials (DDT)	Effective detection reliability and good sizing capability were demonstrated	(Ref. 11)
PWR Primary Coolant Piping, Carbon Steel	Ultrasonics	Fatigue cracking	Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission - Piping Inspection Round Robin	Very good reliability (POD > 90%) demonstrated by all participating teams	(Ref. 12)
BWR Piping, Wrought Stainless Steel	Ultrasonics	Intergranular stress corrosion cracking	Electric Power Research Institute	Early results of an ongoing effort showed poor sizing capability	(Ref. 13)
BWR Piping, Wrought Stainless Steel	Ultrasonics	Intergranular stress corrosion cracking	Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission - Mini Round Robin	Only the best teams demonstrated adequate performance in detecting flaws; the majority of teams had unacceptable performance. All teams were unreliable in sizing flaws	(Ref. 14)
PWR Primary Coolant Piping, Centrifugally Cast Stainless Steel	Ultrasonics	Fatigue cracking	Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission	None of the participating teams demonstrated reliable detection for the coarse grained material	Reference 12
Steam Generator Tubing	Eddy Current (ET). Also ultrasonics and profilometry to limited extent	Pitting, wall thinning, denting and cracking	Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission	Relatively good reliability for detection and sizing of wall thinning and pitting. Relatively poor reliability for cracking	(Ref. 15)
Steam Generator Tubing	Eddy Current	Stress corrosion cracking, Intergranular attack, wastage, pitting	PISC-III (Program for Inspection of Steel Components) European Organization for Economic Cooperation and Development	Multi-year effort with round robin testing underway	
Steam Generator Tubing	Eddy current	Wall thinning, pitting, denting, cracking, etc.	Electric Power Research Institute	Future data will be based on a Round Robin interpretation of existing ET signals from actual steam generator inspections	
Various nuclear and non-nuclear components	Ultrasonics	Cracking, slag inclusions, machined notches and other types of defects	Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission	A wide range of reliability in detection and sizing is indicated by data in this comprehensive survey report	(Ref. 16)
Aircraft structures with emphasis on fastener joints	Ultrasonics	Cracks at fastener holes	Lockheed - Georgia Company with participation of Air Force maintenance facilities. A large scale inspection round robin popularly known as "have cracks will travel".	Best teams demonstrated effective inspections, but large team-to-team variations were noted	(Ref. 17)

reliability code, the uncertainties associated with the POD should be accounted for in any calculation.

Characteristics of POD Curves - Probability of detection is defined as the ratio of the number of flaws actually detected to the number of flaws that would be detected given a perfect NDE system. An example of a POD curve that has been used in probabilistic fracture mechanics calculations with the pc-PRAISE code (Ref. 18) is shown in Figure A5.3. This schematic form is typical of POD curves that have been described in a number of other studies including (Ref. 19). As indicated, flaws must have some minimum size or threshold before detection becomes possible. Above this threshold size, detection increases rapidly as the size of the flaw becomes larger. The POD curve eventually attains a maximum value at which non-detection is governed by other factors (e.g., human errors) that come to dominate the detection processes.

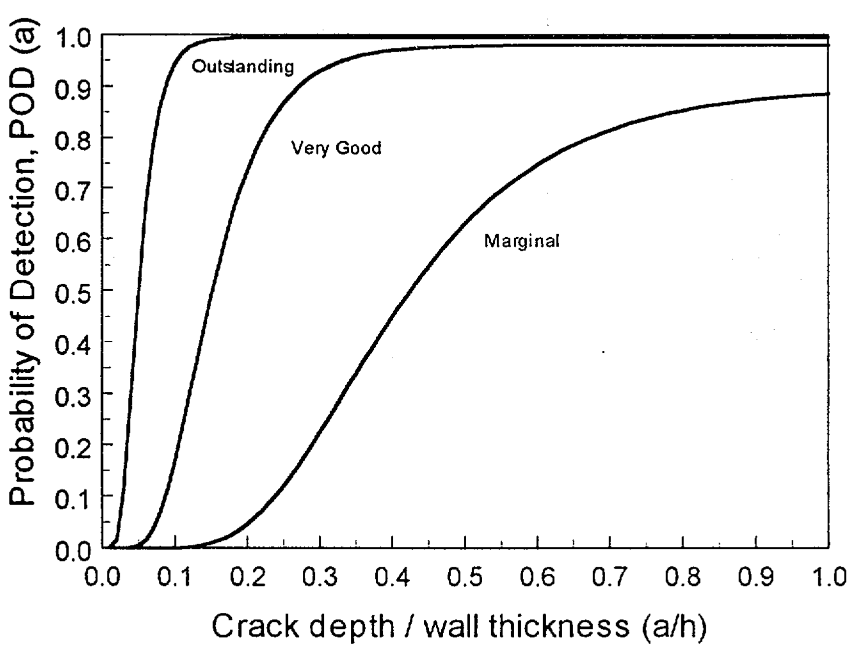

Figure A5.3 Example POD curve used in pc-PRAISE.

Example POD Curve - The specific functional form used in pc-PRAISE is given by

$$P_{ND}(a) = \epsilon + ½ (1-\epsilon) \text{ erfc } (v \ln (A/A^*))$$

where P_{ND} is the probability of non-detection, A is the area of the crack, A^* is the area of crack for 50% P_{ND}, ϵ is the best possible P_{ND} for very large cracks, and v is the "slope" of P_{ND} curve. Based on measured performance for PNNL's mini round robin teams (Reference 13), a range of estimates for A^* (crack area for 50% POD) was provided by the NDE experts. (Ref. 18) assumed that the "slope" parameter v is 1.6. Several POD curves from PNNL studies were analyzed, and it was determined that a value of v = 1.6 is both reasonable and consistent with published curves. While the assigned value of the slope parameter v was held constant, the actual slope of the plotted curves becomes more steep for better POD curves. Thus, the slope of the POD is correlated to the detection threshold parameter A^*. The value of ϵ was assigned such that a smaller value of A^* also implies a smaller value of ϵ.

Example Parameters for POD Curve - An approach taken for the evaluations of candidate inspection strategies has been to consider a range of POD curves that bound the range of performance expected from inspection teams that might actually perform inspections in the field (Ref. 5). This range of POD curves was established in consultation with NDE experts with extensive knowledge of the trends of NDE reliability studies and the performance levels needed to be successful in meeting the criteria of performance demonstration testing. The basic premise was that all teams had passed the ASME Section XI Appendix VIII performance demonstration. It should, however, be recognized that a population of inspection teams (all of which have passed the performance demonstration) operating under either the testing environment of performance demonstration trial or under field conditions can still exhibit a considerable range of POD performance, even though all such teams have successfully completed a performance demonstration. The performance demonstration serves to ensure a minimal level of NDE reliability.

The NDE experts were asked to define POD curves by estimating parameters for the specific form of a POD function used in the pc-PRAISE code given by the above equation. Three POD curves with increasing levels of performance were defined as indicated in Table A5.4:

- *Level 1 Performance*: This curve corresponds to a team that has a level of performance needed to pass an Appendix VIII performance demonstration.
- *Level 2 Performance*: This curve corresponds to the best teams. Such teams significantly exceed the minimum level of performance needed to pass the test.
- *Level 3 Performance*: This curve corresponds to a team that has a level of performance significantly better than expected from any teams that have to date passed an Appendix VIII-type of performance demonstration.

Table A5.4 Parameters of POD curves for three performance levels.

Inspection Performance	a^*(% a/t)	ϵ	v
Level 1	40%	0.10	1.6
Level 2	15%	0.02	1.6
Level 3	5%	.005	1.6

A5.7 Alternative Strategies To Reduce Failure Probabilities

It may be determined in some cases that none of the candidate inspection strategies can provide an adequate reduction in failure probability, or that strategies other than inservice inspection are more cost effective. Some degradation mechanisms can develop unexpectedly, and cause structural failures within time periods shorter than the proposed inservice inspection intervals. Examples are vibrational fatigue and thermal fatigue. New sources of vibrational stresses can develop due to imbalances that develop in rotating equipment or due to changes in the effectiveness of piping supports. Thermal fatigue stresses from the mixing of hot and cold fluids can develop over the life of a plant due to new sources of leakage at valves and thermal sleeves. The needed frequencies for inservice inspections can become unreasonable to detect impending structural failures associated with such new sources of fatigue related stresses. In these cases the most effective strategy can be to monitor the systems for piping vibrations and/or for temperature conditions that indicate the development of thermal fatigue stresses.

Continuous methods involving acoustic emission monitoring or leak monitoring can be used to supplement or replace periodic inservice inspections as a means to detect the progress of degradation in piping system components. Such methods are particularly useful when concern becomes focused on one specific location where degradation is known to exist, and the objective is an early indication that degradation is growing. Such continuous monitoring avoids the need to perform inspections at unreasonably small intervals, such as when calculations and/or measurements of damage (e.g., stress corrosion cracking or erosion/corrosion) indicate potentially high rates of degradation.

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Appendix 6: EXISTING DETERMINISTIC APPROACH AND REGULATORY REQUIREMENTS

A6.1 Introduction

The traditional deterministic ISI program requires extensive examination of the reactor coolant pressure boundary (RCPB), a moderate amount of examination of emergency core cooling and accident mitigation (ECC/AM) systems, and relatively little examination of support systems. Each facet of the examination strategy; level of detail required to define the parts examined, examination method, acceptance standard, and the extent and frequency of examination, is more tightly defined for the RCPB, than for the ECC/AM and support systems. The framework and philosophy of this approach is very prescriptive and is based on the assumption that the RCPB is more "important," and other systems are progressively less important as one moves away from the RCPB.

The basic requirements for deterministic ISIs for a boiling or pressurized water nuclear reactor facility, including inspection intervals, are contained in Section 50.55a, "Codes and Standards," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 1). The requirements specified in 10 CFR 50.55a will remain in effect after development and implementation of the risked-informed methods. Thus, the latter will provide an optional method for performing inservice inspections. The deterministic method can still be used, if the licensee chooses.

The primary objective of 10 CFR 50.55a is to ensure that "*Structures, systems, and components shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.*"

A6.2 Deterministic Decisionmaking Criteria

The sources of requirements for deterministic ISIs are specified in several documents referenced in 10 CFR 50.55a. These documents are listed below and described very briefly. For deterministic analysis the decision criteria are referenced by the requirements. For example, the ASME Boiler and Pressure Vessel Codes Section XI provides acceptance standards that are used to determine if the inspection requirements have been met.

a. ASME Boiler and Pressure Vessel Code - Section XI of this code provides most of the inspection requirements and acceptance criteria for deterministic ISI.

b. Technical Specifications - For some components, the inservice inspection requirements are governed by the plant Technical Specifications rather than the ASME Boiler and Pressure Vessel Code. In addition, the Technical Specifications require amendment if the ISI program revisions required by 10 CFR 50.55a create a conflict.

c. Regulatory Guides - In order to implement the requirements of the ASME Boiler and Pressure Vessel Code, "Code Cases" have been developed by the ASME to explain the intent of the code or provide for alternatives under special circumstances.

d. Nuclear Regulatory Commission Requirements - The Commission may require the licensee to follow an augmented ISI program for systems and components which they decide require added assurance of structural reliability.

For nuclear power plant components, conservative design practices have been successful in precluding anticipated modes of failures. For example, the ASME Boiler and Pressure Vessel Code identifies the following modes of failure:

- excessive elastic deformation, including elastic instability
- excessive plastic deformation
- stress rupture/creep deformation (inelastic)
- plastic instability - incremental collapse
- high strain - low cycle fatigue

Operating reactor experience has raised the issue that other causes not addressed in the design, by the ASME BPVC calculations or otherwise, are most likely to cause structural failures. The two most common examples are intergranular stress corrosion cracking (IGSCC) of stainless steel piping and erosion-corrosion wall thinning of carbon steel piping. Table A6.1 lists a variety of failure mechanisms or causes that should be considered.

The licensees should review industry experience of pipe failures. Available sources of information include NRC documents, EPRI documents, IAEA documents, INPO (Nuclear Plant Reliability Data System), NUMARC (Assessment of Plant Life Extension), ASME BPVC reports, etc. As this data is generically applicable to risk-informed regulatory activities, the industry might want to consider consolidating the information on a computerized system that is updated by individual utilities as they experience failures, detect service related degradation, and identify operational conditions not addressed in the design of the piping components. **Referencing the use of such a data base would provide the NRC assurances that industry experiences are appropriately addressed.**

A6.3 Documents with Deterministic Requirements

As stated in Section C.1, the overall requirements for deterministic ISI are specified in Section 50.55a, "Codes and Standards" of 10 CFR Part 50. Section 50.55a, in turn, references the following documents that contain the detailed requirements:

ASME Boiler and Pressure Vessel Code - The primary inspection requirements and intervals are contained in Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Division 1 of this document contains the requirements for light water cooled reactors.

Regulatory Guides - To implement the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, "Code Cases" have been developed by the ASME to explain the intent of the code or provide for alternatives under special circumstances.

In some cases the plant Technical Specifications will affect the ISI program.

The *(current)* deterministic ISI requirements are described in more detail in the following section.

Table A6.1 Example Failure Causes for LWR Nuclear Power Plant Components (From Ref. 2)

o Stress Corrosion cracking	o Improper or degraded over-pressure protection
o Intergranular attack	o Operation at loads or pressures exceeding design limits
o Thermal fatigue cracking related to: - stratification of fluids - leaking valve seats - thermal sleeve failure	o Excessive rates of heating or cooling (thermal shock)
o Vibrational fatigue cracking	o Structural damage from external forces
o Material or weld defects	o Improper or degraded supports for components
o Flow-assisted corrosion (erosion/corrosion)	o Defective snubbers restraining thermal expansions
o Cavitation and wet steam erosion	o Loose parts - wear and impact damage
o Slurry erosion (raw water lines)	o Loose or missing fasteners
o General corrosion	o Structural damage from maintenance
o Pitting	o Improper repairs or alterations
o Corrosion due to leaking boric acid	o Improper design and fabrication
o Microbe-induced corrosion	o Embrittlement from neutron irradiation
o Fretting	o Embrittlement from thermal aging
o Water hammer	o Improper heat treatment (of bolting materials)
o Over-pressure of system due to leaking or misaligned valves	
o Violations of pressure temperature limits	

A6.4 Inservice Inspection Requirements

Section 50.55a, "Codes and Standards," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, that each operating license for a boiling or pressurized water-cooled nuclear power facility and each construction permit for a utilization facility be subject to the conditions in paragraph (g), "Inservice Inspection Requirements," of § 50.55a. Paragraph (g) requires, in part, that ASME Code Classes 1, 2, and 3 components and their supports meet the requirements of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Boiler and Pressure

Vessel Code or equivalent quality standards. Paragraph 50.55a(b), in part, references the latest editions and addenda in effect of Section XI of the Code and any supplementary requirements to that section of the Code.

Definitions of the ASME Code Classes are given in (Ref. 2). Generally, ASME Code Class 1 includes all reactor coolant pressure boundary (RCPB) components. The RCPB refers to those pressure-containing components of BWRs and PWRs, such as pressure vessels, piping, pumps, and valves that are part of, or connected to, the reactor coolant system. ASME Code Class 2 generally includes systems or portions of systems important to safety that are designed for post-accident containment and removal of heat and fission products. These systems include the reactor shutdown, residual heat removal, and steam and feedwater systems extending from the steam generators to the outermost containment isolation valve. ASME Code Class 3 generally includes those system components or portions of systems important to safety that are designed to provide cooling water and auxiliary feedwater for the front-line systems.

Footnote 6 to § 50.55a references the ASME Code Cases that have been approved for use by the Commission. The footnote also states that the use of other Code Cases may be authorized by the Commission upon request pursuant to paragraph 50.55a(a)(2)(ii) which requires that proposed alternatives to the described requirements or portions thereof provide an acceptable level of quality and safety. The Code Cases applicable to deterministic ISI are contained in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability - ASME Section XI, Division 1."

Paragraph (g)(5)(i) of Section 50.55a requires that the ISI program for a boiling or pressurized water-cooled nuclear power facility be revised by the licensee, as necessary, to comply with Section XI of the ASME Code. If this revision conflicts with the Technical Specification for the facility, paragraph (g)(5)(ii) requires that the licensee apply to the Commission for amendment of the Technical Specifications so that they will conform to the revised program. If the licensee has determined that conformance with certain code requirements is impractical, the licensee must notify the Commission per paragraph (g)(5)(iii) and submit, as specified in § 50.4, information to support the determinations.

Paragraph (g)(6)(ii) of Section 50.55a states that the Commission may require the licensee to follow an augmented ISI program for systems and components which they decide require added assurance of structural reliability.

General Design Criterion 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 requires, in part, that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Where generally recognized codes and standards are used, Criterion 1 requires that they be identified and evaluated to determine their applicability, adequacy, and sufficiency and be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

A6.5 References for Appendix 6[1]

1. Section 50.55a, "Codes and Standards," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

2. USNRC, "Risk-Based Inspection - Development of Guidelines," Vol. 2, Part 1 (Prepared for the NRC by the American Society of Mechanical Engineers), NUREG/GR-0005, July 1993.

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[1]Copies of NUREGs are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-2249); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

Regulatory Analysis

1. Statement of the Problem

During the past several years, both the Commission and the nuclear industry have recognized that probabilistic risk assessment (PRA) has evolved to the point that it can be used increasingly as a tool in regulatory decisionmaking. In August 1995, the Commission published a policy statement that articulated the view that increased use of PRA technology would (1) enhance regulatory decisionmaking, (2) allow for a more efficient use of agency resources, and (3) allow a reduction in unnecessary burdens on licensees. In order for this change in regulatory approach to occur, guidance must be developed describing acceptable means for increasing the use of PRA information in the regulation of nuclear power reactors.

2. Objective

To provide guidance to power reactor licensees and NRC staff reviewers on acceptable approaches for utilizing risk information (PRA) to support requests for changes in a plant's current licensing basis (CLB). It is intended that the regulatory changes addressed by this guidance should allow both industry and NRC staff resources to be focused on the most important regulatory areas while providing for a reduction in burden on the resources of licensees. Specifically, guidance is to be provided in several areas that have been identified as having potential for this application. This application includes risk-informed inservice inspection programs of piping.

3. Alternatives

The increased use of PRA information as described in the draft regulatory guide being developed for this purpose is voluntary. Licensees can continue to operate their plants under the existing procedures defined in their CLB. It is expected that licensees will choose to make changes in their current licensing bases to use the new approaches described in the draft regulatory guide only if it is perceived to be to their benefit to do so.

4. Consequences

Acceptance guidelines included in the draft regulatory guide state that only small increases in overall risk are to be allowed under the risk-informed program. Reducing the inspection frequency of piping identified to represent low risk and low failure potential as provided for under this program is an example of a potential contributor to a small increase in plant risk. However, the program also requires increased emphasis on piping categorized as high-safety -significant and high-failure-potential that may not be inspected under current programs. This is an example of a potential contributor to decreases in plant risk. An improved prioritization of industry and NRC staff resources, such that the most important areas associated with plant safety receive increased attention, should result in a corresponding contributor to a reduction in risk. Some of the possible impacts on plant risk cannot be readily quantified using present PRA techniques and must be evaluated qualitatively. The staff believes that the net effect of the risk changes associated with the risk-informed programs, as allowed using the guidelines in the draft regulatory guide, should result in a very small increase in risk, maintain a risk-neutral condition, or result in a net risk reduction in some cases.

5. Decision Rationale

It is believed that the changes in regulatory approach provided for in the regulatory guide being developed will result in a significant improvement in the allocation of resources both for the NRC and for the industry. At the same time, it is believed that this program can be implemented while maintaining an adequate level of safety at the plants that choose to implement risk-informed programs.

6. Implementation

It is intended that the risk-informed regulatory guide on inservice inspection of piping (DG-1063) be published in final form by early to mid 1998.

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