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NUCLEAR ENERGY INSTITUTE

Stephen D. Floyd
DIRECTOR LICENSING AND
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REGULATION

April 15, 1996

Dr. Brian W. Sheron
Director, Division of Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

PROJECT NUMBER: 689

Dear Dr. Sheron:

This letter transmits Draft A of NEI 96-XX, "Industry Guideline for Risk-Based Inservice Inspection" and supporting documents described below, in support of industry risk-based ISI (RB-ISI) initiatives.

Pursuant to our discussions with your staff on RB-ISI, the following are enclosed for your review. These documents are being forwarded in support of the industry generic RB-ISI relief program including the three pilot plants. We respectfully request a waiver of any fees associated with the review of the following documents. These documents are developed by the nuclear industry through NEI, utilities, WOG, and EPRI and are being submitted to the NRC as a generic approach for implementation of the RB-ISI program at operating nuclear plants. We believe this waiver request satisfies criterion three of footnote four to 10 CFR 170.21. This footnote states that fees will not be assessed if the review request is made "as a means of exchanging information between industry organizations and the NRC for the purpose of supporting generic regulatory improvements or efforts."

Enclosure 1 is a copy of Draft A of NEI 96-XX, "Industry Guideline for Risk-Based Inservice Inspection," dated April 12, 1996. Enclosure 2 is fifteen (15) copies of WCAP-14572 (non-proprietary), "WOG Application of Risk-Based Methods to Piping Inservice Inspection Topical Report," dated March 1996, and one (1) copy of the Copyright Notice (Enclosure 2a). This WCAP supports the pilot application for RB-ISI for the Surry plant. Enclosure 3 is two copies of the EPRI Report TR-106218, "Risk-Informed Inservice Inspection Evaluation Procedure," dated January 1996 (twelve (12) more copies will be forwarded later). Please note that the EPRI

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report was previously sent to Dr. Goutam Bagchi for your staff with a request for withholding from public disclosure under 10 CFR 2.790(a)(4). A copy of this letter is included as Enclosure 3a. This EPRI report supports the pilot application for the RB-ISI for ANO-2 and Fitzpatrick plants. Enclosure 4 includes copies of the industry master plan on RB-ISI and the schedule for submittal of the guideline, supporting technical documents and pilot plant applications.

We would appreciate the opportunity to meet with you at your earliest convenience to make a formal presentation on the industry RB-ISI program and the above reports, including the NEI guideline on RB-ISI.

If you have any questions or comments on the enclosures, or would like to arrange a meeting, please call Mehdi Sarram (202-739-8186) of the NEI staff or me at (202) 739-8078.

Sincerely,

AMarion for

Stephen D. Floyd

MS/SDF/jes
Enclosures

Enclosure 1

NEI 96-XX (DRAFT A)

NUCLEAR ENERGY INSTITUTE

INDUSTRY GUIDELINE FOR RISK-BASED INSERVICE INSPECTION

April 12, 1996

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1 INTRODUCTION

Effective inservice inspection programs can play a significant role in reducing equipment and structural failures. This document provides the guiding principles and two alternative methods for applying risk-based techniques to inservice inspection programs for piping.

Risk-based regulation is defined as follows:

> "A regulatory approach in which operating experience and engineering judgment are used in concert with the analytical insights derived from probabilistic safety assessments to focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety."

1.1 Background

Currently, 10 CFR 50.55a requires each licensee of an operating boiling water or pressurized water-cooled nuclear power facility to implement a program for inservice inspection (ISI) of safety-related equipment. The specific requirements for an ISI program are contained in Section XI of the ASME Boiler and Pressure Vessel Code (B&PVC) and are incorporated into 10 CFR 50.55a by reference.

1.2 Discussion

The design of major reactor systems and components is governed by the ASME Boiler and Pressure Vessel Code (B&PVC). The primary criteria used to establish the ASME B&PVC Classification are the US NRC's Regulatory Guide 1.26 and NUREG-0800, Section 3.2.2. Three classifications are currently considered in the ASME Code: Class 1, Class 2 and Class 3. These classifications are based on implicit considerations of risk. A given system can have multiple classifications.

Generally, Class 1 systems include all reactor coolant pressure boundary (RCPB) components. The RCPB refers to all those pressure-containing components of BWRs and PWRs, such as pressure vessels, piping, pumps, and valves that are part of the reactor coolant system or connected to the reactor coolant system. For BWRs, the reactor coolant system extends to and includes the outermost containment isolation valve in the main steam and feedwater piping. 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. Class 3 generally includes those system components or portions of systems important to safety that are designed to provide cooling water and auxiliary feedwater to the front-line systems.

The ASME B&PVC Section XI requires the most stringent inspections for Class 1 components and the less stringent requirements of components assigned to Class 2 and 3 categories. Inspections are focused toward welds because potential defects and failures have been believed to be more likely at welds than at other locations. Other qualitative considerations further focus ASME B&PVC Section XI inspections, for example, by identifying terminal ends of piping runs, dissimilar metal welds, and locations of high calculated fatigue usage are identified as important components.

In the past 20 years, the probabilistic safety assessment (PSA) technology and applications have evolved and matured. The first significant regulatory application of the PSA technology occurred in the preparation of WASH-1400, "Reactor Safety Study, Assessment of Accident Risk in U.S. Commercial Nuclear Power Plants," dated 1975. Subsequently, to incorporate the significant advancements in technology since WASH-1400, the NRC conducted another comprehensive assessment of public risk in the United States from nuclear plant operation. The results of this study were published in 1987 as NUREG-1150, "Reactor Risk Reference Document." This document included sensitivity studies which showed that public risk was dominated by a small set of plant equipment, much smaller than the set of equipment initially designated as safety-related. Further confirmation of these risk insights were provided by licensees during conduct of the Individual Plant Examination (IPE) in response to the NRC Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," dated 1988.

2 PURPOSE AND SCOPE

The purpose of this document is to provide guidance for optimizing regulatory mandated and other docketed inspections by applying risk-based technology to piping components. This document provides guidance for structuring an inspection program that is premised upon deterministic, operational experience and risk-based concepts. It is expected that the added application of risk technology to the site inspection programs will significantly reduce the resources applied to the inspection processes while maintaining a high level of safety. The guidance provided in this document will enable a utility to develop and implement inspection programs which focus available resources on the components in a manner that is commensurate with the associated risk significance of the component. If a

component is identified as having a high safety impact, but is not addressed by current inspection programs, it would be inspected commensurate with its safety significance.

The approaches described in this guideline to implement a risk-based inspection program include:

- Continued assurance that plant components will perform their intended safety functions.
- Utilization of deterministic and risk-based methodologies, tempered with operating experience feedback.
- Licensee flexibility to implement cost-effective inspection methods while maintaining a high level of safety.

This document delineates guidance for the development and implementation of an alternate to the regulatory mandated and currently docketed inspections at Nuclear Power Plants, based upon deterministic and risk-based regulatory concepts. This document does not redefine a plant's licensing basis and the identification of a component as high safety impact is for component inspections only and is not intended to require any design changes to the plant. This document does not address the methodology of how to conduct examinations.

The approach used in this document is consistent with the EPRI PSA Applications Guide (Ref. 2) and is integrated with ongoing Maintenance Rule activities. When applying risk-based inspection criterion to a plant, the plant should consider all plant piping systems within the scope of ASME Section XI and consider the portions of the plant PSA applicable to the risk-based inservice inspection (RBISI) evaluation.

3 RESPONSIBILITY

If a licensee elects to implement a RBISI program, the guidance described in this document is an acceptable method of implementation.

4 APPLICABILITY

This guidance is applicable to licensees holding an operating license issued in accordance with 10 CFR 50.21(b), 50.22, or 10 CFR Part 52, Subpart C.

5 GENERAL REQUIREMENTS

5.1 Requirements of 10 CFR 50.55a

10 CFR 50.55a, "Codes and Standards," requires each licensee of an operating boiling or pressurized water-cooled nuclear power facility to be subject to the conditions in Paragraph (g), "Inservice Inspection Requirements." 10 CFR 50.55a(g) indicates that the specific requirements for inservice inspection (ISI) of Class 1,2, and 3 components (including supports) are contained in ASME Section XI. The latest NRC-approved edition and addenda of ASME Section XI is listed in 10 CFR 50.55a. Footnote 6 of 10 CFR 50.55a currently states that Code cases related to ISI that have been determined suitable for use by the NRC are listed in Regulatory Guide 1.147. This footnote also states that the use of other ISI Code cases not listed in the most recent revision of 1.147 may be authorized by the NRC upon request pursuant to 10 CFR 50.55a(a)(3).

In addition, 10 CFR 50.55a, Subsection (a)(3) permits licensees to propose alternative approaches to those listed in Section XI under certain circumstances. It states, in part, that:

Proposed alternatives to the requirements of paragraph (g) of this section 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, or

(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

5.2 Current Inspection Requirements

ASME Section XI provides detailed listings of what nondestructive examination technique or combination of techniques (i.e., volumetric, surface, or visual) shall be used for each component. The components to be examined and the appropriate examination techniques have evolved over the 28 years since the "draft" ISI Section XI Code was published in 1968. The approach has typically been to require a larger percentage of ASME Item Numbers to be examined in Class 1 system than in Class 2 system and similarly for Class 2 than Class 3. In addition, Class 3 components receive only a visual

examination while Class 1 and 2 components typically receive volumetric and/or surface examinations.

5.3 Relationship Between RBISI and Current Section XI Programs

Any plant which decides to use RBISI techniques will be required to show that the resulting ISI program is an acceptable alternative to the Section XI requirements as permitted by 10 CFR 50.55a, Subsection (a) (3).

It should be noted that RBISI is only an option to the licensees and that an individual plant may determine that it is best to continue an ISI program that uses existing Section XI requirements as permitted by 10 CFR 50.55a, Subsection (a).

6 METHODOLOGY/APPROACH

6.1 Overview and Scope

A RBISI evaluation process can be used to define the safety significance of piping segments and identify the locations to be examined.

The risk-based evaluation process is illustrated in Figure 6-1. The following is a summary of the items considered in the risk-based evaluation process illustrated in Figure 6-1. Each item is discussed in more detail in sections 6.2 through 6.5.

Piping System Identification

- The inspection scope should include systems and systems' boundaries in Class 1, 2, and 3 piping in the Licensee's existing ISI program.
- The piping system safety functions should be defined.
- Applicable portions of the plant PSA may be used to evaluate piping within the related systems and system's boundaries.

Piping Segment Assessment

Systems are divided into piping segments based on consequence, and may be further refined based on failure potential, as applicable. Each piping segment receives the following evaluations:

- Consequence Evaluation
  - The consequence evaluation shall be performed for all piping segments within the system.
  - All modes of operation shall be included in the evaluation.
  - The evaluation uses a blend of deterministic and probabilistic evaluation methods and inputs.
  - Direct and indirect effects of piping segment failure shall be considered in the evaluation.
- Failure Potential Assessment
  - A systematic evaluation of potential degradation mechanisms/failure potential shall be performed for all piping segments in the RBISI scope.
  - Industry and plant specific service experience, and plant specific operational or postulated conditions shall be factored into the program.
- Risk Categorization
  - The risk categorization shall be based on the combination of failure consequence and failure potential for piping segments.
  - The pipe segments shall be ranked according to their safety impact.

<u>Piping Element and Location Assessment</u>

- Examination locations in the high safety impact segments shall be considered for examination based on potential degradation mechanisms and loading conditions within the segment.
- Elements in low safety impact segments will continue to be inspected via visual examination and leak tests as part of the ISI program.
- Examination locations shall be selected for examination considering access, radiation exposure, cost and relative severity of the postulated degradation mechanism.

<u>Examination Methods Determination</u>

- Examination methods shall be selected based on identified degradation mechanisms and affected regions.

The remaining paragraphs in this section provide additional guidance for performing the RBISI evaluation. Specific, detailed procedures for performing the evaluation are presented in References 3 and 4.

6.2 Piping System Identification

The licensee shall define the system boundaries included in the scope of the RBISI evaluation. Each system boundary should be defined consistent with the FSAR and other design documents. System safety functions should be defined for all operating modes. Within each system boundary, the RBISI evaluation shall include Class 1, 2, and 3 piping in the licensee's ASME Section XI program. The portions of the PSA applicable to the RBISI evaluation shall also be identified.

Licensees should be aware that the potential exists for differences between the PSA and ASME Section XI system boundaries.

6.3 Piping Segment Assessment

Systems are divided into piping segments based on insights from a consequence evaluation, and may be further subdivided using both insights from a consequence evaluation and a failure potential assessment. The piping segments shall incorporate both the consequence evaluation and failure potential assessment into the overall risk categorization process.

6.3.1 Consequence Evaluation

The consequence evaluation shall focus on the impact of a pipe segment failure (loss of pressure boundary integrity) on plant operations. Both direct and indirect effects shall be included in the consequence evaluation. The consequences of pipe segment failure can be characterized using PSA related parameters such as the conditional probability of core damage, or deterministic impacts such as loss of system(s), loss of train(s), or combinations of these parameters.

The following parameters should be considered in the consequence evaluation:

- <u>Break Size</u>: the break size to be addressed in the analysis should be evaluated.

- <u>Isolability of the Break</u>: the possibility of isolating a break should be evaluated.

- Spatial Effects: spatial effects are an example of indirect effects of the pressure boundary failure. The effects of flood, spray, and pipe whip on equipment located in the vicinity of the break should be evaluated.

- Initiating Events: the potential for pressure boundary failure resulting in an initiating event or forced plant shutdown should be evaluated.

- Loss of System/Recovery: the potential for pressure boundary failure to result in a loss of a system or possibly multiple systems should be evaluated.

- Loss of Train/Recovery: the potential for pressure boundary failure to result in a loss of a train or possibly multiple trains should be evaluated.

6.3.1.1 Direct Effects

The direct effects to be considered include:

- failures that cause an initiating event such as a LOCA or reactor trip
- failures that disable a single train or system
- failures that disable multiple trains or systems
- failures that cause any combination above

6.3.1.2 Indirect Effects

Indirect effects evaluations should include consideration of pipe whip, jet impingement, and flooding. The information sources that should be considered to identify indirect effects include the plant's hazard evaluation to meet the requirements of the NRC's Standard Review Plan and the PSA's internal flooding events analysis. A plant walkdown of key areas should also be conducted. The impact of the indirect effects to be considered should be the same as stated above for the direct effects.

6.3.2 Failure Potential Assessment

A failure potential assessment shall be performed to identify the likelihood that active degradation mechanisms are present in piping segments and to

determine the potential for a pipe failure or large leak due to the degradation mechanism.

For nuclear power plant components, conservative design practices have been successful in precluding most anticipated modes of failure. The majority of experienced piping system degradation results from relatively slow growth type degradation mechanisms. In these cases, a defect which initiates and grows over time may, if undetected, progress through the pipe wall. Unless some abnormal loading (e.g., water hammer) occurs in combination with these mechanisms, degradation will occur over an extended period of time and result in a relatively small leak, rather than a large leak or pipe break. These mechanisms lend themselves well to the periodic inservice examinations as a means of managing the structural integrity of the component. Table 6-1 provides an example of degradation mechanisms that should be considered in the RBISI process.

The following design, fabrication and operational conditions should be considered for degradation mechanism identification.

- Design Characteristics:

  Design conditions include material selection, pipe size and schedule, component type (fitting type, ANSI standard, etc.) and other attributes unique to the system layout. It should be taken into account that design conditions will vary between systems, and can occasionally vary within a system.

- Fabrication Practices:

  Fabrication practices may include material selection, weld wire, heat treatment, etc. It is expected that piping elements subjected to nuclear standards will not be exposed to damage mechanisms due to fabrication practices. However, past experience has shown that even nuclear standards have not prevented damage from phenomena unknown at the time of installation.

- Operating Conditions:

  Operating conditions determine the piping elements' internal and external conditions that impact material degradation. These include operating temperatures and pressures, fluid conditions (stagnant, laminar, turbulent flow), fluid quality (primary water, raw water, dry

steam, etc.), chemical control, and service environment (humidity, radiation, etc.).

- Service Experience:

  Industry and plant specific service experience provide information that can be used to identify the potential for the presence of degradation mechanisms, and confirmation that damage mechanisms identified for a specific location are appropriate and complete.

  Additional guidelines for identifying the presence of actual degradation mechanisms in piping segments are presented in references 3 and 4.

6.3.3 Risk Categorization

Risk or safety categorization of pipe segments shall be determined by the combination of the likelihood of pipe segment degradation or failure, and the consequence of pipe failure.

The likelihood of pipe segment failure in this analysis is estimated based on the likelihood that an active degradation mechanism is present in the pipe segment, and the potential for a large leak or the probability of pipe failure given the potential loading conditions on the piping segment.

Consequences of failure may be based on the conditional core damage probability determined from a PSA analysis, or on the impact on plant safety determined from a traditional deterministic analysis.

Safety categorization of specific pipe segments for various combinations of consequence of segment failure, and of segment degradation or failure may be made using the risk measures and procedures described in References 3 and 4.

6.4 Piping Element and Location Assessment

Once the high safety impact segments are identified, the individual piping elements and locations for examination are selected based on potential degradation mechanisms and plant specific/industry experience.

Inspection locations shall be determined considering the following:

- Portions of the element that are susceptible to the specific degradation mechanism.

- Access - There should be adequate access to the element to ensure the examination method defined in this section for the relevant damage mechanism can be used effectively for the defined examination volumes.
- Radiation exposure - Elements should be selected with consideration of potential personnel radiation exposure during inspection.
- Relative degradation severity for specific degradation mechanisms.
- Elements having break or consequence limiting devices, (e.g., pipe whip restraints) need not be inspected, if not considered in the consequence evaluation.
- Plant specific inservice degradation experience (e.g., cracking, thinning, etc.).

6.5 Examination Methods Determination

For the high safety impact elements and locations, examination methods, inspection volumes, and acceptance and evaluation criteria determined specifically for the degradation mechanisms active at the inspection location shall be used for the examination. Table 6-1 provides recommended methods.

Alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods in Table 6-1, provided the authorized nuclear inservice inspector (ANII) is satisfied that the results are demonstrated to be equivalent or superior to those specified.

Elements in low safety impact segments will continue to be inspected via visual examination and leak tests.

The examination should be completed during each inspection interval. Currently the interval is 10 years. The inspections are generally distributed across periods such that a portion of the inspections are conducted in each period such that all high safety impact segments are inspected over the 10-year interval.

6.6 Implementation and Feedback

6.6.1 Implementation

The results of the RBISI process are used for selection and examination of piping elements as an alternative to the licensee's current ASME Section XI program. Remaining requirements of Section XI still apply.

6.6.2 Feedback

The RBISI program should be reevaluated periodically as new information becomes available. Such information may result, for example, from changes to the plant design, from inspection results, from new failure modes experienced by the industry, from replacement activities, from repair activities, or operational changes. The new information should be included at the appropriate level of the analysis and the analysis should be conducted to identify the changes to the RBISI program.

7 Documentation

Documentation developed for implementation of this guideline should be collected and retained in accordance with the licensee's established procedures. Sufficient documentation should be collected and retained so that the effectiveness of the implementation of the RBISI program can be reviewed and evaluated. This documentation should be available for internal and external review, but is not required to be submitted to the NRC.

8 Glossary (Later)

9 References

1. ASME RB-ISI Development of Guidelines, Vol. 2, Part 1,1992, and Volume 2, Part 2, February 1996(draft).
2. EPRI PSA Applications Guide, TR-105396, August 1995
3. Westinghouse Report WCAP 14572, "Westinghouse Owners' Group Application of Risk-Based Methods to Piping Inservice Inspection Topical Report,", March 1996.
4. EPRI Report TR-106218, "Risk-Based Inservice Inspection Evaluation Procedure," February 1996

TABLE 6-1

EXAMINATION CATEGORIES FOR HIGH SAFETY IMPACT PIPING ELEMENTS

Parts Examined(3)	Examination Requirements Fig. No. *	Examination Method	Acceptance Standard *
Elements Subject to Thermal Fatigue	IWB-2500-8(c)[2] IWB-2500-9, 10, 11 IWC-2500-7(a)[2]	Volumetric	IWB-3514
Elements Subject to High Cycle Mechanical Fatigue	–	Visual, VT-2	IWB-3522
Elements Subject to Corrosion Cracking	IWB-2500-8(c)	Surface Volumetric	IWB-3514 IWB-3514
Elements Subject to Crevice Corrosion Cracking	Note 4	Volumetric	IWB-3514
Elements Subject to PWSSC(1)	–	Visual, VT-2	IWB-3522
Elements Subject to IGSCC	IWB-2500-8(c) IWB-2500-9, 10, 11	Volumetric	IWB-3514
Elements Subject to MIC	–	Visual, VT-1 or Volumetric[5]	IWA-5250(b)
Elements Subject to Flow Accelerated Corrosion (FAC)	Note 6	Note 6	Note 6

* Representative Examination Requirements and Acceptance Standards can be found in the referenced sections of the 1989 ASME B&PVC, Section XI. Alternative Code Editions and Addenda, as appropriate, may be used.

NOTES:

1. Applies to mill annealed Alloy 600 nozzle welds and HAZ w/o stress relief.
2. The length for the examination volume shall be increased to include 0.5 inch beyond each side of the base metal thickness transition.

3. Includes 100% of the inspection location. When the required examination volume or area cannot be examined due to interference by another component or part geometry, limited examinations are acceptable provided they are documented on the applicable NDE report. Areas with limited examinations shall be identified in the inservice summary report.
4. Examination volume shall include the volumes 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.
5. The examination volume shall include base metal, welds and weld heat affected zones in the affected regions of carbon and low alloy steel, and the welds and weld heat affected zones of austenitic steel. The examination region should be sufficient to characterize the extent of the MIC degradation.
6. In accordance with the Owner's existing flow accelerated corrosion (FAC) program.

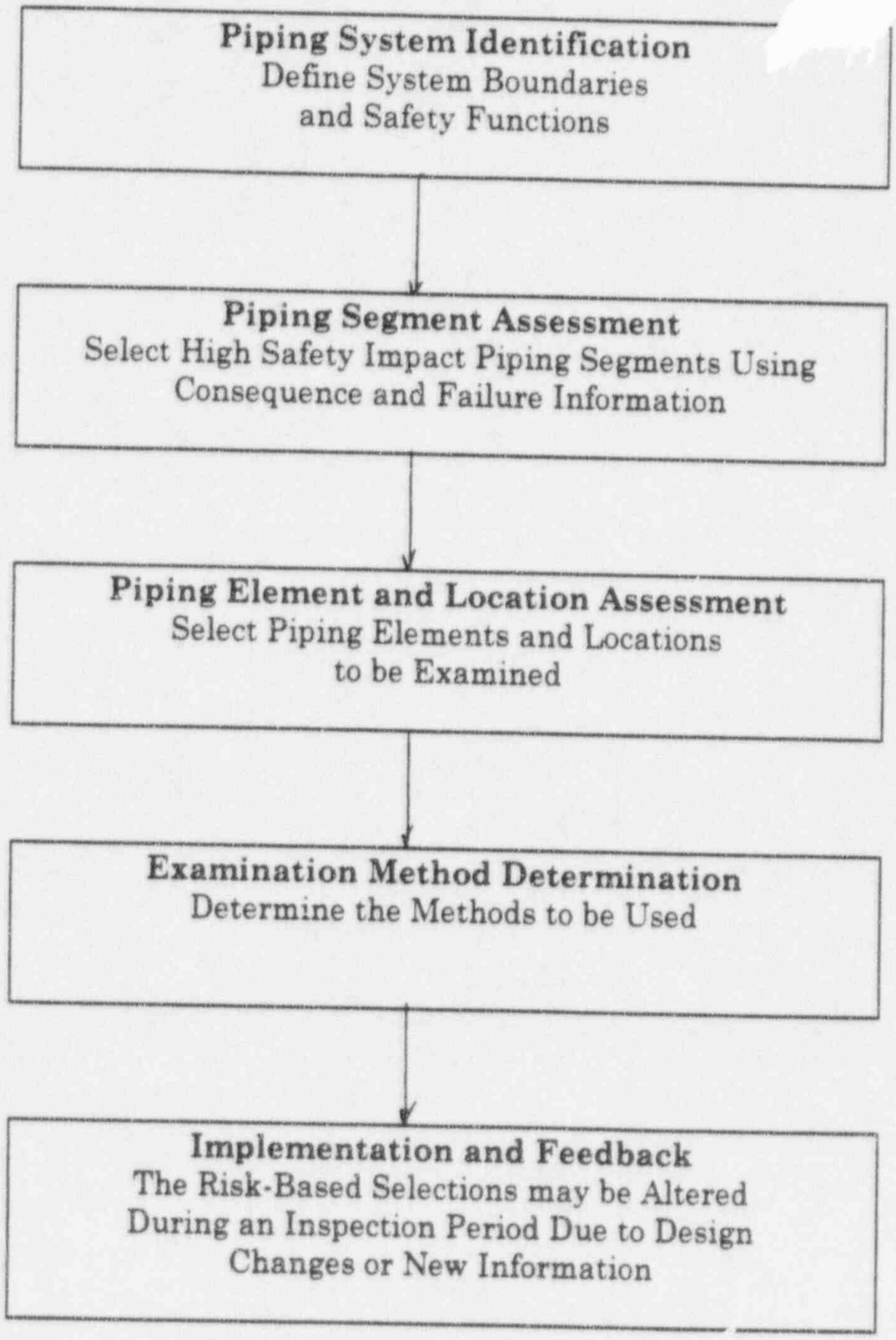

Figure 6-1
Overview of RBISI Selection Process

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