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WESTINGHOUSE OWNERS GROUP APPLICATION OF RISK-BASED METHODS TO PIPING INSERVICE INSPECTION TOPICAL REPORT

Revision 0

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EXECUTIVE SUMMARY

Inservice inspections are intended to play a key role in minimizing structural failures. The objective of inservice inspection (ISI) is to identify conditions, such as flaw indications, that are precursors to leaks and ruptures, which violate pressure boundary integrity principles for plant safety.
All aspects of inspection, including where, when, and how to inspect, affect the benefits of the inspection for enhancing component structural reliability. In addition, accept/reject criteria and repair procedures have a significant influence. Inspections are currently performed based on mandated requirements, such as those for nuclear power plant components in the ASME Boiler and Pressure Vessel Code, insurance requirements, company policy, etc. Most inspection requirements are based on past experience and engineering judgment and have only an implicit consideration of risk-based information, such as failure probability and consequence impacts for the specific material, operation and loading conditions.

Technologies for risk assessment of systems and components have been developing rapidly over the past two decades concurrently with progress in inspection technology and methods for assessment of component structural reliability and the effects of inspection. In fact, all nuclear power plants have been required to perform an Individual Plant Examination (IPE) per the requirements of NRC Generic Letter 88-20 to determine plant vulnerabilities to severe accidents such as core damage and large early release from containment. These developments provide the capability of selecting between candidate inspection programs based on quantitative estimates of the risks associated with component failure, including related inspection and failure costs. Both the probability and the consequence of component failure enter into the evaluation of risk, and inspection programs can be formulated based on managing these risks and related costs.

This report describes a program for showing the benefits of using risk-based technologies to reduce overall operation and maintenance costs associated with the inspection of nuclear power plant components while maintaining a high level of safety. Risk-based inspection processes were applied to evaluate the impact of requirements and methods currently being developed for component/system inspection on overall operation and maintenance costs.

The overall risk-based inservice inspection process is described. The focus of this report is on the identification of the inspection locations using a risk-ranking process. The risk-ranking process includes the following steps:

- Scope Definition
- Segment Definition
- Consequence Evaluation
- Failure Probability Estimation
- ISI Segment Selection
- Structural Element Selection
- Inspection Requirements

The results of the segment risk-ranking for representative WOG plant shows that approximately 37% of the 259 piping segments were determined to be more safety-significant, or to have a high safety impact.

These piping segments were then subjected to a detailed review to determine the structural elements that should be inspected. A comparison of the elements of the risk-based ISI program and those structural elements defined by the current ASME Code requirements was made. This comparison showed that under current ASME code, a total of 753 welds would be inspected while the risk-based ISI program would inspect only 119 structural elements. The current ASME Code inspection locations address approximately 44% of the piping pressure boundary core damage frequency while the risk-based ISI program addresses approximately 98% of the piping pressure boundary failure core damage frequency with less locations being inspected. The total piping core damage frequency, however, is only a small fraction of the total core damage frequency for the plant. While these results represent an application to a plant designed to ASME Section III, the risk-based ISI process should yield similar results to piping systems in plants not designed to ASME Section III.

This program has been of benefit by providing a sound technical approach and process for developing a risk-based inservice inspection program for piping that can be implemented in the industry. It is expected that the risk-based ISI program can be implemented at a cost much less than the direct savings that are gained from piping examinations done in one outage. The representative WOG plant application has shown the potential for reducing operation and maintenance costs in the development and implementation of effective piping inspection programs while maintaining a high level of safety.

ACKNOWLEDGEMENTS

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INTRODUCTION

Inservice inspections are intended to play a key role in minimizing structural failures. The objective of inservice inspection (ISI) is to identify conditions, such as flaw indications, that are precursors to leaks and ruptures, which violate pressure boundary integrity principles for plant safety. All aspects of inspection, including where, when, and how to inspect, affect the benefits of the inspection for enhancing component structural reliability. In addition, accept/reject criteria and repair procedures have a significant influence. Inspections are currently performed based on mandated requirements, such as those for nuclear power plant components in the ASME Boiler and Pressure Vessel Code, insurance requirements, company policy, etc. Most inspection requirements are based on past experience and engineering judgment and have only an implicit consideration of risk-based information, such as failure probability and consequence impacts for the specific material, operation and loading conditions.

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This project demonstrates the benefits of using risk-based technologies to reduce overall operation and maintenance costs associated with the inspection of nuclear power plant components while maintaining a high level of safety. Risk-based inspection processes were applied to evaluate the impact of requirements and methods, currently being developed for component/system inspection, on overall operation and maintenance costs.

1.1 PROGRAM OBJECTIVES

The objective of this program is to apply and document the risk-based inservice inspection process as an alternative for selecting and categorizing piping components into more safety-significant[1] and less safety-significant[1] groups for purposes of meeting ASME BPVC Section XI Inservice Inspection (ISI) requirements.

1.2 PROGRAM BENEFITS

Risk-based processes are used to improve the effectiveness of inspection of components; to enhance inspection strategies in some areas by inspecting for cause and reduce inspection requirements in others; to evaluate improvements to plant availability and enhanced safety measures; and to reduce overall operation and maintenance (O&M) costs while maintaining regulatory compliance and maintaining or enhancing the plant safety. The program focuses inspection resources on more safety-significant piping locations and locations where failure mechanisms are likely to be present. Risk-based ISI programs offer the potential to reduce outage times by defining a smaller set of more safety-significant components that must be addressed during critical paths and by defining more effective inspection programs to reduce the impact on plant outages.

Longer term benefits include knowledge of the safety versus economic benefits of inspection and the cost savings resulting from the risk-based optimization of the locations that require inspection at each interval. Additional cost savings may be realized from the elimination of unplanned outages caused by ineffective inspection strategies that miss potential degradation that could lead to piping failures during plant operation.

1.3 PROGRAM TASKS

This program involves risk-ranking of piping locations to determine the more safety-significant locations to focus the inspection efforts. It also involves the evaluation of various inspection strategies

[1] The terminology for grouping components is still under discussion. For purposes of this document, the term "more safety-significant" is equivalent to "high safety impact," and the term "less safety-significant" is the same as "low safety impact".

and providing recommendations on the appropriate inspection strategy for a given piping type and function.

1.4 REPORT ORGANIZATION

Section 1 of this report provides an overview of the objectives, benefits and tasks of the program. Section 2 provides the background as to why research and applications on risk-based technology with inservice inspection are being performed and other industry factors that potentially impact this program. Section 3 describes the process, results and insights gained from the application. Section 4 provides guidance on inspection methods and compares the risk-based results with the current ASME Section XI inspection locations. Section 5 describes the process to be applied on a plant-specific basis while Section 6 summarizes the findings and recommendations for the application.

SECTION 2

BACKGROUND

This section describes the background and other industry activities associated with applying risk-based methods to inservice inspection.

2.1 CURRENT ASME CODE REQUIREMENTS

The current inspection requirements for nuclear components are found in the ASME Boiler and Pressure Vessel Code (BPVC) Section XI.

Class 1 components include piping and components whose failure would prevent orderly reactor shutdown and cause a loss of coolant in excess of normal makeup capability. This includes the principal fluid systems components of the reactor coolant pressure boundary heat transfer loops and also includes portions meeting this criterion of the piping, fittings and valves leading to connecting systems.

Class 2 components are associated with the reactor containment and include those valves and components of closed systems used to effect isolation of the reactor containment atmosphere, components of the reactor coolant pressure boundary not covered in Class 1 and safety system components of the following: residual heat removal system, portions of the reactor coolant auxiliary systems that form a reactor coolant letdown and makeup loop, reactor containment heat removal systems, emergency core cooling system including injection and recirculation portions, air cleanup systems used to reduce radioactivity within the reactor containment, containment hydrogen control system and portions of the steam and feedwater systems.

Class 3 components include safety system components. This includes portions of the reactor auxiliary systems that provide boric acid, emergency feedwater system, portions of components and process cooling systems (electrical and/or compressed air) that cool other safety systems including the spent pool cooling system, on-site emergency power supply and auxiliary systems, some air cleanup systems that reduce radioactivity released in an accident.

There are several examination techniques specified in the code. These are described below.

A visual examination of piping components is performed to determine the general conditions (such as cracks, wear, corrosion, erosion or physical damage) of the part, component or surface examined by direct or remote observation (VT-1), to search for evidence of leakage from pressure retaining components, or abnormal leakage from components with or without leakage collection systems (VT-2) and to assess the general mechanical and structural conditions of components and their supports, such as the verification of clearances, seatings, physical displacements, loose or missing parts, debris, corrosion, wear, erosion, or the loss of integrity at bolted or welded connections (VT-3).

A direct visual examination requires a sufficient space to place the eyes within 24 inches from the surface with an angle not less than 30 degrees from the surface. Mirrors or lenses can be used to improve vision.

Remote visual examination using telescopes, optical fibers, cameras, television systems and other instruments shall have a minimum resolution capability that cannot be less than that seen by direct visual examination.

A visual examination is required to detect leakage during hydro testing.

A surface examination is performed to detect and size surface or near-to-surface flaws. It may be conducted by either a magnetic particle (MT) or a liquid penetrant (PT) method, eddy current, or other newly developed techniques.

A volumetric examination is performed to detect and size flaws throughout the volume of material. It may be conducted by radiographic (RT), ultrasonic (UT), eddy current, a combination of methods or newly developed techniques.

Effectiveness of Current Requirements

One ASME committee recently reported on the current inspection requirements for class 1 piping welds (category B-J). The following is extracted in part and in some instances directly quoted from "Evaluation of Inservice Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds in Piping," developed by the ASME Section XI Task Group on ISI Optimization (ASME 1995).

"The current ASME BPVC Section XI requirements for class 1 piping welds were established in 1978. Inspection was focused on critical welds presumed to have the highest potential of failure. Inspections were concentrated on terminal ends, dissimilar metal welds, and welds with higher stress levels and fatigue usage factors. Twenty years of service experience, however, has shown no correlation between the welds selected for examination using current criteria (Category B-J) and actual reported problems. The majority of flaws found in Category B-J piping welds have been caused by factors outside the scope of the current selection criteria (e.g., Intergranular Stress Corrosion Cracking (IGSCC), thermal stratification). This is due in part to the fact that stress analyses are dependent on design conditions such as seismic events more than actual service conditions. A recent industry survey, which included 50 nuclear plants representing 733 cumulative years of reactor operation, confirmed this conclusion. The results are summarized as follows:

1) Of all the survey responses, only 156 Category B-J welds were found to contain service induced flaws.

2) Of the 156 welds containing flaws, the degradation mechanism for 151 of them was IGSCC. Only five welds had flaws attributed to other failure mechanisms.

3) Of the 156 welds containing flaws, 55 were detected by ASME Section XI examinations. The remaining were detected by augmented methods (i.e., U.S. Nuclear Regulatory Commission requirements), visual inspections or leakage.

4) Two of the welds contained flaws caused by "general corrosion" because boric acid from a different system had dripped onto the subject piping.

5) The total population of Category B-J welds addressed in this survey is 37,332. Assuming 25% of the total population was inspected per ASME Section XI, the number of welds inspected would be 9333. Using this number, the following percentages can be calculated:

   - 1.67% of the welds inspected contained flaws.

- 0.05% of the welds inspected contained flaws caused by a mechanism other than IGSCC.

- 0.6% of the welds inspected were found to contain flaws by ASME Section XI examinations.

6) None of the flawed welds fell into the category of "high stress/high fatigue" welds. Therefore, there is no apparent relationship between flaws detected and welds selected for inspection due to high design stresses or high fatigue usage factor considerations. (It is recognized that one of the contributing factors to this trend is that many of the older plants cannot categorize by high stress/high fatigue locations because their construction code ANSI B31.1.0 analysis of record is not location specific.)

Given the twenty-plus years of operating nuclear power plant experience in the U.S. and overseas, and the fact that no new plants or plant types, which may be prone to some new degradation mechanism, are being placed in service, it is logical to ask whether a more efficient and technically meaningful means of selecting welds for inservice inspection is possible. Also, recent advances in risk-based inspection approaches have illustrated that the consequences of failure at a piping location, in terms of threat to reactor safety, ought to play at least as important role in selecting inspection locations as the probability of failure at that location."

This report captures the need to reevaluate the current requirements if improved safety, detection of flaws in components and optimization of critical resources are to be a reality. The need for this improvement is self-evident as the utility industry comes nearer to market deregulation in conjunction with continued safe operational requirements. The incorporation of risk-based technology into inservice inspection programs will benefit both the utility industry and regulatory bodies in that the "mechanical integrity" of plant components utilizing risk-based methods in performing inservice inspections will be more accurately known. The utility industry and regulatory bodies will be better able to deal with "aging and life-extension" than under today's rules. In addition, both the regulatory bodies and utility industry will be able to better focus and allocate limited resources to the risk-

significant components. Utilities should experience a reduction in plant operating and maintenance costs associated with risk-based inservice inspection while maintaining a high level of safety.

2.2 ASME RISK-BASED RESEARCH AND CODE EFFORTS

The ASME has recognized the need for the use of risk-based methods in the formulation of policies, codes and standards. In 1985, under the direction of the ASME Council on Engineering, a Risk Analysis Task Force was formed to provide recommendations on how this need could be met.

At the suggestion of the Risk Analysis Task Force, the ASME Codes and Standards Research Planning Committee recommended in 1986 that a research program be initiated to determine how risk-based methods could be used to establish inspection requirements and guidelines for systems and components of interest to the engineering community. Since late 1988, a multi-disciplined ASME Research Task Force on Inspection has been evaluating and integrating these technologies in order to recommend and describe appropriate approaches for establishing risk-based inspection guidelines. The task force is comprised of members from private industry, government, and academia representing a variety of industries. Figure 2.2-1 shows the relationship of this ASME research program to the ASME Code.

The research task force published its first document titled, "Risk-Based Inspection - Development of Guidelines, Volume 1, General Document" (ASME 1991). This document describes general risk-based processes and methods that can be used to develop inspection programs for any industrial facility or structural system. Specific applications of this general methodology to particular industries are being addressed by a subsequent series of supplemental volumes. Volume 2 - Part 1, which is the first document of this series, is directed to the inspection of light water reactor nuclear power plant components (ASME 1992). Volume 3 (ASME 1994) addresses the inservice inspection of components in fossil fuel-fired electric power generating stations and includes numerous examples from several applications.

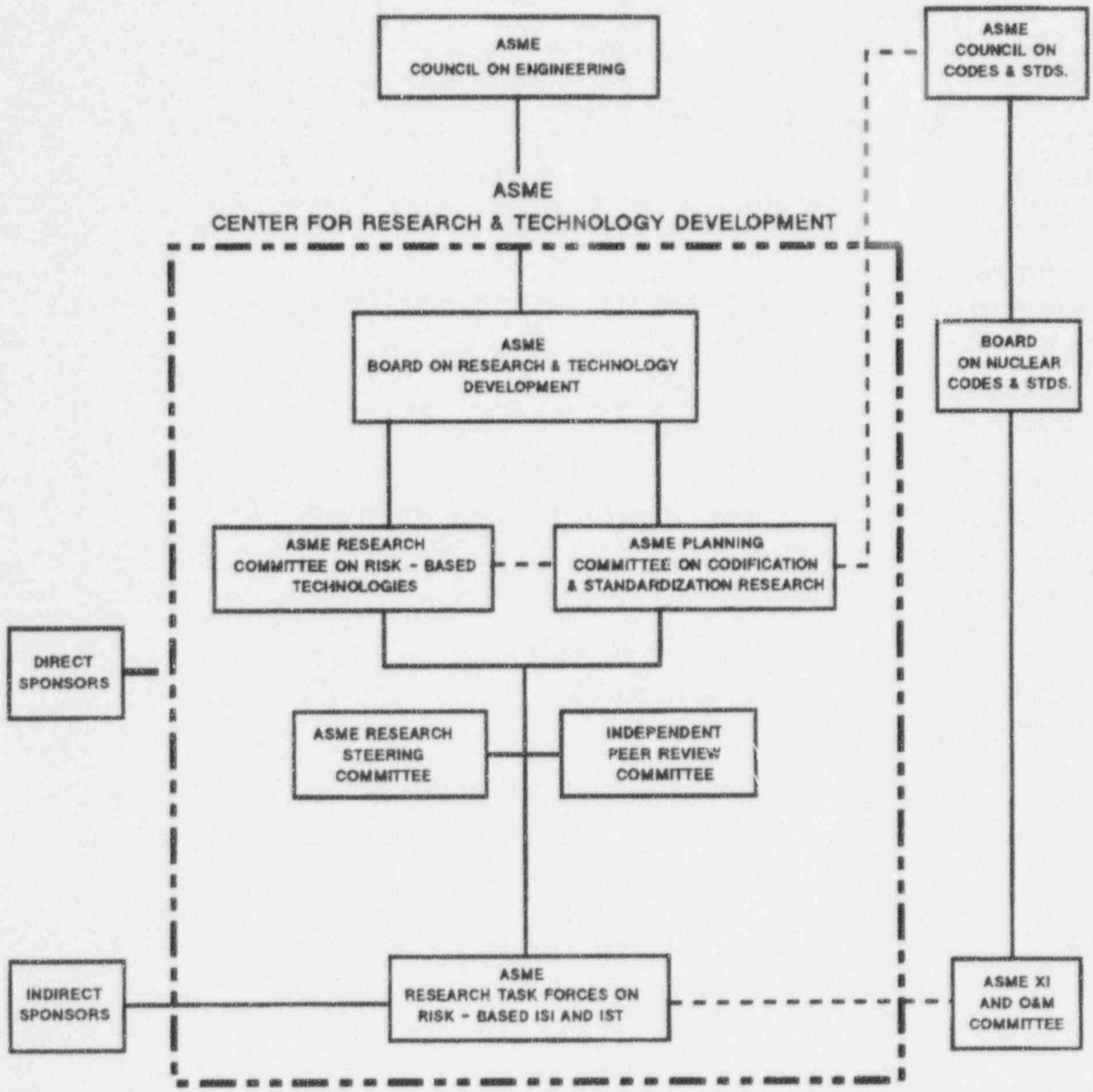

Figure 2.2-1. Organization of ASME Risk-Based Inservice Inspection and Testing Research Projects

The U.S. Nuclear Regulatory Commission, as part of the research effort, applied this technology in pilot studies of inspection requirements for both PWR and BWR plant systems (NUREG/CR-6151, Vo et. al., 1994). Virginia Power's Surry Unit 1 was studied in more detail regarding the effectiveness of ASME Section XI inspections versus a risk-based inspection approach (NUREG/CR-6181, Vo et. al., 1994). The Surry study evaluated selected nuclear steam supply and balance of plant piping systems and components.

At a review meeting in June 1994, U.S. NRC senior management requested the ASME Research Task Force to make the risk-based ISI process consistent with other PSA applications. Building on the Surry study results, the use of risk-importance measures and review by an expert panel were included to enhance the risk-based ISI process. This enhanced process, which is being applied in this process, is being incorporated into a Volume 2 - Part 2 ASME Research Document (ASME, to be published), along with other research developments to make comprehensive recommendations to ASME Section XI.

ASME Section XI has formed a Working Group on Implementation of Risk-Based Examination to begin making Code changes based on risk for inservice inspection of passive, pressure boundary components. The first efforts of this organization have been to develop a Code Case providing risk-based selection rules for Class 1, 2, and 3 piping. The above ASME research work has been used to support this Code development effort.

2.3 INDUSTRY ACTIVITIES

The nuclear industry recognizes that the current operational, regulatory, and economic environment in the United States presents a unique opportunity to apply probabilistic risk assessment (PRA) or probabilistic safety assessment (PSA) technology. Risk-based technology provides unique tools that can aid in focusing resources more effectively in areas of true safety significance. Industry experience indicates that utilities have the potential to enhance safety while lowering overall operation and maintenance (O&M) costs through the utilization of insights obtained from routine application of risk-based technology processes. To this end, the Nuclear Energy Institute (NEI) worked with EPRI to develop a PSA Applications Guide (EPRI 1995) to enhance and expand such processes.

EPRI PSA Applications Guide

As stated in the executive summary of the EPRI guide, "the purpose of the PSA Applications Guide is to provide utilities with guidance on the preparation, utilization, interpretation, and maintenance of plant-specific PSAs for regulatory and non-regulatory applications. The intent of this guide is to provide a framework, within which PSA methodologies can be used to address regulatory and non-regulatory issues associated with plant safety. This guide is general in nature and does not focus on any one application or application type. In this regard, the guide is intended to provide the overall framework within which utility and industry PSA applications can be developed and evaluated." The EPRI PSA Applications Guide suggested criteria for risk-significance is shown in Table 2.3-1.

Table 2.3-1

EPRI PSA APPLICATIONS GUIDE

GENERAL APPROACH TO OVERALL RISK SIGNIFICANCE DETERMINATION

Risk Importance Measure	Criteria
Risk Reduction Worth (RRW) • System Level • Component Level	 > 1.05 > 1.005
Fussell - Vesely Importance (F-V) • System Level • Component Level	 > 0.05 > 0.005
Risk Achievement Worth (RAW) (Component/Train Level)	> 2

NRC Initiatives

In response to industry initiatives, the NRC has issued "Final Policy Statement, on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities;" SECY-95-126 (NRC 1995). The NRC also issued "Proposed Agency-Wide Implementation Plan for Probabilistic Risk Assessment (PSA)," SECY-94-219 (NRC 1994). The overall objectives of the policy statement are to "improve the regulatory process through improved risk-effective safety decision-making, through more efficient use of agency resources, and through a reduction in unnecessary burden on licensees." In the implementation plan, the NRC identifies pilot applications of risk-based concepts to specific regulatory initiatives. The NRC specifically identifies inservice inspection and testing requirements in the list of initiatives.

Nuclear Energy Institute Risk-Based Inspection Task Force

In January 1995, the Nuclear Energy Institute (NEI) formed a Risk-based Inservice Inspection and Testing (ISI/IST) Task Force, comprised of industry representatives. The mission of the task force is to assist NEI in coordination of industry development of performance and risk-based methodologies and implementation of an industry regulatory plan for ISI/IST. The goals of the task force are to "support development of methodologies for ISI/IST that are amenable to performance and risk-based concepts and to support development and implementation of a regulatory plan for resolution of generic performance and risk-based ISI/IST." The risk-based ISI task force (split from the IST portion in summer 1995) is developing an industry guideline document that will be submitted for NRC endorsement. Meetings with the NRC are planned to be held to further the implementation of risk-based ISI.

2.3.1 Maintenance Rule

The Maintenance Rule and the supporting industry guideline document provide the first true application of risk-based technology in the regulatory process and provides an excellent foundation and starting point for the application of risk-based methods to select locations for piping inservice inspection.

As stated in NRC Regulatory Guide 1.160 (NRC 1993), the NRC published the maintenance rule on July 10, 1991, as Section 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (NRC 1991). The NRC's determination that a maintenance rule was necessary arose from the conclusion that proper maintenance is essential to plant safety.

10CFR50.65 requires that power reactor licensees monitor the performance or condition of structures, systems, and components (SSCs) against licensee-established goals in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. Such goals are to be established commensurate with safety and, where practical, take into account industry-wide operating experience. When the performance or condition of an SSC does not meet established goals, appropriate corrective action must be taken.

Performance and condition monitoring activities and associated goals and preventive maintenance activities must be evaluated at an interval associated with every refueling outage (but not to exceed 2 years), taking into account, where practical, industry-wide operating experience.

An industry document, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," NUMARC 93-01 (NUMARC 1993), was developed by the NUMARC[2] Maintenance Working Group, Ad Hoc Advisory Committees for the Implementation of the Maintenance Rule, and an Ad Hoc Advisory Committee for the Verification and Validation of the Industry Maintenance Guideline. The NUMARC 93-01 industry guide was endorsed by the NRC in Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (NRC, 1993).

As stated in NUMARC 93-01, "this industry guideline has been developed to assist the industry in implementing the final Maintenance Rule and to build on the significant progress, programs and facilities established to improve maintenance. The guideline provides a process for deciding which of the many SSCs that make up a commercial nuclear power plant are within the scope of the Maintenance Rule. It then describes the process of establishing plant-specific risk significant and performance criteria to be used to decide if goals need to be established for specific structures, systems, trains and components covered by the Maintenance Rule that do not meet their performance criteria.

As of July 10, 1996, all SSCs that are within the scope of the Maintenance Rule will have been placed in (a)(2) (of 10CFR50.65) and be part of the preventive maintenance program. To be placed in (a)(2), the SSC will have been determined to have acceptable performance. In addition, those SSCs with unacceptable performance will be placed in (a)(1) with goals established. This determination is made by considering the risk significance as well as the performance of the SSCs against plant-specific performance criteria. Specific performance criteria are established for those SSCs that are either risk significant or standby mode; the balance are monitored against the overall plant level performance criteria."

[2] The Nuclear Management and Resource Council (NUMARC) has since been integrated with other industry organizations to become the Nuclear Energy Institute (NEI).

In general, most utilities have completed identification of risk significant SSCs by exercising the PSA models that were initially used to meet the generic IPE requirements of NRC Generic Letter 88-20. The total contribution to core damage frequency (CDF) and large early release frequency (LERF) are used as a basis for establishing plant-specific risk significant criteria.

The NUMARC 93-01 suggested criteria for the determination of risk-significance is:

- Risk-Reduction Worth (RRW) of greater than 1.005
- Risk-Achievement Worth (RAW) of greater than 2
- Components included in cutsets that cumulatively account for about 90 percent of the core damage frequency.

Expert Panel. When the PSA is utilized for the Maintenance Rule application, a panel of individuals experienced with the plant PSA and with operations and maintenance (usually from the utility) is also used in the decision-making process. The panel utilizes their expertise and PSA insights to develop the final list of risk significant systems. NUREG/CR-5424, "Eliciting and Analyzing Expert Judgement," (Meyer and Booker 1990) or NUREG/CR-4692, "Methods for the Elicitation and Use of Expert Opinion in Risk Assessment," (Murphy and Cletcher 1987) are used as a guideline in structuring the panel. The panel's judgments usually consider the risk achievement worth and risk reduction worth risk importance calculational methods shown previously and further described in Sections 9.3.1.1 and 9.3.1.3 of NUMARC 93-01. Each method is useful in providing insights into selecting those SSCs that will be included in the maintenance rule and consideration is given to using both of them in the decision-making process.

The use of the expert panel process compensates for the limitations of PSA implementation approaches resulting from the PSA structure and limitations in the meaning of the importance measures. The expert panel process that is used for the maintenance rule should also be used for the risk-based ISI application. However, additional experts should be considered, particularly those cognizant of current ISI requirements, component failure data, and results of any previously performed inservice inspections or maintenance.

SECTION 3

APPLICATION OF RISK-BASED METHODS TO ISI

This section describes the process and how it was applied to a representative WOG plant.

3.1 OVERVIEW

The overall recommended risk-based inservice inspection process (as defined by ASME Research) includes four major parts as shown in Figure 3.1-1. The four major parts of the process shown in Figure 3.1-1 include:

- **Scope Definition** - Definition of system boundaries and success criteria using a plant PSA that was initially developed to meet Individual Plant Examination (IPE) requirements by the U.S. Nuclear Regulatory Commission (NRC 1988);

- **Risk Ranking** - ranking of components into more safety-significant and less safety-significant categories, by applying risk importance measures and deterministic insights with the plant expert panel making the final selection of where to focus ISI resources;

- **ISI Program Development** - determination of an effective ISI program that defines when and how to appropriately inspect the two categories of components; and

- **Perform ISI** - performance of the ISI program to verify component reliability and then updating the risk rankings and/or inspection methods based on the inspection results.

The above process can also be applied to inservice testing as shown on the figure. This report focuses primarily on the first two parts that involve the identification of where to inspect. Guidance is also provided on the inspection methods.

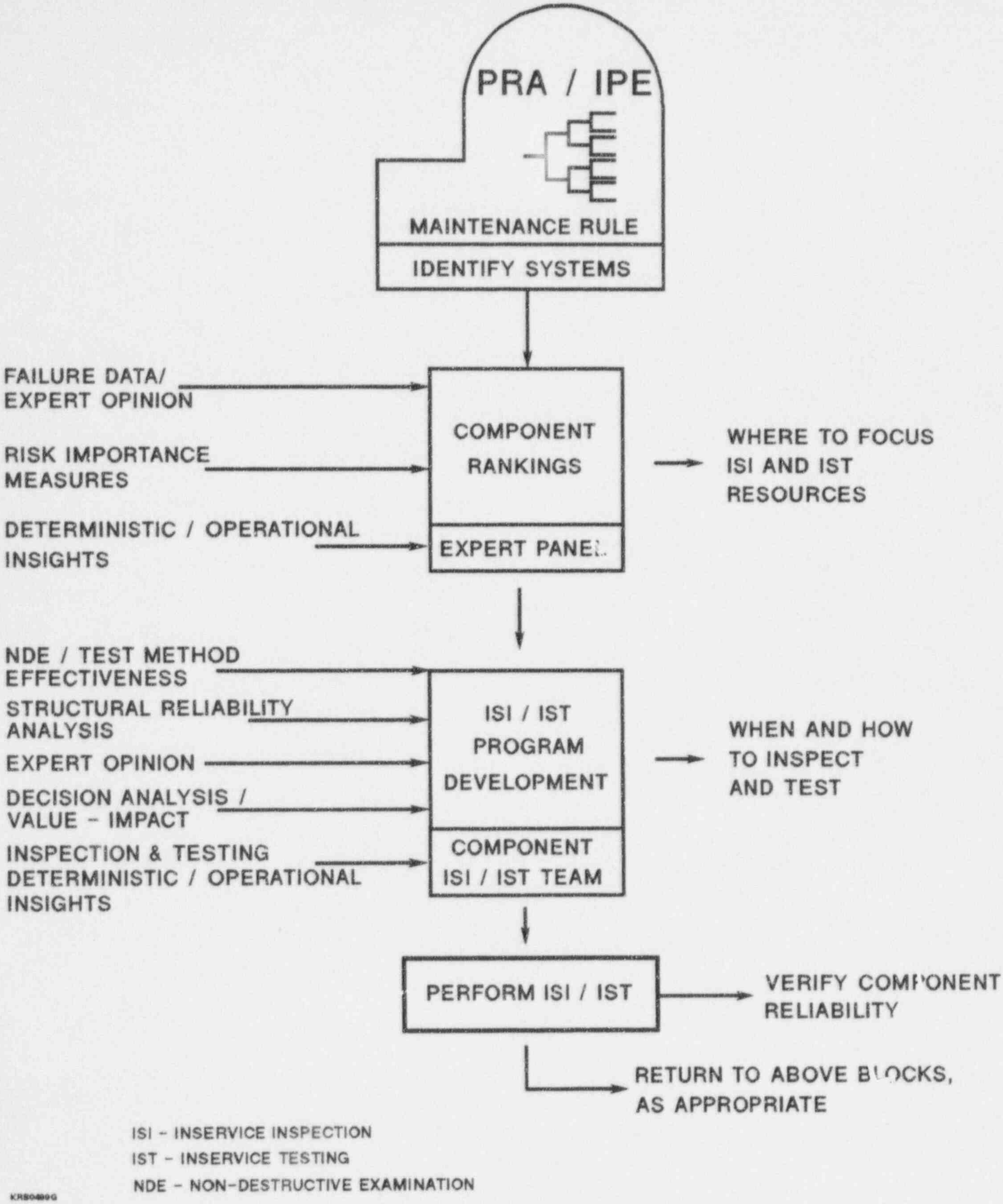

Figure 3.1-1. General ASME Research Risk-Based Inservice Inspection and Testing Process

3.1.1 Risk-Ranking Process

The overall risk-ranking process is shown in Figure 3.1-2. The process involves the following steps:

- **Scope Definition** - The fluid systems contained in the plant, modeled in the PSA and considered as risk-significant for the Maintenance Rule, are identified and compared with the current classifications and required ISI examinations, and with the existing stress analyses (if available). This review, along with other plant documentation, is used to determine which systems, or portions of systems, should be evaluated as part of the risk-based ISI process. Given that system boundaries involve system functions and may also involve interfaces between different types of 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.

- **Segment Definition** - This task involves the development of piping segments for the risk-ranking. A piping segment is defined as a portion of piping for which a failure at any point in the segment results in the same consequence (e.g., loss of a system, loss of a pump train, etc.) and includes piping structural elements between major discontinuities such as pumps and valves.

- **Consequence Evaluation** - The consequences given the failure of a piping segment are identified through PSA insights, engineering evaluations and plant design and operations review. Consequences that must be considered include both direct effects (failure of a train in which the piping segment is contained) and indirect effects (such as those due to flooding, pipe whip, or jet impingement).

- **Failure Probability Assessment** - The task of estimating component failure probabilities for each piping segment can be challenging. In most cases though, consideration of failure probabilities, however uncertain their estimated values, leads to more effective allocation of inspection resources compared to present practices. Although absolute values of failure probabilities may have large uncertainties, the relative values (e.g., from location to location in a given piping system) are generally better known. Structural

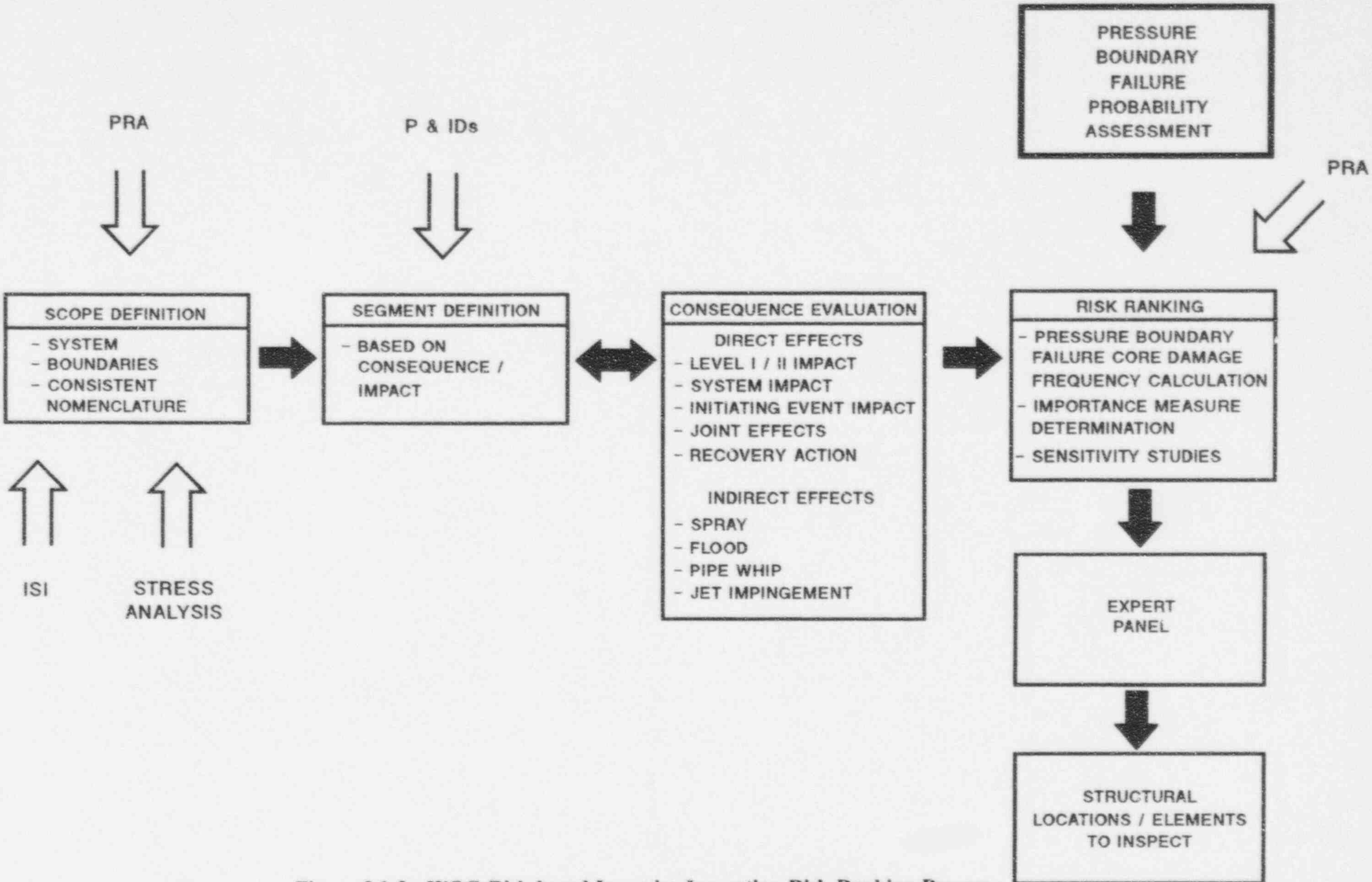

Figure 3.1-2. WOG Risk-based Inservice Inspection Risk-Ranking Process (Expansion of the First Two Parts of the General ASME Research Process in Figure 3.1-1)

reliability/risk assessment (SRRA) models, based on probabilistic structural mechanics methods, are used to estimate failure probabilities for important components.

- **Risk Ranking** - This task is to identify and prioritize the important components (or pipe segments). The approach calculates the relative importance for each component within the systems of interest. This risk-importance is based on the core damage frequency (and large early release frequency, if available) resulting from the structural failure of the component in a given segment and the total pressure boundary core damage frequency. The results are then used to calculate the risk-importance for each segment within the system.

- **Expert Panel Review** - An expert panel (such as the expert panel used for the Maintenance Rule supplemented by appropriate ISI-related disciplines) evaluates the risk-based results and makes a final review to determine the more safety-significant pipe segments for ISI. The identification of potential inspection locations within a more safety-significant pipe segment is obtained by a further review of the structural elements and postulated failure mechanisms.

The output of this process defines the structural elements selected for inspection. The method and frequency of the inspection is then determined by a focused ISI team comprised of materials, ISI and NDE expertise. The selections are then reviewed and approved by the expert panel.

3.1.2 Use of the EPRI PSA Applications Guide

In order to apply this risk-based ranking process, the EPRI PSA Applications Guide, as discussed in Section 2, provides the framework and guidance on using PSA for applications. The application of risk-based ISI is built from PSA modeling efforts used initially to meet NRC IPE requirements and within the framework outlined in the EPRI PSA Applications Guide. The guide discusses some specific questions and considerations, which are addressed prior to a specific PSA application, in three areas: Application Planning, Analysis, and Results Interpretation. General guidance is also presented for PSA Maintenance and Updates that is necessary for risk-based ISI applications. With respect to the guide, the following phases are considered for risk-based ISI.

Application Planning

The first phase involves problem definition, scope assessment, and identification of the figures of merit to be used in the quantitative analysis. Each element is described below in relation to risk-based ISI.

- **Problem Definition** - PSA can be used in the ranking or prioritization of components (including pipe segments) and structural elements for ISI to identify where resources should be focused in order to justify a change to nuclear utility ISI plans. For ISI ranking, the PSA is to be supplemented with further specific failure data (using experience data sources, expert judgement, and/or structural reliability methods) to represent pressure boundary failures which are not usually modeled in detail in current PSA models. The change in ISI scope based on the PSA will be evaluated on a relative basis to assess the degree of risk significance (importance) of components and structural elements independent of any changes to the plant. However, the scope is to be updated on a regular basis to reflect the results of inservice inspections, plant design or operation changes, PSA model and data updates, and new industry findings, as appropriate. The ISI ranking results are to also be reviewed by a utility expert panel (peer review group) to include deterministic insights and to make the final prioritization.

- **Scope Assessment** - Since many U.S. plants do not have full scope PSAs (full Level 3 PSA), ISI ranking should initially focus on systems, components, and structural elements involved in PSA internal event scenarios. External event scenarios (fires, earthquakes, etc.) and shutdown considerations should be addressed by the expert panel if external events and shutdown PSAs are unavailable. For pressure boundary components that protect containment integrity, Level 2 PSA insights are to be addressed in the ISI ranking of these components by explicit PSA modeling, if available, and the expert panel.

- **Figures of Merit** - Core damage frequency (CDF) due to pressure boundary failures is the preferred Level 1 PSA figure of merit. Large, early release frequency (LERF) due to pressure boundary failures is the preferred Level 2 PSA figure of merit for pressure boundary components that are needed to protect containment integrity. For risk-based ISI prioritization or ranking, two measures of risk importance have been found to be quite useful in characterizing risk properties in aiding decision-making. The two measures are

termed "Risk Achievement Worth" (RAW) and "Risk Reduction Worth" (RRW). The risk achievement worth of a feature (system, component, or structural element) is a measure of how the figure of merit (CDF or LERF) could increase if the feature were guaranteed to fail at all times. The risk reduction worth is a measure of how much the figure of merit could decrease if the feature were guaranteed to succeed at all times.

Fussell-Vesely (F-V) Importance may be used in lieu of RRW because of the mathematical relationship between the measures. The following relationship allows translation of F-V results to RRW:

$$RRW = \frac{1}{[1-(F-V)]}$$

Technical Analysis

The technical analysis phase contains three key aspects: assessment of the adequacy of the PSA, establishing the cause and effect relationship associated with the change being evaluated, and defining the overall technical approach.

- **Adequacy of PSA Model** - Section 3.1 of the EPRI PSA Guide outlines a number of guiding principles for a PSA application to be successful. The PSA model should accurately reflect the current plant configuration and operational practices. For ISI ranking, the PSA model or its results can be modified to represent pressure boundary integrity failures, as previously noted. In addition, care must be taken in defining and insuring agreement of system boundaries and definitions between the PSA and those currently used in ISI plans.

  The PSA model will need to assess and possibly incorporate plant design and operation changes, results of inservice inspections and new industry findings, as appropriate. Given that ISI plans are currently implemented over a 10-year interval, in general, this should be considered as the longest response time for the PSA model for this application. Shorter response times may be necessary for systems, components, and structural elements that have the potential to be subjected to aggressive degradation mechanisms as identified in the risk-ranking process for ISI.

The assumptions and limitations of the base PSA can have a strong impact on the overall risk-ranking results. The EPRI PSA Guide addresses this issue by providing a checklist for technical adequacy in Appendix B of the guide which identifies the key Level 1 internal events PSA elements that have been found in past PSA applications to have the most significant potential for influencing results.

- **Establishing a Cause-Effect Relationship** - A key step in performing the risk-ranking of components and structural elements involves the identification of the portions of the PSA affected by this ISI application. General guidance for determining elements of a PSA that may need to be modified for successful application are provided as a list of questions, which are grouped by PSA model element in Tables 3-1 and 3-2 of the EPRI guide. Key general considerations regarding this cause-effect relationship for risk-based ISI are discussed below for Level-1, internal events PSAs (only for portions of the PSA that are affected). However, these questions should be revisited by each user because of variabilities in PSA models across the industry and if Level-2 or External Events PSAs are going to be exercised for this application.

  Initiating Events - The risk-based ISI application requires the consideration of component pressure boundary failures as initiating events throughout key plant systems. While some industry failure rates have been established for pressure boundary failures, focused effort is required to obtain failure probabilities at the component (or pipe segment) level using existing failure data, expert opinion, and/or structural reliability modeling.

  System Reliability Models - This application may require the introduction of new branches to represent components and pipe segments that have not been explicitly modeled in the PSA. However, because pressure boundary failures are low probability events, these failures may be simulated by including the pressure boundary failure probability with the failure probability of an already-modeled component that would result in the same impact on the system operation (i.e., same consequence), or using a surrogate component.

  Parameter Data Base - Failure probabilities for pressure boundary failures must be obtained. This can be done via several methods including industry databases, expert opinion/elicitation or structural reliability methods.

Human Reliability Analysis - Recovery actions may be necessary in order to isolate a pressure boundary failure in order to mitigate or reduce the consequences. These actions are treated on a case-by-case basis and an estimated failure probability is based on discussions with the plant staff and calculated via human reliability techniques.

Quantification - This application requires the calculation of the core damage frequency and large early release frequency (if available) due to pressure boundary failures and the calculation of importance measures based on these frequencies.

Analysis of Results - This application uses an importance analysis to rank segments based on pressure boundary failures and their consequences. Quantitative sensitivity studies should be included in the evaluation.

- <u>**Technical Approach**</u> - The risk-based ISI application requires manipulation of the PSA model to evaluate the figure of merit. The PSA model and/or results are used as input to a model to determine the core damage frequency due to pressure boundary integrity failures. These failures and their related consequences should be evaluated in a realistic manner to obtain useful risk-ranking values for purposes of ISI.

  A blended approach is used for the risk-based ISI application. Reviews of operational experience, engineering judgement, and/or structural reliability engineering analyses are used to obtain pressure boundary integrity failure probabilities for use with the evaluations using the PSA model. An expert panel is also utilized to review the PSA risk-rankings and to make the final prioritization groups for ISI.

Results Interpretation

The reporting and interpretation of PSA application results can be divided into three distinct elements: qualitative assessment of results, quantitative assessment of results, and reporting requirements.

- <u>**Quantitative Criteria**</u> - The baseline piping pressure boundary failure PSA results can be used to assess the degree of risk significance of components for purposes of inservice inspection. The criteria in Table 4-2 of the EPRI Guide have been found to provide useful

results on a component level basis. However, one key modification should be made for the risk-based ISI piping application as follows.

The total CDF (and LERF) in the above risk significance evaluation should only account for those associated with piping pressure boundary failures. If the total CDF for all plant internal events is used, none of the pressure boundary components will be more safety-significant. Thus, the PSA results will be useless in helping to determine where to focus priorities for ISI. Modeling the piping pressure boundary failures and then assessing the relative risk significance to a total CDF related to just pressure boundary failures renders more meaningful results. In other words, the PSA model has been used to assist in defining ISI programs that will ensure that piping pressure boundary failures do not become major contributors to total plant risk as a result of age degradation mechanisms. Risk-based ISI programs will help to keep the assumptions that piping pressure boundary failures are low probability events valid in the PSA.

Table 3.1-1 summarizes criteria for risk significance determination for ISI. The table includes appropriate criteria from Table 4-2 of the EPRI Guide and the above modifications. Section 4.2.6 of the EPRI guide provides further discussion on the risk importance measures for prioritization and ranking. This section also discusses the combined ranking or prioritization of results when more than one figure of merit is used.

- **Qualitative Assessment** - Section 4.3 of the EPRI Guide provides a general discussion on the qualitative review of the results. Sensitivity studies are used to evaluate the impacts.

- **Reporting** - Section 4.4 of the EPRI Guide outlines some minimum general practices for documentation of a PSA application. Section XI of the ASME Boiler and Pressure Vessel Code also defines requirements for records and reports that will apply in the documentation of the process used to select ISI locations.

Table 3.1-1 APPROACH TO OVERALL RISK SIGNIFICANCE DETERMINATION FOR ALTERNATIVE RISK-BASED SELECTION PROCESS FOR INSERVICE INSPECTION[a]	
RISK IMPORTANCE MEASURE	CRITERIA[b] Pipe Segment Level
Risk Reduction Worth (RRW)	>1.005
Fussell-Vesely Importance (FV)	>0.005

(a) Adapted from EPRI PSA Applications Guide (EPRI 1995)[c]

(b) These criteria apply to the use of a total CDF_{PIPING}, which is the total core damage frequency attributed to pressure boundary failure in plant piping systems.

(c) Piping failure probabilities are typically very small compared to other component failures modeled in the PSA. When the failure probability is set to 1.0 for the RAW calculation, large RAW values typically result. Therefore, the EPRI guideline classifying a segment as more safety-significant for RAW values greater than 2 does not provide meaningful results. Instead, the safety-significance determination focused on the RRW values, and RAW values were used on a relative basis to help differentiate segments which had similar RRW values.

3.1.3 Representative WOG Plant

In order to apply the risk-based ISI process, a plant was identified to apply this process. Northeast Utilities volunteered the Millstone Unit 3 plant to be the representative WOG plant for this application.

The Millstone Nuclear Power Station Unit 3 (MP3) is located on a site in the town of Waterford, New London County, Connecticut, on the north shore of the Long Island Sound. The plant was designed and constructed by Stone and Webster and features a pressurized water reactor (PWR) by Westinghouse Electric Corporation and a turbine generator furnished by General Electric. It incorporates a 4-loop closed-cycle type nuclear steam supply system (NSSS). The reactor is operated inside a reinforced concrete containment structure maintained at a subatmospheric pressure between 10.6 and 14.0 psia. The reactor core is designed for a warranted power output of 3,411 MWt, which is the license application rating. This output, combined with the reactor coolant pump heat output of 14 MWt, gives the NSSS warranted output of 3,425 MWt. The gross calculated electrical output of approximately 1,153 MWe.

The Millstone Unit 3 current Inservice Inspection (ISI) plan for the first 10-year interval consists of ASME Class 1, 2, and 3 systems and components (and their supports) and was developed and has been updated during the interval by giving due consideration to the following documents:

- 10CFR50.55a - Title 10: Code of Federal Regulations Part 50 Revised as of January 1, 1995

- Section XI of the ASME Code, 1983 Edition through the Summer 1983 Addenda - Rules for Inservice Inspection of Nuclear Power Plant Components

- Section XI of the ASME Code, 1983 Edition through the Winter 1985 Addenda - Rules for Inservice Inspection of Nuclear Power Plant Components

- Section III of the ASME Code - Rules For Construction of Nuclear Power Plant Components

- Section V of the ASME Code - Nondestructive Examination

- USNRC Standard Review Plan (SRP 6.6, Section II-7)

- USNRC Regulatory Guides:

  Regulatory Guide 1.14, Rev. 1, August 1975 - Reactor Coolant Pump Flywheel Integrity

  Regulatory Guide 1.26, Rev. 3, February 1976 - Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants

  Regulatory Guide 1.65, Rev. 0, October 1973 - Materials and Inspections for Reactor Vessel Closure Studs

  Regulatory Guide 1.83, Rev. 1 July 1975 - Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes

Regulatory Guide 1.147, Rev. 11, October 1994 - Inservice Inspection Code Case Acceptability - ASME Section XI Division 1

Regulatory Guide 1.150, Rev. 1, February 1983 - Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examination

- Millstone Unit 3 FSAR

- Millstone Unit 3 PSI Inspection Plan (PSI-2.01)

- Millstone Unit 3 Technical Specifications

At present the unit is in the process of updating and developing the ISI plan for its second 10-year interval to the ASME Code Section XI, 1989 Edition. The ASME Code Section XI, Editions and Addenda used for piping requirements in the first 10-year interval are essentially the same as those now being referenced for the second 10-year interval. The following paragraphs help explain the relationships in these requirements as they have been used in the past and are currently being updated. Except for minor plant changes and some clarifications in the requirements provided by current Code interpretations the ISI plans for the first and second 10-year intervals as they relate to piping examinations will be the same.

During the first 10-year interval, Class 1 examination requirements for piping were taken from the ASME Code Section XI, 1983 Edition up to and including the Summer 1983 Addenda.

These Class 1 requirements are described under Table IWB-2500-1, Examination Category B-F, Pressure Retaining Dissimilar Metal Welds and Examination Category B-J, Pressure Retaining Welds in Piping. Other than some minor editorial changes in these tables all the requirements are identical to the ASME Code 1989 Edition and the requirements state that the welds initially selected during the first 10-year interval will be reexamined during the next 10-year interval and are being scheduled accordingly.

Class 2 examination requirements for piping used during the beginning of the first 10-year interval were originally taken from the alternative rules provided in Code Case N-408, Alternative Rules for

Examination of Class 2 Piping Section XI, Division 1 and later updated to the ASME Code Section XI, 1983 Edition with the Winter 1985 Addenda. Under this Code Case and the Winter 1985 Addenda requirements both the Examination Category C-F-1, Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping and the Examination Category C-F-2, Pressure Retaining Welds in Carbon or Low Alloy Steel Piping were applied identically. As with the Class 1 examination requirements only minor editorial changes have been made in the ASME Code Section XI, 1989 Edition requirements for these welds and the same requirement exists to select welds for examination during the second 10-year interval that were selected for examination during the first 10-year interval.

In August of 1983, a Level 3 Probabilistic Safety Study for Millstone Unit 3 was completed by a combined effort of Westinghouse and Northeast Utilities. This study also included an examination of external events. Six substantial updates were performed before the Individual Plant Examination (IPE) was submitted in 1990. Millstone Unit 3 received the SER on the IPE in May of 1992 (NRC 1992). The IPE submittal also included a section addressing External Events; however, MP3 is still awaiting an IPEEE SER. Since the IPE submittal, a major update of the Level 1 PSA was completed in 1995 to incorporate plant history, design changes, NRC IPE recommendations and change in methodology from support states to large fault trees. This updated PSA model was used as a basis for this project.

The base plant PSA core damage frequency due to internal events is 5.87E-05/yr. Table 3.1-2 provides the core damage contribution of each internal events initiator for Millstone Unit 3. The dominant accident sequences include a total loss of Service Water with failure to recover leading to a consequential small LOCA, and small LOCA with failure of recirculation. The core damage frequency due to internal flooding is 8.5E-07/yr. The core damage contribution due to external events is dominated by seismic and fire events. The CDF due to seismic events is 9.08E-06/yr, and is 4.85E-06/yr due to fire.

Table 3.2-1

MILLSTONE UNIT 3 RBI SYSTEM IDENTIFICATION

System ID	System	Basis
BDG	Steam Generator Blowdown	High Energy Line Break Concerns
CCE	Charging Pump Cooling	PSA (1)
CCI	Safety Injection Pump Cooling	PSA (1)
CCP	Reactor Plant Component Cooling	PSA & ASME Section XI
CHS	Chemical & Volume Control	PSA & ASME Section XI
CNM	Condensate	PSA (2)
DTM	Turbine Plant Misc. Drains	ASME Section XI (3)
ECCS	Emergency Core Cooling (4)	PSA & ASME Section XI
EGF	Emergency Diesel Fuel	PSA
FWA	Auxiliary Feedwater	PSA & ASME Section XI
FWS	Feedwater	PSA (2) & ASME Section XI
HVK	Control Bldg. Chilled Water	PSA
MSS	Main Steam	PSA & ASME Section XI
QSS	Quench Spray	PSA & ASME Section XI
RCS	Reactor Coolant	PSA & ASME Section XI
RHS	Residual Heat Removal	PSA & ASME Section XI
RSS	Containment Recirculation	PSA & ASME Section XI
SFC	Fuel Pool Cooling and Purification	PSA (5)
SIH	High Pressure Safety Injection	PSA & ASME Section XI
SIL	Low Pressure Safety Injection	PSA & ASME Section XI
SWP	Service Water System	PSA & ASME Section XI

Notes:
(1) Included in PSA boundary, but exempt by ASME Section XI pipe size.
(2) Modeled indirectly in the PSA
(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 - Charg
Quench Spray
(5) Not included in PSA internal events model, important to shutdown risk

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3.2 SCOPE DEFINITION

The first step in the program is to define the systems to be evaluated in the scope of the program. Currently, the scope of this program is limited to nuclear plant piping. The piping boundaries of the plant PSA and the current ASME Section XI inservice inspection program Class 1, 2, and 3 examination boundaries are reviewed for possible inclusion in the scope of the program. The piping boundaries that are included in the plant PSA, but are outside the current ASME Section XI boundary, may also be included. The systems identified under the Maintenance Rule are also used to identify piping systems for inclusion in the scope of the program.

In addition to defining the systems to be included in the scope of the program, the piping structural elements to be included in the program are identified.

For the Millstone 3 application, the scope included ASME Class 1, 2 and 3 piping systems and various balance of plant (non-nuclear Code Class) systems. Table 3.2-1 provides a list of the systems included in the scope of the program. This list was reviewed by the expert panel (Maintenance Rule panel supplemented by appropriate ISI-related disciplines) to determine its completeness for this application. The systems had been selected based on three criteria:

- All Class 1, 2, and 3 systems currently within the ASME Section XI program;
- Piping systems modeled in the PSA; and
- Various balance of plant fluid systems determined to be of importance (mainly based on Maintenance Rule ranking).

Twenty-one systems had been selected to be evaluated in more detail. The basis for exclusion of the other plant piping systems was provided and reviewed by the expert panel for their concurrence. A sample is provided in Table 3.2-2.

Table 3.1-2 BASE PLANT PSA CORE DAMAGE FREQUENCY PERCENT CONTRIBUTION BY INITIATOR	
Initiating Event	**Percent Contribution to Overall Base Plant PSACDF**
Large Loss of Coolant Accident (LOCA)	3.3
Medium LOCA	7.0
Small LOCA	3.8
Steam Generator Tube Rupture (SGTR)	2.0
Incore Instrument Tube Rupture	2.6
Steamline Break Inside Containment	2.8
Steamline Break Outside Containment	2.8
General Plant Transient	9.5
Loss of Main Feedwater (MFW)	1.3
Loss of Offsite Power (LOSP)	2.3
Station Blackout	1.5
Loss of 1 Service Water Train	3.1
Total Loss of Service Water	10.4
Loss of 1 DC Bus A (B)	<.01
Total Loss of DC	<.01
Loss of Vital AC 1 or 2	.06
Loss of Vital AC 3 or 4	<.01
Anticipated Transient Without Scram (ATWS)	8.7
Consequential Small LOCA	23.6
Consequential Steamline Break Inside Containment	<.01
Consequential Steamline Break Outside Containment	<.01
Interfacing Systems LOCA (ISLOCA)	3.9

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HVK	Control Bldg. Chilled Water	PSA
MSS	Main Steam	PSA & ASME Section XI
QSS	Quench Spray	PSA & ASME Section XI
RCS	Reactor Coolant	PSA & ASME Section XI
RHS	Residual Heat Removal	PSA & ASME Section XI
RSS	Containment Recirculation	PSA & ASME Section XI
SFC	Fuel Pool Cooling and Purification	PSA (5)
SIH	High Pressure Safety Injection	PSA & ASME Section XI
SIL	Low Pressure Safety Injection	PSA & ASME Section XI
SWP	Service Water System	PSA & ASME Section XI

Notes:
(1) Included in PSA boundary, but exempt by ASME Section XI pipe size.
(2) Modeled indirectly in the PSA
(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 PSA internal events model, important to shutdown risk

Table 3.2-2 EVALUATION OF PIPING SYSTEMS FOR EXCLUSION FROM THE RBI PROGRAM		
System ID	**System Description**	**Resolution**
DSM	Moisture Separator Drains & Vents	Determined to be non-risk significant as part of the Maintenance Rule*
DSR	Main Steam Separator Reheater Drains and Vents	Determined to be non-risk significant as part of the Maintenance Rule*
EGD	Emergency Diesel Fuel Exhaust & Comb. Air	Determined to be non-risk significant as part of the Maintenance Rule
EGS	Emergency Diesel Jacket Water	Included with the Diesel Generator boundary in Maint. Rule; however, Expert Panel determined that the DG would function without this system but with slower start time. No need to evaluate.
ESS	Extraction Steam	Determined to be non-risk significant as part of the Maintenance Rule*
GMC	Stator Cooling Water	Determined to be non-risk significant as part of the Maintenance Rule
GMH	Generator Hydrogen & CO2	Determined to be non-risk significant as part of the Maintenance Rule
GMO	Generator Seal Oil	Determined to be non-risk significant as part of the Maintenance Rule
HDH	H.P. Feedwater Heater Drains	Determined to be non-risk significant as part of the Maintenance Rule*
HDL	L.P. Feedwater Heater Drains	Determined to be non-risk significant as part of the Maintenance Rule*
IAC	Containment Instrument Air	Determined to be non-risk significant as part of the Maintenance Rule
IAS	Instrument Air	Determined to be non-risk significant as part of the Maintenance Rule
SWT	Traveling Screen Wash & Disposal	Determined to be non-risk significant as part of the Maintenance Rule
TMB	Turbine Control System	Determined to be non-risk significant as part of the Maintenance Rule
CCS	Turbine Plant Component Cooling	Determined to be non-risk significant as part of the Maintenance Rule*

* 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.

The structural elements considered for the Millstone 3 application included the examination items presently included under Examination Categories B-F, B-J, C-F-1, C-F-2 and D-A only as it relates to Class 3 piping systems that would be included under this category in the 1992 and later editions of ASME Section XI. The process also included evaluation of additional areas and volumes of base material and examination zones such as weld counterbore areas and fitting material with consideration to all piping welds to nozzles, valves and fittings such as tees, elbows, branch connections and safe ends. Welded attachments and piping supports were not included in the program. However, possible snubber degradation was given consideration as a factor which may increase piping fatigue effects.

3.3 SEGMENT DEFINITION

In order to evaluate the importance of the piping contained in each system, piping segments were defined. Piping segments can be defined on many levels: piping between welds; train level piping, etc. The approach used to define piping segments was based on:

- Piping which have same consequence as determined from the plant PSA and other considerations (e.g., loss of train A of residual heat removal (RHR), loss of refueling water storage tank (RWST), inside or outside containment consequences, etc.);

- Where flow splits or joins (traditional PSA modeling points);

- Includes piping to a point in which a pipe break could be isolated (e.g., check valve, motor- or air-operated valve, but no credit for manual valves); and

- Pipe size changes.

Thus, a piping segment is primarily defined as a portion of piping for which a failure at any point in the segment results in the same consequence. Distinct segment boundaries are identified at such branching points or size changes where there could be a significant difference in consequence, or the break probability is expected to be markedly different due to material properties. The consequences that should be considered are defined in the next section. The segment definition process is an iterative process with the determination of the consequences and identification of any potential operator recovery actions.

An example of a system and its defined piping segments is shown in Figure 3.3-1. In this example, the ECCS segments are defined. ECCS segment 1 is defined as piping between check valve 8847A (from the RHR pumps), check valve 8819A (from the HPSI pumps) and check valve 8818A (which isolates this pathway from the accumulator pathway). This piping segment is postulated to result in a loss of RWST inside containment resulting in an earlier transfer to recirculation and loss of high and low pressure safety injection to one cold leg. Similarly, segments 2, 3, and 4 are defined for the other injection points into the RCS cold legs. ECCS piping segment 5 (6, 7, and 8) is defined as piping between check valves 8818A, 8956A, and 8948A. These piping segments are postulated to result in loss of RWST inside containment resulting in an earlier transfer to recirculation and a loss of high and low pressure injection and accumulator injection to one cold leg. The ten-inch RCS piping downstream of check valve 8948A and the main reactor coolant loop piping defines other segments that are postulated to result in a large loss of coolant accident (LOCA).

For the representative WOG plant, the total number of segments defined and the systems are shown in Table 3.3-1.

3.4 CONSEQUENCE EVALUATION

The consequence from a pressure boundary failure should focus on safety consequences. Nevertheless, economic consequence can also be a secondary consideration resulting in additional inspection locations chosen to reduce economic risk. A risk-based evaluation may go beyond the ASME BPVC and regulatory requirements for inspections and thereby also improve plant reliability and availability factors.

In many risk-based applications, safety consequence has been measured in terms of core damage and large early release. These measures should also be applied for risk-based inspection. The impact on core damage due to pressure boundary failures can be both direct and indirect. A direct consequence would be the loss of a system, whereby the ruptured pipe can no longer provide fluid flow that is essential to the safe shutdown of the plant. An example of an indirect consequence may be a disabling of a critical electrical component by flooding associated with a ruptured pipe.

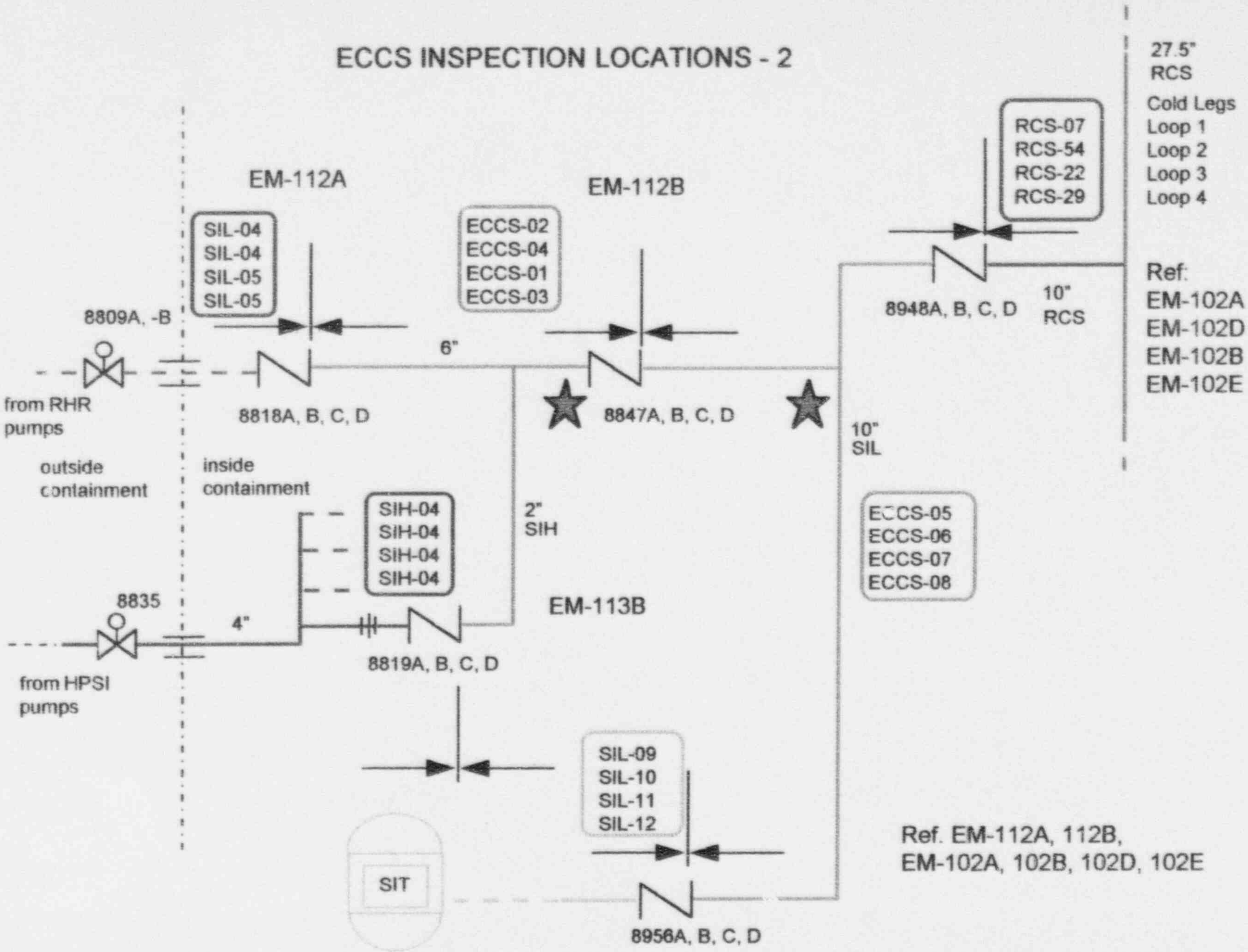

Figure 3.3-1. Example of Piping Segments

Note: The * indicates the selected location for the failure probability estimation described in Section 3.5.

Table 3.3-1

NUMBER OF SEGMENTS DEFINED FOR REPRESENTATIVE WOG PLANT

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.

3.4.1 Direct Consequences

PSAs can be used to gain insights into the consequences of pressure boundary failures. 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

The initial focus for the consequences should be related to the events considered in the PSA internal events scenarios. Table 3.4-1 provides an example of the direct consequences postulated for several piping segments, considering possible operator actions and their impact on the consequences.

3.4.2 Indirect Consequences

PSAs can be applied to establish indirect or spatial consequences, once based on detailed knowledge of the plant those systems and components (if any) affected by the pressure boundary failure are identified. Indirect effects evaluations include consideration of pipe whip, jet impingement and flooding. The information sources that are considered to identify indirect effects include the plant hazard evaluation to meet the requirements of the NRC's Standard Review Plan, the final safety analysis report (FSAR) and the PSA internal flooding events analysis. In addition, the expert panel may provide input on indirect consequences. The impact of the indirect effects to be considered should be the same as stated above for the direct effects. A plant walkdown of key areas should also be conducted.

The process used for conducting the walkdown at Millstone 3 is described below:

Table 3.4-1 EXAMPLE CONSEQUENCES FOR PIPING SEGMENTS			
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 RWST	Loss of RWST
ECCS-1*	From CV8819C and CV 8818C 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 which could be taken would result in closure of MV8835 (no HPSI to any paths) and closure of MV8809A or B (loss of 2 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 expert panel 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.

Pre-Walkdown

Existing documents which examine the local effects of pipe breaks for the systems were reviewed. Other systems/trains affected by a break in the area were identified. The plant layout drawings, for areas not covered by the documentation review, were also examined. In addition, plant areas for which documentation was not clear, specific equipment was not listed, or modifications should have been made were identified. The results of the evaluation were documented, reviewed, and used as an aid in organizing the walkdown. Table A-1 in Appendix A contains the results of the pre-walkdown review process.

Walkdown

Participants in the walkdown included team members from the PSA, piping, ISI, and operations groups. The walkdown covered the specific areas listed in Appendix A, Table A-1, in the ESF Building and the Auxiliary Building. The walkdown also included the intake structure for the circulating and service water pumphouse and the Turbine Building. An example of a walkdown worksheet documenting the information gathered is presented in Table 3.4-2.

Post-Walkdown

The walkdown results were documented for use in the risk-based ISI program. The more significant findings of the walkdown were:

- Interactions of postulated AFW pipe breaks in the motor-driven auxiliary feedwater pump rooms can affect cable trays,
- Reactor plant component cooling water pipe breaks can affect one train of AFW,
- Pipe shrouds had been installed (as prescribed by the hazards evaluation) to mitigate the interactions of a postulated pipe break in one train of reactor plant component cooling water disabling the pump in the other train,

Table 3.4-2 MILLSTONE 3 RISK-BASED INSPECTION EXPERT PANEL EVALUATION 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 rupture 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. Hazards 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 unisolatable break
2. Near pump
3. Located at postulated break location
4. Located at far end of room away pump and postulated break

- Motor control centers in the service water pump cubicles could be affected by a service water pipe break, and

- Postulated pipe breaks in the turbine building would lead to a reactor trip, notably the turbine plant component cooling water system. Also, a break in the condensate pump discharge header could potentially disable all three plant air compressors.

A summary of the indirect effects identified at Millstone 3 are shown in Table 3.4-3.

Additional information on the walkdown performed at Millstone 3, examples of completed walkdown worksheets, and a discussion of the major findings for the Millstone 3 plant walkdown are included in Appendix A.

3.5 FAILURE MODES AND FAILURE PROBABILITY ESTIMATION

Once the consequences for each segment are defined, the failure probability for a postulated pipe break and pipe leak must be determined. Information relating to the expected failure modes and causes, industry experience and plant specific characteristics are necessary inputs to this determination. These elements are discussed in the following sections and Figure 3.5-1 summarizes the process for this effort.

3.5.1 Failure Modes and Causes

The number of possible degradation mechanisms and loading conditions is large and this section is not intended to provide a full treatment of the details of their occurrence. It does provide an overview of many typical mechanisms and describes the typical process by which they can be evaluated for a specific plant.

Table 3.4-3

SUMMARY OF INDIRECT EFFECTS

Segment ID	Segment Description	Indirect Effect Consequence
CCP-13	Containment penetration cooler supply and return lines	Postulated break disables train A AFW pump due to spray
CCP-14	Containment penetration cooler supply and return lines	Postulated break disables train B AFW pump due to spray
FWA-1	Demin. water storage tank through motor-driven pump P1A to check valves V12 and V7	Postulated break may spray overhead cable tray - loss of HVAC to Train A RHR, QSS, and SI areas
FWA-4	Demin. water storage tank through motor-driven pump P1B to check valves V21 and V26	Postulated break may spray overhead cable tray - loss of HVAC to Train B RHR, QSS, and SI areas
FWA-12,-18	Check valves to cavitating venturi	Postulated break may spray overhead cable tray - loss of HVAC to Train A RHR, QSS, and SI areas
FWA-14,-16	Check valves to cavitating venturi	Postulated break may spray overhead cable tray - loss of HVAC to Train B RHR, QSS, and SI areas
SWP-1,-2	Service water pump discharge check valve to MOV	Flooding of other pump in area, loss of MCC which powers SW Train B equipment
SWP-3,-4	Service water pump discharge check valve to MOV	Flooding of other pump in area, loss of MCC which powers SW Train A equipment
SWP-13	Tee connection near CV 706B through SI pump cooler E1B and 3HVQ*ACUS1B & 2B	Spray could result in a loss of MCC in ESF Room which powers valves needed for operation of one train of recirc.
SWP-15	Tee connection near V63 through cooler CCE-E1B	Postulated break disables both trains of charging due to loss of train B charging pump cooling from the break, and loss of train A charging pump cooling from spray on the train A cooling water pump
SWP-20	Tee connection near CV 705 through SI pump cooler E1A & E2A and residual heat removal vent units ACUS1A & ACUS2A	Spray could result in a loss of MCC in ESF Room which powers valves needed for operation of one train of recirc.

Table 3.4-3 (cont) SUMMARY OF INDIRECT EFFECTS		
Segment ID	Segment Description	Indirect Effect Consequence
SWP-22	Tee connection near V31 through cooler CCE-E1A	Postulated break disables both trains of charging due to loss of train A charging pump cooling from the break, and loss of train B charging pump cooling from spray on the train B cooling water pump
SWP-26,-27	Service water pump to CV and back to pump.	Flooding of other pump in area, loss of MCC which powers SW Train B equipment
SWP-28,-29	Service water pump to CV and back to pump.	Flooding of other pump in area, loss of MCC which powers SW Train A equipment

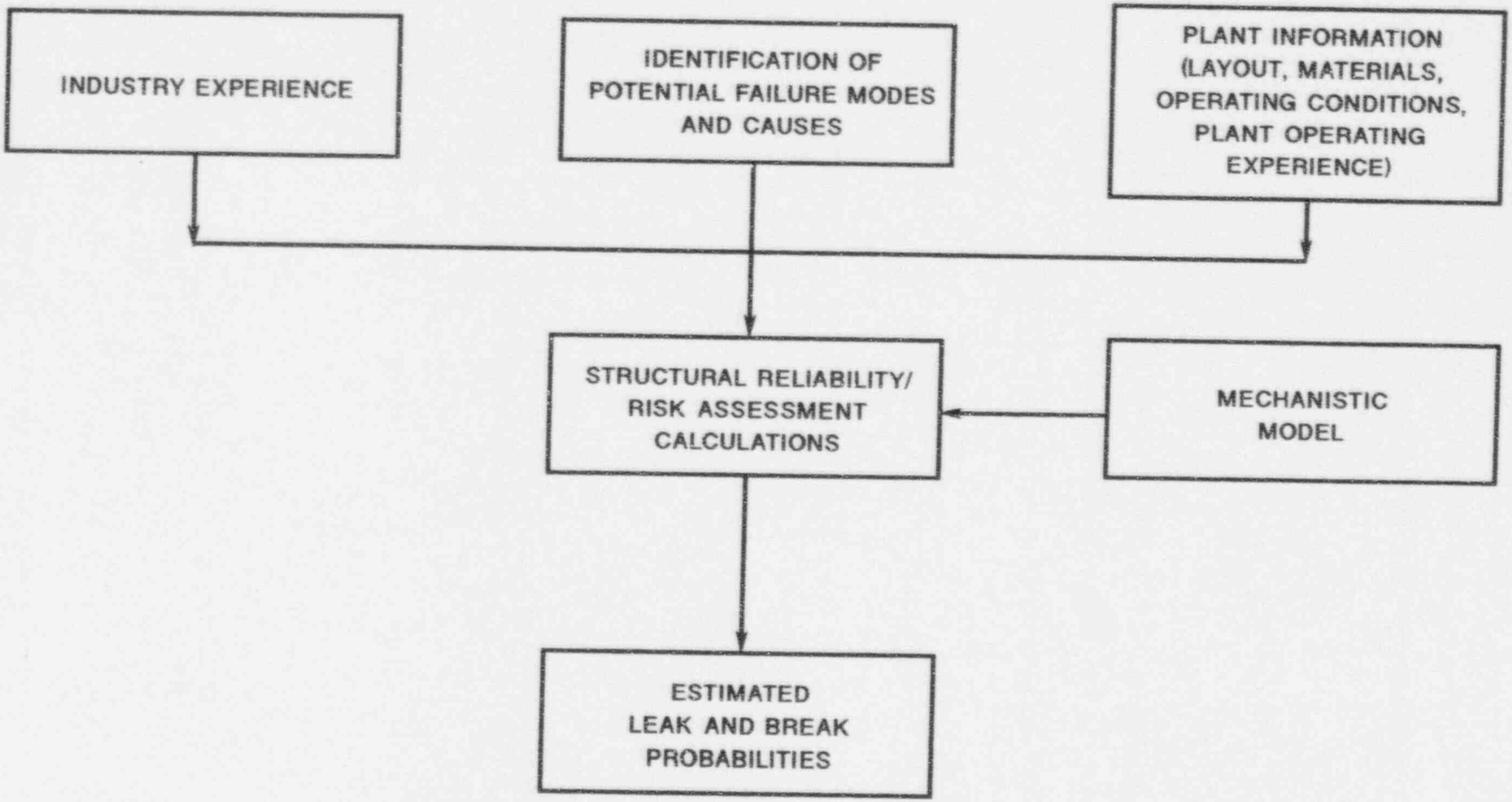

Figure 3.5-1. Failure Probability Estimation Process

The occurrence of a large pressure boundary failure may be considered a two stage process. In the first stage there is physical degradation of the piping element,[3] caused by pitting, crack growth, loss of wall thickness, loss of ductility, etc. The second stage comes into play when loading events occur that challenge the remaining structural integrity of the degraded element. Examples of the additional loads causing failure include pressure surges, water hammer, inadvertent thermal transients, earthquakes, and failure of a support. Some loadings occur randomly while others are related to system operation. Whether the structural failure is limited (causing a leak) or unstable (causing a rupture) depends on the material properties, flaw configuration and nature of the loading. If the degradation mechanism is progressive, then eventually, normal operating loads within the design basis, such as a pump startup, may be sufficient to initiate a limited structural failure.

Based on the above discussion, the piping failure mode is either leakage or rupture depending on the combination of degradation mechanism and initiating loading. For this application, the specific failure event considered is a postulated rupture of the pressure boundary that results in the loss of safety function for the piping segment. Leakage cracks may or may not precede the break. If the leakage could significantly affect the operation of the system or have significant indirect effects, it is also considered a failure event.

For nuclear power plant components, conservative design practices have been successful in addressing most anticipated modes of failure. For example, the ASME Boiler and Pressure Vessel Code identifies the following modes of failure:

- excessive elastic deformation, including elastic instability
- excessive plastic deformation
- brittle fracture
- stress rupture/creep deformation (inelastic)
- plastic instability – incremental collapse
- high strain – low-cycle fatigue

[3] "Piping" for this program includes straight pipe, elbows, tees, other piping components and their interconnecting welds.

The ASME Code rules for design and construction are generally considered effective in precluding these failure modes. It is generally believed within the nuclear industry, however, that other causes not anticipated in the original design 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. The possibility of piping failure due to such unanticipated causes is the basic motivation underlying the ASME Section XI Code rules for inservice examination.

Table 3.5-1 lists a variety of failure causes that should be considered. It can be used as the starting point for a plant-specific evaluation; some listed mechanisms may be discounted while others may need to be added based upon plant-specific experience.

The table includes thermal fatigue as a single item representing several mechanisms such as thermal transients, flow stratification, striping and inadequate design flexibility. The dissimilar metal weld item is not a mechanism in itself but is a significant location for possible weld defects or inservice degradation due to other mechanisms.

Vibration fatigue is one degradation mechanism that can both degrade the structural element and help drive it to its ultimate failure. The issue with vibration is that if it occurs in its most severe form, it can cause failure within a matter of hours or minutes, and there are no precursor indications that can be detected prior to failure. However, the most severe vibration does not usually exist. The driving force may be unsteady in amplitude or frequency, or be intermittent. Also, the resonant amplification that usually contributes to the issue may vary with temperature, piping contents, or growth of any flaws. Vibration fatigue failure is therefore not always intermediate, and its fatigue cracks could in some cases be detected.

Table 3.5-1 EXAMPLE FAILURE CAUSES FOR LWR NUCLEAR POWER PLANT PIPING COMPONENTS	
Erosion	Fatigue (high or low cycle)
Erosion	Mechanical
Erosion/Corrosion	Thermal
Mechanical wear	Vibrational
Fretting	Corrosion
Cavitation	Bulk corrosion
Embrittlement	Crevice corrosion
Irradiation	Pitting corrosion
Thermal aging	Galvanic corrosion
Corrosion/Cracking	Microbiologically influenced
Intergranular	Pitting
Transgranular	Mechanical Damage
Fabrication/Maintenance	Water hammer
Improper heat treatment	Improper or degraded supports
Improper repairs or alterations	Improper or degraded restraints
Dissimilar metal weld	External loads/impact

If vibration is determined to be a possible cause of failure, it must be determined if the piping inservice inspection program would be effective in identifying it prior to failure. In cases where significant vibration is known to be present in normal operation, it should be addressed through the normal technical problem resolution and design change process as it is unlikely that ISI programs will detect vibration fatigue cracking before failure occurs. If vibration is possible, but may continue for a significant time without discovery, then inservice examination for it will probably also be ineffective.

If the potential piping vibration is expected to be induced by equipment vibration (such as degraded pump bearing), and the equipment is operated only occasionally, then inservice examination for fatigue cracking may be effective if it is scheduled to follow equipment testing. For example, if a proposed examination location is at a safety injection pump discharge, then examination following scheduled pump testing may be effective.

After all the possible degradation mechanisms are considered it may be judged that no degradation mechanism is credible for some segments. The piping is expected to retain its full strength and integrity for its entire operating lifetime. For such segments, the only conceivable failure mode is the occurrence of loads greatly in excess of the design basis loads. An example of such a load may be a large mass being dropped on the pipe during nearby maintenance activities, or the occurrence of a greater than design basis earthquake that may shift equipment from its foundations. The failure mode for such segments is classified as "external loads" and such segments are retained in the overall process. The risk assessment may determine the segment to be more safety-significant based only on the severity of its failure consequence. Any examinations ultimately scheduled for such segments would have the value of confirming that indeed no mechanism is active for that piping segment.

3.5.2 Review of Industry Experience

Known failures at other plants should be considered and evaluated for applicability. Available information sources include NRC and EPRI published documents regarding reported failures or operating occurrences, such as flow stratification, which may be applicable to plants. Other useful sources of information include: the Nuclear Plant Reliability Data System (NPRDS), Licensee Event Reports (LERs), NRC Nuclear Plant Aging Research (NPAR) reports, NUMARC (now NEI) Assessments of Plant Life Extension, ASME BPVC Section XI Task Group report on fatigue, NRC pipe crack studies, EPRI Materials Degradation and environmental effects studies, and EPRI/industry erosion-corrosion work.

3.5.3 Information Requirements

To properly evaluate possible failure modes for a given piping segment, specific system information is required. This includes: piping materials, system thermal operating modes (pressure, temperature, and number of cycles), the presence of any thermal transients, the presence of any extended system layup periods or intermittent system operation, system water chemistry, and previous ISI experience.

Plant operating experience should be sought including cracks, leaks, repairs, corrosion, valve leakage, vibration observed during normal operation or during test modes, pipe support issues (including snubber drag loads or lockup, spring hangers topped out or bottomed out), high steam condensate flows, and inadvertent or unexpected system transients.

The best source of qualitative information regarding piping operation and past history is typically the "system engineer" who has full responsibility for the design basis and maintenance of one or more assigned systems. Piping ISI program inspectors and engineers assigned to evaluate identified flaws are also good sources of information for active degradation mechanisms.

3.5.4 Considerations for Selection of Likely Failure Locations

Selection of possible failure modes and their likely location can have a significant influence on the estimated failure probability. One approach for identifying possible failure modes and locations is to classify the pipe segment along the following lines:

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

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

- Materials/chemistry dependent. The IGSCC susceptibility of 304 stainless steel is the most common example. Dissimilar material welds are another example.

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

Interactions among the factors are of course common.

Determination of the most probable break location should involve consideration of all the likely degradation mechanisms for the piping segment under review. Component dependent failure modes are easily 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 most likely location. Load dependent failure

modes would 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) not considered in the design analysis, or external loading. A location where such loads could have the greatest impact can often be determined.

3.5.5 Consideration of Other Piping Reliability Programs

There are several existing programs and activities that positively affect piping reliability. For example, the use of solvents containing chlorides is restricted to help prevent degradation of stainless steel piping. Another example is the set of restrictions on system operation that prevent loading the piping outside its design basis. A third is the regular walkdowns performed on piping by system engineers and plant equipment operators. A fourth example is the erosion-corrosion control program implemented for carbon steel piping at many plants. All of these programs provide a positive contribution to piping reliability. There are also periodic system performance tests that are designed to verify equipment performance but have an additional effect of demonstrating piping reliability.

The beneficial effects of such programs should be considered when estimating failure probabilities of piping elements. If the programs are not considered, the risk-based inspections could become overly weighted to piping that is less safety-significant, as the plant is actually maintained and operated. It is better to rank the segments and select inspection locations and methods that clearly enhance safety by recognizing all of the effects of the existing programs used to ensure piping reliability.

These considerations apply most directly to piping affected by flow-assisted-corrosion (FAC). When properly implemented, the inspections, chemistry control, wall-thinning predictions and component replacements comprising the FAC program result in highly reliable piping. The maintenance of wall thickness above the minimums helps to ensure that failure is by leakage rather than catastrophic rupture. Other mechanisms for failure may be present in some locations; therefore, the piping cannot be excluded altogether from the risk-based inspection program. This piping should be reviewed to determine the most probable failure location for the other mechanisms and a failure probability should be calculated in a manner similar to piping unaffected by FAC.

The presence of the FAC degradation mechanism, however, should not be neglected. Since the FAC program is only intended to manage the wall loss and avoid catastrophic rupture of the pipe, some

pipe wall erosion can take place. The effect of this moderate wall thickness reduction (and any slight imperfections of program implementation) should be incorporated by selecting a "moderate" value for the material wastage parameter in the SRRA failure probability estimation program (see Section 3.5.6 below). The program will thus determine the probability of failure (leak or break) considering FAC and all other relevant causative factors.

If the proposed inspection locations are determined after ranking to be more safety-significant, inspections should be performed at the specified element. An evaluation may be required if there are any additional failure modes beyond the expected FAC wall thinning to ensure the FAC examinations are adequate. Other Code requirements such as inspector qualifications may need to be satisfied.

A similar approach may be taken with other degradation mechanisms for which mechanism-specific programs have been developed. Examples include programs to manage service water piping degradation due to erosion or microbiologically induced corrosion. When such programs are in place and determined to be effective, the failure probability estimation may take them into account. On the other hand, if credit for such programs is taken, then reevaluation of the affected segments must be performed when the programs are changed or discontinued.

3.5.6 Failure Probability Determination

Several approaches can be used to categorize and prioritize the likelihood of failure. A qualitative rating (high, medium, and low likelihood) can be used. However, a quantitative approach can provide further refinement of these general categories.

The task of estimating component failure probabilities can be challenging. In most cases though, consideration of failure probabilities, however uncertain their estimated values, leads to a more effective allocation of inspection resources compared to present practices. Although absolute values of failure probabilities may have large uncertainties, the relative values (e.g., from location to location in a given piping system) are generally better known.

There is no one "best method" to estimate failure probabilities for nuclear components. There will inevitably exist a large uncertainty associated with the estimated probabilities. Catastrophic structural failures rarely occur, and thus little historical data exists to validate estimated failure rates.

Some of the methods available include:

- Historical Data. A number of reports have been published, e.g., by Bush (1988), Jamali (1992), Thomas (1981), and Wright, et al.(1984), with estimates of failure probabilities for nuclear power plant systems and components based on the few occurrences of pipe and vessel rupture events that have actually occurred in related situations. This information is useful as benchmarks of estimates obtained from the other methods.

- Expert Judgement. Elicitation of expert opinion has gained acceptance as a means to quantify input to PSAs and risk-based studies. A systematic procedure, as described by Wheeler, et al.(1989) and the U.S. NRC (1990), has been developed for conducting such elicitations. Generally, the process calls for enlisting and training a suitable team of experts. The team provides responses to a collection of structured questions, allowing sufficient time for the experts to document their rationale. The ASME Research Risk-Based Inspection Development of Guidelines, Volume 2 - Part 1, Light Water Reactor Nuclear Power Plant Components (1992) provides details of this process along with example results for ISI. However, the expert judgement process can be laborious and require the use of several experienced people beyond utility personnel.

- SRRA Predictions. Structural reliability/risk assessment (SRRA) models are usually based on probabilistic structural mechanics methods to estimate failure probabilities for important components. SRRA estimates provide a higher level of detail than estimates based on historical data or expert judgment. Locations within a system with varying failure probabilities can be defined to focus ISI resources. SRRA models can 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 trends can be useful for selecting appropriate intervals over the service life of the components for periodic ISI examination. Some simplified SRRA models have been developed, (e.g., by Chapman and Davers (1987) and by Bishop and Phillips (1993)), that are currently being used in demonstration studies. These simplified models are built from more detailed SRRA models, such as the PRAISE Code which was developed earlier by Harris, et al. (1981).

For the Millstone Unit 3 application, structural reliability and risk assessment (SRRA) tools were used to estimate the failure probabilities of the structural elements most likely to fail in each of the piping segments. The SRRA tools were developed by Westinghouse for Idaho National Engineering Laboratory (INEL) to address the aging of passive components for NRC and DOE (Bishop and Phillips 1993). The SRRA tools are a set of executable personal computer programs to specify input, calculate and plot failure probability of piping with time for the selected input values of key design, operational, and inspection parameters. The computer tool for structural reliability uses Monte-Carlo simulation with importance sampling to calculate the probability of leak or break of type 304 or 316 stainless steel piping (due to fatigue crack growth and stress corrosion cracking) and of carbon steel piping (due to fatigue crack growth and loss of thickness due to wastages, such as erosion-corrosion-wear).

The SRRA code is described in detail in Appendix D. Appendix D discusses the code inputs, guidelines for selecting limiting locations and estimating failure probabilities, guidelines on expertise and information required, and sample outputs. Benchmarking of the SRRA code (such as against the PRAISE code) is discussed in Appendix E.

In cases in which the SRRA tool could not be applied (such as pipe segments containing copper-nickel material or pipe internally coated with epoxy), expert judgement was used to provide a failure probability estimate.

Table 3.5-2 identifies example piping segment small leak and full break failure probabilities for the representative WOG plant, Millstone Unit 3. The piping failure modes are either exceeding the limiting crack depth (leak) or exceeding the flow stress in the remaining uncracked section (break).

Table 3.5-2

EXAMPLE CALCULATED PIPE FAILURE PROBABILITIES

SEGMENT ID	SEGMENT DESCRIPTION	FAILURE PROBABILITY*			
		SMALL LEAK		FULL BREAK	
		NO ISI	WITH ISI**	NO ISI	WITH ISI**
EMERGENCY CORE COOLING-ECCS					
ECCS-1	From CV 8819C and CV 8819C to CV 8847C	0 (6.4E-08)	0 (6.4E-08)	0 (2.3E-12)	0 (2.3E-12)
ECCS-2	From CV 8819A and CV 8819A to CV 8847A	0 (6.4E-09)	0 (6.4E-09)	0 (2.3E-12)	0 (2.3E-12)
ECCS-3	From CV 8819D and CV 8819D to CV 8847D	0 (6.4E-09)	0 (6.4E-09)	0 (2.3E-12)	0 (2.3E-12)
ECCS-4	From CV 8819B and CV 8819B to CV 8847B	0 (6.4E-09)	0 (6.4E-09)	0 (2.3E-12)	0 (2.3E-12)
ECCS-5	From CV 8847A and CV 8956A to CV 8948A	9.2E-09	8.7E-09	1.4E-13	1.4E-13
ECCS-6	From CV 8847B and CV 8956B to CV 8948B	0 (6.4E-09)	0 (6.4E-09)	0 (2.3E-12)	0 (2.3E-12)
ECCS-7	From CV 8847C and CV 8956C to CV 8948C	9.2E-09	8.7E-09	1.5E-15	1.5E-15
ECCS-8	From CV 8847D and CV 8956D to CV 8948D	1.4E-08	9.9E-09	7.5E-15	6.6E-15
MAIN FEEDWATER/CONDENSATE SYSTEM					
FWS-1	From MOV 35A to FCV 510	1.1E-03	6.2E-06	0 (3.5E-11)	0 (3.5E-11)
FWS-2	From FCV 510 and LV 550 to CTV 41A	1.1E-03	6.2E-06	0 (3.5E-11)	0 (3.5E-11)
FWS-13	From main feedwater pumps P1, P2A, P2B to MOVs 35A, B, C, D	1.4E-03	7.1E-05	2.5E-07	2.1E-08
FWS-18	From condenser pipe connections 3A, 3B, 3C to MOVs 49A, B, C	1.2E-03	1.7E-04	6.8E-07	2.3E-08

Table 3.5-2 (cont)
EXAMPLE CALCULATED PIPE FAILURE PROBABILITIES

SEGMENT ID	SEGMENT DESCRIPTION	FAILURE PROBABILITY* SMALL LEAK NO ISI	SMALL LEAK WITH ISI**	FULL BREAK NO ISI	FULL BREAK WITH ISI**
REACTOR COOLANT SYSTEM-RCS					
RCS-7	LPSI connection from Loop A cold leg tee to CV 8948A	1.9E-06	1.3E-06	4.1E-09	3.4E-09
RCS-22	LPSI connection from Loop C cold leg tee to CV 8948C	1.9E-06	1.3E-06	4.2E-09	3.4E-09
RCS-29	LPSI connection from Loop D cold leg tee to CV 8948D	1.9E-06	1.3E-06	4.1E-09	3.4E-09
RCS-54	LPSI connection from Loop B cold leg tee to CV 8948B	0 (2.1E-08)	0 (2.1E-08)	1.2E-12	1.2E-12
HIGH PRESSURE SAFETY INJECTION					
SIH-4	From MOVs 8821A and 8821B to CVs 8819A,B,C,D	0 (2.2E-09)	0 (2.2E-09)	0 (8.1E-13)	0 (8.1E-13)
SIH-5	From MOVs 8920 and 8814 to RWST	5.4E-08	4.1E-08	1.2E-10	5.9E-12
LOW HEAD SAFETY INJECTION-SIL					
SIL-4	From MOV 8809A to CVs 8818A,B	0 (2.5E-08)	0 (2.5E-08)	0 (9.2E-12)	0 (9.2E-12)
SIL-5	From MOV 8809B to CVs 8818C,D	0 (2.5E-08)	0 (2.5E-08)	0 (9.2E-12)	0 (9.2E-12)
SERVICE WATER SYSTEM-SWP					
SWP-1	Service water pump P1D to MOV 102D and return to pump	1.7E-03	1.3E-04	2.6E-08	5.6E-11
SWP-2	Service water pump P1B to MOV 102B and return to pump	1.7E-03	1.3E-04	2.6E-08	5.6E-11
SWP-3	Service water pump P1C to MOV 102C and return to pump	1.7E-03	1.3E-04	2.6E-08	5.6E-11
SWP-4	Service water pump P1A to MOV 102A and return to pump	1.7E-03	2.6E-05	2.6E-08	3.2E-11
SWP-5	Service water pumps P1B & P1D discharge to MOV 54B, 54D, 71B and 50B	6.6E-05	8.8E-06	0 (3.7E-13)	0 (3.7E-13)

*For weld most likely to fail at end of life in the segment (see Appendix D for description of failure modes)
For the cases in which 0 failures are predicted, the values in parentheses are those calculated assuming one half failure in 5000 trials, corrected for importance sampling.

**The failure probabilities shown "with ISI" reflect the inspection interval and inspection accuracy associated with the inspection method recommended for each respective location.

The full break failure probabilities without ISI were used in the calculations to determine the total segment core damage frequency. The small leak probabilities were used as part of a sensitivity study.

Based on the full break failure probabilities calculated using the SRRA code, a threshold pipe failure probability of 1E-08 was selected for use in the consequence calculations. This value was used for piping segments in which a credible failure mechanism could not be postulated. This threshold was used to account for the possibility of an incredible pipe failure and to account for consequence-driven pipe segments.

3.6 SELECTION OF ISI SEGMENTS

This section discusses how the segments are categorized into two risk categories: more safety-significant and less safety-significant. There are three phases to the risk categorization process: 1) application of the PSA to calculate the total pressure boundary core damage frequency (CDF) and LERF (if possible) and importance measures and evaluation of other PSA-related factors, 2) integration of other deterministic considerations, and 3) expert panel evaluation. The segment risk-ranking process is shown in Figure 3.6-1.

3.6.1 Risk-ranking

Because plant PSA models do not explicitly include piping pressure boundary failures except for LOCAs, SGTR, steamline/feedline breaks, and Reactor Vessel Rupture, another method for evaluating pressure boundary failures in terms of risk was required.

First, a means to determine the relative risk significance of piping segments was necessary. Because piping failure probabilities are low, if the total CDF for all plant internal events is used, none of the pressure boundary piping components would be more safety-significant via RRW (all RRWs would be equal to 1.0). Thus, the PSA results will be useless in helping to determine where to focus priorities for piping ISI. Modeling the piping pressure boundary failures and then assessing the relative risk significance to a total CDF related to just piping pressure boundary failures renders more meaningful results. Therefore, it was decided that the total CDF used in the risk significance evaluation should only account for those associated with piping pressure boundary failures.

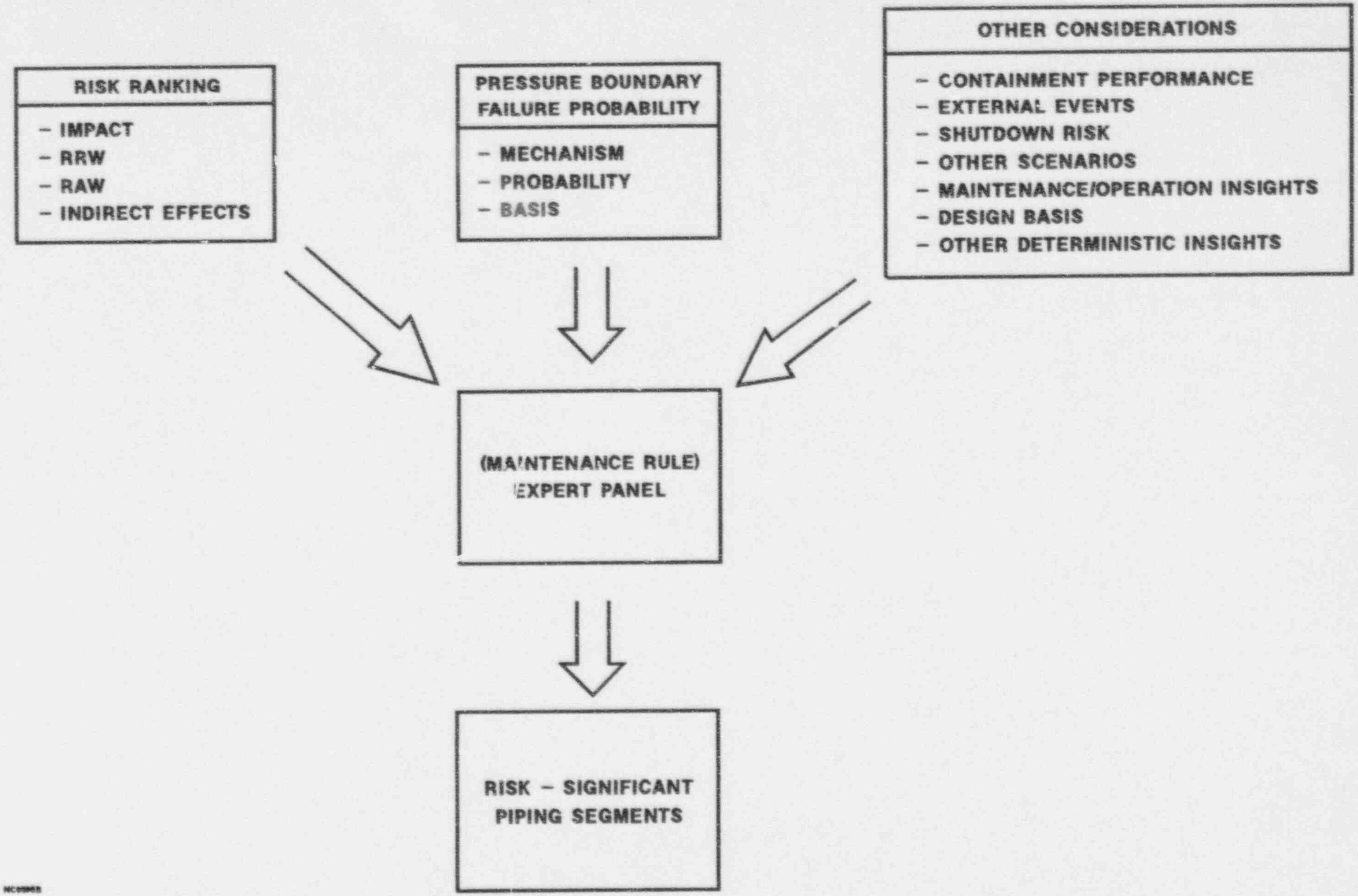

Figure 3.6-1. Pipe Segment Risk-Ranking Process

Secondly, how to determine the CDF due to piping pressure boundary failures was evaluated. The inclusion of piping segments directly into the PSA models was considered but not adopted since: 1) the effort would be too labor intensive and 2) the pipe segment failure probabilities are sufficiently lower than already-modeled components, that the pipe segments would in all likelihood fall below the truncation limits used in quantifying PSA models. Therefore, the approach identified was to quantify the CDF due to piping pressure boundary failures outside the PSA model but to use the plant PSA model as input. To determine the CDF for each piping segment, a surrogate component (basic event or set of basic events, such as a pump or valve) or an initiator that is already modeled in the plant PSA is identified in which the consequence or impact on the CDF matches the postulated consequence for the piping pressure boundary failure. The surrogate component is assumed to fail with a failure probability of 1.0 for use in obtaining the conditional core damage frequency (or probability). The conditional core damage frequency/probability results are combined with segment failure probability/rate to obtain the CDF contribution for each segment. The CDF contributions from all piping segments are then summed to obtain total piping pressure boundary failure CDF.

From this information, the risk importance measures can then be calculated to provide a relative ranking of piping segments.

In order to use the plant PSA as input to the pressure boundary failure CDF calculations, the postulated consequences of the failure must be identified as described in previous sections. Then based on the postulated consequences, the PSA model must be manipulated to obtain the required information. The consequences to be considered from both direct effects and indirect effects include:

- Failures that cause an initiating event such as a LOCA or reactor trip
- Failures that disable a single component, train or system
- Failures that disable multiple components, trains or systems
- Failures that cause any combination of the above

Because the consequences can vary and the correct PSA and failure probability information is necessary for the CDF (or LERF) calculations, the process requires different manipulations for each type of consequence. The process is outlined in Figure 3.6-2. Different equations were developed to ensure the proper calculation for each type of consequence. Care must be taken to ensure that the

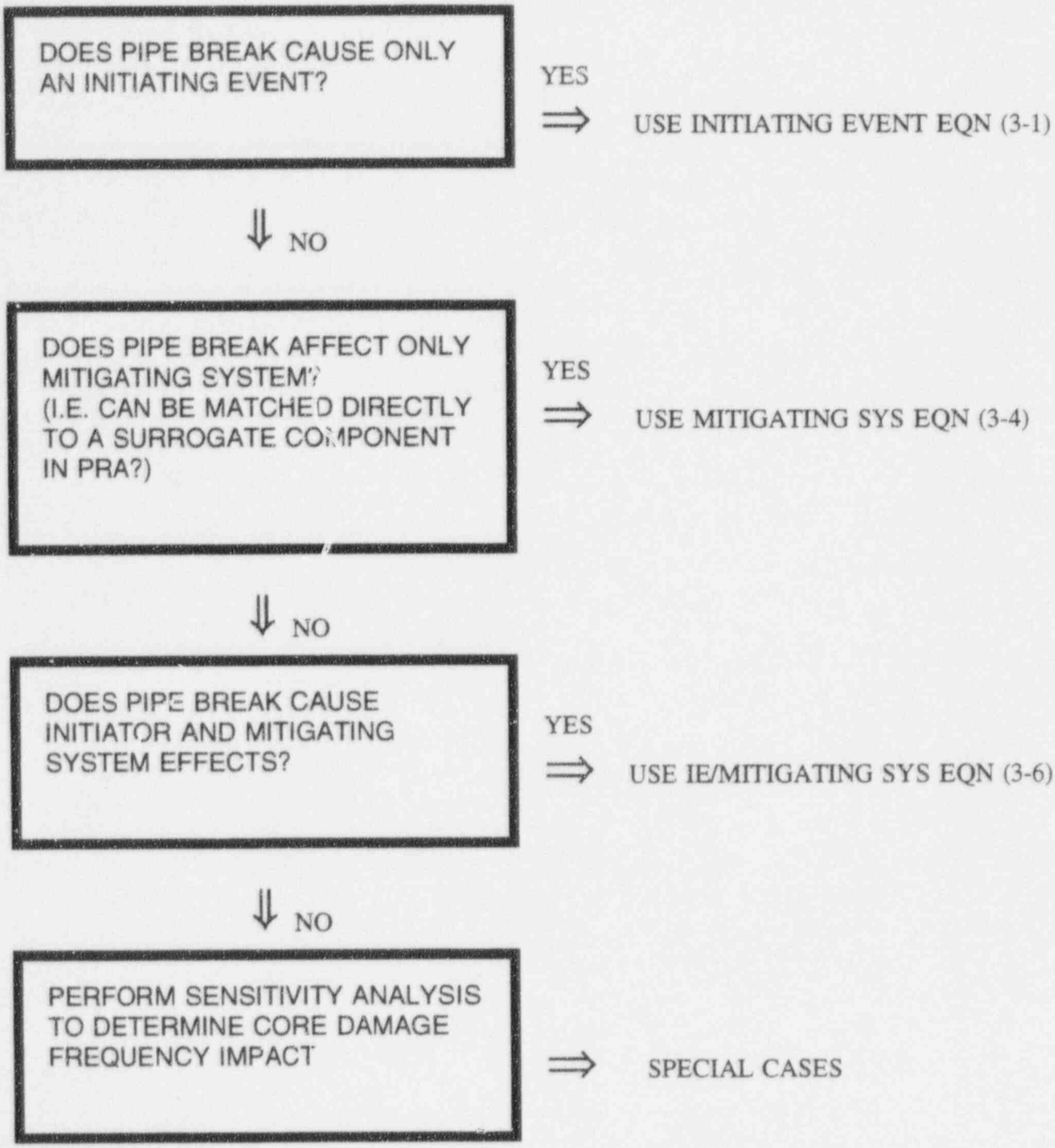

Figure 3.6-2. Core Damage Frequency Calculation Process

correct units are applied both from the failure probability calculations and the core damage frequency calculations to obtain a core damage frequency (in units of events per year) for each piping segment. The results of conditional core damage frequency/probability are combined with the results of the segment failure probability/rate to obtain core damage frequency for each segment. The different impacts are described below.

<u>Initiating Event Consequence</u>

For piping failures that cause an initiating event only, the portion of the PSA model that is impacted is the initiating event and its frequency. For a piping segment, the core damage frequency from the piping failure is calculated by:

$$CDF_{PB} = FR_{PB} * CCDP_{IE} \qquad (3\text{-}1)$$

where:

CDF_{PB} = Core Damage Frequency from a piping failure (events per year)
$CCDP_{IE}$ = Conditional core damage probability for the initiator
FR_{PB} = Piping failure rate assuming no ISI (in events per year)

The conditional core damage probability is determined from existing base PSA results. The core damage frequency contribution from the initiating event postulated for the piping failure is identified along with the base PSA initiating event frequency. Dividing the CDF by the initiating event frequency yields the conditional core damage probability as shown by:

$$CCDP_{IE} = CDF_{IE} \ / \ FREQ_{IE} \text{ (dimensionless)} \qquad (3\text{-}2)$$

where:

CDF_{IE} = Base PSA Core Damage frequency from the initiating event (in events per year)
$FREQ_{IE}$ = Initiating event frequency from base PSA (in events per year)

An example of a piping failure resulting in an initiating event is the failure of a piping segment in the main feedwater system near the main feedwater pumps (that does not also cause a loss of auxiliary feedwater). This piping failure is equated to a loss of main feedwater initiating event. Given that the base PSA CDF contribution from the loss of main feedwater is 7.67E-07 events/year and the initiating event frequency is 0.64 events/year,

$$\begin{aligned} CCDP_{IE} &= CDF_{IE} \;/\; FREQ_{IE} \\ &= 7.67E\text{-}07/\text{year} \;/\; 0.64/\text{year} \\ &= 1.20E\text{-}06 \end{aligned}$$

Similarly, for a piping failure in the RCS which results in a large LOCA with a large LOCA CDF of 1.90E-06/year and an initiating event frequency of 2.03E-04/year, the conditional core damage probability is determined to be 9.36E-03.

The piping failure rate (in events per year) is obtained from the SRRA model assuming no ISI. Because the SRRA model generates a probability, the probability must be transformed into a failure rate. The cumulative break probability at end of license is divided by the number of years at end of license. In other words,

$$FR_{PB} = FP_{PB}/EOL \tag{3-3}$$

where:

FP_{PB} = Piping failure probability from SRRA model assuming no ISI (dimensionless)
EOL = Number of years used in SRRA model from beginning to end of license (usually assumed to be 40 years)

For a piping segment in the main feedwater system in which a failure probability from the SRRA model (no ISI) is identified to be 6.80E-07, the failure rate would be (6.80E-07 / 40 years) or 1.7E-08/year Similarly, for a piping segment in the RCS in which a failure probability from the SRRA model is determined to be 8.38E-06, the failure rate would be (8.38E-06 / 40 years) or 2.10E-07/year.

Using the above information for both the conditional core damage probability and piping failure rate, the piping segment core damage frequency can be calculated. For the RCS piping segment described above, the core damage frequency from the piping failure is calculated by:

$$\begin{aligned} CDF_{PB} &= FR_{PB} * CCDP_{IE} \\ &= 2.10E\text{-}07/year * 9.36E\text{-}03 \\ &= 1.96E\text{-}09/year \end{aligned}$$

For the main feedwater piping segment, the core damage frequency would be (1.70E-08/year * 1.20E-06) or 2.04E-14/year.

Mitigating System(s) Consequence

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

$$CDF_{PB} = FP_{PB} * CCDF_{PB} \tag{3-4}$$

where:

CDF_{PB} = Core Damage Frequency from a piping failure (in events/year)
$CCDF_{PB}$ = Conditional CDF with segment failed (=1.0) (in events/year)
FP_{PB} = Pipe break failure probability (dimensionless)

To obtain the conditional CDF, a surrogate component (basic event or set of basic events, such as a pump or valve) that is already modeled in the plant PSA 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 to obtain a new total plant conditional core damage frequency. In order to determine the conditional core damage frequency for the piping segment only, the base total plant PSA CDF is subtracted from the new total plant CDF as shown by:

$$CCDF_{PB} = CDF_{PB=1.0} - CDF_{BASE} \tag{3-5}$$

where:

$CDF_{PB=1.0}$ = new total plant CDF with surrogate component = 1.0 (in events/year)

CDF_{BASE} = base total plant CDF (in events per year)

For example, for a piping segment in the high pressure injection system which causes a loss of the RWST, the $CDF_{PB=1.0}$ was calculated to be 4.74E-02/year. Given a base total plant CDF of 5.87E-05/year, the $CCDF_{PB}$ is (4.74E-02/year - 5.87E-05/year) or 4.73E-02/year. Similarly, for a piping segment that causes a loss of one high pressure injection system train, the $CDF_{PB=1.0}$ was determined to be 8.85E-05/year. Therefore, the $CCDF_{PB}$ is (8.85E-05/year - 5.87E-05/year) or 2.98E-05/year.

Using the above information for both the conditional core damage frequency and piping failure probability, the piping segment core damage frequency can be calculated. For the loss of RWST piping segment described above with a failure probability of 1E-08, the core damage frequency from the piping failure is calculated by:

$$\begin{aligned} CDF_{PB} &= FP_{PB} * CCDF_{PB} \\ &= 1E\text{-}08 * 4.73E\text{-}02/year \\ &= 4.73E\text{-}10/year \end{aligned}$$

For the loss of one high pressure safety injection train with a failure probability of 1E-08, the CDF_{PB} is 1E-08 * 2.98E-05/year = 2.98E-13/year.

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 with the surrogate component for the mitigating system assumed to fail with a probability of 1.0. For piping failures that cause an initiating event and system degradation, the following equation is applied:

$$CDF_{PB} = FR_{PB} * CCDP_{IE,\ seg=1.0} \quad (3\text{-}6)$$

where

CDF_{PB} = Core Damage Frequency from a piping failure (events per year)

$CCDP_{IE, seg=1.0}$ = Conditional core damage probability for the initiator with mitigating system component assumed to fail

FR_{PB} = Piping failure rate (in events per year)

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

$$CCDP_{IE, seg=1.0} = CDF_{IE, seg=1.0} / FREQ_{IE} \quad (3\text{-}7)$$

where:

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

$FREQ_{IE}$ = Initiating event frequency

For example, a piping failure in a segment in the charging system may result in a reactor trip and a loss of the RWST. A surrogate component for the RWST is assumed to fail with a probability of 1.0 and the reactor trip initiator event tree sequences are requantified to obtain the new core damage frequency for that initiator (an alternative would be to set both initiating event frequency and surrogate component to 1.0 and requantifying). For this example, the new CDF for the reactor trip initiator with the segment failed was determined to be 5.61E-03/year. With an initiating event frequency of 3.38/year, the conditional core damage probability is:

$$\begin{aligned} CCDP_{IE, seg=1.0} &= CDF_{IE, seg=1.0} / FREQ_{IE} \\ &= 5.61E\text{-}03/\text{year} / 3.38/\text{year} \\ &= 1.66E\text{-}03 \end{aligned}$$

Assuming a piping failure probability of 1E-08, the failure rate is:

$$\begin{aligned} FR_{PB} &= FP_{PB}/EOL \\ &= 1.00E\text{-}08 / 40 \text{ years} = 2.50E\text{-}10/\text{year} \end{aligned}$$

The core damage frequency contribution can then be calculated as:

$$\begin{aligned} CDF_{PB} &= FR_{PB} * CCDP_{IE,\ seg\ =\ 1.0} \\ &= 2.50E\text{-}10/year * 1.66E\text{-}03 \\ &= 4.15E\text{-}13/year \end{aligned}$$

Special Cases

Not all piping segments fit into the three categories described above. Each piping segment is analyzed separately to determine the best method of calculation. Some segments may fall into several of these categories depending on the circumstance. For example, a failure in the piping segment in the charging system is postulated to result in a reactor trip and subsequent loss of the RWST. This segment has two separate cases considered that are then added together to obtain the total core damage frequency for that segment. First, the segment is modeled as a reactor trip and loss of RWST using equation 3-6 and then the segment is modeled as a loss of RWST for the remaining initiating events using equation 3-4.

Total Pressure Boundary Core Damage Frequency

Each piping segment within the scope of the program is evaluated to determine its core damage frequency due to piping failure. Once this fs completed, the total pressure boundary core damage frequency is calculated by summing across each individual segment. This now provides the baseline from which to determine the risk importance measures.

For the representative WOG plant, the total piping pressure boundary core damage frequency was estimated to be 2.28E-08/year. The results by system are shown in Table 3.6-1. The piping CDF does not result in an increase in the total plant CDF for all events (5.87E-05 per year).

The same process described for CDF can also be applied to determine the importance to LERF.

Table 3.6-1 NUMBER OF SEGMENTS DEFINED AND CDF CONTRIBUTIONS BY SYSTEM		
SYSTEM	**NUMBER OF SEGMENTS**	**AT-POWER PRESSURE BOUNDARY CDF (events/year)**
BDG (SG Blowdown)	4	1.61E-15
CCE (CCP Cool)	2	1.44E-11
CCI (SI Cool)	with SIH	–
CCP (CCW)	14	2.25E-12
CHS (CVCS)	23	2.25E-09
CNM (Condensate)	with FWS	–
DTM (Turbine Plant Drains)	with MSS	–
ECCS	9	5.33E-10
EGF (DG Fuel)	4	3.21E-12
FWA (Aux Feed)	15	4.25E-09
FWS (Feedwater)	19	3.75E-14
HVK (Control Bldg Chilled Water)	1	4.21E-11
MSS (Main Steam)	30	3.20E-13
QSS (Quench)	5	4.79E-10
RCS	66	3.08E-09
RHS (RHR)	with SIL	–
RSS (Recirc)	11	5.98E-10
SFC (Fuel Pool)	4	*
SIH (HPI)	10	2.66E-09
SIL (LPI)	13	2.39E-09
SWP (SW)	29	6.49E-09
TOTAL	259	2.28E-08

*Not modeled as part of at power PSA.

Risk Importance Calculations

Risk categorization involves calculating the relative importance of a component to a pre-defined consequence measure, such as core damage frequency (CDF). Two importance measures are generally calculated for each component: Risk-Reduction Worth and Risk Achievement Worth Importance.

- Risk Reduction Worth (RRW) measures how much the core damage frequency will decrease if the unavailability of the component of interest is set to 0 (that is, the component is always available/perfectly reliable). The equation used to calculate RRW is:

$$RRW = CDF_{base} / CDF_0 \tag{3-8}$$

where:

CDF_0 = Core Damage Frequency when the component failure probability is set to 0
CDF_{base} = Base Core Damage Frequency

- Risk Achievement Worth (RAW) measures the increase in core damage frequency (CDF) when the component failure probability is set to 1.0. In other words, the RAW computes a increase in CDF when the component of interest is guaranteed to fail. The equation used to calculate RAW is:

$$RAW = CDF_1 / CDF_{base} \tag{3-9}$$

where:

CDF_1 = Core Damage Frequency when the component failure probability is set to 1.0
CDF_{base} = Base Core Damage Frequency

Fussell-Vesely (F-V) Importance may be used in lieu of RRW because of the mathematical relationship between the measures. Fussell-Vesely Importance (F-V) measures the decrease in CDF if

the components failure probability is set to 0.0. In other words, the F-V computes the decrease in CDF when the component of interest is perfectly reliable. The equation used to calculate F-V is:

$$F\text{-}V = (CDF_{base} - CDF_0) / CDF_{base} \quad (3\text{-}10)$$

where:

CDF_0 = Core Damage Frequency when the component failure probability is set to 0.0

CDF_{base} = Base Core Damage Frequency

In assessing the safety significance for the piping segments, RAW and RRW values were calculated. Piping failure probabilities are typically very small compared to other component failures modeled in the PSA. Therefore, when the failure probability is set to 1.0 for the RAW calculation, large RAW values typically result. Table 2.3-1 (based on the EPRI PSA Applications Guide) suggests that RAW values greater than 2 should be considered more safety-significant. This EPRI criteria was not used for the Millstone 3 application because the majority of the calculated RAW values were above 2. Instead, the safety-significance determination focused on the RRW values, and RAW values were used on a relative basis to help differentiate segments which had similar RRW values.

A summary of results of the calculations are shown in Table 3.6-2. Segments with a RRW value greater than 1.00 are shown along with the RAW values. The RRW values range from 1.001 to 1.044 while the RAW valves range from 1.85E+05 to 3.67E+06.

Sensitivity Studies

In addition to quantitatively comparing the risk importance measure results to the screening criteria, the results are reviewed qualitatively as prescribed by the EPRI PSA Applications Guide (EPRI 1995). Sensitivity studies are conducted to determine if changes in key assumptions or data can impact the categorization of the piping segments. These sensitivity studies address the potential changes in component rankings by varying the estimates of the piping pressure boundary failure probabilities/rates and estimates of the conditional core damage frequency/probability. Several sensitivity studies were conducted during the program on the total piping pressure boundary CDF results. The base CDF for Millstone 3 piping generally assumes no operator recovery actions in the conditional CDF calculations

Table 3.6-2 PIPING SEGMENTS WITH RRW > 1.00		
SEGMENT	**RRW**	**RAW**
CHS-3	1.026	2.65E+06
CHS-5	1.026	2.65E+06
CHS-7	1.026	2.65E+06
CHS-23	1.021	2.08E+06
ECCS-0	1.021	2.08E+06
ECCS-5	1.001	5.22E+04
ECCS-6	1.001	5.22E+04
ECCS-8	1.001	5.22E+04
FWA-1	1.002	3.66E+06
FWA-4	1.002	3.66E+06
FWA-7	1.038	3.66E+06
FWA-12	1.038	3.66E+06
FWA-14	1.038	3.66E+06
FWA-16	1.038	3.66E+06
FWA-18	1.038	3.66E+06
HVK-1	1.002	1.85E+05
QSS-2	1.021	2.08E+06
RCS-1	1.004	4.11E+05
RCS-2	1.004	4.11E+05
RCS-3	1.001	4.11E+05
RCS-5	1.004	4.11E+05
RCS-6	1.004	4.11E+05
RCS-8	1.006	4.11E+05
RCS-9	1.005	4.11E+05
RCS-10	1.005	4.11E+05
RCS-11	1.007	4.11E+05
RCS-13	1.004	4.11E+05
RCS-14	1.005	4.11E+05
RCS-16	1.004	4.11E+05
RCS-17	1.004	4.11E+05
RCS-18	1.044	4.11E+05

Table 3.6-2 (cont) PIPING SEGMENTS WITH RRW > 1.00		
SEGMENT	RRW	RAW
RCS-20	1.044	4.11E+05
RCS-21	1.005	4.11E+05
RCS-23	1.005	4.11E+05
RCS-24	1.005	4.11E+05
RCS-25	1.007	4.11E+05
RCS-27	1.004	4.11E+05
RCS-28	1.005	4.11E+05
RCS-43	1.001	4.11E+05
RCS-56	1.001	4.11E+05
RSS-11	1.026	2.64E+06
SIH-1	1.021	2.08E+06
SIH-2	1.039	2.08E+06
SIH-3	1.039	2.08E+06
SIH-4	1.021	2.08E+06
SIL-1	1.021	2.08E+06
SIL-2	1.021	2.08E+06
SIL-3	1.021	2.08E+06
SIL-4	1.021	2.08E+06
SIL-5	1.021	2.08E+06
SWP-1	1.036	1.32E+06
SWP-2	1.036	1.32E+06
SWP-3	1.035	1.30E+06
SWP-4	1.035	1.30E+06
SWP-5	1.013	1.32E+06
SWP-6	1.013	1.32E+06
SWP-7	1.013	1.30E+06
SWP-8	1.013	1.30E+06
SWP-23	1.013	1.30E+06
SWP-25	1.013	1.32E+06
SWP-26	1.018	6.69E+05
SWP-27	1.018	6.69E+05
SWP-28	1.017	6.64E+05
SWP-29	1.017	6.55E+05

and uses the break probabilities from the SRRA calculation, but with a 1E-08 truncation for piping segments with no active failure mechanisms. This base case is shown as "BASE PIPING CDF" in Figure 3.6-1, with a value of 2.28 E-08/year, along with the results of the sensitivity studies that are discussed below in detail. Figure 3.6-2 shows a breakdown by system of the CDF changes for the sensitivities.

The risk importance measures were calculated for each sensitivity study and a comparison of these results with the piping segments chosen by the expert panel is discussed in Section 3.6.4.

- Credit for Operator Recovery Action - For Millstone 3, the only operator action credited in the base calculations was for auxiliary feedwater piping segments FWA-1 and FWA-4 where the operator would align the condensate storage tank (CST) to the intact auxiliary feedwater trains to reduce the consequences from a loss of the demineralized water storage tank (DWST) to all trains to the loss of one auxiliary feedwater motor-driven pump train. Credit for this operator action was taken based on the advice of the expert panel (which included a plant operator familiar with the emergency operator procedures) and time availability based on safety analysis calculations.

  A sensitivity was performed in which postulated operator recovery actions that would change the consequences in each piping segment were considered. The operator recovery action was credit with a success probability of 1.0 (failure probability of 0). For a majority of piping segments, an operator recovery action could not be postulated. The total piping pressure boundary core damage frequency was calculated to be 1.14E-08/year (CDF W/OP ACTION), which is only a factor of 2 lower than the base piping CDF of 2.28E-08/year.

- Use of Actual SRRA failure probabilities - The use of a threshold value of 1E-08 for piping failure probabilities was examined by using the actual SRRA code failure probabilities. An example of the actual SRRA code failure probabilities was shown in Table 3.5-2. The use of these failure probabilities would be expected to give a lower bound of the piping failure probabilities. The CDF was recalculated to be 9.96E-09/year (CDF USING ACTUAL PROB), which is more than a factor of 2 reduction in the base piping CDF. The risk importance measures were also recalculated.

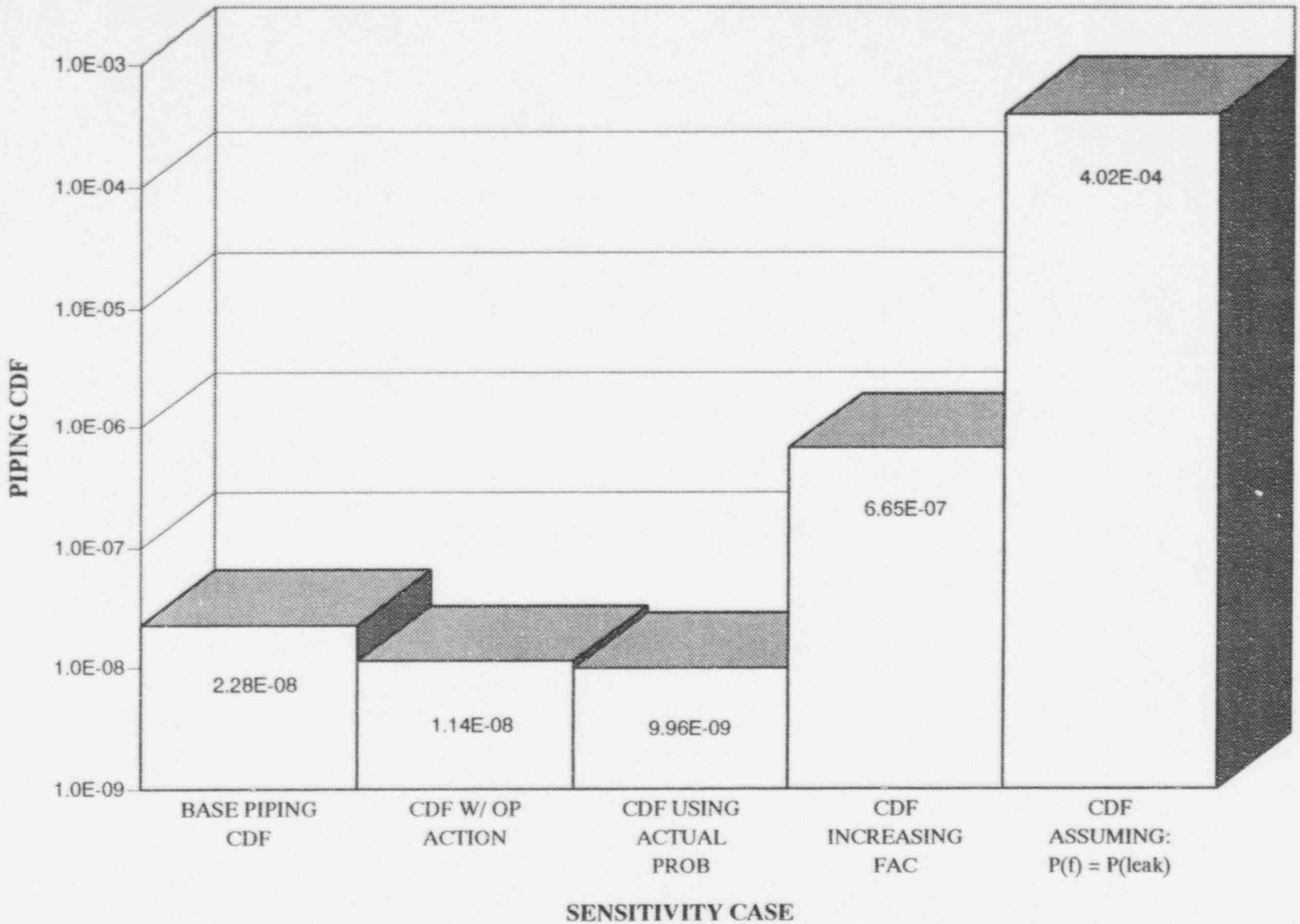

Figure 3.6-1. Comparison of Piping CDF for Various Sensitivity Studies

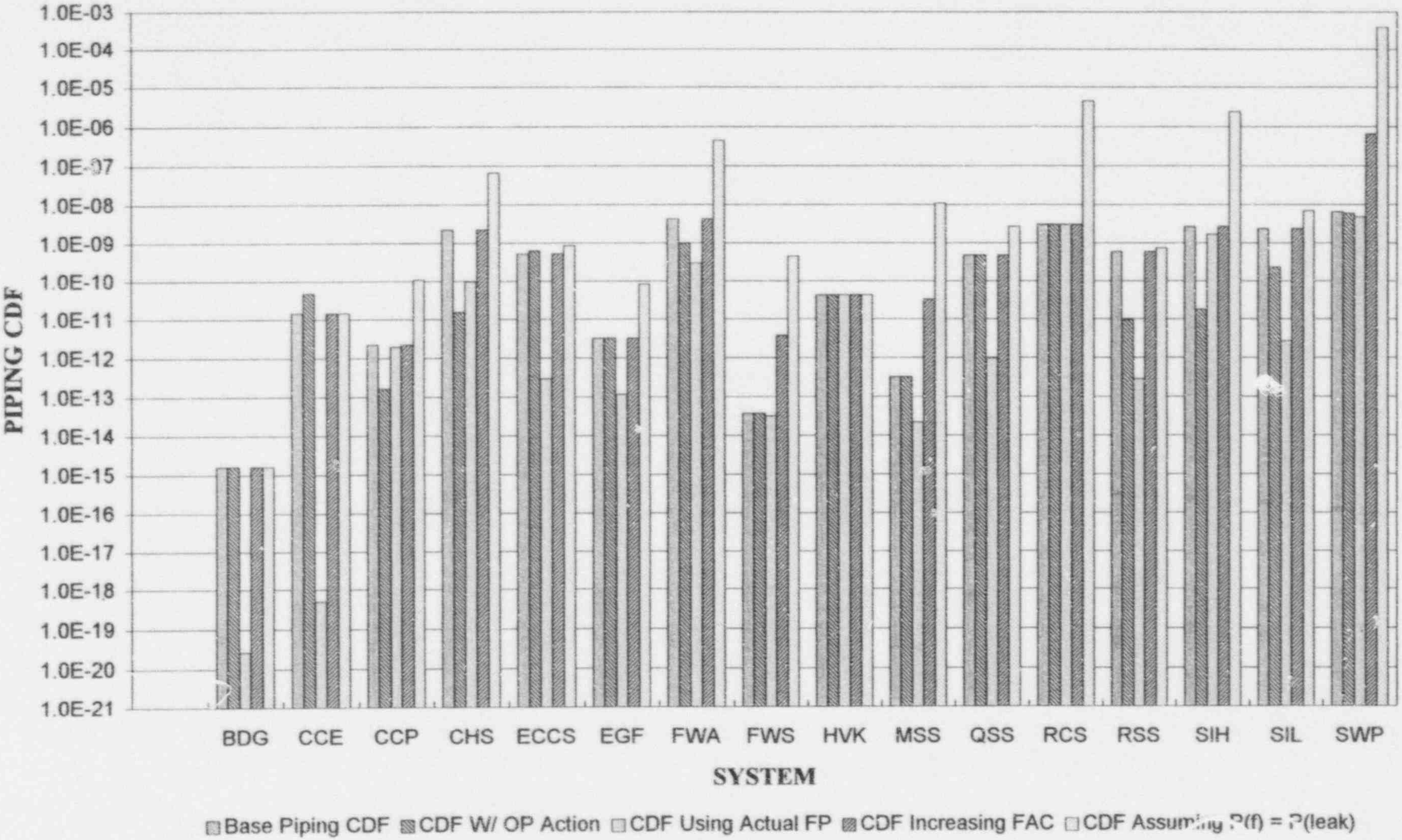

Figure 3.6-2. Comparison by System of Piping CDF for Various Sensitivity Studies

- Credit for Plant FAC Program - The Millstone application assumed that the plant's FAC program was adequate in finding piping degradation with respect to this failure mode. This was considered in the development of the piping failure probabilities for several systems affected by FAC. A sensitivity study was conducted that increased the failure probabilities for those piping segments (in the main feedwater, main steam, and service water systems) by a factor of 100 and the resultant core damage frequency and risk importance measures were recalculated. The CDF from this sensitivity study (CDF Increasing FAC) was determined to be 6.65E-07/year, which is a factor of 30 increase in the base piping CDF. This assumption results in the affected systems dominating the CDF, and inspection resources, which should be used for other plant piping systems, would be misallocated.

- Use of Leak Probabilities - The SRRA code (described in Section 3.5 and Appendix D) generates small leak probabilities in addition to full break probabilities. The use of the these leak probabilities provides an upper bound estimate of the piping pressure boundary failure probabilities. The use of the leak probabilities in lieu of the break probabilities in the core damage frequency and risk importance measure calculations may provide an additional differentiation between the piping segments even though the small leak does not disable the safety function of the piping segment and thus would actually result in significantly reduced consequences. The core damage frequency calculated for this case was 4.02E-04/year (CDF Assuming P(f) = P(leak)). This result provides an extremely conservative upper bound of the expected CDF contribution due to piping failures. The risk importance measures were also recalculated as discussed in Section 3.6.4.

These sensitivity studies showed that although variation exists in the numerical results, most piping segments have the same relative ranking (as discussed in section 3.6.4). This result is similar to that obtained from the uncertainty/sensitivity analyses performed by ASME Research and Pacific Northwest Laboratories, as documented in NUREG/CR-6181 (NRC 1994) and NUREG/GR-005 (ASME 1993), in earlier work performed at the Surry nuclear plant.

3.6.2 Deterministic Considerations

The risk importance measures provide a sound basis for determining the plant risk for normal power operation and the required response to internal initiating events; however, there are other considerations which also should be incorporated into the piping segment safety significance assessment. These considerations are not directly related to the probabilistic assessments and include the segment importance for external events (seismic, fire and external flood), safety function performance during shutdown modes, the importance to design basis analysis and other accident scenarios, and operation and maintenance insights which should be taken into account. These considerations are described below.

External Events Evaluation

The importance measures calculated using the plant PSA identify the safety significance for internal events. Similar calculations for external events such as seismic, fire, and flood can not be performed unless a PSA model exists for these types of events. If a PSA model does not exist, to determine whether a segment is more or less safety-significant for an external event, the expert panel considers the segment function in mitigating the consequences of the external event, as well as the likelihood of the event.

Shutdown Risk Evaluation

A process to evaluate component importance to safe plant shutdown is used. The shutdown risk process is based upon three objectives: shutting down the reactor to the cold shutdown condition, maintaining the cold shutdown condition, and mitigating the consequences of an accident. There are five safety functions associated with the shutdown objectives: reactivity control, decay heat removal, pressure control, inventory control, and containment integrity. (Power availability should also be considered.) Each piping segment is evaluated to determine its possible importance in performing any of the safety functions. The process considers flow paths used or isolated during shutdown; when flow paths are used or isolated; the importance of component operation in performing a safety function; the length of time the plant is in a configuration that requires component operation; and the availability of other systems to provide functional redundancy. These factors should be considered when determining component safety significance to shutdown operations. In addition, the impact of

this mode of plant operation on the failure modes and causes and failure probability (lower temperatures, etc.).

Importance to Other Accident Scenarios

Piping failures which could lead to other accident scenarios, such as radioactive releases, are considered to identify any accident scenarios not previously accounted for in the safety significance assessment.

Component Maintenance and Operations Insights

Plant operation and maintenance experience may show that some piping segments have a history of design or operating issues. Information provided by the maintenance staff is used to identify any piping segments that would cause safety concerns.

Importance to Design Basis Analysis

Segment importance in the plant design basis analysis performed for the Final Safety Analysis Report, used to license the plant, is considered to identify any design basis concerns.

Other Deterministic Insights

This category is used to document important information regarding other deterministic aspects of the piping segment failure which does not readily fit into any of the above categories.

Piping Segment Worksheets

To aid the expert panel, segment information worksheets are prepared for the expert panel. Table 3.6-3 shows the format for the worksheet while Table 3.6-4 provides a description of each section of the worksheet. The worksheets are used as a mechanism for documenting and reviewing the comments and exchanges of information that are part of an expert panel process. The structure of the worksheet is based largely upon the three phases of the risk categorization process. Therefore, a completed worksheet contains all the information, calculation results, evaluations, and the risk categorization for a segment. Supplemental information is attached to each worksheet to provide a complete package for each piping segment.

Table 3.6-3
EXPERT PANEL EVALUATION
SEGMENT RANKING WORKSHEET

Section 1 System & Pipe Segment Identification
System & Segment Description:
Location/P&ID Drawing:
System Function(s):

Section 2 Risk Ranking Information		
Failure Effect on System Without Operator Action:		
Failure Effect on System With Operator Action:		
PSA Initiating Events Impact:		
PSA Containment Performance Impact:		
Conditional Core Damage Frequency due to Pressure Boundary Failure:	**Without OA**	**With OA**
Total Segment Pressure Boundary Failure Core Damage Frequency (FP * CDF_{cond})		
CDF_{pb} Importance Measure Values	**RAW** **RRW**	
Comments		

Section 3 Pressure Boundary Failure Probability	
Segment Elements (welds, tees, elbows, etc.):	
Pressure Boundary Failure Mechanism(s):	
Pressure Boundary Failure Probability	**Small Leak:** **Full Break:**
Basis for Pressure Boundary Failure Probability	
Comments	

Table 3.6-3 (cont)
EXPERT PANEL EVALUATION
SEGMENT RANKING WORKSHEET

Section 4 Indirect Effects Evaluation
Indirect Effects **(spray, flood, pipe whip, jet impingement)**
Pressure Boundary Failure Impact **on Other Systems**
Core Damage Frequency Contribution **due to Indirect Effects**

Section 5 Other Considerations		
External Events Evaluation		
Seismic:	**Fire:**	**External Flood:**
Shutdown Risk Evaluation:		
Importance to Other Accident Scenarios:		
Component Maintenance and Operation Insights:		
Importance to Design Basis Analysis:		
Other Deterministic Insights:		

Section 6 Final Risk Category	
Category: More Safety Significant	**Less Safety Significant**
Basis	

Table 3.6-4 SEGMENT RANKING WORKSHEET SECTION DEFINITIONS	
Section	Definition
SECTION 1 SYSTEM & SEGMENT IDENTIFICATION	This section contains information describing the segment, the location of the segment on plant P&IDs, and the system function(s).
SECTION 2 RISK RANKING INFORMATION	This section contains the risk categorization results developed using the plant PSA model and core damage frequency as the consequence measure. This section also contains the results of the at-power importance measure calculations.
A. Failure Effect on System Without Operator Action	This subsection identifies the failure effect from a pipe break in the defined segment without consideration of operator action, such as loss of train A pump, loss of entire system, etc.
B. Failure Effect on System With Operator Action	This subsection identifies the failure effect from a pipe break in the defined segment with operator action, such as loss of train A pump, loss of entire system, etc.
C. PSA Initiating Events Impact	A review of each initiating event (LOCAs, steamline/feedline breaks, etc.) is conducted to see if the pipe segment failure results in an initiating event.
D. PSA Containment Performance Impact	This subsection identifies pipe segments that are important to containment performance (PSA level 2 analysis), such as segments that penetrate containment or whose failure would cause a release path outside containment.
E. Conditional Core Damage Frequency due to Pressure Boundary Failure	This section contains the plant conditional core damage frequency or probability from a requantification of the PSA.
F. Total Segment Pressure Boundary Failure Core Damage Frequency	This section contains the total segment core damage frequency due to piping pressure boundary failures calculated by using the plant PSA to obtain the conditional core damage frequency and the failure probabilities (from section 3) determined for each segment.
G. CDF_{pb} Importance Measure Values	This subsection contains the PSA-power risk categorization based upon the computed importance measures (Risk Achievement Worth and Risk Reduction Worth). These can then be compared to guidelines to aid in identifying more safety-significant components.

Table 3.6-4 (Continued) **SEGMENT RANKING WORKSHEET SECTION DEFINITIONS**	
Section	Definition
I. Comments	This subsection lists any comments that are important in describing information contained in this section. This information may identify where the pipe segment is modeled in the PSA and an explanation of the importance measure results.
SECTION 3. PRESSURE BOUNDARY FAILURE PROBABILITY	This section describes the postulated pressure boundary failure, the postulated failure mechanism and the basis for the failure probability obtained for the pressure boundary failure.
A. Segment Elements	This subsection describes the segment element(s) for which the failure probability is calculated
B. Pressure Boundary Failure Mechanism(s)	This subsection identifies the failure mechanisms postulated for the failure of the pressure boundary, such as stress.
C. Pressure Boundary Failure Probability	This subsection identifies the pressure boundary failure probability calculated.
D. Basis for Pressure Boundary Failure Probability	This subsection provides a summary of the basis for the pressure boundary failure probability.
E. Comments	This subsection provides any comments association with the pressure boundary failure probability calculation.
SECTION 4. INDIRECT EFFECTS EVALUATION	This section describes any indirect effects resulting from the pressure boundary failures (such as loss of adjacent equipment).
A. Indirect Effects	This section describes the postulated mechanism for the indirect effect including spray, flood, pipe whip, jet impingement, etc.
B. Pressure Boundary Failure Impact on Other Systems	This section defines the impact on other systems of the pressure boundary failure, such as loss of a motor control center, loss of one train of solid state protection, etc.
C. Core Damage Frequency Contribution	This section identifies the core damage frequency contribution due to the indirect effect.

Table 3.6-4 (Continued)
SEGMENT RANKING WORKSHEET SECTION DEFINITIONS

Section	Definition
SECTION 5. OTHER CONSIDERATIONS	There are other considerations, such as operational issues and external events, that could affect risk categorization. Risk importance, beyond quantifiable measures, is determined by expert engineering judgement. This section documents the information provided and considered by utility experts from a variety of disciplines to completely evaluate component performance for other plant events and operating conditions.
A. External Events Evaluation	This section identifies any specific concerns with regard to mitigation of external events such as seismic, fire and flood. The expert panel considers the function in mitigating the consequences of events, as well as the likelihood of events.
B. Shutdown Risk Evaluation	Shutdown risk is based upon shutting down the reactor to the cold shutdown condition, maintaining the cold shutdown condition, and mitigating the consequences of an accident. Reactivity control, decay heat removal, pressure control, inventory control, and containment integrity are the safety functions required to meet the shutdown objectives. Power availability should also be considered. This section documents the evaluation performed for each pipe segment to determine its importance in performing any safety function during shutdown modes.
C. Importance to Other Accident Scenarios	This subsection identifies failures that could lead to other accident scenarios, such as radioactive releases.
D. Component Maintenance and Operational Insights	This subsection provides plant operation and inspection insights for segments that would cause safety or operational concerns.
E. Importance to Design Basis Analysis	This subsection identifies, if any, the segment importance to the plant design bases analysis or licensing impact (from the Final Safety Analysis Report).
F. Other Deterministic Insights	This subsection documents important information that does not fit into any of the other deterministic sections.
SECTION 6. FINAL RISK CATEGORY	This section contains the final risk category for a pipe segment and the worksheet section(s) that, primarily, provide the basis for the categorization. The information in this section is the culmination of the analysis, information, and results presented in the entire worksheet and the expert panel process.

3.6.3 Expert Panel

The primary focus of the expert panel sessions is to review all pertinent information and determine the final safety-significant category for each of the piping segments. The expert panel is responsible for the review and approval of all risk-based selection results by utilizing their expertise (including knowledge of prior inspection results, industry data, and any available stress and fracture mechanics results) and probabilistic safety assessment insights to develop the final categories of more safety-significant and less safety-significant items to be included for inservice inspection. In order to provide a comprehensive review, many of the panel members are also members of the expert panel that was established to implement the Maintenance Rule. The risk-based ISI expert panel includes expertise in the following fields:

- Probabilistic Safety Assessment
- Plant Operations
- Plant Maintenance
- Plant Engineering
- Safety Analysis

In addition, panel members with expertise in the following areas should be included:

- Inservice Inspection
- Nondestructive Examination
- Stress and Materials Considerations

At the initial expert panel session, information on each of following topics should be presented:

- Overview of risk-based inservice inspection methodology

- Goals of the study

- PSA Model: Information on the plant PSA model (scope, events analyzed, total core damage frequency, dominant contributors to core damage frequency, and Level 2 results). Information on the method(s) for modeling the segment piping failures in the PSA.

- Scope of Program: The systems included in the scope of the program should be presented and concurrence obtained.

- Importance Measures: Importance measure calculations, specifically the RAW and RRW, what each measure indicates, and how the measures will be used in the risk categorization process.

- Information Gathering and At-Power Results: Results of the information gathering and at-power importance measure calculations should be presented. This information is provided to the expert panel with an initial indication of possible safety-significant piping segments.

- Deterministic Consideration: Other considerations such as external events, shutdown risk, other accident scenarios, design basis, and component maintenance and operation insights should be discussed.

- Expert Panel Process: The performance of the expert panel in determining safety-significant piping segments, including the panel composition, participation, and expectations should be discussed. A facilitator can be used to open expert panel discussions and to aid the panel in proceeding through the information and to reach a consensus on safety-significance categorization of each piping segment.

To aid the expert panel, piping segment information worksheets and simplified drawings should be prepared for the expert panel meetings to facilitate the safety-significance categorization process. The worksheets are described in Section 3.6.2. Examples of completed worksheets are shown in Appendix B. In addition, information on piping reliability should be provided and discussed. An example is provided in Table 3.6-5.

Table 3.6-5

EMERGENCY CORE COOLING SYSTEM (ECCS)

PIPING RELIABILITY REMARKS

System Design and Operation

Segment 0 is the RWST suction line to the RHR pumps. The remaining segments are portions of the safety injection pathways from the accumulators, high head pumps or RHR pumps through to the RCS cold legs. Segments 5 through 8 are ASME Class 1; the others are ASME Class 2.

Except for the RWST suction line, the piping is of heavy wall type 316 stainless steel construction. The 24" suction line is type 304 standard wall near the tank and heavy wall for the buried yard piping.

Except in accident response scenarios, the system is operated only during system test and RHR shutdown cooling.

Segment 0 sees no significant cyclic loading. The system normal operating cycles for the remaining segments are primarily due to RHR operation, and the temperature range is about 250°F.

The accumulator lines (ECCS-5 -> ECCS-8) see a static pressure head with no flow during normal operation. They see static bending loads once per RHR and plant heatup cycle.

During test of the accumulators there is transient loading as the lines discharge very quickly. During accumulator discharge following a LOCA, there may be some RCS chugging which may impose transient loading on the accumulator lines.

Possible Failure Modes

Segment 0 could be damaged by an earthquake severe enough (beyond the design basis) to displace or deform the RWST near the piping connection, or the buried portion could be subject to corrosion despite being coated.

For eight segments in the RHR flow path the most likely failure mode is thermal fatigue causing undetected crack growth. Excess thermal fatigue might be caused by snubbers locking up. Back leakage through check valve 8948 combined with bonnet leakage through an upstream check could cause thermal cycling.

No known failures.

Failure Scenario and Break Probabilities

Possible pipe break failure on demand during LOCA accident due to system startup transient loading or earthquake, causing undetected crack to propagate into full break.

For the stainless steel piping material of this system it is likely that leakage would precede a break. Leakage in segments ECCS-5 -> 8 would be detected by a drop in accumulator level.

Pipe break failure probabilities are calculated to be very low by probabilistic fracture mechanics program, and this result is confirmed by experience and judgment.

3.6.4 Representative WOG Plant Results

The results of the categorization of the piping segments for the representative WOG plant are shown in Table 3.6-6. A summary of the more safety-significant piping segments and the basis for the determination is provided in Table 3.6-7. A comparison of the results of the expert panel safety significance determination and the risk importance measure determination are shown in Table 3.6-8.

This comparison shows good agreement between the piping segments chosen by the expert panel and those identified in the various sensitivity studies. The piping segments that were not identified as more safety-significant by the expert panel were those piping segments with RRW values below the 1.005 threshold value. The piping segments chosen by the expert panel show that the expert panel incorporates deterministic considerations.

3.7 STRUCTURAL ELEMENT SELECTION

The risk-based selection process includes assessments and evaluations of the piping structural elements in each of the more-safety-significant piping segments. These structural elements include the following examination items:

(1) all piping welds, including those to nozzles, valves and fittings such as elbows, [illegible] reducers, branch connections, and safe ends

(2) areas and volumes of base material and examination zones, such as weld counterbore areas and fitting material of the items given in (1), as appropriate.

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

All of the piping structural elements have already been evaluated in the risk-based piping segment selection process. Each piping segment has been reviewed, as discussed in Section 3.5, to select the most likely failure location(s) in each segment and to calculate a probability of failure for each of those locations. These values have been used in the risk-importance calculations for each piping segment, as discussed in Section 3.6, which have been reviewed by the expert panel along with other deterministic and operational insights to select the more safety-significant piping segments.

Table 3.6-6

NUMBER OF SEGMENTS DEFINED FOR EACH SYSTEM AND MORE SAFETY-SIGNIFICANT SEGMENTS DEFINED BY EXERT PANEL

SYSTEM	NUMBER OF SEGMENTS	MORE SAFETY SIGNIFICANT SEGMENTS
BDG (SG Blowdown)	4	0
CCE (CHS Cool)	2	0
CCI (SI Cool)	with SIH	--
CCP (CCW)	14	4
CHS (CVCS)	23	4
CNM (Condensate)	with FWS	--
DTM (Turbine Plant Drains)	with MSS	--
ECCS	9	1
EGF (DG Fuel)	4	0
FWA (Aux Feed)	15	5
FWS (Feedwater)	19	0
HVK (Control Bldg. Chilled Water)	1	0
MSS (Main Steam)	30	0
QSS (Quench)	5	1
RCS	66	55
RHS (RHR)	with SIL	--
RSS (Recirc)	11	1
SFC (Fuel Pool)	4	0
SIH (HPI)	10	4
SIL (LPI)	13	5
SWP (SW)	29	16
TOTAL	259	96 (37%)

Table 3.6-7 SUMMARY OF MORE SAFETY-SIGNIFICANT SEGMENTS		
Segment IDs	**Description of Segments**	**Basis for Safety Significance**
CCP-1, 2, 4, 5	Suction and discharge lines for the CCP pumps and heat exchangers	Loss of the only operable train of CCP during shutdown (CCP cools RHR heat exchangers)
CHS-3	Charging pump suction and discharge piping, mini flow lines	Loss of all charging. loss of RWST outside containment at-power and shutdown, reactor trip
CHS-5	Charging to RCP seal injection	Loss of all charging. loss of RWST outside containment at-power and shutdown, reactor trip
CHS-7	Normal charging line	Loss of all charging. loss of RWST outside containment at-power and shutdown, reactor trip
CHS-23	Cold leg safety injection	Loss of RWST outside containment at-power and shutdown
ECCS-0	Piping between RWST and splits to LPSI, HPSI and charging	Loss of RWST outside containment at-power and shutdown
FWA-7	Piping from DWST through turbine driven FWA pump to SG feed split	Loss of DWST within short period of time. No operator action for aligning CST credited
FWA-12,14,16,18 Note: FWA-13, 15, 17, 19 are less safety-significant, but the feedwater nozzles should be in the inspection program	From check valves to cavitating venturi before SG	Loss of DWST within short period of time. No operator action for aligning CST credited
QSS-2	V32 to containment boundary past MOV34A&B and to V42 & V43	Loss of RWST outside containment at power and shutdown
RCS-1, 8, 9, 16, 23	Hot leg from vessel to loop isolation valve	High failure probability, high consequences from a large loss of coolant accident (LOCA)
RCS-2, 10, 17, 24	Hot leg from loop isolation valve to steam generator	High failure probability, high consequences from a large LOCA
RCS-3, 11, 18, 25	Crossover leg from steam generator to RCP	High failure probability, high consequences from a large LOCA
RCS -4, 12, 19, 26	From crossover leg tee to loop fill line isolation valve	High consequences from a medium LOCA, unisolable break
RCS-5, 13, 20, 27	Cold leg from RCP to loop isolation valve	High failure probability, high consequences from a large LOCA
RCS-6, 14, 21, 28	Cold leg from loop isolation valve to reactor vessel	High failure probability, high consequences from a large LOCA
RCS-7, 22, 29, 54	ECCS cold leg injection line from last check valve to cold leg	High consequences from a large LOCA

Table 3.6-7 (cont) SUMMARY OF MORE SAFETY-SIGNIFICANT SEGMENTS		
Segment IDs	**Description of Segments**	**Basis for Safety Significance**
RCS-15, 49, 60, 66	Charging line from last check valve to cold leg	High consequences from a large LOCA
RCS-30	Pressurizer surge line	High consequences from a large LOCA
RCS-31, 32, 33	From pressurizer to pressurizer safety valves	High consequences from a medium LOCA, unisolable break
RCS-34	From pressurizer to PORV block valves	High consequences from a medium LOCA, unisolable break
RCS-35, 36	Lines between PORV block valves and PORVs	High consequences from a medium LOCA
RCS-38	From loop drain line isolation valves to crossover leg and cold leg	High consequences from a medium LOCA, unisolable break
RCS-40	Pressurizer spray lines from last valves to pressurizer	High consequences from a medium LOCA, unisolable break
RCS-42, 61	Hot leg high/low pressure safety injection line from last isolation valve to hot leg	High consequences from a large LOCA
RCS-43, 51, 56, 62	Cross-connect line from hot loop isolation valve to cold leg loop isolation valve	High failure probability (43, 56 only), high consequences from a large LOCA
RCS-45, 53	Pressurizer spray line	High consequences from a medium LOCA, unisolable break
RCS-47, 64	Charging line from last check valve to cold leg	High consequences from a medium LOCA, unisolable break
RCS- 50,55	From hot leg to high/low pressure safety injection check valve	High consequences from a LOCA, unisolable break
RCS-58	Normal letdown line to first isolation valve	High consequences from a medium LOCA, unisolable break
RSS-11	Cross-connect between SIL and SIH pumps	Loss of all charging and loss of RWST at power and shutdown
SIH-1	From RWST isolation MOV to SI pump suction isolation MOV	High consequence, loss of RWST outside containment
SIH-2, 3	From SI pump suction isolation MOV to SI pump discharge isolation MOV	Loss of RWST outside containment at power and shutdown
SIH-4	From high pressure SI header isolation MOVs to injection line check valves	Loss of RWST, both SI pumps injecting to break location
SIL-1, 2	From RWST isolation MOV though RHR pump and HX to RHR discharge isolation MOVs	High consequence, loss of RWST outside containment, loss of RHR (shutdown impact)

Table 3.6-7 (cont)

SUMMARY OF MORE SAFETY-SIGNIFICANT SEGMENTS

Segment IDs	Description of Segments	Basis for Safety Significance
SIL-3	From RHR discharge header isolation MOVs to hot leg injection isolation MOV	High consequence, loss of RWST outside containment (no shutdown impact)
SIL-4, 5	From cold leg injection isolation MOV to injection line check valves	High consequence, loss of RWST inside containment, loss of RHR (shutdown impact)
SWP-1, 2, 3, 4, 26, 27, 28, 29	Service water pump discharge	Importance of one SW pump train during shutdown (results in loss of operating RHR train and a loss of cooling to the Diesel Generator)
SWP-5, 6, 7, 8	Service water pump discharge	Loss of the only operable SW pump train during shutdown (results in loss of operating RHR train and a loss of cooling to the Diesel Generator)
SWP-15, 22	Service water pipe to CCE heat exchangers	Loss of charging, reactor trip
SWP-23,25	Service water through CCP heat exchangers to circ. water discharge	Loss of the only operable SW pump train during shutdown (results in loss of operating RHR train and a loss of cooling to the Diesel Generator)

Table 3.6-8
COMPARISON OF EXPERT PANEL AND RISK CALCULATIONS
IN SAFETY SIGNIFICANCE DETERMINATION
SEGMENTS WITH RRW > 1.0

SYSTEM	EXPERT PANEL	BASE MODEL WITHOUT OPERATOR ACTION	WITH OPERATOR ACTION	USING LEAK PROBABILITIES	USING ACTUAL FAILURE PROBABILITIES	INCREASING FAC FOR MSS,FWS & SWP
BDG	NONE	NONE	NONE	NONE	NONE	NONE
CCE	NONE	NONE	NONE	NONE	NONE	NONE
CCP	CCP-1,2,4.5	NONE	NONE	NONE	NONE	NONE
CHS	CHS-3,5,7,23	CHS-3,5,7,23	NONE	NONE	CHS-3	CHS-3,5,7,23
ECCS	ECCS-0	ECCS-0,5,6,8	ECCS-0,1,2,3, 4,5,6,7,8	NONE	NONE	ECCS-0
EG[illegible]	NONE	NONE	NONE	NONE	NONE	NONE
FWA	FWA-7, 12,14,16,18	FWA-1,4,7, 12,14,16,18	FWA-7, 12,14,16,18	FWA-7	FWA-1,4,7	FWA-7,12,14,16,18
FWS	NONE	NONE	NONE	NONE	NONE	NONE
HVK	NONE	HVK-1	HVK-1	NONE	HVK-1	NONE
MSS	NONE	NONE	NONE	NONE	NONE	NONE
QSS	QSS-2	QSS-2	QSS-2	NONE	NONE	QSS-2
RSS	RSS-11	RSS-11	NONE	NONE	NONE	RSS-11
SFC	-	-	-	-	-	-
SIH	SIH-1,2,3,4	SIH-1,2,3,4	NONE	SIH-2,3	SIH-2,3	SIH-1,2,3,4

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Table 3.6-8 (cont.)
COMPARISON OF EXPERT PANEL AND RISK CALCULATIONS IN SAFETY SIGNIFICANCE DETERMINATION
SEGMENTS WITH RRW > 1.0

SYSTEM	EXPERT PANEL	BASE MODEL WITHOUT OPERATOR ACTION	WITH OPERATOR ACTION	USING LEAK PROBABILITIES	USING ACTUAL FAILURE PROBABILITIES	INCREASING FAC FOR MSS, FWS & SWP
SWP	SWP-1,2,3,4,5, 6,7,8,15,22,23,25,26, 27,28,29	SWP-1,2,3,4,5, 6,7,8,23,25,26,27, 28,29	SWP-1,2,3,4,5, 6,7,8,15,22, 26,27,28,29	SWP-1,2,3,4,5, 6,7,8,23,25,26,27, 28,29	SWP-1,2,3,4, 6,8,26,27,28,29	SWP-1,2,3,4,5, 6,7,8,15,22,23,25, 26,27,28,29
RCS	RCS-1,2,3,4,5, 6,7,8,9,10,11,12, 13,14,15,16,17, 18,19,20,21,22, 23,24,25,26,27, 28,29,30,31,32, 33,34,35,36,38, 40,42,43,45,47, 49,50,51,53,54, 55,56,58,60,61, 62,64,66	RCS-1,2,3,5, 6,8,9,10,11,13,14, 16,17,18,20,21,23, 24,25,27,28,43,56	RCS-1,2,3,5, 6,8,9,10,11,13,14, 16,17,18,20,21,23, 24,25,27,28,43,56	RCS-11,17,18,25	RCS-1,2,3,5, 6,8,9,10,11,13,14, 16,17,18,20,21,23, 24, 25,27,28,43,56	RCS-18

*Segment with RRW values > 1.0 from the base model without operator action that were not chosen to be more safety-significant by the expert panel include:

Segment	RRW
ECCS-5	1.001
ECCS-6	1.001
ECCS-8	1.001
FWA-1	1.002
FWA-4	1.002
HVK-1	1.002

In reviewing the piping segment selection information, each of the more-safety-significant piping segments has one of the two following conditions:

(1) The segment has a high consequence of failure, but no known failure mechanism is present, i.e., failure probability < 1.0 E-8

(2) The segment has a high consequence of failure, but a potential failure mechanism is expected to exist, i.e., failure probability > 1.0 E-8

A subpanel (i.e., component ISI team or focused structural element expert panel) with the following expertise:

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

performs the review of the more-safety-significant segments to select the important structural elements for inspection. Figure 3.7-1 displays how this expertise and information is brought together in the structural element selection process.

For Condition (1), the subpanel must select a minimum of one location for inspection in the segment based on the factors that may affect the integrity of the piping segment in order to insure that no unexpected mechanism is present.

For Condition (2), the subpanel must select the limiting location(s) that resulted in the segment being more-safety-significant, as a minimum. In addition, the subpanel must review the remaining structural elements in the segment to capture any other locations that may also be potential failure sites, thereby contributing to risk. SRRA sensitivity studies may be performed to assist in this determination.

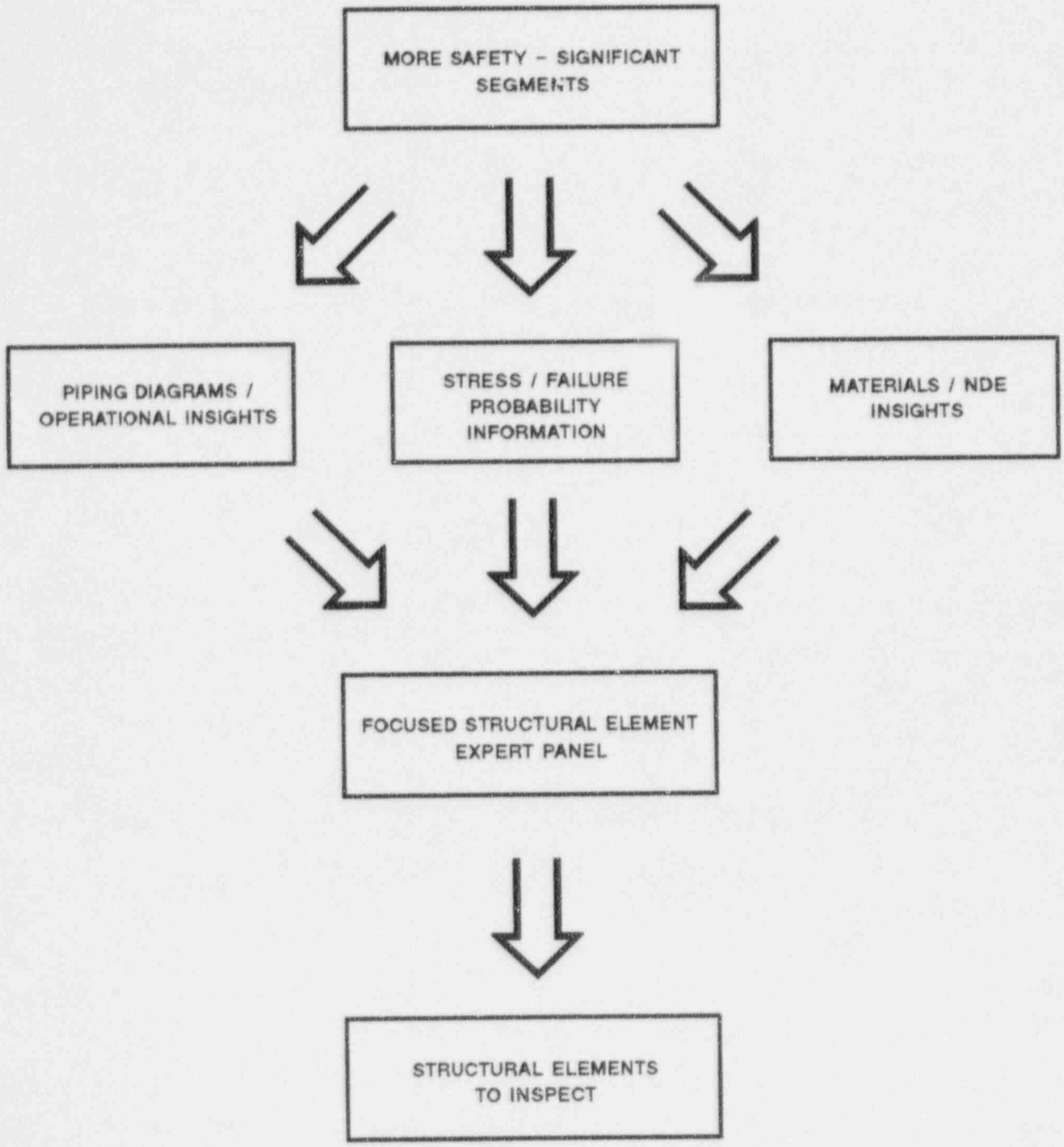

Figure 3.7-1 WOG Structural Element Selection Process

Essentially, the subpanel is further evaluating the information that has been previously generated to verify and come to a consensus, using sound engineering judgement and discussion, that the most likely locations for potential structural failure within the more safety-significant piping segments are identified and documented. The following process is used by the subpanel to perform this evaluation:

(1) Simplified P&IDs showing the segment boundaries and selected break location(s) are reviewed along with piping isometrics, the inputs/outputs of the SRRA worksheets, and any materials/NDE insights. RRW values for the segment and the failure probabilities for the element(s) are also reviewed with the following questions being asked:

- Does the subpanel believe that the location(s) identified have the highest failure potential within the segment?

- Does the subpanel believe that the failure mode(s) have been adequately identified based on the available information?

- Are there any additional insights, such as industry failure history or plant-specific conditions, that need to be addressed?

(2) Based on the postulated failure mode(s), the subpanel reviews the selected NDE from the SRRA worksheets and determines the appropriate visual examination, NDE method, or continuous monitoring technique that should be used at the selected structural element location(s). (Guidance for this determination is provided later in Section 4.1).

(3) The results of the subpanel assessments and conclusions are documented, and presented to the full expert panel for final review and approval.

For example, only one segment, ECCS-0, is considered to be more-safety-significant in the emergency core cooling system. The selection of this segment is primarily based on the consequence of failure because the selected element SRRA failure probability was less than 1.0 E-08. The subpanel reviewed

the structural elements within the segment and concurred that the element location that was selected is considered to have the highest failure potential. The location of concern is the base metal of a 24" pipe at ground surface that may be subjected to cracking because of outside diameter corrosion and external loads. Since the area being examined at this selected element location is base material; not currently addressed in ASME Section XI, Figure 3.7-2 has been developed to identify the area to be inspected by VT-2 and eddy current examination.

QSS-2 is the only segment that is considered to be more-safety-significant in the quench spray system. The selection of this segment is primarily based on consequence of failure. However, the failure probabilities in this segment were based on prior SRRA evaluations of two locations, both of which are less than 1.0E-8. The subpanel reviewed all the elements in the segment and concurred that the two selected locations have the highest failure potential. Both locations are pipe-to-elbow welds in the 12" pipe that may be subjected to cracking from vibrational fatigue caused by pump operation. Both UT and VT-2 examinations are recommended for these two locations.

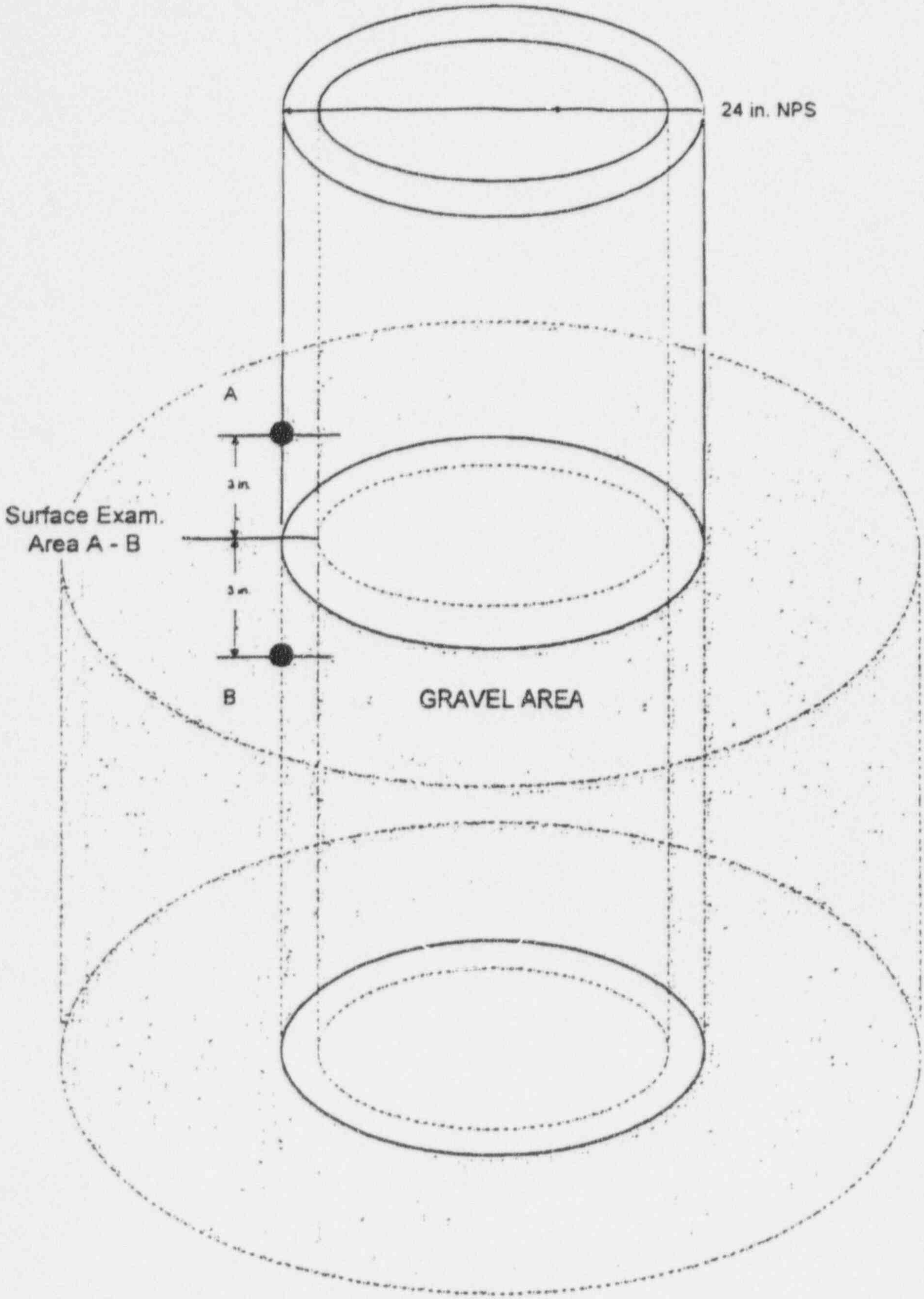

Figure 3.7-2 Base Metal Examination Location for ECCS-0

SECTION 4
INSPECTION REQUIREMENTS

Inservice examinations and system pressure tests are performed during either system operation or plant outages, such as refueling outages or maintenance outages. The required examinations are completed during each inspection interval. Currently the interval is 10 years. The inspections are generally distributed across periods such that one third of the inspections are conducted in each period.

4.1 MORE SAFETY-SIGNIFICANT LOCATIONS

Once the piping structural elements are identified for the more safety-significant piping segments, the areas and/or volumes of concern for each more safety-significant piping structural element are determined. This determination is based on the postulated failure modes and the configuration of each piping structural element. The examination requirements may include areas or volumes such as the counterbore of a valve body, outside the normal weld inspection area.

The examination methods or monitoring techniques for these identified areas and/or volumes of concern should be based on the guidance provided in Table 4.1-1. If an examination method can not be applied to a structural element due to some limitations, alternative methods that provide assurance of structural integrity may be considered. The basis for the use of these alternative methods should be provided.

4.2 LESS SAFETY-SIGNIFICANT LOCATIONS

The less safety-significant locations do not receive as rigorous of an examination in the inservice inspection program. However, leakage from these piping segments would be identified by general operational walkdowns and pressure test requirements.

Pressure test requirements and VT-2 visual examinations should continue to be performed on all Class 1, 2, and 3 systems regardless of whether the segments contain locations that have been identified as more or less safety-significant.

Table 4.1-1

GUIDANCE FOR VISUAL EXAMINATION METHODS, MONITORING TECHNIQUES, AND NDE METHODS ASSOCIATED WITH POSTULATED FAILURE MODES

Potential Piping Inside Surface Initiated Flaws or Relevant Conditions (1)		
Piping Structural Elements	Postulated Failure Modes	Suggested NDE Method, Monitoring Technique or Visual Exam Method
Butt Welds (2) *≥ .237 in. Nominal Wall Thickness for Piping ≥ NPS 2*	Cracking *Thermal Fatigue, Mechanical Fatigue, or Corrosion*	Ultrasonic Examination (3) or Continuous Temperature and/or Stress Monitoring *For Thermal Fatigue*
Butt Welds (2) *< .237 in. Nominal Wall Thickness*	Cracking *Thermal Fatigue, Mechanical Fatigue, or Corrosion*	Radiographic Examination (4) or Continuous Temperature and/or Stress Monitoring *For Thermal Fatigue*
Butt Welds (2) *Essentially Limited to Raw Water Cooling Systems*	FAC *Microbiologically Influenced Corrosion, Heat Affected Zone Washout, and General Erosion*	Combinations of Ultrasonic Examination (5) and Radiographic Examination (4)
Branch Connection Welds *Branch Pipe ≤ NPS 2 Connected to Main Run Pipe ≤ NPS 4*	Cracking *Thermal Fatigue, Mechanical Fatigue, Corrosion, or Vibrational Fatigue (6)*	Radiographic Examination (4) or Continuous Temperature and/or Stress Monitoring *For Thermal Fatigue*
Branch Connection Welds *Branch Pipe > NPS 2 Connected to ≥ .237 in. Nominal Wall Thickness Main Run Pipe > NPS 4*	Cracking *Thermal Fatigue Mechanical Fatigue, Corrosion, or Vibrational Fatigue (6)*	Ultrasonic Examination (3) *Main Run Pipe Base Material Adjacent to The Weld* and Radiographic Examination (4) *Weld and Branch Fitting Base Material Adjacent to The Weld to The Extent Possible* or Continuous Temperature and/or Stress Monitoring *For Thermal Fatigue*
Socket Welds *≥ .237 in. Nominal Wall Thickness*	Cracking *Thermal Fatigue Mechanical Fatigue, Corrosion, or Vibrational Fatigue (6)* FAC *General Wastage from Flow or Oxidation*	Radiographic Examination (4) Supplemented By Ultrasonic Examination (3) *Pipe Base Material Adjacent to The Weld* or Continuous Temperature and/or Stress Monitoring *For Thermal Fatigue*

Table 4.1-1 (cont)

GUIDANCE FOR VISUAL EXAMINATION METHODS, MONITORING TECHNIQUES, AND NDE METHODS ASSOCIATED WITH POSTULATED FAILURE MODES

Piping Structural Elements	Postulated Failure Modes	Suggested NDE Method, Monitoring Technique, or Visual Exam Method
Potential Piping Inside Surface Initiated Flaws or Relevant Conditions (1)		
Socket Welds *< .237 in. Nominal Wall Thickness*	Cracking *Thermal Fatigue Mechanical Fatigue, Corrosion, or Vibrational Fatigue (6)* FAC *General Wastage from Flow or Oxidation*	Radiographic Examination (4) or Continuous Temperature and/or Stress Monitoring *For Thermal Fatigue*
Pipe Runs or Areas *Base Material and Welds*	FAC *General Wastage from Flow or Oxidation*	Ultrasonic Examination (5), Radiographic Examination (4), or Infra-Red Thermography (7)
Pipe Fittings *Such as Elbows, Tees, Reducers, or Expanders*	FAC *General Wastage from Flow or Oxidation*	Ultrasonic Examination (5), Radiographic Examination (4), or Infra-Red Thermography (7)
Potential Piping Outside Surface Initiated Flaws or Relevant Conditions		
Piping Structural Elements	Postulated Failure Modes	Suggested NDE Method, Monitoring Technique, or Visual Exam Method
All Piping Structural Elements *Such as Butt Welds, Branch Connection Welds, Socket Welds, Pipe Runs, or Pipe Fittings*	Cracking *Thermal Fatigue Mechanical Fatigue, Corrosion, or Vibrational Fatigue (6)*	Liquid Penetrant Examination or Eddy Current Examination *For Austenitic Stainless Steels, Non-Ferritic High Alloy Materials, and Dissimilar Metal Welds* or Magnetic Particle Examination or Eddy Current Examination *For Carbon Steel, Ferritic Low Alloy Steel Materials and Welds*
All Piping Structural Elements *Such as Butt Welds, Branch Connection Welds, Socket Welds, Pipe Runs, or Pipe Fittings*	Corrosion *General Wastage from Oxidation*	Visual, VT-3 Examination (8)

Table 4.1-1 (cont)

GUIDANCE FOR VISUAL EXAMINATION METHODS, MONITORING TECHNIQUES, AND NDE METHODS ASSOCIATED WITH POSTULATED FAILURE MODES

NOTES:

(1) Inside surface examinations of piping structural elements subject to cracking may be performed if they become accessible in lieu of the suggested volumetric examinations of this table. Examination methods such as liquid penetrant examination, eddy current examination, or magnetic particle examination for appropriate materials may be used. For piping structural elements subject to FAC, a general VT-3 visual examination may be performed from the inside surface of the piping, but it may be necessary to supplement this general visual examination with other examination methods to determine the extent of the erosion or corrosion.

(2) Butt welds include circumferential welds and longitudinal welds. The examination methods suggested for these welds include methods for welds of all materials, dissimilar metal welds, or portions thereof except for those welds that are made from austenitic cast stainless steel materials. Radiographic examination should be used for welds that include austenitic cast stainless steel materials.

(3) An ultrasonic angle beam examination sensitive to flaws initiating at the inside diameter surface of a weld or heat affected zone should be used.

(4) Radiographic examination is a sensitive examination for identifying flaws parallel to the radiation beam used in the technique. The method is good for the detection of pits, slag, and thermal fatigue cracks. Intergranular stress corrosion cracking, stress corrosion cracking, and off angle cracks are not reliably detected with this method. This examination method provides an accurate plan view for the location of flaws that it can detect and is extremely helpful used in conjunction with ultrasonic examination to evaluate localized areas of pitting, flow erosion, or microbiologically influenced corrosion attacks.

(5) An ultrasonic straight beam examination is used here for accurate measurements of material thickness. This method is used to assess erosion/corrosion material loss.

(6) Cracking resulting from vibrational fatigue is not usually detectable by NDE methods prior to leaking. Guidance for assessment of vibrational fatigue conditions may be found in Part 3 of the ASME OM-S/G-1990 GUIDE.

(7) Infra-red thermography is a useful examination method for overall FAC assessments to locate general areas of wall loss in steam or hot fluid systems. This method should be combined with ultrasonic examination or radiographic examination for accurate wall loss measurements.

(8) This general VT-3 visual examination method is good for location of general wastage from oxidation, but if severe oxidation is identified other examination methods may have to be used to quantify the amount of material loss.

4.3 COMPARISON OF RESULTS TO CURRENT ASME XI INSPECTION LOCATIONS

This section discusses the comparison of the results of the risk-based process to the current ASME Section XI piping inspection locations.

4.3.1 Comparison of Examination Locations

Table 4.3-1 provides a comparison of the structural element/location selections by system for the representative WOG plant. The risk-based ISI program results are compared against the existing ISI program weld selections based on the 1989 Edition of the ASME Code Section XI requirements.

The first column of the table represents the systems that were evaluated under the risk-based ISI program. This list is also shown in Table 3.2-1 and includes all the ASME Code Class 1, 2, and 3 piping systems of the existing ISI program, piping systems modeled in the PSA, and various balance of plant (non-nuclear Code Class) systems.

The second column of the table identifies the piping segments determined to be more safety-significant by the expert panel previously shown in Table 3.6-6. These more safety-significant piping segments include all the piping structural elements that were evaluated for inclusion in the risk-based ISI program by the expert panel.

The third column divides the number of the structural elements selected for examination by the expert panel into each of the applicable ASME Code Classifications for each system. This column shows the number of elements that were selected for examination in accordance with the risk-based ISI program within the ASME Code Class 1, 2, and 3 piping systems, and no exemptions were applied from IWB-1220, IWC-1220, or IWD-1220 of Section XI.

No element selections were determined to be applicable outside the existing ASME Code Class boundaries at Millstone Unit 3, but this may not be the case at all plants that apply this process. Section XI currently addresses only weld selections, and under a risk-based ISI program, this may not always be the case. Since the process identifies the segments of piping that are more safety-significant in relation to their possible failure affecting core damage, the use of existing Section XI exemptions and examination criteria has been shown at Millstone Unit 3 not to be appropriate. Additionally, the

Table 4.3-1

MILLSTONE UNIT 3 PRELIMINARY STRUCTURAL ELEMENT SELECTION RESULTS AND COMPARISON TO ASME SECTION XI 1989 EDITION REQUIREMENTS

SYSTEMS EVALUATED	MORE-SAFETY SIGNIFICANT SEGMENTS	RISK-BASED ISI PROGRAM MORE-SAFETY-SIGNIFICANT STRUCTURAL ELEMENTS			ASME SECTION XI ISI PROGRAM 1989 EDITION EXAMINATION CATEGORY WELD SELECTIONS			
		CLASS 1	CLASS 2	CLASS 3	B-F	B-J	C-F-1	C-F-2
BDG (SG Blowdown)	0	-	-	-	-	-	-	-
CCE (CHS Cool)	0	-	-	-	-	-	-	-
CCI (SI Cool)	with SIH	-	-	-	-	-	-	-
CCP (CCW)	4	0	0	5	0	0	0	0
CHS (CVCS)	4	0	6	0	0	9	10	0
CNM (Condensate)	with FWS	-	-	-	-	-	-	-
DTM (Turbine Plant Drains)	with MSS	-	-	-	-	-	-	-
ECCS (1)	1	0	1	0	0	0	0	0
EGF (DG Fuel)	0	-	-	-	-	-	-	-
FWA (Aux Feed)	5	0	8 (2)	1	0	0	0	3
FWS (Feedwater)	0	0	0	0	0	0	0	41
HVK (Control Bld Chill)	0	-	-	-	-	-	-	-
MSS (Main Steam)	0	0	0	0	0	0	0	32
QSS (Quench spray)	1	0	2	0	0	0	64	0

Table 4.3-1 (cont)

MILLSTONE UNIT 3 PRELIMINARY STRUCTURAL ELEMENT SELECTION RESULTS AND COMPARISON TO ASME SECTION XI 1989 EDITION REQUIREMENTS

SYSTEMS EVALUATED	MORE-SAFETY SIGNIFICANT SEGMENTS	RISK-BASED ISI PROGRAM MORE-SAFETY-SIGNIFICANT STRUCTURAL ELEMENTS			ASME SECTION XI ISI PROGRAM 1989 EDITION EXAMINATION CATEGORY WELD SELECTIONS			
		CLASS 1	CLASS 2	CLASS 3	B-F	B-J	C-F-1	C-F-2
RCS	55	67 (3)	0	0	22	318	0	0
RHS (RHR)	with SIL	-	-	-	-	-	-	-
RSS (Recirc)	1	0	1	0	0	0	23	0
SFC (Fuel Pool)	0	-	-	-	-	-	-	-
SIH (HPI)	4	0	4	0	0	57	28	0
SIL (LPI)	5	0	6	0	0	40	106	0
SWP (SW)	16	0	0	18 (3)	0	0	0	0
TOTAL (4)	96	67	28	24	22	424	231	76

NOTES:
(1) Section XI weld selections are included in the SIH and SIL systems.
(2) Includes 4 Feedwater Pipe to Nozzle welds that were not determined to be More-Safety-Significant.
(3) Eight RCS and 4 Service Water More-Safety-Significant elements/segments will require VT-2 exams only.
(4) Total RBI Elements Requiring NDE = 107   Total Section XI Welds = 753   **86% REDUCTION**

following specific information about some of these element selections is provided to show that, under a risk-based ISI program, the current Section XI requirements may not be applicable to the elements selected for examination:

- for the Chemical and Volume Control System (CHS), six Class 2 elements are shown to have been selected for examination under the risk-based ISI program. Of these six elements, five are currently exempt from NDE by Section XI because of their pipe sizes under IWC-1220;

- the element selected for examination under the Class 2 column of the Emergency Core Cooling System (ECCS), is not a weld location, but is limited to base metal and is identified in Figure 3.7-2;

- in the Auxiliary Feedwater System (FWA), the Class 3 element that was selected for examination is located on a line that is currently exempt from NDE by pipe size under IWD-1220;

- in the Low Pressure Safety Injection System (SIL), one of the six Class 2 elements selected for examination is also exempt from NDE by pipe size under IWC-1220; and

- for the Service Water System (SWP), selected Class 3 elements, two of the 18 selected are also exempt from NDE by pipe size under IWD-1220.

The fourth column shows the current weld selections under the requirements of the existing Millstone Unit 3 ISI program for Class 1 and 2 piping. These selections are determined under the requirements of Table IWB-2500-1 for Class 1 piping, Examination Categories B-F – Pressure Retaining Dissimilar Metal Welds and B-J – Pressure Retaining Welds in Piping; and Table IWC-2500-1 for Class 2 piping, Examination Categories C-F-1 – Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping and C-F-2 – Pressure Retaining Welds in Carbon or Low Alloy Steel Piping. For Class 3 piping, there are no current requirements to examine welds, but the piping itself receives system pressure tests. For purposes of identifying Class 3 piping subject to examination, the rules of Table IWD-2500-1, Examination Category D-A under the 1992 Edition of ASME Section XI, have been used.

Table 4.3-1 shows that 119 elements were selected for some type of examination under the Millstone Unit 3 risk-based ISI program. 107 of these elements will receive some type of NDE, Vibration Monitoring, or ID Visual VT3 examination. All the remaining elements in the risk-based ISI program and those currently included in the Section XI ISI program will continue to receive Visual VT-2 examinations during system pressure tests.

4.3.2 Risk/Safety Evaluation

Figure 4.3-1 shows a comparison of the core damage frequency being addressed by examination of the 119 structural elements in the risk-based ISI program versus the 753 weld locations that are examined per current ASME Section XI requirements. Examination of the current ASME Code weld locations addresses a CDF of 1.00E-08/yr (44%) while examination of the risk-based ISI structural elements addresses a CDF of 2.25E-08/yr (98%) for pressure boundary piping failures (out of a total piping CDF of 2.28E-08/year). Thus, safety is enhanced with far less locations being inspected.

This figure shows the comparison by the systems as defined in the risk-based program. For example, Table 4.3-1 shows no risk-based ISI locations for the FWS system, but it shows ISI locations for current ASME Section XI requirements. However, because of the system definition used in the risk-based ISI program, several locations classified under FWS in ASME Section XI are the same as those classified in the FWA system under the risk-based ISI program (piping that is common to both the FWA and FWS systems was assigned to the FWA system in the risk-based program).

This comparison also assumes 100% effectiveness in detection of precursors to failures for both the Section XI and risk-based ISI locations in the more safety-significant segments. Credit for leak testing in finding these precursors by either program in both the more safety-significant and less safety-significant piping segments is not taken in this evaluation.

The total piping core damage frequency is a small fraction of the total plant core damage frequency of 5.87E-05/yr. Examination of the plant piping at the risk-based locations, however, will verify that the risk of piping pressure boundary failure remains a small contributor to total risk as the unit ages over its licensed life.

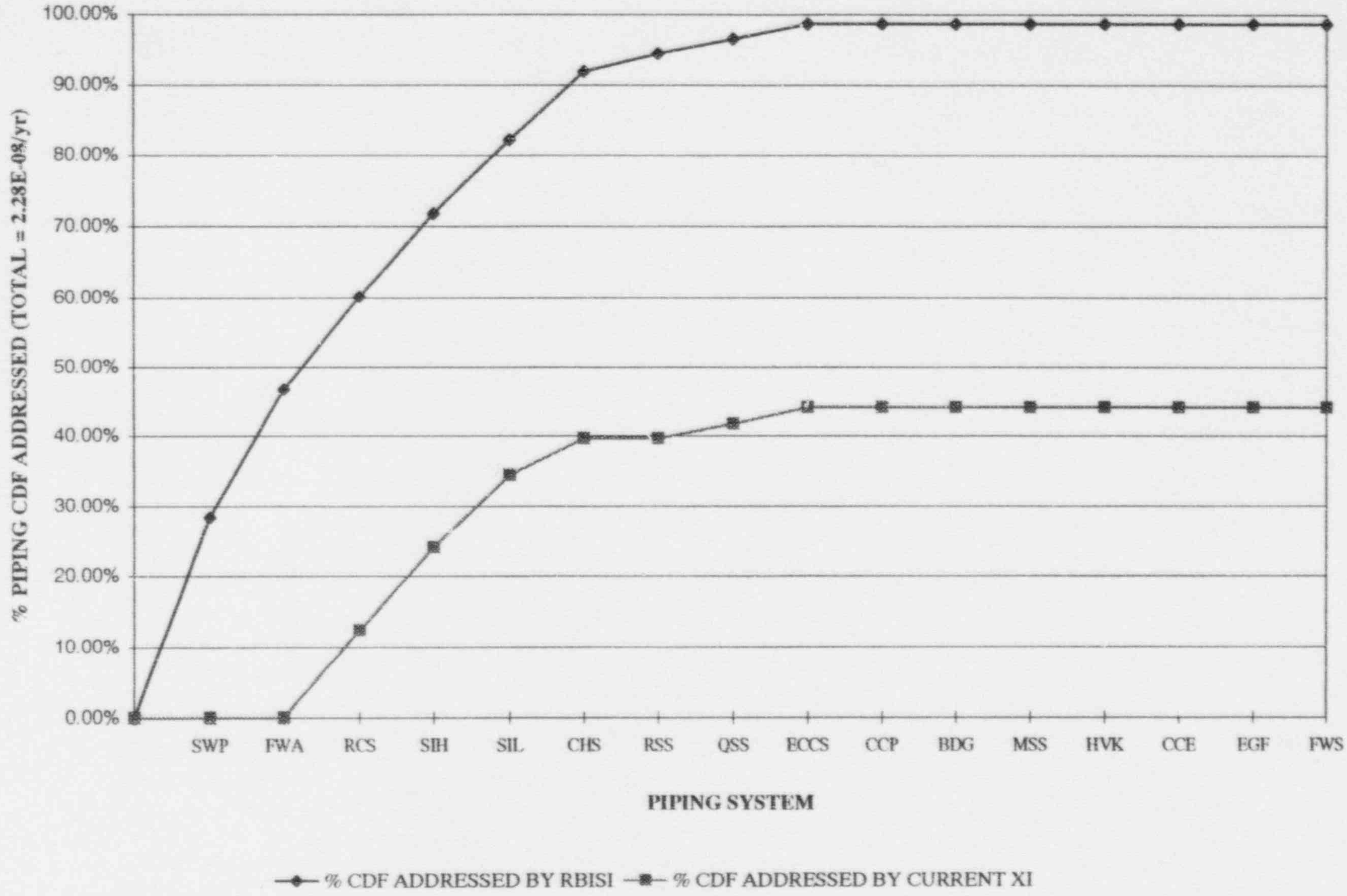

Figure 4.3-1. Comparison of CDF Results on a Piping System Level

4.3.3 Cost-Benefit Evaluation

Northeast Utilities has provided estimated savings from implementation of a risk-based inservice inspection program to the piping systems at Millstone Unit 3 in the Supplemental Information enclosed within this topical report. This section builds on this information to provide an indication of the cost-benefit for all WOG member plants.

An estimated savings of $332,000 per outage in direct inspection related costs has been identified for Millstone Unit 3. A savings of 15 person-rem per outage has also been estimated for inspection of Millstone Unit 3 piping using a risk-based approach.

The Westinghouse Owners Group has established estimated standard cost factors for parameters that are impacted by their programs using a blending of information from the membership. These factors are used in this cost-benefit evaluation, where applicable.

Table 4.3-2 shows net present values of estimated savings from implementation of a risk-based inspection program for nuclear plant piping systems. As shown in the table, significant savings can be achieved in direct costs. Other indirect cost savings are also expected to be significantly reduced. These indirect cost savings are expected to include:

- Outage critical path reduction (which is becoming more important as utilities continue to reduce outage length)
- Program administration cost reduction
- Insurance premium reduction
- Cost reduction associated with evaluating flaw indications in less safety-significant piping

In addition, a risk-based ISI program should enhance the finding of precursors to potential failures because inspection resources are focused on locations of highest failure potential in more safety-significant piping segments. The identification of these precursors should help minimize events like leaks, which result in significant business interruption losses. In summary, the development and implementation of a risk-based ISI program provides the opportunity to significantly reduce burden while maintaining or enhancing safety.

Table 4.3-2 ESTIMATED SAVINGS FROM RISK-BASED INSPECTION FOR TYPICAL 4-LOOP PLANT*		
Description	**Considerations**	**Net Present Value of Savings****
Direct Costs		
Actual Inspection Costs	Includes NDE, scaffolding and insulation removal	$1,889,660
ALARA Costs	Assuming approximately 15 REM per outage savings and using $10,000/REM	$846,650
	TOTAL DIRECT COST SAVINGS	$2,736,310
Indirect Costs		
Administrative Costs	Paper work including work orders, surveillances and clearances	Not estimated
Outage Critical Path	Reduction of 1-2 days of outage time anticipated as outages become shorter (NPV savings assumes 0.5 day at $340,000 per day)	$1,314,170
Insurance Premiums		Not estimated
Analysis Costs	From flaw indication evaluations in less safety-significant piping segments	Not estimated
	TOTAL ESTIMATED DIRECT AND INDIRECT SAVINGS	> $4,050,480

* The estimated savings for 2-loop and 3-loop units will obviously be lower than these values depending on the number of piping locations currently being inspected to the requirements of ASME Section XI. The effort to perform a risk-based ISI program, however, will require less resources relative to the number of piping system segments to be addressed.

** Assumes discount rate of 7.5% and estimated savings at each outage over the remaining 30 years of operating license life.

The total effort to perform the risk-based ISI program for the representative WOG plant exceeded the direct savings that would be gained during one outage at that unit. However, more than half of that cost was associated with learning and adapting the methodology to be applied across all the piping systems at a large nuclear plant, which is a first-of-a-kind application. In addition, there were considerable costs associated with interfacing with ASME, NEI, and the NRC on this project.

It is believed by the team members that the risk-based ISI program can be applied in the future at a cost much less than the direct savings that are gained from piping examinations done in one outage from implementation of the program.

SECTION 5

PLANT-SPECIFIC APPLICATION PROCESS

This section provides the framework for applying the risk-based methods to a specific plant for piping inservice inspection. The tasks required to develop a comprehensive risk-based inservice inspection program for piping are provided below. The tasks are:

- Scope Definition
- Segment Definition
- Consequence Evaluation
- Failure Probability Estimation
- ISI Segment Selection
- Structural Element Selection
- Inspection Requirements

Figure 5-1 provides a roadmap of the process along with the appropriate section of the report which contains more detail.

Once an ISI program has been established, a feedback loop is included based on the actual results of inspection of piping.

Figure 5-2 identifies the skills necessary for a successful program.

5.1 SCOPE DEFINITION

The fluid systems contained in the plant, modeled in the PSA and considered as part of the Maintenance Rule, are identified and compared with the current classifications and required ISI examinations, and with the stress analysis. This review, along with other plant documentation, is used to determine which systems, or portions of systems, should be evaluated as part of the risk-based ISI process. Given that system boundaries involve system functions and may also involve interfaces between different types of 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

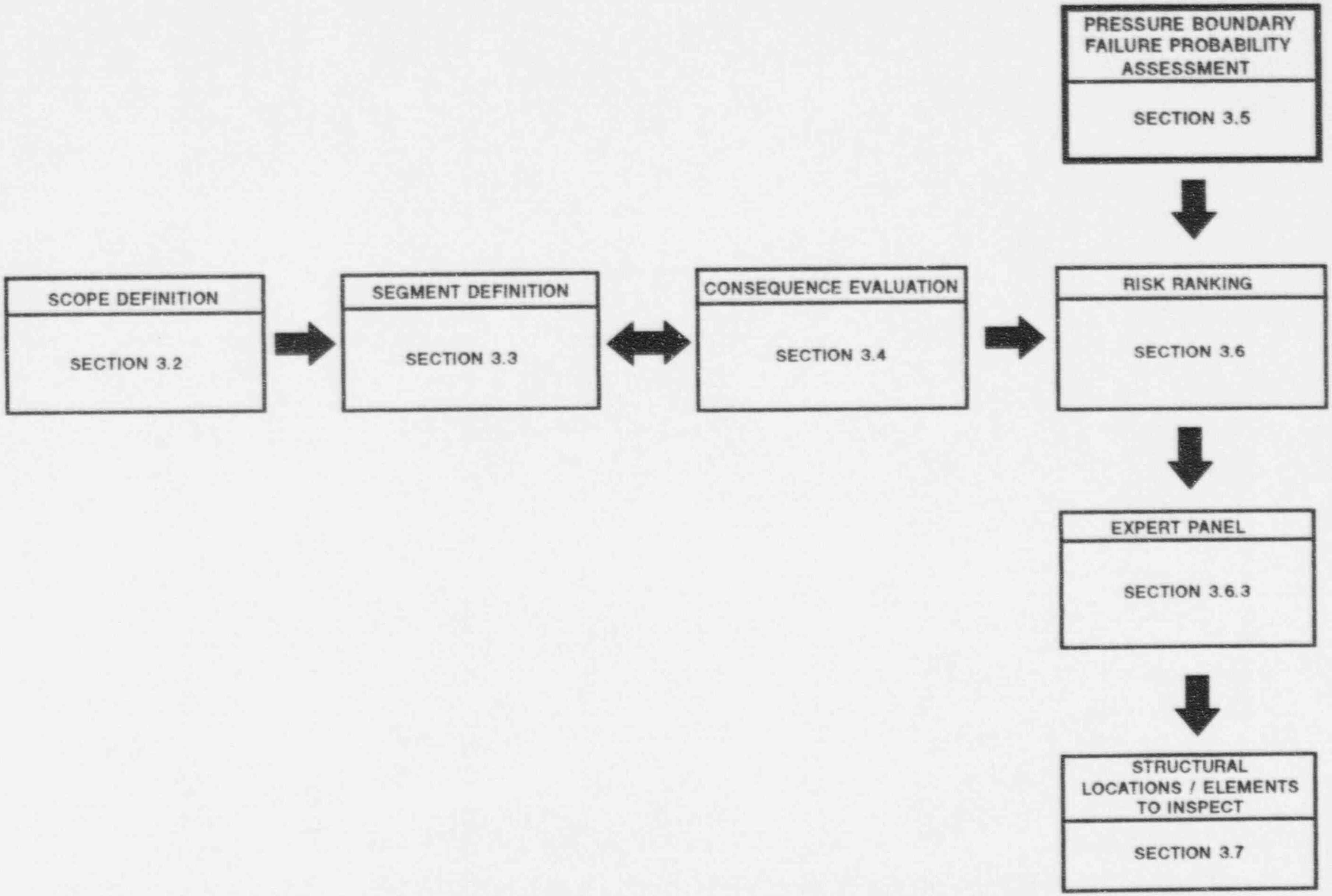

Figure 5-1. WOG Risk-Based ISI Process and Roadmap

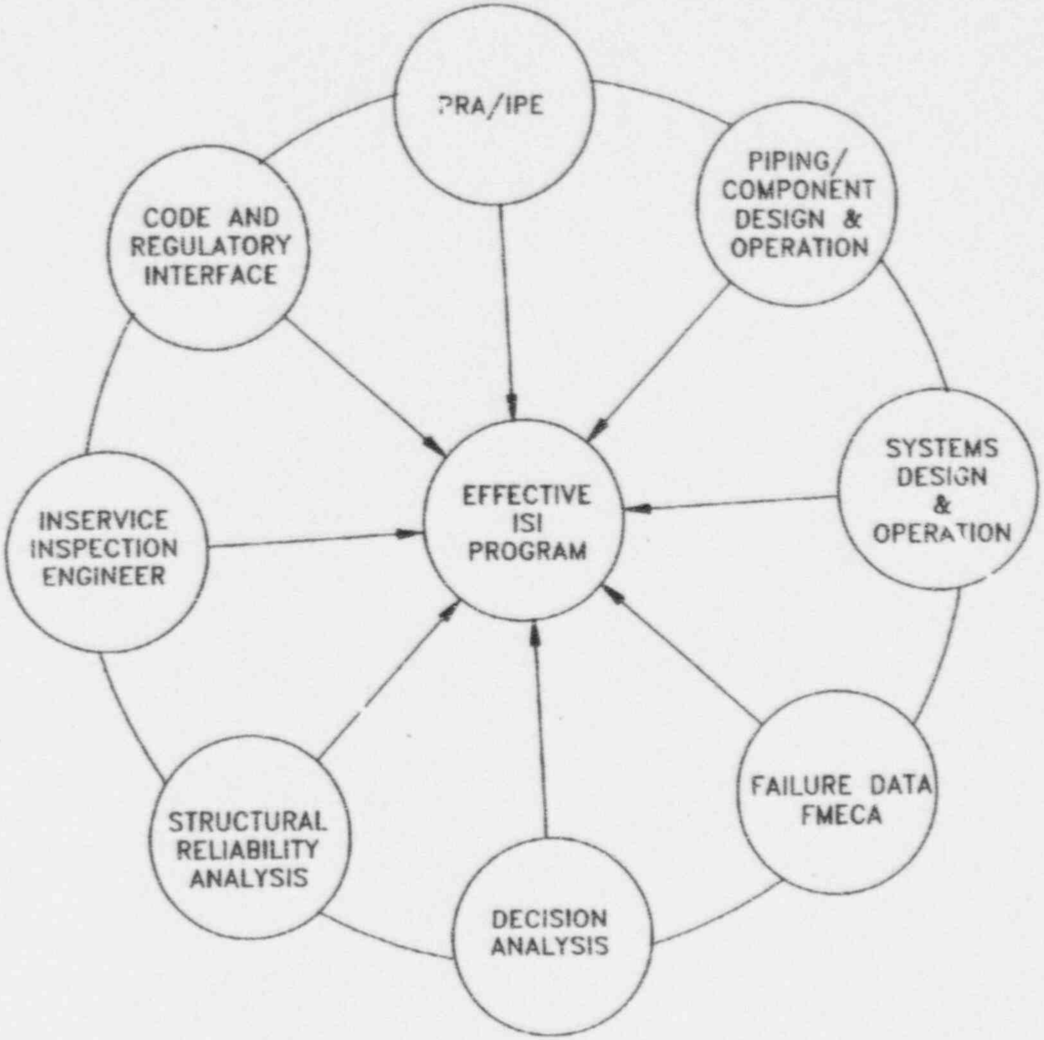

Figure 5-2 Required Skills for Risk-Based Inspection

boundaries, when viewed from the systems on either side of each boundary, and that no safety functions are overlooked.

5.2 SEGMENT DEFINITION

This task involves the development of piping segments for the risk-ranking. A piping segment is defined as a portion if piping for which a failure at any point in the segment results in the same consequence (e.g., loss of a system, loss of a pump train, etc.) and includes piping structural elements between major discontinuities such as pumps and valves.

5.3 CONSEQUENCE EVALUATION

The consequences given the failure of a piping segment are identified through PSA insights, engineering evaluations and plant design and operations. Consequences that must be considered include both direct effects (failure of a train in which the piping segment is contained) and indirect effects (such as those due to flooding, pipe whip, or jet impingement).

5.4 FAILURE MODES AND FAILURE PROBABILITY ASSESSMENT

The overall process of identifying potential failure modes, selecting locations and calculating failure probabilities proceeds by system, and includes preliminary activities for the system as a whole, and detailed assessments and data gathering for each segment. This includes the following steps:

- Gather design basis information
- Review industry experience
- Discuss system operations with system engineer and gain further insights into any potential piping problems
- Determine likely failure mode(s)
- Select candidate location(s)

- Gather detailed data for probability of failure analysis

- Calculate probabilities of failure

- Document locations and probabilities

5.5 SELECTION OF ISI SEGMENTS AND STRUCTURAL ELEMENTS

This task is to identify and prioritize the important components (or pipe segments). The approach calculates the relative importance for each component within the systems of interest. This risk-importance is based on the frequency of core damage (or LERF, if available) resulting from the structural failure of the component in a given segment and the total piping pressure boundary core damage frequency (and LERF, if available). The results are then used to calculate the risk-importance for each component within the system.

The following outlines the steps of the process:

1. Apply PSA to calculate piping pressure boundary core damage frequency (and LERF, if available)
    - Identify impact on PSA model (using EPRI PSA Applications Guide)
    - Identify surrogate component
    - Obtain conditional core damage frequency/probability
    - Integrate pressure boundary failure probability/rate
    - Calculate segment piping pressure boundary core damage frequency
    - Calculate total piping pressure boundary core damage frequency
2. Calculate importance measures
    - Calculate segment Risk Reduction Worth importance measure
    - Calculate segment Risk Achievement Worth measure
3. Evaluate other PSA-related factors through sensitivity studies
4. Integrate other deterministic considerations
    - Shutdown risk evaluation
    - External events evaluation
    - Other accident scenarios
    - Component operating history

- Plant operation and maintenance insights
- Design basis analysis
- Other deterministic insights

5. Conduct expert panel sessions and document results

An expert panel (such as the expert panel used for the Maintenance Rule) evaluates the risk-based results and makes a final review to determine the more safety-significant pipe segments for ISI.

The selection of inspection locations within each more safety-significant pipe segment is obtained by further review by a subpanel, comprised of materials, ISI and NDE expertise, using the following steps.

1. Verify that the locations with the highest failure potential within a segment are identified.
2. Determine appropriate examination method(s) at the selected structural element locations.
3. Document the results and present to the full expert panel for final review and approval.

The output of this process defines the structural elements selected and the associated examination method and frequency for inspection.

5.6 INSPECTION REQUIREMENTS

The inspection requirements defined in Section 4 should be consulted to define the type of inspection to be performed on the structural elements.

5.7 IMPLEMENTATION AND FEEDBACK

Implementation

Once the risk-based process is completed, the inspection program can be implemented. The required examinations are scheduled over the 10 year inspection interval in periods. If, during the interval, a reevaluation of the risk-based process is conducted and scheduled items are no longer required, the items may be eliminated. If items are identified for inclusion in the program, the items should be

added and distributed across the remaining periods in the interval. Each subsequent 10 year interval should include, as a minimum, a reevaluation of the risk-based process.

For examinations that reveal flaws or relevant conditions exceeding ASME acceptance standards, additional examinations should be conducted. The additional examinations should include the same type of piping structural element(s) with the same postulated failure mode(s).

If piping structural elements are accepted for continued service, the areas containing flaws or relevant conditions should be reexamined during the next three inspection periods. If the reexaminations reveal that flaws or relevant conditions remain essentially unchanged for three successive inspection periods, the piping examination schedule may revert to the original schedule.

The examination qualification and methods requirements and personnel qualification requirements should be the same as under the plant's current inservice inspection program.

Feedback

The risk-based inservice inspection program should be reevaluated periodically as new information becomes available. Such information may result for example from changes to the PSA, from inspection results, from new failure modes experienced by the industry, from replacement activities, from repair activities, or plant design or operational changes. The effect of the new information on the risk-based process should be determined. Each phase of the risk-based process should be reevaluated to determine where the new information impacts the process and/or the results. The new information should be included at the appropriate level of the analysis (consequence evaluation, failure probability estimation, etc.) and the analysis should be conducted to identify the changes to the risk-based inspection program.

SECTION 6
SUMMARY OF RESULTS AND CONCLUSIONS

6.1 RESULTS OF APPLICATION

Table 3.6-6 summarizes the results of the segment risk-ranking for Millstone Unit 3. It identifies the number of piping segments for each system and the number of more safety-significant segments. The table shows that approximately 37% of the piping segments were determined to be more safety-significant.

These piping segments were then subjected to a detailed review to determine the structural elements that should be inspected. Table 4.3-1 summarizes the structural elements for each system to be inspected under the risk-based ISI program and compares these elements by system with those defined by the current 1989 ASME Code requirements. As shown in Figure 4.3-1, the current ASME Code inspection locations address approximately 44% of the piping pressure boundary core damage frequency while the risk-based ISI program addresses approximately 98% of the piping pressure boundary failure core damage frequency with less locations being inspected. The total piping core damage frequency is a small fraction of the total plant core damage frequency. While these results represent an application to a plant designed to ASME Section III, the risk-based ISI process should yield similar results to piping systems in plants not designed to ASME Section III.

6.2 CONCLUSIONS FROM APPLICATION

This program has been of benefit by providing a sound technical approach and process for developing a risk-based inservice inspection program for piping that can be implemented in the industry. It is expected that the risk-based ISI program can be implemented at a cost much less than the direct savings that are gained from piping examinations done in one outage. The representative WOG plant application has shown the potential for reducing operation and maintenance costs in the development and implementation of effective piping inspection programs while maintaining a high level of safety.

6.3 INSIGHTS FOR APPLICATIONS TO OTHER EQUIPMENT

While the effort for this application focused on the use of risk-based methods for the inservice inspection of piping, several insights have been obtained for possible application to other equipment. The process described and the steps can be applied to all types of components, such as vessels, tanks, heat exchangers, snubbers and other equipment under ASME Section XI.

SECTION 7
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APPENDIX A
PLANT WALKDOWN INFORMATION

The appendix discusses the review of the plant hazards evaluation and the conduct of the plant walkdown to identify potential indirect effects from piping failure.

PRE-WALKDOWN

The Millstone 3 Hazards Review Program Summary Sheets were reviewed for systems interactions due to postulated pipe breaks. The summary sheets examine the effects of spray wetting, flooding, temperature, pipe whip, jet impingement, rotating machinery, and pressure boundary ejected missiles. Because the risk-based inspection program is concerned only with the effects due to pipe breaks and leaks, the rotating machinery and pressure boundary missiles evaluations were not reviewed. Note that the pressure boundary missiles are primarily from valves, which are not part of this program. In addition, Section 3.6 of the Millstone FSAR, "Protection Against Dynamic Effects Associated with the Postulated Ruptures of Piping," was reviewed. A summary of the review is provided in Table A-1.

The Hazards Evaluation examined the containment, the ESF building, the auxiliary building, the diesel generator building, the fuel building, the circulating and service water pumphouse, and the hydrogen recombiner building. Because only two cubicles in the circulating and service water pumphouse were mentioned in the Hazards Evaluation, it was decided to include the entire pumphouse in the walkdown. The turbine building was also included because the Hazards Evaluation did not address the building, and because of the amount of the high energy piping in the building.

WALKDOWN

The walkdown was performed and included members from the PRA, piping, and operations groups at Northeast Utilities, and members of risk and structural reliability groups at Westinghouse. The walkdown covered the specific areas listed in Table A-1 in the ESF building and the auxiliary building. The walkdown also included all of the circulating and service water pumphouse and the turbine building. Two of the walkdown worksheets documenting the information gathered are presented in Tables A-2 and A-3.

INSIGHTS FROM THE WALKDOWN

The following summarizes the insights from the plant walkdown for the various areas investigated.

Auxiliary Feedwater System

There were numerous valves near the discharge of the motor auxiliary feedwater pump. An AFW piping failure could disable some of these valves, but the effect would still be a loss of one train. Two concerns noted were the spray onto overhead cable trays, and a postulated reactor plant component cooling water (CCP) break which targets the AFW pump and some valve controllers. These sections of piping were not in the original program scope for CCP. Based on the interaction possibility with the AFW system, two CCP segments were added for risk evaluation and the cable trays were investigated for their effects. (Table A-1 Item 5)

Component Cooling Water

It was verified that pipe shrouds had been placed on the discharge piping of CCP pumps 3CCP*P1A and P1C. These shrouds were placed to mitigate the interactions of a break in one train disabling the pump in the other train (as noted in the Hazards Evaluation). No other unique interactions were noted for these areas. (Table A-1 Item 16)

Service Water

There are vital and non-vital motor control centers (MCCs) in the service water pump cubicles. Large drains were noted in each cubicle to prevent flooding problems. The implications of a pipe break spraying on the MCCs was noted for further review. (Table A-1 Item 29) (Note: the expert panel considered this and decided to not take credit for drains and considered this as an indirect effect.)

Turbine Building

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.

Table A-1

HAZARDS REVIEW SUMMARY

Item	Building	Cubicle/Area	Equipment / Pipe Segment	Indirect Effects	Consequences	Walkdown?	Shutdown?	Comments
1	ESF	001, 002, 021, 022	3FWA-004-126-3/128-3	Pipe Whip	Potential loss of "B" electrical division	No	No	Eval concludes no damage
2	ESF	003, 004, 005	FWA, SWP, CCP, RHS piping	Flooding	None	No	Yes	
3	ESF	006, 007, 008, 009, 019, 020	Moderate Energy Cracks	Temperature/Humidity	Potential loss of equip for 1 RHS or SIH Train (same train/system as break)	No	No	
4	ESF	010	QSS-P1A/B	Flooding	Bounded by 12179-PR-1194	No	No	
5	ESF	011, 012, Rev. 1	FWA*P1B	Water Spray	Loss of Train "B" Equipment in cubicle	Yes	No	Check other equip in cubical
6	ESF	011, 012, Rev. 1	FWA*P1B	Jet Impingement	Cable trays 3TC7520, 3TC7610, TK7520 RHS*P1A cooling	No	Yes	Eval concludes no damage
7	ESF	013, 014	SW & CCW Piping	Flooding	Bounded by 12179-PR-1157	No	No	
8	ESF	013, 014	3FWA-004-126, -128	Pipe Whip	Could cause start of AFW TD pump	No	No	
9	ESF	015, 016, 017, 018	HVQ*ACUS1A/B & HVQ*SCUS2A/B	Water Spray	3EHS*MCC1A4 RHR operation	No	Yes	Eval. concludes no damage
10	AB1	23A, B, E	3CHS-003-8-2	Jet Impingement	3CHS-002-283-2	No	Maybe per T.S.	Letdown line damages seal return line

Table A-1 (Continued)

HAZARDS REVIEW SUMMARY

Item	Building	Cubicle/Area	Equipment / Pipe Segment	Indirect Effects	Consequences	Walkdown?	Shutdown?	Comments
11	AB-1	23C, 23D, 24, 25	30" SW	Flooding	Bounded by 12179-PR-1071	No	No	
12	AB-1	23F	30" SW	Flooding	Bounded by 12179-PR-1071	No	No	
13	AB-1	AB26, 27, 28, 89, 90, 99B, 112	-	-	-	-	-	No piping in risk-based ISI scope
14	AB-1	33, 34, 35	CHS piping	Flooding	Bounded by 12179-PR-1071	No	No	
15	AB-1	29, 91 Rev. 1	-	-	-	-	-	No piping in risk-based ISI scope
16	AB-2	86, 87, 88	3CCP*P1C/A	Water Spray	Two CCP Trains	Yes	Yes	Check for CCP pipe shroud
17	AB-2	36	3" CHS Letdown Exchanger Inlet Piping	Pipe Whip	6" CCP inlet or outlet lines	No	Yes	Eval concludes no damage
			3" CHS Letdown Exchanger Inlet Piping	Flooding	Bounded by 12179-PR-1071	No	No	
18	AB-2	38 thru 53, 55 thru 78	CHS piping	Pipe Whip	None - redundant trains in individual cubicles	No	No	
			3" CHS Letdown Exchanger Inlet Piping	Flooding	Bounded by 12179-PR-1071	No	No	
19	AB-2	54, 79, 80, 81	CHS piping	Pipe Whip	Redundant trains in individual cubicles	No	No	

Table A-1 (Continued)

HAZARDS REVIEW SUMMARY

Item	Building	Cubicle/Area	Equipment / Pipe Segment	Indirect Effects	Consequences	Walkdown?	Shutdown?	Comments
20	AB-2	92, 93, 94	CHS alt. mini-flow piping	Jet Impingement	One service water train	No	No	
21	AB-2	30, 31, 32, 95, 96, 97	-	-	-	-	-	No piping in risk-based ISI scope
22	AB-2	98 Rev. 1	CCP Piping	Flooding	Bounded by 12179-PR-1071	No	Maybe, per TS	
23	EGE	175 - 181 Rev. 1	Service Water	Flooding	Bounded by 12179-PR-1073 Loss of single Generator Train	No	Maybe, per TS	
24	HP	182 - 187 Rev. 1	-	-	-	-	-	No piping in risk-based ISI scope
25	FB	188, 197, 198	SFC, FPW, CCP Piping	Flooding	Bounded by 12179-PR-1038	No	No	
26	FB	191	CCP, FPW piping	Flooding	Bounded by 12179-PR-1038	No	No	
27	FB	194	SFC pump discharge	Water Spray	Bounded by 12179-NMS-793-DM	No	No	
28	FB	195, 196, 200	SFC piping	Flooding	Bounded by 12179-PR-1038	No	No	
29	CW	201, 202 Rev. 1	SW Pump Discharge Piping	Water Spray	Loss of single electrical train 3EJS*US1A due to spray on 3EHS*MCC1A5 or 3EHS*MCC1B5	Yes	Yes	

Table A-1 (Continued)
HAZARDS REVIEW SUMMARY

Item	Building	Cubicle/Area	Equipment / Pipe Segment	Indirect Effects	Consequences	Walkdown?	Shutdown?	Comments
30	AB-3	99A	SW Piping, 3SWP*P3A suction or discharge	Water Spray	3SWP*P3A suction or discharge spray on 3SWP*P3B	No	Maybe, per TS	SW pumps are drip protected; No consequential damage
31	AB-3	99C, 110, 111	CCP piping	Water Spray	None	No	Maybe per TS	
32	AB-3	99D	CHS piping	Water Spray	None	No	No	
33	AB-3	100, 118 - 121	-	-	-	-	-	No piping in risk-based ISI scope
34	AB-3	101, 102	CCP piping	Water Spray	None	No	No	
35	AB-3	103 - 109	CCP piping	Water Spray	None	No	No	
36	AB-3	113 - 117	CHS, SWP piping	Water Spray, Flooding	None	No	No	
37	AB-3	Elev. 66'-6"	-	-	-	-	-	Hazards addressed are for fans in systems outside risk-based ISI scope
38	CS-1	131A - F, 132A - H, 138	Moderate energy cracks in all piping	Flooding	Bounded by 12179-NS(B)-249	No	No	
39	CS-1	133A, 133B, 135, 142A, 144	3RCS-003-171-1	Pipe Whip	Conduit damage resulting in closing letdown and isolation valves	No	Yes	Break postulated to isolate itself due to valve closure
40	CS-1	133C, D Rev. 1	3-CHS-003-662-2	Jet Impingement	Seal Water return line 3-CHS-002-618-2	No	No	

Table A-1 (Continued)

HAZARDS REVIEW SUMMARY

Item	Building	Cubicle/Area	Equipment / Pipe Segment	Indirect Effects	Consequences	Walkdown?	Shutdown?	Comments
41	CS-1	134A - F Rev. 1	3-CHS-025-304-2	Jet Impingment	Seal Water return line 3-CHS-002-618-2	No	No	Note event description for BDG line breaks
42	CS-1	136 Rev. 1	RCS piping	Jet Impingement	Bounded by 12179-NSB-177	No	Yes	
43	CS-2	137 Rev. 2	RCS piping	Pipe Whip/Jet Impinge.	Bounded by 12179-NSB-177	No	Yes	
44	CS-2	139, 146 Rev. 2	RCS piping	Pipe Whip/Jet Impinge.	Bounded by various calcs	No	Yes	
45	CS-2	140 Rev. 2	RCS Piping	Pipe Whip/Jet Impinge.	Bounded by various calcs	No	Yes	
46	CS-2	141 Rev. 1	RCS piping	Pipe Whip/Jet Impinge.	Bounded by various calcs	No	Yes	
47	CS-2	142B - F	FWS, MSS, FWA piping	Pipe Whip/Jet Impinge.	Bounded by various calcs	No	Yes	
48	CS-2	145A - F, 143, 147	Intermediate Break in 30" MSS line at upstream elbow	Axial Jet	Loss of conduits results in loss of radiation monitors, 3RMS*RIY05 & 3RMS*RIY42 and loss of power to 3RMS*RM42	No	No	
			Intermediate Break in 30" MSS line at downstream elbow	Radial Jet	Loss of one MSS line to FWA TD pump	No	Yes	

Table A-2

MILLSTONE 3 RISK-BASED INSPECTION EXPERT PANEL EVALUATION

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 rupture 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*P1A	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

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

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. 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 affect FWA Train A. Need to consider the CCP interaction for inclusion in the segments analyzed.

Table A-3
MILLSTONE 3 RISK-BASED INSPECTION EXPERT PANEL EVALUATION
INDIRECT EFFECTS WALKDOWN WORKSHEET

Item #: NA

Building: Turbine

Cubicle/Area:

Elevation: 14' - 6"

Indirect Effect of Concern:

Components/Equipment in Cubicle/Area					
System	Comp. Type	Tag No.	Train	Needed for Safe Shutdown?	Support System?
IAS	Compressor	3IAS-C1A	-	N	Y
IAS	Compressor	3IAS-C1B	-	N	Y
SAS	Compressor	3SAS-C1	-	N	Y

Comments

The three compressors are located side by side near the condensate pump discharge header. A break in the header could potentially fail all three compressors which would cause a reactor trip.

Conclusions

Needs to be considered along with other possible breaks in turbine building.

APPENDIX B

SAMPLE EXPERT PANEL WORKSHEETS

Contained in this appendix are sample segment worksheets which were used by the expert panel review. Section 6 of the worksheet contains the final safety-significance category (more or less safety-significant) determined by the expert panel. Below is a brief summary of the segments represented by the worksheets.

FWS-1: This segment is the main feedwater piping to steam generator A, between motor-operated valve 35A and gate valve FCV 510. A break in this line causes a loss of main feedwater (feedline break), modeled in the PSA as an initiating event. The calculated full break probability is 0 (1.0E-08 was assumed). The RRW value calculated is 1.00 and the RAW value is relatively low. The segment was designated less safety-significant because of the low failure probability and the relatively low consequence.

ECCS-1: This segment is one of the four safety injection lines and it is located between check valves 8818A and 8819A and 8847A (inside containment). A break in this line causes a partial loss of injection, and the eventual loss of the RWST inside containment. The calculated full break probability is 0 (1.0E-08 was assumed). The RRW and RAW values were relatively high, however, the expert panel believed the PSA modeling was too conservative because the RWST inventory would be available for recirculation. The time to switch to recirculation would however be shorter. This segment was designated less safety-significant because of the low failure probability and the expert panel's assessment that the consequence would be lower than calculated.

RCS-7: This segment is the safety injection line from check valve 8948A to the tee on the loop A cold leg. A break in this segment causes a large LOCA, modeled in the PSA as an initiating event. The calculated full break probability is 4.1E-09 (the threshold value of 1.0E-08 was used). The RRW value calculated is 1.00 but the RAW value is relatively high. The segment was designated more safety-significant due to the relatively high RAW value and because of the high consequence of a large LOCA.

RCS-15: This segment is the high pressure safety injection connection from the cold leg tee to check valve 8900B. A break in this segment causes a small LOCA, modeled in the PSA as an initiating

event. The calculated full break probability is 1.5E-12 (1.0E-08 was assumed). The RRW value calculated is 1.00 but the RAW value is relatively high. The segment was designated more safety-significant due to the relatively high RAW value and because the pipe failure results in an unisolable break in the RCS.

SIL-9: This segment is from accumulator TK1A to check valve 8956A. A break in this line results in the loss of accumulator TK1A. The calculated full break probability is 0 (1.0E-08 was assumed). The RRW value is 1.00 and the RAW is in a medium range. This segment was designated less safety-significant due to the low failure probability, benign normal operating conditions, and low consequence.

WOG RISK-BASED INSPECTION
EXPERT PANEL EVALUATION
SEGMENT RANKING WORKSHEET

Section 1 System & Pipe Segment Identification	
System & Segment Description:	FWS-1 Main Feedwater/Condensate System From motor valve MOV-35A(V14) to gate valve FCV-510(V15)
Location/P&ID Drawing:	EM-130C
System Function(s):	Provides feedwater to steam generators

Section 2 Risk Ranking Information		
Failure Effect on System Without Operator Action:	Loss of main feedwater flow to steam generator A	
Failure Effect on System With Operator Action:	Loss of main feedwater flow to steam generator A	
PSA Initiating Events Impact:	Loss of Main Feedwater	
PSA Containment Performance Impact:	None	
Conditional Core Damage Frequency Due to Pressure Boundary Failure	Without OA 1.20E-06	With OA 1.20E-06
Total Pressure Boundary Failure Core Damage Frequency ($FP*CDF_{cond}$)	3.00E-16	3.00E-16
CDF_{pb} Importance Measure Values	RAW 5.38 RRW 1.000	106 1.000
Comments:		

Section 3 Pressure Boundary Failure Probability	
Segment Elements (welds, tees, elbows, etc.):	Pipe to valve V14 weld
Pressure Boundary Failure Mechanism(s):	Thermal fatigue, erosion/corrosion
Pressure Boundary Leak Probability:	Small Leak: 1.1E-03 Full Break : 0 (use 1.0E-08)
Basis for Pressure Boundary Failure Probability:	High temperature at pipe weld, large nominal pipe size, high normal operating pressure
Comments:	Break exclusion zone. No EC trending LOC 040-016 US

WOG RISK-BASED INSPECTION
EXPERT PANEL EVALUATION
SEGMENT RANKING WORKSHEET

Segment: FWS-1 (Sheet 2)

Section 4 Indirect Effects Evaluation	
Indirect Effect (Spray, flood, pipe whip, jet impingement)	None identified
Pressure Boundary Failure Impact on Other Systems	None identified
Core Damage Frequency Contribution due to Indirect Effects	None

Section 5 Other Considerations		
External Events Evaluation		
Seismic:	Fire:	External Flood:
Shutdown Risk Evaluation	Feedline break during cooldown. No impact at shutdown.	
Importance to Other Accident Scenarios		
Component Maintenance and Operation Insights:	Review of reports conducted, no major problems found	
Importance to Design Basis Analysis:	Decrease in heat removal by the secondary system, per FSAR Chapter 15.	
Other Deterministic Insights:		

Section 6 Final Risk Category			
Category:	More Safety Significant	Less Safety Significant	X
Basis	Low failure probability, relatively low consequence - loss of MFW		

WOG RISK-BASED INSPECTION
EXPERT PANEL EVALUATION
SEGMENT RANKING WORKSHEET

Section 1 System & Pipe Segment Identification	
System & Segment Description:	ECCS-1 Emergency Core Cooling System From CV8819C (V24) and CV8818C (V13) to CV8847C (V985)
Location/P&ID Drawing:	EM-112A, 112B & 113B
System Function(s):	Provides water from the RWST and the containment sump for core cooling during a LOCA

Section 2 Risk Ranking Information		
Failure Effect on System Without Operator Action:	Loss of RWST inside containment	
Failure Effect on System With Operator Action:	Loss of all RHR & HPSI flow	
PSA Initiating Events Impact:	None	
PSA Containment Performance Impact:	None	
Conditional Core Damage Frequency due to Pressure Boundary Failure:	Without OA 4.73E-02* (3.00E-04)	With OA 2.09E-03
Total Segment Pressure Boundary Failure Core Damage Frequency (FP*CDF_{cond}):	4.73E-10* (3.00E-12)	2.09E-11
CDF_{pb} Importance Measure Values:	RAW 1.50E+05* (1.32E+4) RRW 1.002* (1.00)	1.83E+05 1.002
Comments:	*Based on Expert Panel discussion, the consequence is much less than this – will be requantified (shown in parentheses) – would result in draindown of RWST and earlier transfer to recirc.	

Section 3 Pressure Boundary Failure Probability	
Segment Elements (welds, tees, elbows, etc.):	Weld at V985
Pressure Boundary Failure Mechanism(s):	Thermal fatigue
Pressure Boundary Failure Probability:	Small Leak: 0 (use 1.0E-08 per demand) Full Break: 0 (use 1.0E-08 per demand)
Basis for Pressure Boundary Failure Probability:	High normal operating pressure, Maximum residual stress level, High fatigue transient frequency
Comments:	Valve is located on branch line within 2 feet of run pipe connection; Many nearby branch line snubbers exist which potentially may lockup causing break potential

WOG RISK-BASED INSPECTION
EXPERT PANEL EVALUATION
SEGMENT RANKING WORKSHEET

Segment: ECCS - 1 (Sheet 2)

Section 4 Indirect Effects Evaluation	
Indirect Effect: (Spray, flood, pipe whip, jet impingement)	None Identified
Pressure Boundary Failure Impact on Other Systems:	None identified
Core Damage Frequency Contribution due to Indirect Effects:	None

Section 5 Other Considerations		
External Events Evaluation		
Seismic:	**Fire:**	**External Flood:**
Shutdown Risk Evaluation:	Failure results in possible reduced flow for emergency core cooling; loss of RHR flow and LOCA during shutdown if RHR is not isolated	
Importance to Other Accident Scenarios:		
Component Maintenance and Operation Insights:	Review of reports conducted, no major problems found	
Importance to Design Basis Analysis:		
Other Deterministic Insights:		

Section 6 Final Risk Category	
Category: More Safety Significant	Less Safety Significant X
Basis Low Failure Probability and lower consequence given draindown of RWST	

WOG RISK-BASED INSPECTION
EXPERT PANEL EVALUATION
SEGMENT RANKING WORKSHEET

Section 1 System & Pipe Segment Identification		
System & Segment Description:	RCS-7 Reactor Coolant System LPSI Connection from Loop A Cold Leg Tee to CV 8948A (V30)	
Location/P&ID Drawing:	EM-102A	
System Function(s):	Reactor heat removal	

Section 2 Risk Ranking Information		
Failure Effect on System Without Operator Action:	Large loss of coolant accident	
Failure Effect on System With Operator Action:	Large loss of coolant accident	
PSA Initiating Events Impact:	Large LOCA initiator	
PSA Containment Performance Impact:	None	
Conditional Core Damage Frequency due to Pressure Boundary Failure	Without OA 9.36E-03	With OA 9.36E-03
Total Segment Pressure Boundary Failure Core Damage Frequency (FP * CDF_{cond})	2.34E-12	2.34E-12
CDF_{pb} Importance Measure Values	RAW 4.12E+05 RRW 1.000	8.22E+05 1.000
Comments		

Section 3 Pressure Boundary Failure Probability		
Segment Elements (welds, tees, elbows, etc.):	10" Pipe weld at connection to RCS cold leg	
Pressure Boundary Failure Mechanism(s):	Thermal fatigue	
Pressure Boundary Failure Probability:	Small Leak: 1.9E-06 Full Break: 4.1E-09 (Use 1E-08)	
Basis for Pressure Boundary Failure Probability:	High temperature at pipe weld, Maximum residual stress level, High steady state stress level	
Comments	High usage factor. Branch is on fatigue watch list	

Segment: RCS-7 (Sheet 2)

Section 4 Indirect Effects Evaluation	
Indirect Effects (spray, flood, pipe whip, jet impingement)	None Identified
Pressure Boundary Failure Impact on Other Systems	None Identified
Core Damage Frequency Contribution due to Indirect Effects	None

Section 5 Other Considerations		
External Events Evaluation Seismic:	Fire:	External Flood:
Shutdown Risk Evaluation	Failure results in Large LOCA at shutdown	
Importance to Other Accident Scenarios		
Component Maintenance and Operation Insights:	Review of reports conducted, no major problems found	
Importance to Design Basis Analysis	Large LOCA, per FSAR Chapter 15	
Other Deterministic Insights		

Section 6 Final Risk Category	
Category: More Safety Significant X	Less Safety Significant
Basis Relatively High RAW Value, High consequence - Large LOCA	

WOG RISK-BASED INSPECTION
EXPERT PANEL EVALUATION
SEGMENT RANKING WORKSHEET

Section 1 System & Pipe Segment Identification	
System & Segment Description:	RCS-15 Reactor Coolant System HPSI Connection from Cold Leg Tee to CV 8900B (V70)
Location/P&ID Drawing:	EM-102D
System Function(s):	Reactor heat removal

Section 2 Risk Ranking Information		
Failure Effect on System Without Operator Action:	Small loss of coolant accident	
Failure Effect on System With Operator Action:	Small loss of coolant accident	
PSA Initiating Events Impact:	Small LOCA initiator	
PSA Containment Performance Impact:	None	
Conditional Core Damage Frequency due to Pressure Boundary Failure	Without OA 8.61E-04	With OA 8.61E-04
Total Segment Pressure Boundary Failure Core Damage Frequency (FP * CDF_{cond})	2.15E-13	2.15E-13
CDF_{pb} Importance Measure Values	RAW 3.79E+04 RRW 1.000	7.56E+04 1.000
Comments		

Section 3 Pressure Boundary Failure Probability	
Segment Elements (welds, tees, elbows, etc.):	Weld to V70
Pressure Boundary Failure Mechanism(s):	Thermal fatigue
Pressure Boundary Failure Probability:	Small Leak: 0 (Use 1.0E-08) Full Break: Use 1.0E-08
Basis for Pressure Boundary Failure Probability:	High temperature at pipe weld, High normal operating pressure, Maximum residual stress level
Comments	Area of maximum bending stress. SR EL @ 535/540 & Tee @ 550 are on fatigue watch list

WOG RISK-BASED INSPECTION
EXPERT PANEL EVALUATION
SEGMENT RANKING WORKSHEET

Segment: RCS-15 (Sheet 2)

Section 4 Indirect Effects Evaluation	
Indirect Effects (spray, flood, pipe whip, jet impingement)	None Identified
Pressure Boundary Failure Impact on Other Systems	None Identified
Core Damage Frequency Contribution due to Indirect Effects	None

Section 5 Other Considerations		
External Events Evaluation Seismic:	Fire:	External Flood:
Shutdown Risk Evaluation	Failure results in Small LOCA at shutdown	
Importance to Other Accident Scenarios		
Component Maintenance and Operation Insights:	Review of reports conducted, no major problems found	
Importance to Design Basis Analysis	Small LOCA, per FSAR Chapter 15	
Other Deterministic Insights		

Section 6 Final Risk Category	
Category: More Safety Significant X	Less Safety Significant
Basis Relatively large RAW value, Unisolable break	

WOG RISK-BASED INSPECTION
EXPERT PANEL EVALUATION
SEGMENT RANKING WORKSHEET

Section 1 System & Pipe Segment Identification	
System & Segment Description:	SIL-9 Low Pressure Safety Injection SI Accumulator Tank TK1A to CV8956A (V15)
Location/P&ID Drawing:	EM-112B
System Function(s):	Provides borated water to core during design basis accidents

Section 2 Risk Ranking Information		
Failure Effect on System Without Operator Action:	Loss of Accumulator A water flow to cold leg 1	
Failure Effect on System With Operator Action:	Loss of Accumulator A water	
PSA Initiating Events Impact:	None	
PSA Containment Performance Impact:	None	
Conditional Core Damage Frequency due to Pressure Boundary Failure	Without OA 6.61E-04	With OA 6.61E-04
Total Segment Pressure Boundary Failure Core Damage Frequency ($FP*CDF_{cond}$)	6.61E-12	6.61E-12
CDF_{pb} Importance Measure Values	RAW 2.91E+04 RRW 1.000	5.80E+04 1.001
Comments		

Section 3 Pressure Boundary Failure Probability	
Segment Elements (welds, tees, elbows, etc.):	Valve/pipe weld
Pressure Boundary Failure Mechanism(s):	Thermal fatigue
Pressure Boundary Failure Probability:	Small Leak: 0 (use 1E-08 per demand) Full Break: 0 (use 1E-08 per demand)
Basis for Pressure Boundary Failure Probability:	Maximum Residual Stress
Comments	Location based on potential check valve leakage causing thermal cycling. Choked flow consideration during DBA not considered to be a significant loading concern (thick stainless steel piping).

WOG RISK-BASED INSPECTION
EXPERT PANEL EVALUATION
SEGMENT RANKING WORKSHEET

Segment: SIL-9

Section 4 Indirect Effects Evaluation	
Indirect Effects (spray, flood, pipe whip, jet impingement)	None
Pressure Boundary Failure Impact on Other Systems	None
Core Damage Frequency Contribution due to Indirect Effects	None

Section 5 Other Considerations		
External Events Evaluation		
Seismic:	Fire:	External Flood:
None		
Shutdown Risk Evaluation		Accumulators isolated during shutdown, do not provide function during shutdown, redundant accumulators available if necessary
Importance to Other Accident Scenarios		
Component Maintenance and Operation Insights		Review of reports conducted; no major problems found
Importance to Design Basis Analysis		
Other Deterministic Insights		

Section 6 Final Risk Category	
Category: More Safety Significant	Less Safety Significant X
Basis	Reliable piping, benign normal conditions, minimal consequence.

APPENDIX C

SAMPLE FAILURE PROBABILITY WORKSHEETS

This appendix contains sample SRRA code input worksheets and the code output. The piping segments presented are the same as those in Appendix B. The piping segments are ECCS-1 (Tables C-1 through C-3), FWS-1 (Tables C-4 through C-6), RCS-7 (Tables C-7 through C-9), RCS-15 (Tables C-10 through C-12), and SIL-9 (Tables C-13 through C-15). For a given segment, the input worksheet is shown first, followed by the small leak probability calculation output then the full break output. For the cases in which 0 failures are predicted, the values in parentheses on the worksheets are those calculated assuming one half failure in 5000 trails, corrected for importance sampling. Appendix D discusses the SRA code and its input and output parameters in detail.

Table C-1 **ECCS-1**					
Piping Structural Reliability Estimates for Millstone Unit No. 3					
System: ECCS		Segment: 1			Sheet of
P&ID No.: EM-112A, B & 113B		Data Point: 165 of X7003B			
Pipe Stress Calculation Number: X7003B 831, X10705		PSI/Const. Method: VT-2, PT, UT/Hydro, RT			
Piping Stress Isometric No.: SIL-6, 159 & 165		Proposed ISI Method: VT-2, UT			
Piping Component/Segment Element (weld, tee, elbow, etc.): Weld at valve V985					
No.	**Input Parameter Description**	**Check Input Choice (for Table 1 Value)**			**Set Value***
1	Type of Piping Material	304 SS	316 SS	Carbon Steel	---
2	Temperature at Pipe Weld	Low (150)	Medium (350)	High (550)	350
3	Nominal Pipe Size	Small (2)	Medium (5)	Large (16)	6
4	Pipe Wall Thickness	Thin (.06)	Normal (.14)	Thick (.22)	.12
5	Normal Operating Pressure	Low (0.5)	Medium (1.3)	High (2.1)	2.5
6	Residual Stress Level	None (0.0)	Moderate (0.1)	Maximum (0.2)	.2
7	Initial Flaw Size	Small (.05)	Medium (.11)	Large (.17)	.05
8	Steady-State Stress Level	Low (.05)	Medium (.11)	High (.17)	.17
9	Stress Corrosion Potential	None (0.0)	Moderate (0.5)	Maximum (1.0)	0
10	Material Wastage Potential	None (0.0)	Moderate (0.5)	Maximum (1.0)	0
11	High Cycle Fatigue Loads	None (0.0)	Moderate (.08)	Maximum (.16)	0
12	Fatigue Transient Loads	Low (.10)	Medium (.22)	High (.34)	.28
13	Fatigue Transient Frequency	Low (5)	Medium (13)	High (21)	17
14	Design-Limiting Stress (Break Only)	Low (.10)	Medium (.26)	High (.42)	.22
15	Optional Crack Inspection Interval	Low (6)	Medium (10)	High (14)	10
16	Optional crack Inspection Accuracy	High (.16)	Medium (.24)	Low (.32)	.24
*For optional numeric input, use a value (and associated units) from the standard range given in Table 1.					
Small Leak Probability, No ISI: 0 (6.4E-09)		Optional Leak Probability With ISI: 0 (6.4E-09)			
Full Break Probability, No ISI: 0 (2.3E-12)		Optional Break Probability With ISI: 0 (2.3E-12)			
Comments: Valve is located on branch line within 2 ft. of run pipe connection. Many nearby branch line snubbers exist which potentially may lockup causing break potential.					

Table C-2

ECCS-1 Small Leak Probability

```
                       STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE              PROBABILITY OF FAILURE PROGRAM SPFMPROF     ESBU-NTD
===========================================================================
     INPUT VARIABLES FOR CASE 34: 316 STAINLESS STEEL PIPE WELD LEAK

       NCYCLE =     40            NFAILS =  1000            NTRIAL =  5000
       NOVARS =     29            NUMSET =     6            NUMISI =     5
       NUMSSC =      7            NUMTRC =     7            NUMFMD =     4
```

VARIABLE NO.	NAME	DISTRIBUTION TYPE	LOG	MEDIAN VALUE	DEVIATION OR FACTOR	SHIFT MV/SD	USAGE NO.	SUB
1	PIPE-DIA	NORMAL	NO	6.0000D+00	3.0000D-02	.00	1	SET
2	WALL/DIA	NORMAL	NO	1.1000D-01	3.3000D-03	.00	2	SET
3	SRESIDUAL	NORMAL	YES	1.2357D+01	1.4125D+00	1.00	3	SET
4	INT%DEPTH	NORMAL	YES	5.0000D+00	1.4125D+00	1.00	4	SET
5	L/D-RATIO	NORMAL	YES	6.0000D+00	1.4125D+00	1.00	5	SET
6	PROB/VOL	- CONSTANT -		1.0000D-04			6	SET
7	FIRST-ISI	- CONSTANT -		5.0000D+00			1	ISI
8	FREQ-ISI	- CONSTANT -		1.0000D+01			2	ISI
9	EPST-PND	- CONSTANT -		1.0000D-03			3	ISI
10	ASTAR-PND	- CONSTANT -		-4.8000D-01			4	ISI
11	ANUU-PND	- CONSTANT -		1.6000D+00			5	ISI
12	HOURS/CY	NORMAL	YES	7.4473D+03	1.0500D+00	1.00	1	SSC
13	PRESSURE	NORMAL	NO	2.5000D+00	1.5000D-02	.00	2	SSC
14	STRESS-SS	NORMAL	YES	1.0503D+01	1.2589D+00	.00	3	SSC
15	SCC-COEFF	NORMAL	YES	3.2310D-12	2.3714D+00	1.00	4	SSC
16	SCC-EXPNT	- CONSTANT -		2.1610D+00			5	SSC
17	SCC-TIMEI	- CONSTANT -		1.0000D+00			6	SSC
18	ECW-RATE	NORMAL	YES	1.2740D-11	2.3714D+00	.00	7	SSC
19	NOFTRS/HR	- CONSTANT -		6.0000D+01			1	TRC
20	STRESS-FT	NORMAL	YES	6.1783D-02	1.4125D+00	.00	2	TRC
21	NOSTRS/CY	- CONSTANT -		1.5000D+01			3	TRC
22	STRESS-ST	NORMAL	YES	1.7917D+01	1.2589D+00	.00	4	TRC
23	FCG-COEFF	NORMAL	YES	9.1401D-12	2.8508D+00	1.00	5	TRC
24	FCG-EXPNT	- CONSTANT -		4.0000D+00			6	TRC
25	FCG-THOLD	- CONSTANT -		4.6000D+00			7	TRC
26	LIMIT-DSL	NORMAL	NO	-9.7000D-01	1.0000D-02	.00	1	FMD
27	LIMIT-PBS	- CONSTANT -		0.0000D+00			2	FMD
28	STRESS-DL	- CONSTANT -		0.0000D+00			3	FMD
29	FREQ-DLTR	- CONSTANT -		0.0000D+00			4	FMD

PROBABILITIES OF FAILURE MODE: EXCEED LIMITING DEPTH FOR SMALL LEAK

NUMBER FAILED = 0 NUMBER OF TRIALS = 5000

END OF CYCLE	FAILURE PROBABILITY WITHOUT FOR PERIOD	CUM. TOTAL	AND WITH IN-SERVICE INSPECTION FOR PERIOD	CUM. TOTAL
.0	6.37720D-09	6.37720D-09	6.37720D-09	6.37720D-09
40.0	0.00000D+00	6.37720D-09	0.00000D+00	6.37720D-09

Table C-3

ECCS-1 Full Break Probability

```
                    STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE            PROBABILITY OF FAILURE PROGRAM SPFMPROF          ESBU-NTD
==========================================================================
    INPUT VARIABLES FOR CASE 35: 316 STAINLESS STEEL PIPE WELD BREAK

    NCYCLE =     40          NFAILS =  1000          NTRIAL =  5000
    NOVARS =     29          NUMSET =     6          NUMISI =     5
    NUMSSC =      7          NUMTRC =     7          NUMFMD =     4
```

VARIABLE NO.	NAME	DISTRIBUTION TYPE	LOG	MEDIAN VALUE	DEVIATION OR FACTOR	SHIFT MV/SD	USAGE NO.	SUB
1	PIPE-DIA	NORMAL	NO	6.0000D+00	3.0000D-02	.00	1	SET
2	WALL/DIA	NORMAL	NO	1.1000D-01	3.3000D-03	.00	2	SET
3	SRESIDUAL	NORMAL	YES	1.2357D+01	1.4125D+00	1.00	3	SET
4	INT%DEPTH	NORMAL	YES	5.0000D+00	1.4125D+00	1.00	4	SET
5	L/D-RATIO	NORMAL	YES	6.0000D+00	1.4125D+00	1.00	5	SET
6	PROB/VOL	- CONSTANT -		1.0000D-04			6	SET
7	FIRST-ISI	- CONSTANT -		5.0000D+00			1	ISI
8	FREQ-ISI	- CONSTANT -		1.0000D+01			2	ISI
9	EPST-PND	- CONSTANT -		1.0000D-03			3	ISI
10	ASTAR-PND	- CONSTANT -		-4.8000D-01			4	ISI
11	ANUU-PND	- CONSTANT -		1.6000D+00			5	ISI
12	HOURS/CY	NORMAL	YES	7.4473D+03	1.0500D+00	1.00	1	SSC
13	PRESSURE	NORMAL	NO	2.5000D+00	1.5000D-02	.00	2	SSC
14	STRESS-SS	NORMAL	YES	1.0503D+01	1.2589D+00	.00	3	SSC
15	SCC-COEFF	NORMAL	YES	3.2310D-12	2.3714D+00	1.00	4	SSC
16	SCC-EXPNT	- CONSTANT -		2.1610D+00			5	SSC
17	SCC-TIMEI	- CONSTANT -		1.0000D+00			6	SSC
18	ECW-RATE	NORMAL	YES	1.2740D-11	2.3714D+00	.00	7	SSC
19	NOFTRS/HR	- CONSTANT -		6.0000D+01			1	TRC
20	STRESS-FT	NORMAL	YES	6.1783D-02	1.4125D+00	.00	2	TRC
21	NOSTRS/CY	- CONSTANT -		1.5000D+01			3	TRC
22	STRESS-ST	NORMAL	YES	1.7917D+01	1.2589D+00	.00	4	TRC
23	FCG-COEFF	NORMAL	YES	9.1401D-12	2.8508D+00	1.00	5	TRC
24	FCG-EXPNT	- CONSTANT -		4.0000D+00			6	TRC
25	FCG-THOLD	- CONSTANT -		4.6000D+00			7	TRC
26	LIMIT-DSL	- CONSTANT -		0.0000D+00			1	FMD
27	LIMIT-PBS	NORMAL	NO	6.1783D+01	3.2000D+00	-1.00	2	FMD
28	STRESS-DL	NORMAL	YES	1.4210D+01	1.4125D+00	1.00	3	FMD
29	FREQ-DLTR	- CONSTANT -		1.0000D-03			4	FMD

PROBABILITIES OF FAILURE MODE: EXCEED FLOW STRESS LIMIT FOR FULL BREAK

NUMBER FAILED = 0 NUMBER OF TRIALS = 5000

END OF CYCLE	FAILURE PROBABILITY WITHOUT FOR PERIOD	CUM. TOTAL	AND WITH IN-SERVICE INSPECTION FOR PERIOD	CUM. TOTAL
.0	2.34604D-12	2.34604D-12	2.34604D-12	2.34604D-12
40.0	0.00000D+00	2.34604D-12	0.00000D+00	2.34604D-12

Table C-4
FWS-1

Piping Structural Reliability Estimates for Millstone Unit No. 3

System: FWS	Segment: 1	Sheet of
P&ID No.: EM-130C	Data Point: 410	
Pipe Stress Calculation Number: X1709	PSI/Const. Method: VT-2/Hydro, RT	
Piping Stress Isometric No.: C.I. FWS-11	Proposed ISI Method: VT-2, UT	

Piping Component/Segment Element (weld, tee, elbow, etc.): Pipe to valve (V14) weld

No.	Input Parameter Description	Check Input Choice (for Table 1 Value)			Set Value*
1	Type of Piping Material	304 SS	316 SS	Carbon Steel	---
2	Temperature at Pipe Weld	Low (150)	Medium (350)	High (550)	446
3	Nominal Pipe Size	Small (2)	Medium (5)	Large (16)	18
4	Pipe Wall Thickness	Thin (.06)	Normal (.14)	Thick (.22)	.06
5	Normal Operating Pressure	Low (0.5)	Medium (1.3)	High (2.1)	1.8
6	Residual Stress Level	None (0.0)	Moderate (0.1)	Maximum (0.2)	0.1
7	Initial Flaw Size	Small (.05)	Medium (.11)	Large (.17)	.05
8	Steady-State Stress Level	Low (.05)	Medium (.11)	High (.17)	.08
9	Stress Corrosion Potential	None (0.0)	Moderate (0.5)	Maximum (1.0)	0
10	Material Wastage Potential	None (0.0)	Moderate (0.5)	Maximum (1.0)	0.5
11	High Cycle Fatigue Loads	None (0.0)	Moderate (.08)	Maximum (.16)	0
12	Fatigue Transient Loads	Low (.10)	Medium (.22)	High (.34)	0.1
13	Fatigue Transient Frequency	Low (5)	Medium (13)	High (21)	13
14	Design-Limiting Stress (Break Only)	Low (.10)	Medium (.26)	High (.42)	.16
15	Optional Crack Inspection Interval	Low (6)	Medium (10)	High (14)	10
16	Optional crack Inspection Accuracy	High (.16)	Medium (.24)	Low (.32)	.24

*For optional numeric input, use a value (and associated units) from the standard range given in Table 1.

Small Leak Probability, No ISI: 1.09E-3	Optional Leak Probability With ISI: 6.21E-06
Full Break Probability, No ISI: 0 (3.5E-11)	Optional Break Probability With ISI: 0 (3.5E-11)

Comments: Break exclusion zone. No EC trending, LOC 040-016 US.

Table C-5

FWS-1 Small Leak Probability

```
                STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE          PROBABILITY OF FAILURE PROGRAM SPFMPROF          ESBU-NTD

================================================================================
      INPUT VARIABLES FOR CASE  2: CARBON STEEL PIPE WELD SMALL LEAK

      NCYCLE =     40            NFAILS =  1000            NTRIAL =  5000
      NOVARS =     29            NUMSET =     6            NUMISI =     5
      NUMSSC =      7            NUMTRC =     7            NUMFMD =     4
```

VARIABLE NO.	VARIABLE NAME	DISTRIBUTION TYPE	DISTRIBUTION LOG	MEDIAN VALUE	DEVIATION OR FACTOR	SHIFT MV/SD	USAGE NO.	USAGE SUB
1	PIPE-DIA	NORMAL	NO	1.8000D+01	9.0000D-02	.00	1	SET
2	WALL/DIA	NORMAL	NO	6.0000D-02	1.8000D-03	.00	2	SET
3	SRESIDUAL	NORMAL	YES	6.4337D+00	1.4125D+00	.00	3	SET
4	INT%DEPTH	NORMAL	YES	5.0000D+00	1.4125D+00	1.00	4	SET
5	L/D-RATIO	NORMAL	YES	6.0000D+00	1.4125D+00	1.00	5	SET
6	PROB/VOL	- CONSTANT -		1.0000D-04			6	SET
7	FIRST-ISI	- CONSTANT -		5.0000D+00			1	ISI
8	FREQ-ISI	- CONSTANT -		1.0000D+01			2	ISI
9	EPST-PND	- CONSTANT -		5.0000D-03			3	ISI
10	ASTAR-PND	- CONSTANT -		-2.4000D-01			4	ISI
11	ANUU-PND	- CONSTANT -		3.0000D+00			5	ISI
12	HOURS/CY	NORMAL	YES	7.4473D+03	1.0500D+00	1.00	1	SSC
13	PRESSURE	NORMAL	NO	1.8000D+00	1.5000D-02	.00	2	SSC
14	STRESS-SS	NORMAL	YES	5.1470D+00	1.2589D+00	.00	3	SSC
15	SCC-COEFF	NORMAL	YES	3.5900D-13	2.3714D+00	.00	4	SSC
16	SCC-EXPNT	- CONSTANT -		2.1610D+00			5	SSC
17	SCC-TIMEI	- CONSTANT -		1.0000D+00			6	SSC
18	ECW-RATE	NORMAL	YES	6.3700D-07	2.3714D+00	.00	7	SSC
19	NOFTRS/HR	- CONSTANT -		6.0000D+01			1	TRC
20	STRESS-FT	NORMAL	YES	6.4337D-02	1.4125D+00	.00	2	TRC
21	NOSTRS/CY	- CONSTANT -		1.3000D+01			3	TRC
22	STRESS-ST	NORMAL	YES	6.4337D+00	1.2589D+00	1.00	4	TRC
23	FCG-COEFF	NORMAL	YES	1.2017D-11	2.8508D+00	1.00	5	TRC
24	FCG-EXPNT	- CONSTANT -		3.7000D+00			6	TRC
25	FCG-THOLD	- CONSTANT -		3.5000D+00			7	TRC
26	LIMIT-DSL	NORMAL	NO	-9.7000D-01	1.0000D-02	.00	1	FMD
27	LIMIT-PBS	- CONSTANT -		0.0000D+00			2	FMD
28	STRESS-DL	- CONSTANT -		0.0000D+00			3	FMD
29	FREQ-DLTR	- CONSTANT -		0.0000D+00			4	FMD

PROBABILITIES OF FAILURE MODE: EXCEED LIMITING DEPTH FOR SMALL LEAK

NUMBER FAILED = 316   NUMBER OF TRIALS = 5000

END OF CYCLE	FAILURE PROBABILITY WITHOUT: FOR PERIOD	CUM. TOTAL	AND WITH IN-SERVICE INSPECTION: FOR PERIOD	CUM. TOTAL
6.0	1.41843D-08	1.41843D-08	7.09612D-11	7.09612D-11
7.0	8.24432D-07	8.38616D-07	4.34590D-09	4.41686D-09
8.0	4.48267D-07	1.28688D-06	2.25107D-09	6.66793D-09
9.0	1.86245D-05	1.99114D-05	5.15727D-07	5.22395D-07
10.0	1.62683D-07	2.00740D-05	1.18905D-09	5.23584D-07
11.0	2.59319D-05	4.60059D-05	9.03992D-07	1.42758D-06

Table C-5 (Continued)

FWS-1 Small Leak Probability

12.0	8.95441D-07	4.69014D-05	3.93405D-08	1.46692D-06
14.0	3.21027D-06	5.01116D-05	1.22563D-06	2.69255D-06
15.0	2.74723D-08	5.01391D-05	6.15578D-09	2.69871D-06
16.0	2.95454D-06	5.30936D-05	6.75335D-09	2.70546D-06
17.0	1.58686D-05	6.89622D-05	5.10427D-08	2.75650D-06
18.0	5.31092D-07	6.94933D-05	1.32861D-09	2.75783D-06
19.0	6.22227D-05	1.31716D-04	1.89205D-07	2.94704D-06
20.0	1.34045D-05	1.45121D-04	1.82564D-08	2.96529D-06
21.0	8.13526D-06	1.53256D-04	2.86175D-08	2.99391D-06
22.0	6.98358D-06	1.60239D-04	4.41429D-08	3.03805D-06
23.0	1.05365D-04	2.65604D-04	1.05385D-06	4.09190D-06
24.0	1.05498D-04	3.71102D-04	4.55720D-07	4.54762D-06
25.0	8.28412D-05	4.53943D-04	1.51379D-06	6.06141D-06
26.0	1.63160D-05	4.70259D-04	7.16360D-10	6.06212D-06
27.0	2.23614D-04	6.93873D-04	3.49592D-08	6.09708D-06
28.0	1.09478D-04	8.03351D-04	2.32728D-08	6.12036D-06
29.0	1.08010D-05	8.14152D-04	9.21074D-10	6.12128D-06
30.0	1.78803D-05	8.32032D-04	3.64537D-09	6.12492D-06
31.0	4.47131D-06	8.36503D-04	9.58423D-10	6.12588D-06
32.0	7.85007D-05	9.15004D-04	4.43658D-08	6.17025D-06
33.0	9.77842D-06	9.24782D-04	4.74730D-09	6.17499D-06
34.0	1.75473D-05	9.42330D-04	1.57386D-08	6.19073D-06
35.0	2.58613D-05	9.68191D-04	1.96033D-08	6.21034D-06
36.0	3.97057D-05	1.00790D-03	2.22109D-10	6.21056D-06
37.0	4.21448D-05	1.05004D-03	1.58531D-10	6.21072D-06
38.0	7.24170D-06	1.05728D-03	5.41483D-11	6.21077D-06
39.0	1.53097D-05	1.07259D-03	1.93248D-10	6.21096D-06
40.0	1.44083D-05	1.08700D-03	2.48961D-10	6.21121D-06
	DEVIATION ON CUMULATIVE TOTALS =		5.91907D-05	4.62194D-06

Table C-6

FWS-1 Full Break Probability

```
                    STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE           PROBABILITY OF FAILURE PROGRAM SPFMPROF         ESBU-NTD

==============================================================================
     INPUT VARIABLES FOR CASE  3: CARBON STEEL PIPE WELD FULL BREAK

     NCYCLE =     40            NFAILS =  1000             NTRIAL =  5000
     NOVARS =     29            NUMSET =     6             NUMISI =     5
     NUMSSC =      7            NUMTRC =     7             NUMFMD =     4
```

VARIABLE NO.	NAME	DISTRIBUTION TYPE	LOG	MEDIAN VALUE	DEVIATION OR FACTOR	SHIFT MV/SD	USAGE NO.	SUB
1	PIPE-DIA	NORMAL	NO	1.8000D+01	9.0000D-02	.00	1	SET
2	WALL/DIA	NORMAL	NO	6.0000D-02	1.8000D-03	.00	2	SET
3	SRESIDUAL	NORMAL	YES	6.4337D+00	1.4125D+00	.00	3	SET
4	INT%DEPTH	NORMAL	YES	5.0000D+00	1.4125D+00	1.00	4	SET
5	L/D-RATIO	NORMAL	YES	6.0000D+00	1.4125D+00	1.00	5	SET
6	PROB/VOL	- CONSTANT -		1.0000D-04			6	SET
7	FIRST-ISI	- CONSTANT -		5.0000D+00			1	ISI
8	FREQ-ISI	- CONSTANT -		1.0000D+01			2	ISI
9	EPST-PND	- CONSTANT -		5.0000D-03			3	ISI
10	ASTAR-PND	- CONSTANT -		-2.4000D-01			4	ISI
11	ANUU-PND	- CONSTANT -		3.0000D+00			5	ISI
12	HOURS/CY	NORMAL	YES	7.4473D+03	1.0500D+00	1.00	1	SSC
13	PRESSURE	NORMAL	NO	1.8000D+00	1.5000D-02	.00	2	SSC
14	STRESS-SS	NORMAL	YES	5.1470D+00	1.2589D+00	.00	3	SSC
15	SCC-COEFF	NORMAL	YES	3.5900D-13	2.3714D+00	.00	4	SSC
16	SCC-EXPNT	- CONSTANT -		2.1610D+00			5	SSC
17	SCC-TIMEI	- CONSTANT -		1.0000D+00			6	SSC
18	ECW-RATE	NORMAL	YES	6.3700D-07	2.3714D+00	.00	7	SSC
19	NOFTRS/HR	- CONSTANT -		6.0000D+01			1	TRC
20	STRESS-FT	NORMAL	YES	6.4337D-02	1.4125D+00	.00	2	TRC
21	NOSTRS/CY	- CONSTANT -		1.3000D+01			3	TRC
22	STRESS-ST	NORMAL	YES	6.4337D+00	1.2589D+00	1.00	4	TRC
23	FCG-COEFF	NORMAL	YES	1.2017D-11	2.8508D+00	1.00	5	TRC
24	FCG-EXPNT	- CONSTANT -		3.7000D+00			6	TRC
25	FCG-THOLD	- CONSTANT -		3.5000D+00			7	TRC
26	LIMIT-DSL	- CONSTANT -		0.0000D+00			1	FMD
27	LIMIT-PBS	NORMAL	NO	6.4337D+01	3.2000D+00	-1.00	2	FMD
28	STRESS-DL	NORMAL	YES	1.0294D+01	1.4125D+00	1.00	3	FMD
29	FREQ-DLTR	- CONSTANT -		1.0000D-03			4	FMD

PROBABILITIES OF FAILURE MODE: EXCEED FLOW STRESS LIMIT FOR FULL BREAK

NUMBER FAILED = 0 NUMBER OF TRIALS = 5000

END OF CYCLE	FAILURE PROBABILITY WITHOUT: FOR PERIOD	CUM. TOTAL	AND WITH IN-SERVICE INSPECTION: FOR PERIOD	CUM. TOTAL
.0	3.50552D-11	3.50552D-11	3.50552D-11	3.50552D-11
40.0	0.00000D+00	3.50552D-11	0.00000D+00	3.50552D-11

Table C-7
RCS-7

Piping Structural Reliability Estimates for Millstone Unit No. 3

System: Reactor Coolant System	Segment: RCS-7	Sheet of
P&ID No.: 12179-EM-102A R10	Data Point: 1021	
Pipe Stress Calculation Number: X7001B	PSI/Const. Method: VT-2, PT, UT/Hydro, PT,RT	
Piping Stress Isometric No.:	Proposed ISI Method: VT-2, UT	

Piping Component/Segment Element (weld, tee, elbow, etc.): Pipe weld at conn RCL

No.	Input Parameter Description	Check Input Choice (for Table 1 Value)			Set Value*
1	Type of Piping Material	304 SS	316 SS	Carbon Steel	---
2	Temperature at Pipe Weld	Low (150)	Medium (350)	High (550)	600
3	Nominal Pipe Size	Small (2)	Medium (5)	Large (16)	10
4	Pipe Wall Thickness	Thin (.06)	Normal (.14)	Thick (.22)	.1
5	Normal Operating Pressure	Low (0.5)	Medium (1.3)	High (2.1)	2.5
6	Residual Stress Level	None (0.0)	Moderate (0.1)	Maximum (0.2)	.2
7	Initial Flaw Size	Small (.05)	Medium (.11)	Large (.17)	.05
8	Steady-State Stress Level	Low (.05)	Medium (.11)	High (.17)	.14
9	Stress Corrosion Potential	None (0.0)	Moderate (0.5)	Maximum (1.0)	0
10	Material Wastage Potential	None (0.0)	Moderate (0.5)	Maximum (1.0)	0
11	High Cycle Fatigue Loads	None (0.0)	Moderate (.08)	Maximum (.16)	.08
12	Fatigue Transient Loads	Low (.10)	Medium (.22)	High (.34)	.25
13	Fatigue Transient Frequency	Low (5)	Medium (13)	High (21)	5
14	Design-Limiting Stress (Break Only)	Low (.10)	Medium (.26)	High (.42)	.22
15	Optional Crack Inspection Interval	Low (6)	Medium (10)	High (14)	10
16	Optional crack Inspection Accuracy	High (.16)	Medium (.24)	Low (.32)	.24

*For optional numeric input, use a value (and associated units) from the standard range given in Table 1.

Small Leak Probability, No ISI: 1.85E-06	Optional Leak Probability With ISI: 1.30E-06
Full Break Probability, No ISI: 4.15E-09	Optional Break Probability With ISI: 3.44E-09

Comments: High usage factor. Branch is on Fatigue watch list.

Table C-8

RCS-7 Small Leak Probability

```
                         STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE                PROBABILITY OF FAILURE PROGRAM SPFMPROF              ESBU-NTD

==================================================================================
      INPUT VARIABLES FOR CASE 53: 316 STAINLESS STEEL PIPE WELD LEAK

      NCYCLE =     40            NFAILS =  1000           NTRIAL =  5000
      NOVARS =     29            NUMSET =     6           NUMISI =     5
      NUMSSC =      7            NUMTRC =     7           NUMFMD =     4
```

VARIABLE NO.	NAME	DISTRIBUTION TYPE	LOG	MEDIAN VALUE	DEVIATION OR FACTOR	SHIFT MV/SD	USAGE NO.	SUB
1	PIPE-DIA	NORMAL	NO	1.0000D+01	5.0000D-02	.00	1	SET
2	WALL/DIA	NORMAL	NO	9.0000D-02	2.7000D-03	.00	2	SET
3	SRESIDUAL	NORMAL	YES	1.0318D+01	1.4125D+00	1.00	3	SET
4	INT%DEPTH	NORMAL	YES	5.0000D+00	1.4125D+00	1.00	4	SET
5	L/D-RATIO	NORMAL	YES	6.0000D+00	1.4125D+00	1.00	5	SET
6	PROB/VOL	- CONSTANT -		1.0000D-04			6	SET
7	FIRST-ISI	- CONSTANT -		5.0000D+00			1	ISI
8	FREQ-ISI	- CONSTANT -		1.0000D+01			2	ISI
9	EPST-PND	- CONSTANT -		1.0000D-03			3	ISI
10	ASTAR-PND	- CONSTANT -		-4.8000D-01			4	ISI
11	ANUU-PND	- CONSTANT -		1.6000D+00			5	ISI
12	HOURS/CY	NORMAL	YES	7.4473D+03	1.0500D+00	1.00	1	SSC
13	PRESSURE	NORMAL	NO	2.7000D+00	1.5000D-02	.00	2	SSC
14	STRESS-SS	NORMAL	YES	7.7003D+00	1.2589D+00	.00	3	SSC
15	SCC-COEFF	NORMAL	YES	3.2310D-12	2.3714D+00	1.00	4	SSC
16	SCC-EXPNT	- CONSTANT -		2.1610D+00			5	SSC
17	SCC-TIMEI	- CONSTANT -		1.0000D+00			6	SSC
18	ECW-RATE	NORMAL	YES	1.2740D-11	2.3714D+00	.00	7	SSC
19	NOFTRS/HR	- CONSTANT -		6.0000D+01			1	TRC
20	STRESS-FT	NORMAL	YES	4.1068D+00	1.4125D+00	.00	2	TRC
21	NOSTRS/CY	- CONSTANT -		5.0000D+00			3	TRC
22	STRESS-ST	NORMAL	YES	1.2834D+01	1.2589D+00	.00	4	TRC
23	FCG-COEFF	NORMAL	YES	9.1401D-12	2.8508D+00	1.00	5	TRC
24	FCG-EXPNT	- CONSTANT -		4.0000D+00			6	TRC
25	FCG-THOLD	- CONSTANT -		4.6000D+00			7	TRC
26	LIMIT-DSL	NORMAL	NO	-9.7000D-01	1.0000D-02	.00	1	FMD
27	LIMIT-PBS	- CONSTANT -		0.0000D+00			2	FMD
28	STRESS-DL	- CONSTANT -		0.0000D+00			3	FMD
29	FREQ-DLTR	- CONSTANT -		0.0000D+00			4	FMD

PROBABILITIES OF FAILURE MODE: EXCEED LIMITING DEPTH FOR SMALL LEAK

NUMBER FAILED = 38 NUMBER OF TRIALS = 5000

END OF CYCLE	FAILURE PROBABILITY WITHOUT FOR PERIOD	CUM. TOTAL	AND WITH IN-SERVICE INSPECTION FOR PERIOD	CUM. TOTAL
2.0	8.94271D-10	8.94271D-10	8.94271D-10	8.94271D-10
3.0	1.01876D-08	1.10818D-08	1.01876D-08	1.10818D-08
4.0	5.04658D-08	6.15476D-08	5.04658D-08	6.15476D-08
5.0	9.95457D-08	1.61093D-07	9.95457D-08	1.61093D-07
6.0	3.65580D-08	1.97651D-07	7.38916D-09	1.68482D-07
7.0	1.34157D-09	1.98993D-07	8.06409D-10	1.69289D-07

Table C-8 (Continued)

RCS-7 Small Leak Probability

8.0	2.70362D-09	2.01697D-07	2.26866D-09	1.71558D-07
9.0	2.09142D-10	2.01906D-07	1.21163D-10	1.71679D-07
10.0	4.51808D-07	6.53714D-07	4.44936D-07	6.16614D-07
11.0	2.28950D-07	8.82664D-07	2.25890D-07	8.42504D-07
12.0	1.13720D-08	8.94036D-07	1.11866D-08	8.53691D-07
14.0	5.01018D-08	9.44137D-07	4.95022D-08	9.03193D-07
18.0	3.38555D-08	9.77993D-07	1.21196D-08	9.15312D-07
19.0	1.49986D-09	9.79493D-07	8.09079D-10	9.16122D-07
21.0	5.88162D-09	9.85374D-07	4.88857D-09	9.21010D-07
26.0	3.87838D-07	1.37321D-06	7.42869D-08	9.95297D-07
28.0	2.32675D-08	1.39648D-06	2.60714D-09	9.97904D-07
35.0	3.64726D-07	1.76121D-06	2.94839D-07	1.29274D-06
36.0	8.99882D-08	1.85119D-06	4.59765D-09	1.29734D-06
40.0	0.00000D+00	1.85119D-06	0.00000D+00	1.29734D-06
	DEVIATION ON CUMULATIVE TOTALS =		2.99190D-07	2.50752D-07

Table C-9

RCS-7 Full Break Probability

```
                     STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE             PROBABILITY OF FAILURE PROGRAM SPFMPROF           ESBU-NTD

================================================================================
     INPUT VARIABLES FOR CASE 54: 316 STAINLESS STEEL PIPE WELD BREAK

     NCYCLE =     40            NFAILS =  1000            NTRIAL =  5000
     NOVARS =     29            NUMSET =     6            NUMISI =     5
     NUMSSC =      7            NUMTRC =     7            NUMFMD =     4
```

VARIABLE NO.	NAME	DISTRIBUTION TYPE	LOG	MEDIAN VALUE	DEVIATION OR FACTOR	SHIFT MV/SD	USAGE NO.	SUB
1	PIPE-DIA	NORMAL	NO	1.0000D+01	5.0000D-02	.00	1	SET
2	WALL/DIA	NORMAL	NO	9.0000D-02	2.7000D-03	.00	2	SET
3	SRESIDUAL	NORMAL	YES	1.0318D+01	1.4125D+00	1.00	3	SET
4	INT%DEPTH	NORMAL	YES	5.0000D+00	1.4125D+00	1.00	4	SET
5	L/D-RATIO	NORMAL	YES	6.0000D+00	1.4125D+00	1.00	5	SET
6	PROB/VOL	- CONSTANT -		1.0000D-04			6	SET
7	FIRST-ISI	- CONSTANT -		5.0000D+00			1	ISI
8	FREQ-ISI	- CONSTANT -		1.0000D+01			2	ISI
9	EPST-PND	- CONSTANT -		1.0000D-03			3	ISI
10	ASTAR-PND	- CONSTANT -		-4.8000D-01			4	ISI
11	ANUU-PND	- CONSTANT -		1.6000D+00			5	ISI
12	HOURS/CY	NORMAL	YES	7.4473D+03	1.0500D+00	1.00	1	SSC
13	PRESSURE	NORMAL	NO	2.7000D+00	1.5000D-02	.00	2	SSC
14	STRESS-SS	NORMAL	YES	7.7003D+00	1.2589D+00	.00	3	SSC
15	SCC-COEFF	NORMAL	YES	3.2310D-12	2.3714D+00	1.00	4	SSC
16	SCC-EXPNT	- CONSTANT -		2.1610D+00			5	SSC
17	SCC-TIMEI	- CONSTANT -		1.0000D+00			6	SSC
18	ECW-RATE	NORMAL	YES	1.2740D-11	2.3714D+00	.00	7	SSC
19	NOFTRS/HR	- CONSTANT -		6.0000D+01			1	TRC
20	STRESS-FT	NORMAL	YES	4.1068D+00	1.4125D+00	.00	2	TRC
21	NOSTRS/CY	- CONSTANT -		5.0000D+00			3	TRC
22	STRESS-ST	NORMAL	YES	1.2834D+01	1.2589D+00	.00	4	TRC
23	FCG-COEFF	NORMAL	YES	9.1401D-12	2.8508D+00	1.00	5	TRC
24	FCG-EXPNT	- CONSTANT -		4.0000D+00			6	TRC
25	FCG-THOLD	- CONSTANT -		4.6000D+00			7	TRC
26	LIMIT-DSL	- CONSTANT -		0.0000D+00			1	FMD
27	LIMIT-PBS	NORMAL	NO	5.1336D+01	3.2000D+00	-1.00	2	FMD
28	STRESS-DL	NORMAL	YES	1.1807D+01	1.4125D+00	1.00	3	FMD
29	FREQ-DLTR	- CONSTANT -		1.0000D-03			4	FMD

PROBABILITIES OF FAILURE MODE: EXCEED FLOW STRESS LIMIT FOR FULL BREAK

NUMBER FAILED = 40 NUMBER OF TRIALS = 5000

END OF CYCLE	FAILURE PROBABILITY WITHOUT FOR PERIOD	CUM. TOTAL	AND WITH IN-SERVICE INSPECTION FOR PERIOD	CUM. TOTAL
3.0	3.32838D-12	3.32838D-12	3.32838D-12	3.32838D-12
4.0	4.56267D-14	3.37400D-12	4.56267D-14	3.37400D-12
5.0	1.11528D-09	1.11865D-09	1.11528D-09	1.11865D-09
6.0	1.80913D-12	1.12046D-09	8.92447D-14	1.11874D-09
7.0	5.08248D-10	1.62871D-09	1.35968D-10	1.25471D-09
8.0	8.65115D-13	1.62957D-09	2.01630D-13	1.25491D-09

Table C-9 (Continued)

RCS-7 Full Break Probability

9.0	3.43633D-12	1.63301D-09	2.85746D-12	1.25777D-09
10.0	1.16420D-11	1.64465D-09	9.38850D-12	1.26716D-09
11.0	3.90819D-10	2.03547D-09	3.83862D-10	1.65102D-09
13.0	9.94750D-11	2.13495D-09	9.90966D-11	1.75011D-09
14.0	1.13095D-12	2.13608D-09	1.00875D-12	1.75112D-09
15.0	2.06633D-12	2.13814D-09	2.05977D-12	1.75318D-09
17.0	1.40478D-12	2.13955D-09	6.68420D-14	1.75325D-09
19.0	3.61956D-11	2.17574D-09	8.23173D-12	1.76148D-09
20.0	2.13062D-11	2.19705D-09	8.09302D-12	1.76957D-09
22.0	3.36388D-12	2.20041D-09	1.92871D-12	1.77150D-09
24.0	1.90910D-09	4.10951D-09	1.66636D-09	3.43786D-09
26.0	3.11303D-11	4.14064D-09	1.30261D-12	3.43917D-09
30.0	8.01516D-12	4.14866D-09	1.98107D-12	3.44115D-09
40.0	0.00000D+00	4.14866D-09	0.00000D+00	3.44115D-09
	DEVIATION ON CUMULATIVE TOTALS =		6.53396D-10	5.95488D-10

Table C-10
RCS-15

Piping Structural Reliability Estimates for Millstone Unit No. 3					
System: Reactor Coolant System		Segment: RCS-15			Sheet of
P&ID No.: 12179-EM-102D R4		Data Point: 530			
Pipe Stress Calculation Number: X10702		PSI/Const. Method: VT-2, PT/Hydro, PT, RT			
Piping Stress Isometric No.:		Proposed ISI Method: VT-2, RT			
Piping Component/Segment Element (weld, tee, elbow, etc.): Weld to V70					
No.	**Input Parameter Description**	**Check Input Choice (for Table 1 Value)**			**Set Value***
1	Type of Piping Material	304 SS	316 SS	Carbon Steel	---
2	Temperature at Pipe Weld	Low (150)	Medium (350)	High (550)	600
3	Nominal Pipe Size	Small (2)	Medium (5)	Large (16)	1.5
4	Pipe Wall Thickness	Thin (.06)	Normal (.14)	Thick (.22)	.14
5	Normal Operating Pressure	Low (0.5)	Medium (1.3)	High (2.1)	2.5
6	Residual Stress Level	None (0.0)	Moderate (0.1)	Maximum (0.2)	.2
7	Initial Flaw Size	Small (.05)	Medium (.11)	Large (.17)	.05
8	Steady-State Stress Level	Low (.05)	Medium (.11)	High (.17)	.11
9	Stress Corrosion Potential	None (0.0)	Moderate (0.5)	Maximum (1.0)	0
10	Material Wastage Potential	None (0.0)	Moderate (0.5)	Maximum (1.0)	0
11	High Cycle Fatigue Loads	None (0.0)	Moderate (.08)	Maximum (.16)	0
12	Fatigue Transient Loads	Low (.10)	Medium (.22)	High (.34)	.16
13	Fatigue Transient Frequency	Low (5)	Medium (13)	High (21)	5
14	Design-Limiting Stress (Break Only)	Low (.10)	Medium (.26)	High (.42)	.22
15	Optional Crack Inspection Interval	Low (6)	Medium (10)	High (14)	10
16	Optional crack Inspection Accuracy	High (.16)	Medium (.24)	Low (.32)	.16
*For optional numeric input, use a value (and associated units) from the standard range given in Table 1.					
Small Leak Probability, No ISI: 0 (1.7E-10)		Optional Leak Probability With ISI: 0 (1.7E-10)			
Full Break Probability, No ISI: 1.47E-12		Optional Break Probability With ISI: 1.47E-12			
Comments: Area of maximum bending stress. SR el at 535/540 & tee at 550 are on fatigue watch list.					

Table C-11

RCS-15 Small Leak Probability

```
                          STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE                 PROBABILITY OF FAILURE PROGRAM SPFMPROF            ESBU-NTD
 ==========================================================================================
      INPUT VARIABLES FOR CASE 67: 316 STAINLESS STEEL PIPE WELD LEAK

      NCYCLE =      40            NFAILS =  1000            NTRIAL =  5000
      NOVARS =      29            NUMSET =     6            NUMISI =     5
      NUMSSC =       7            NUMTRC =     7            NUMFMD =     4

    VARIABLE         DISTRIBUTION        MEDIAN        DEVIATION     SHIFT      USAGE
   NO.   NAME        TYPE      LOG        VALUE        OR FACTOR     MV/SD    NO.  SUB

    1  PIPE-DIA      NORMAL     NO     1.5000D+00    7.5000D-03      .00      1   SET
    2  WALL/DIA      NORMAL     NO     1.5000D-01    4.5000D-03      .00      2   SET
    3  SRESIDUAL     NORMAL     YES    1.0318D+01    1.4125D+00     1.00      3   SET
    4  INT%DEPTH     NORMAL     YES    5.0000D+00    1.4125D+00     1.00      4   SET
    5  L/D-RATIO     NORMAL     YES    6.0000D+00    1.4125D+00     1.00      5   SET
    6  PROB/VOL      - CONSTANT -      1.0000D-04                             6   SET
    7  FIRST-ISI     - CONSTANT -      5.0000D+00                             1   ISI
    8  FREQ-ISI      - CONSTANT -      1.0000D+01                             2   ISI
    9  EPST-PND      - CONSTANT -      1.0000D-03                             3   ISI
   10  ASTAR-PND     - CONSTANT -     -3.2000D-01                             4   ISI
   11  ANUU-PND      - CONSTANT -      1.6000D+00                             5   ISI
   12  HOURS/CY      NORMAL     YES    7.4473D+03    1.0500D+00     1.00      1   SSC
   13  PRESSURE      NORMAL     NO     2.7250D+00    1.5000D-02      .00      2   SSC
   14  STRESS-SS     NORMAL     YES    5.6469D+00    1.2589D+00      .00      3   SSC
   15  SCC-COEFF     NORMAL     YES    3.2310D-12    2.3714D+00     1.00      4   SSC
   16  SCC-EXPNT     - CONSTANT -      2.1610D+00                             5   SSC
   17  SCC-TIMEI     - CONSTANT -      1.0000D+00                             6   SSC
   18  ECW-RATE      NORMAL     YES    1.2740D-11    2.3714D+00      .00      7   SSC
   19  NOFTRS/HR     - CONSTANT -      6.0000D+01                             1   TRC
   20  STRESS-FT     NORMAL     YES    5.1336D-02    1.4125D+00      .00      2   TRC
   21  NOSTRS/CY     - CONSTANT -      5.0000D+00                             3   TRC
   22  STRESS-ST     NORMAL     YES    8.7271D+00    1.2589D+00      .00      4   TRC
   23  FCG-COEFF     NORMAL     YES    9.1401D-12    2.8508D+00     1.00      5   TRC
   24  FCG-EXPNT     - CONSTANT -      4.0000D+00                             6   TRC
   25  FCG-THOLD     - CONSTANT -      4.6000D+00                             7   TRC
   26  LIMIT-DSL     NORMAL     NO    -9.7000D-01    1.0000D-02      .00      1   FMD
   27  LIMIT-PBS     - CONSTANT -      0.0000D+00                             2   FMD
   28  STRESS-DL     - CONSTANT -      0.0000D+00                             3   FMD
   29  FREQ-DLTR     - CONSTANT -      0.0000D+00                             4   FMD

    PROBABILITIES OF FAILURE MODE: EXCEED LIMITING DEPTH FOR SMALL LEAK

               NUMBER FAILED =     0             NUMBER OF TRIALS =  5000

  END OF      FAILURE PROBABILITY WITHOUT  AND  WITH IN-SERVICE INSPECTION
   CYCLE        FOR PERIOD       CUM. TOTAL       FOR PERIOD        CUM. TOTAL

      .0      1.66284D-10      1.66284D-10      1.66284D-10      1.66284D-10
    40.0      0.00000D+00      1.66284D-10      0.00000D+00      1.66284D-10
```

Table C-12

RCS-15 Full Break Probability

STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)

WESTINGHOUSE PROBABILITY OF FAILURE PROGRAM SPFMPROF ESBU-NTD

INPUT VARIABLES FOR CASE 68: 316 STAINLESS STEEL PIPE WELD BREAK

NCYCLE =	40	NFAILS =	1000	NTRIAL =	5000
NOVARS =	29	NUMSET =	6	NUMISI =	5
NUMSSC =	7	NUMTRC =	7	NUMFMD =	4

VARIABLE NO.	NAME	DISTRIBUTION TYPE	LOG	MEDIAN VALUE	DEVIATION OR FACTOR	SHIFT MV/SD	USAGE NO.	SUB
1	PIPE-DIA	NORMAL	NO	1.5000D+00	7.5000D-03	.00	1	SET
2	WALL/DIA	NORMAL	NO	1.5000D-01	4.5000D-03	.00	2	SET
3	SRESIDUAL	NORMAL	YES	1.0318D+01	1.4125D+00	1.00	3	SET
4	INT%DEPTH	NORMAL	YES	5.0000D+00	1.4125D+00	1.00	4	SET
5	L/D-RATIO	NORMAL	YES	6.0000D+00	1.4125D+00	1.00	5	SET
6	PROB/VOL	- CONSTANT -		1.0000D-04			6	SET
7	FIRST-ISI	- CONSTANT -		5.[illegible]00D+00			1	ISI
8	FREQ-ISI	- CONSTANT -		1[illegible]0D+01			2	ISI
9	EPST-PND	- CONSTANT -		[illegible]00D-03			3	ISI
10	ASTAR-PND	- CONSTANT -		-3.2000D-01			4	ISI
11	ANUU-PND	- CONSTANT -		1.6000D+00			5	ISI
12	HOURS/CY	NORMAL	YES	7.4473D+03	1.0500D+00	1.00	1	SSC
13	PRESSURE	NORMAL	NO	2.7250D+00	1.5000D-02	.00	2	SSC
14	STRESS-SS	NORMAL	YES	5.6469D+00	1.2589D+00	.00	3	SSC
15	SCC-COEFF	NORMAL	YES	3.2310D-12	2.3714D+00	1.00	4	SSC
16	SCC-EXPNT	- CONSTANT -		2.1610D+00			5	SSC
17	SCC-TIMEI	- CONSTANT -		1.0000D+00			6	SSC
18	ECW-RATE	NORMAL	YES	1.2740D-11	2.3714D+00	.00	7	SSC
19	NOFTRS/HR	- CONSTANT -		6.0000D+01			1	TRC
20	STRESS-FT	NORMAL	YES	5.1336D-02	1.4125D+00	.00	2	TRC
21	NOSTRS/CY	- CONSTANT -		5.0000D+00			3	TRC
22	STRESS-ST	NORMAL	YES	8.7271D+00	1.2589D+00	.00	4	TRC
23	FCG-COEFF	NORMAL	YES	9.1401D-12	2.8508D+00	1.00	5	TRC
24	FCG-EXPNT	- CONSTANT -		4.0000D+00			6	TRC
25	FCG-THOLD	- CONSTANT -		4.6000D+00			7	TRC
26	LIMIT-DSL	- CONSTANT -		0.0000D+00			1	FMD
27	LIMIT-PBS	NORMAL	NO	5.1336D+01	3.2000D+00	-1.00	2	FMD
28	STRESS-DL	NORMAL	YES	1.1294D+01	1.4125D+00	1.00	3	FMD
29	FREQ-DLTR	- CONSTANT -		1.0000D-03			4	FMD

PROBABILITIES OF FAILURE MODE: EXCEED FLOW STRESS LIMIT FOR FULL BREAK

NUMBER FAILED = 1 NUMBER OF TRIALS = 5000

END OF CYCLE	FAILURE PROBABILITY WITHOUT FOR PERIOD	CUM. TOTAL	AND WITH IN-SERVICE INSPECTION FOR PERIOD	CUM. TOTAL
1.0	1.46947D-12	1.46947D-12	1.46947D-12	1.46947D-12
40.0	0.00000D+00	1.46947D-12	0.00000D+00	1.46947D-12

DEVIATION ON CUMULATIVE TOTALS = 1.46947D-12 1.46947D-12

Table C-13
SIL-9

Piping Structural Reliability Estimates for Millstone Unit No. 3

System: Low Pressure Safety Injection		Segment: SIL-9			Sheet of
P&ID No.: EM-112B		Data Point: 95			
Pipe Stress Calculation Number: 7001B		PSI/Const. Method: VT-2, UT, PT/Hydro, RT			
Piping Stress Isometric No.:		Proposed ISI Method: VT-2, UT			
Piping Component/Segment Element (weld, tee, elbow, etc.): Valve/pipe weld					
No.	**Input Parameter Description**	**Check Input Choice (for Table 1 Value)**			**Set Value***
1	Type of Piping Material	304 SS	316 SS	Carbon Steel	---
2	Temperature at Pipe Weld	Low (150)	Medium (350)	High (550)	350
3	Nominal Pipe Size	Small (2)	Medium (5)	Large (16)	10
4	Pipe Wall Thickness	Thin (.06)	Normal (.14)	Thick (.22)	.1
5	Normal Operating Pressure	Low (0.5)	Medium (1.3)	High (2.1)	.7
6	Residual Stress Level	None (0.0)	Moderate (0.1)	Maximum (0.2)	.2
7	Initial Flaw Size	Small (.05)	Medium (.11)	Large (.17)	.05
8	Steady-State Stress Level	Low (.05)	Medium (.11)	High (.17)	.11
9	Stress Corrosion Potential	None (0.0)	Moderate (0.5)	Maximum (1.0)	0
10	Material Wastage Potential	None (0.0)	Moderate (0.5)	Maximum (1.0)	0
11	High Cycle Fatigue Loads	None (0.0)	Moderate (.08)	Maximum (.16)	0
12	Fatigue Transient Loads	Low (.10)	Medium (.22)	High (.34)	.1
13	Fatigue Transient Frequency	Low (5)	Medium (13)	High (21)	5
14	Design-Limiting Stress (Break Only)	Low (.10)	Medium (.26)	High (.42)	.09
15	Optional Crack Inspection Interval	Low (6)	Medium (10)	High (14)	10
16	Optional crack Inspection Accuracy	High (.16)	Medium (.24)	Low (.32)	.16
*For optional numeric input, use a value (and associated units) from the standard range given in Table 1.					
Small Leak Probability, No ISI: 0 (2.5E-08)		Optional Leak Probability With ISI: 0 (2.5E-08)			
Full Break Probability, No ISI: 0 (9.2E-12)		Optional Break Probability With ISI: 0 (9.2E-12)			
Comments: Location based on potential check valve leakage causing thermal cycling.					

Table C-14

SIL-9 Small Leak Probability

```
                         STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE                PROBABILITY OF FAILURE PROGRAM SPFMPROF            ESBU-NTD

===========================================      ================================
     INPUT VARIABLES FOR CASE 18: 316 STAI  ESS STEEL PIPE WELD LEAK

     NCYCLE =     40              NFAILS =  1000              NTRIAL =  5000
     NOVARS =     29              NUMSET =     6              NUMISI =     5
     NUMSSC =      7              NUMTRC =     7              NUMFMD =     4
```

VARIABLE NO.	NAME	DISTRIBUTION TYPE	LOG	MEDIAN VALUE	DEVIATION OR FACTOR	SHIFT MV/SD	USAGE NO.	SUB
1	PIPE-DIA	NORMAL	NO	1.0000D+01	5.0000D-02	.00	1	SET
2	WALL/DIA	NORMAL	NO	1.0000D-01	3.0000D-03	.00	2	SET
3	SRESIDUAL	NORMAL	YES	1.2357D+01	1.4125D+00	1.00	3	SET
4	INT%DEPTH	NORMAL	YES	5.0000D+00	1.4125D+00	1.00	4	SET
5	L/D-RATIO	NORMAL	YES	6.0000D+00	1.4125D+00	1.00	5	SET
6	PROB/VOL	- CONSTANT -		1.0000D-04			6	SET
7	FIRST-ISI	- CONSTANT -		5.0000D+00			1	ISI
8	FREQ-ISI	- CONSTANT -		1.0000D+01			2	ISI
9	EPST-PND	- CONSTANT -		1.0000D-03			3	ISI
10	ASTAR-PND	- CONSTANT -		-3.2000D-01			4	ISI
11	ANUU-PND	- CONSTANT -		1.6000D+00			5	ISI
12	HOURS/CY	NORMAL	YES	7.4473D+03	1.0500D+00	1.00	1	SSC
13	PRESSURE	NORMAL	NO	7.0000D-01	1.5000D-02	.00	2	SSC
14	STRESS-SS	NORMAL	YES	6.1783D+00	1.2589D+00	.00	3	SSC
15	SCC-COEFF	NORMAL	YES	3.2310D-12	2.3714D+00	1.00	4	SSC
16	SCC-EXPNT	- CONSTANT -		2.1610D+00			5	SSC
17	SCC-TIMEI	- CONSTANT -		1.0000D+00			6	SSC
18	ECW-RATE	NORMAL	YES	1.2740D-11	2.3714D+00	.00	7	SSC
19	NOFTRS/HR	- CONSTANT -		6.0000D+01			1	TRC
20	STRESS-FT	NORMAL	YES	6.1783D-02	1.4125D+00	.00	2	TRC
21	NOSTRS/CY	- CONSTANT -		5.0000D+00			3	TRC
22	STRESS-ST	NORMAL	YES	6.1783D+00	1.2589D+00	.00	4	TRC
23	FCG-COEFF	NORMAL	YES	9.1401D-12	2.8508D+00	1.00	5	TRC
24	FCG-EXPNT	- CONSTANT -		4.0000D+00			6	TRC
25	FCG-THOLD	- CONSTANT -		4.6000D+00			7	TRC
26	LIMIT-DSL	NORMAL	NO	-9.7000D-01	1.0000D-02	.00	1	FMD
27	LIMIT-PBS	- CONSTANT -		0.0000D+00			2	FMD
28	STRESS-DL	- CONSTANT -		0.0000D+00			3	FMD
29	FREQ-DLTR	- CONSTANT -		0.0000D+00			4	FMD

PROBABILITIES OF FAILURE MODE: EXCEED LIMITING DEPTH FOR SMALL LEAK

NUMBER FAILED = 0   NUMBER OF TRIALS = 5000

END OF CYCLE	FAILURE PROBABILITY WITHOUT FOR PERIOD	CUM. TOTAL	AND WITH IN-SERVICE INSPECTION FOR PERIOD	CUM. TOTAL
.0	2.50257D-08	2.50257D-08	2.50257D-08	2.50257D-08
40.0	0.00000D+00	2.50257D-08	0.00000D+00	2.50257D-08

Table C-15

SIL-9 Full Break Probability

```
                    STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE             PROBABILITY OF FAILURE PROGRAM SPFMPROF            ESBU-NTD
=============================================================================
     INPUT VARIABLES FOR CASE 17: 316 STAINLESS STEEL PIPE WELD BREAK

     NCYCLE =     40            NFAILS =  1000           NTRIAL =  5000
     NOVARS =     29            NUMSET =     6           NUMISI =     5
     NUMSSC =      7            NUMTRC =     7           NUMFMD =     4
```

VARIABLE NO.	NAME	DISTRIBUTION TYPE	LOG	MEDIAN VALUE	DEVIATION OR FACTOR	SHIFT MV/SD	USAGE NO.	SUB
1	PIPE-DIA	NORMAL	NO	1.0000D+01	5.0000D-02	.00	1	SET
2	WALL/DIA	NORMAL	NO	1.0000D-01	3.0000D-03	.00	2	SET
3	SRESIDUAL	NORMAL	YES	1.2357D+01	1.4125D+00	1.00	3	SET
4	INT%DEPTH	NORMAL	YES	5.[illegible]000D+00	1.4125D+00	1.00	4	SET
5	L/D-RATIO	NORMAL	YES	6.0000D+00	1.4125D+00	1.00	5	SET
6	PROB/VOL	- CONSTANT -		1.0000D-04			6	SET
7	FIRST-ISI	- CONSTANT -		5.0000D+00			1	ISI
8	FREQ-ISI	- CONSTANT -		1.0000D+01			2	ISI
9	EPST-PND	- CONSTANT -		1.0000D-03			3	ISI
10	ASTAR-PND	- CONSTANT -		-3.2000D-01			4	ISI
11	ANUU-PND	- CONSTANT -		1.6000D+00			5	ISI
12	HOURS/CY	NORMAL	YES	7.4473D+03	1.0500D+00	1.00	1	SSC
13	PRESSURE	NORMAL	NO	7.0000D-01	1.5000D-02	.00	2	SSC
14	STRESS-SS	NORMAL	YES	6.1783D+00	1.2589D+00	.00	3	SSC
15	SCC-COEFF	NORMAL	YES	3.2310D-12	2.3714D+00	1.00	4	SSC
16	SCC-EXPNT	- CONSTANT -		2.1610D+00			5	SSC
17	SCC-TIMEI	- CONSTANT -		1.0000D+00			6	SSC
18	ECW-RATE	NORMAL	YES	1.2740D-11	2.3714D+00	.00	7	SSC
19	NOFTRS/HR	- CONSTANT -		6.0000D+01			1	TRC
20	STRESS-FT	NORMAL	YES	6.1783D-02	1.4125D+00	.00	2	TRC
21	NOSTRS/CY	- CONSTANT -		5.0000D+00			3	TRC
22	STRESS-ST	NORMAL	YES	6.1783D+00	1.2589D+00	.00	4	TRC
23	FCG-COEFF	NORMAL	YES	9.1401D-12	2.8508D+00	1.00	5	TRC
24	FCG-EXPNT	- CONSTANT -		4.0000D+00			6	TRC
25	FCG-THOLD	- CONSTANT -		4.6000D+00			7	TRC
26	LIMIT-DSL	- CONSTANT -		0.0000D+00			1	FMD
27	LIMIT-PBS	NORMAL	NO	6.1783D+01	3.2000D+00	-1.00	2	FMD
28	STRESS-DL	NORMAL	YES	5.5605D+00	1.4125D+00	1.00	3	FMD
29	FREQ-DLTR	- CONSTANT -		1.0000D-03			4	FMD

PROBABILITIES OF FAILURE MODE: EXCEED FLOW STRESS LIMIT FOR FULL BREAK

NUMBER FAILED = 0 NUMBER OF TRIALS = 5000

END OF CYCLE	FAILURE PROBABILITY WITHOUT FOR PERIOD	CUM. TOTAL	AND WITH IN-SERVICE INSPECTION FOR PERIOD	CUM. TOTAL
.0	9.20644D-12	9.20644D-12	9.20644D-12	9.20644D-12
40.0	0.00000D+00	9.20644D-12	0.00000D+00	9.20644D-12

APPENDIX D

SRRA CODE DESCRIPTION

For the Millstone Unit 3 application, structural reliability and risk assessment (SRRA) tools were used to estimate the failure probabilities of the various piping segments. The SRRA tools were developed by Westinghouse for Idaho National Engineering Laboratory (INEL) to address the aging of passive components for NRC and DoE (Bishop and Phillips 1993). The SRRA tools are a set of executable personal computer programs to specify input, calculate and plot failure probability of piping with time for the selected input values of key design, operational, and inspection parameters. The computer tool for structural reliability uses Monte-Carlo simulation with importance sampling to calculate the probability of leak or break of type 304 or 316 stainless steel piping due to fatigue crack growth and stress corrosion cracking and of carbon steel piping due to fatigue crack growth and erosion-corrosion-wear.

In addition to the type of piping material and failure modes, fourteen variable parameters are used for the simplified input for Millstone Unit 3 piping failure probabilities. These parameters are listed in the input form of Table D-1 and their range of standard values are provided in Tables D-2 and D-3. The selection of these key parameters is based on studies of size effects of Thomas (1981), inservice inspection effects of Woo and Simonen (1984) and transient severity and frequency effects of Chapman, Milner and Baker (1991). The effects observed in the piping failure database (e.g., Bush 1992) and correlated by Gamble and Taggart (1991) are also considered. Use of the ratio of the applied stress values to flow stress is from the work of LaPay, et al (1985). Since the flow stress is normally related to design stress limits (yield and ultimate strength), experience with piping design stress analyses can also be used to guide the setting of the input values of stress ratio.

To prepare the input sheet (shown in Table D-1) for each system piping segment for the location with limiting failure probability, the following general instructions were used:

- If the location is not certain, fill out a range of values or use separate sheets for different locations and note it as such in the "comments."

- Normally, the higher the value of the non-ISI input parameters, the higher the failure probability, all other things being equal. Since higher temperature gives a lower flow

stress, stresses would be lower for the same input fraction and the leak (but not break) probability could be lower.

- Calculated stresses from design analyses (per ASME Code) are assumed to be upper bound values with the means (expected values) one-half of those values.

- The stress of input item 14 is the maximum primary stress during the design limiting event, such as earthquake seismic loading or transient pressure loads.

- For the optional inservice inspection input, the failure probability is higher for high intervals and low accuracy.

- The number in parenthesis is the value of the corresponding parameter in Table D-2 that is used for qualitative input (high, medium or low). If more precise input is known, it can be specified in the last column on the input sheet. Input to the computer is simplified if the standard range of values specified in Table D-2 are used.

To assist in selecting the limiting location in each piping system segment, the following general guidance is also provided:

- The purpose of the piping inspection is to detect a small flaw before it becomes large enough to be a potential problem during a postulated design-limiting event, such as a safe-shutdown earthquake or loss of coolant accident.

- Locations to be considered are not only those where a small flaw might occur (mechanistic), but also where you would want to know about it (break potential) if one did occur for an unexpected reason.

- Also consider the effects of adjacent components not working properly on both the mechanistic approach (e.g., snubber lockup or a leaking valve) and the break potential (e.g., snubber not engaging).

- Since the probability of having a flaw and its initial size, which is a fraction of the wall thickness, is greater for larger and thicker pipes, failure probabilities should be higher, all other factors being equal.

More detailed guidance on selecting the limiting location and the input for calculating its failure probability is provided in Table D-4. The expertise and needed information to accomplish this is summarized in Table D-5. Tables D-6 and D-7 show example input and output screens with the default values for the SPFMMENU Program, which was the simplified piping input preprocessor used for the Millstone Unit 3 Risk Based Inspection Program.

To calculate failure probability for the simplified piping model input of Tables D-1 and D-2, the specified values are transformed for use in a more detailed structural reliability model. The input variables for this model and the correspondence to those for the simplified input are shown in Table D-8. The "order" refers to the simplified input variables of Tables D-1 and D-2. If the simplified input ranges and steps of Table D-2 are not adequate, then the "FULL MENU" option can be used to change the input values for any of the variables in Table D-8. The type of statistical distribution of each random variable in the detailed model input is also indicated in Table D-8.

The piping failure modes considered in the structural reliability models are either exceeding the limiting crack depth for a leak during normal operation or exceeding the flow stress in the remaining uncracked section during some design-limiting transient. These failure modes encompass the highest (small leak) to lowest (full break) probabilities typically calculated for piping. The primary cause of failure is intergranular stress-corrosion cracking (IGSCC) growth or fatigue-crack growth of a small undetected crack due to thermal transients or high-cycle loads such as flow-induced vibration. The piping structural reliability model also includes the effects of pipe wall thinning due to wastage, such as erosion, corrosion or wear. The structural reliability models also include log-normally distributed depth and length of the initial crack and growth of the crack in both directions calculated using fracture mechanics analysis methods. The stress intensity factors for a semi-elliptical crack on the inside surface in a uniform stress field, the methods for calculating the effects of inservice inspection (ISI) and the combination of high and low cycle fatigue crack growth are similar to those in the PRAISE code (Harris et al 1986; Harris and Dedhia 1992), which for a number of years had been the nuclear industry's standard for calculating the probability of piping failure as a function of operating time.

The median values of the material related variables of Table D-8 are given in Table D-9. When the default values are the same for all three materials, the value is indicated for carbon steel only in Table D-9. The median is the value at 50% probability (half above and half below this value); it is also the mean (average) value for symmetric distributions, like the normal (bell-shaped curve) distribution.

All calculated piping probabilities are multiplied by the probability of the initial crack being present. This probability is variable number 6 and is estimated as V / 10,000, where V is the volume of weld metal, which is consistent with previous piping reliability analyses (Harris, et al 1986). The IGSCC equation for crack growth rate at a given stress intensity level is modelled as a log-log function with values for variables 15 to 17 and the uncertainty bounds for 304 SS given by an NRC study (Hazelton and Koo 1988). The fraction of the growth rate factor for 316 SS is from PRAISE (Harris, et al 1986) and 1% of the 304 SS rate is used for carbon steel. The rate of erosion-corrosion (variable 18) is assumed to be constant with a maximum value and uncertainty assigned to reflect those rates observed and calculated in the industry (Mattu et al 1988) for carbon steel (CS) piping (Figure D-1 provides a comparison of the probability of violating the ASME minimum wall thickness criteria with the cumulative industry ratio of number replaced to the number inspected). The maximum values for SS are assumed to be only one percent of those for CS. Variables 23 to 25 for the log-log fatigue crack growth equation are taken from the PRAISE code (Harris and Dehia 1992) for stainless steel and from Bamford's (1980) and Phillip's (1992) work for carbon steel. The fast transient frequency of variable 19 is assumed to be one per minute, which is high enough to cause significant growth if the fatigue-crack growth threshold value is exceeded.

The failure modes in the piping structural reliability model include either exceeding the limiting crack depth for a leak during normal operation or exceeding the flow stress in the remaining uncracked section during some design-limiting transient, such as a safe-shutdown earthquake (SSE). The first or second mode of failure is set depending upon whether variable 26 or 27 in Table D-9 is set to zero. The value of flow stress of variable 27 is required to predict full break failures. It is also used for the simplified input of Table D-1, where the applied stresses are specified as a ratio to the flow stress (even for small leak analysis). For stainless steel, a statistical evaluation of the data for various types of welds (EPRI 1986) is used for the change in mean flow stress with temperature. From the summary of licensee's responses to NRC Bulletin 87-01 (Mattu et al 1988), the carbon steel material specifications indicated that the flow stress is higher than that for stainless steel at the same

temperature (Phillips 1992). All other uncertainties are based upon expert engineering judgement and previous structural reliability modeling experience at Westinghouse.

The probability of failure of the piping as a function of operating time is calculated directly for each set of input values using Monte-Carlo simulation with importance sampling. This variance reduction technique, as described by Witt (1984), is used to greatly reduce the number of trials required for calculating small failure probabilities. In importance sampling, the important random values are selected from the more severe high or low regions of their distributions so as to promote failure. However, when failure is calculated, the count is corrected to account for the lower probability of simultaneously obtaining all the more severe random values.

To apply this simulation method to the simplified piping fracture mechanics (SPFM) structural reliability models, the existing Westinghouse PROF (probability of failure) Software System (object library) is used. The PROF library provides standard input and output, including plotting, and probabilistic analysis capabilities (e.g., random number generation, importance sampling). The result is the executable program SPFMPROF.EXE for calculation of piping failure probability with time. The Westinghouse PROF Software Library, which was used to generate the SPFMPROF program, has been verified and benchmarked in a number of ways as described in Appendix E.

The flow chart for the SPFMPROF Program, which is used to calculate the failure probabilities for the Millstone Unit 3 Risk-Based Inspection Program, is shown in Figure D-2. Variables 1 to 6 in Tables D-8 and D-9 are used to "initialize parameters" that do not vary with time in Subroutine SET. Variables 7 to 11 are needed to calculate the "effects of ISI" in Subroutine ISI, 12 to 18 for "steady-state changes" in Subroutine SSC and 13 to 25 for "transient changes" in Subroutine TRC. Finally, variables 26 to 29 are used to "check if failure occurs" in Subroutine FMD.

Table D-10 provides sample output from the SPFMPROF Program for the values of the input variables in Tables D-2 and D-9. The first page of the output describes the input that is used for the calculations. The "SHIFT MV/SD" column indicates how many standard deviations (SD) the median value (MV) is shifted for importance sampling (Witt 1984). The second page of the output provides the change in failure probability per fuel (operating) cycle and the cumulative probability. The deviation on the cumulative total that is output is the deviation only due to the Monte-Carlo simulation. Figure D-3 shows the plot generated by the SRRAPLOT post-processor program. It

compares the calculated piping failure probabilities with and without the effects of inservice inspection for leak of carbon steel piping.

Table D-1
INPUT FORM FOR PIPING STRUCTURAL RELIABILITY ESTIMATES FOR MILLSTONE UNIT 3

System : ______________________________ **PSI/Constr. Method** ________________
P&ID No.: ____________________________ **Segment:** _________________________
Pipe Stress Calculation Number: ________ **Proposed ISI Method** ________________
Piping Stress Isometric No.: ___________ **Data Point:** ______________
Piping Component/Segment Element (weld, tee, elbow, etc.): ___________________

No.	Input Parameter Description	Check Input Choice (for Table 1 Value)			Set Value *
1	Type of Piping Material	304 SS	316 SS	Carbon Steel	---
2	Temperature at Pipe Weld	Low (150)	Medium (350)	High (550)	
3	Nominal Pipe Size	Small (2)	Medium (5)	Large (16)	
4	Pipe Wall Thickness	Thin (.06)	Normal (.14)	Thick (.22)	
5	Normal Operating Pressure	Low (0.5)	Medium (1.3)	High (2.1)	
6	Residual Stress Level	None (0.0)	Moderate (0.1)	Maximum (0.2)	
7	Initial Flaw Size	Small (.05)	Medium (.11)	Large (.17)	
8	Steady-State Stress Level	Low (.05)	Medium (.11)	High (.17)	
9	Stress Corrosion Potential	None (0.0)	Moderate (0.5)	Maximum (1.0)	
10	Material Wastage Potential	None (0.0)	Moderate (0.5)	Maximum (1.0)	
11	High Cycle Fatigue Loads	None (0.0)	Moderate (.08)	Maximum (.16)	
12	Fatigue Transient Loads	Low (.10)	Medium (.22)	High (.34)	
13	Fatigue Transient Frequency	Low (5)	Medium (13)	High (21)	
14	Design-Limiting Stress (Break Only)	Low (.10)	Medium (.26)	High (.42)	
15	Optional Crack Inspection Interval	Low (6)	Medium (10)	High (14)	
16	Optional Crack Inspection Accuracy	High (.16)	Medium (.24)	Low (.32)	

* For optional numeric input, use a value (and associated units) from the standard range given in Table 1.

Small Leak Probability, No ISI: _______________ **Optional Leak Probability With ISI:** ___________
Full Break Probability, No ISI: _______________ **Optional Break Probability With ISI:** __________

Comments: ____

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Table D-2 RANGE OF STANDARD MEDIAN VALUES FOR SIMPLIFIED PIPING STRUCTURAL RELIABILITY INPUT			
No.	**Parameter Description**	**Range**	**Step Size**
2	Degrees (F) at Pipe Weld	50 - 650	50
3	Pipe Outside Diameter (inch)	1 - 20	0.5, $D \leq 5$ 2.0, $D \geq 6$
4	Thickness to Diameter Ratio	0.02 - 0.26	0.02
5	Operating Pressure (ksi)	0.1 - 2.5	0.20
6	Ratio* of Residual to Flow Stress	0.0 - 0.2	0.05
7	Flaw Depth Fraction of Wall	0.02 - 0.20	0.03
8	Ratio* Steady-State to Flow Stress	0.02 - 0.20	0.03
9	Stress Corrosion Cracking Level	0.0 - 1.0	0.10
10	Erosion-Corrosion-Wear Level	0.0 - 1.0	0.10
11	Ratio* Stress Amplitude to Flow Stress for High Cycle Fatigue Loading	0.0 - 0.16	0.04
12	Ratio* Stress Amplitude to Flow Stress for Fatigue Transient Loading	0.04 - 0.40	0.03
13	Fatigue Transients per Operating Cycle	1 - 25	4
14	Ratio* Maximum Stress to Flow Stress for the Design-Limiting Condition	0.02 - 0.50	0.04
15	Operating Cycles Between Inspections	4 - 16	2
16	Wall Fraction for 50% Crack Detection	0.12 - 0.36	0.04

* The stress ratio is the value of the applied stress to the weld flow stress for the specified temperature and the type of material per Table D-3.

Table D-3
VALUE OF WELD FLOW STRESS (KSI) USED FOR SIMPLIFIED PIPING STRUCTURAL RELIABILITY MODELS

Temperature (°F)	304 & 316 SS	Carbon Steel
50	74.32	80.92
100	72.23	78.83
150	70.14	76.74
200	68.05	74.65
250	65.96	72.56
300	63.87	70.47
350	61.78	68.38
400	59.69	66.29
450	57.60	64.20
500	55.51	62.11
550	53.42	60.02
600	51.33	57.93
650	49.24	55.84

Table D-4

GUIDELINES FOR SELECTING LIMITING LOCATIONS AND ESTIMATING FAILURE PROBABILITIES

- MECHANISTIC APPROACH

  - For *poor fabrication and preservice inspection quality (initial flaws)*, look for field vs. shop welds and configurations that would be hard to maintain fabrication tolerances or to inspect. Lack of stress relief or cold springing could also lead to high residual stresses.

  - For *stress-corrosion cracking*, in BWR piping only, high stresses (residual, steady-state, pressure), sensitized material (304 SS) and high coolant conductivity are all required.

  - For *material wastage*, look for locations of relative support motion (wear), high pressure drop or turning losses (erosion-corrosion) or areas of stagnant coolant (microbiological attack) if the piping materials, especially at crevices, are susceptible to any of these wastage mechanisms.

  - For *high cycle fatigue*, look for configurations susceptible to flow induced vibration and flow striping or for vibratory resonance with rotating equipment (pump) frequencies.

  - For *low cycle fatigue*, look for areas with high loads due to thermal expansion (equipment nozzles and other anchor points, near snubbers, dissimilar metal joints) for heat-up and cool-down thermal cycling.

- BREAK POTENTIAL APPROACH

  - Identify source of potentially limiting loads (e.g., seismic, water hammer) and then the location of maximum loading if the source was to occur.

  - If some new unexpected loading were to occur, what is the weakest point in the segment that would be inspected/checked for failure?

  - Look at locations identified in the mechanistic approach to see if potential source loadings would still be high enough to be of concern.

Table D-5

GUIDELINES ON EXPERTISE AND INFORMATION REQUIRED TO SELECT LIMITING LOCATIONS AND ESTIMATE FAILURE PROBABILITIES

- THERMAL-FLUIDS SYSTEM ANALYSIS
  - Potential sources and locations of thermal striping or stratification
  - Areas of high flow velocity or turning losses for vibration/wastage
  - Stagnant flow zones and coolant chemistry for wastage/corrosion
  - Location of high transient pressures or loads (e.g., water hammer)
  - Steady-state and transient temperatures and gradients

- DESIGN STRESS ANALYSIS
  - Location of discontinuities like snubbers, anchors, support lugs and dissimilar metal joints for high operating or cyclic stresses
  - Location of any field welds or cold springing (residual stress)
  - Areas of high thermal stress (low cycle fatigue)
  - Locations with high transient loads (seismic)
  - Sensitized material locations for potential stress corrosion

- INSERVICE INSPECTION
  - Locations with poor preservice inspection (undetected flaws)
  - Inspection locations now required by ASME Code, Section XI
  - History of any indications in this or similar configurations
  - Results of applicable Section XI flaw evaluations
  - Accuracy of potential inservice inspection

- OPERATIONS AND MAINTENANCE
  - Any problems observed during fabrication, installation or hot functional testing of system
  - History of any leaks or repairs in this or similar configurations
  - Any observed failures or areas of high vibration
  - History of snubber retesting or other support problems
  - Any other maintenance problems of concern (valves, bellows, etc.)

Table D-6

SAMPLE INPUT SCREEN FOR THE SPFMMENU PROGRAM

Westinghouse	Program SPFMMENU	ESBU-NTD

Type of SPFMMENU Program Option	Set Input
Type of Piping Steel Material	Carbon
Pipe Weld Failure Mode	Small Leak
Crack Inspection Interval	Medium
Crack Inspection Accuracy	High
Temperature at Pipe Weld	Medium
Nominal Pipe Size	Medium
Pipe Wall Thickness	Normal
Normal Operating Pressure	Medium
Residual Stress Level	Moderate
Initial Flaw Size	Medium
Steady-State Stress Level	Medium
Stress Corrosion Potential	Moderate
Material Wastage Potential	Moderate
High Cycle Fatigue Loads	Moderate
Fatigue Transient Loads	Medium
Fatigue Transient Frequency	Low
Design-Limit Stress Level	Medium

Messages and Input	Use Up, Down, Right or Left Arrows, End, Esc, Enter or Insert Keys to Select Options\Values

Table D-7

SAMPLE OUTPUT SCREEN FOR THE SPFMMENU PROGRAM

Westinghouse	Program SPFMMENU	ESBU-NTD
	Type of SPFMMENU Program Option	Run PROF
	Type of Piping Steel Material	Carbon
	Pipe Weld Failure Mode	Small Leak
	Operating Cycles Between Inspections	10.0
	Wall Fraction for 50% Detection	0.160
	Degrees (F) at Pipe Weld	350.0
	Pipe Outside Diameter (inch)	5.0
	Thickness to Diameter Ratio	0.140
	Operating Pressure (ksi)	1.30
	Ratio of Residual to Flow Stress	0.10
	Flaw Depth Fraction of Wall	0.110
	Ratio Steady-State to Flow Stress	0.11
	Stress Corrosion Cracking Level	0.50
	Erosion-Corrosion-Wear Level	0.50
	Ratio HCFL Amplitude to Flow Stress	0.080
	Ratio FTL Amplitude to Flow Stress	0.220
	Transients per Operating Cycle	5.0
	Ratio Design-Limit to Flow Stress	0.260
	Value of Weld Metal Flow Stress in Ksi	68.35

Messages and Input	Actual input values for current run are shown above. Do you wish to quit after run (Y/N):

Table D-8

VARIABLES FOR PIPING STRUCTURAL RELIABILITY MODEL

ORDER*	NO.	NAME	DESCRIPTION OF MODEL VARIABLE	TYPE OF DISTRIBUTION
3rd	1	PIPE-DIA	PIPE OUTSIDE DIAMETER (INCH)	Normal
4th	2	WALL/DIA	PIPE WALL TO DIAMETER RATIO	Normal
6th	3	SRESIDUAL	WELD I.D. RESIDUAL STRESS (KSI)	Log-Normal
7th	4	INT%DEPTH	INITIAL CRACK DEPTH (% OF WALL)	Log-Normal
	5	L/D-RATIO	INITIAL CRACK LENGTH TO DEPTH RATIO	Log-Normal
	6	PROB/VOL	PROBABILITY OF CRACK PER CUBIC INCH	Constant
15th	7	FIRST-ISI	CYCLE NUMBER FOR FIRST INSPECTION (ISI)	Constant
15th	8	FREQ-ISI	FREQUENCY FOR SUBSEQUENT ISI'S (CYCLES)	Constant
	9	EPST-PND	MINIMUM ISI PROB. OF NONDETECTION (PND)	Constant
16th	10	ASTAR-PND	DEPTH FOR 50% PROB. OF NONDETECTION	Constant
	11	ANUU-PND	PND EXPONENTIAL SLOPE WITH CRACK DEPTH	Constant
	12	HOURS/CY	EFFECTIVE HOURS PER OPERATING CYCLE	Log-Normal
5th	13	PRESSURE	NORMAL OPERATING PRESSURE (KSI)	Normal
8th	14	STRESS-SS	STEADY-STATE OPERATING STRESS (KSI)	Log-Normal
9th	15	SCC-COEFF	STRESS-CORROSION COEFFICIENT (IN/HR)	Log-Normal
	16	SCC-EXPNT	SCC EXPONENT FOR STRESS INTENSITY	Constant
	17	SCC-TIMEI	TIME TO INITIATE STRESS-CORROSION (HRS)	Constant
10th	18	ECW-RATE	EROSION-CORROSION WEAR RATE (IN/HR)	Log-Normal
	19	NOFTRS/HR	NUMBER OF FAST TRANSIENTS PER HOUR	Constant
11th	20	STRESS-FT	FAST TRANSIENT STRESS AMPLITUDE (KSI)	Log-Normal
12th	21	NOSTRS/CY	NUMBER OF SLOW TRANSIENTS PER CYCLE	Constant
13th	22	STRESS-ST	SLOW TRANSIENT STRESS AMPLITUDE (KSI)	Log-Normal
1st	23	FCG-COEFF	FATIGUE CRACKING COEFFICIENT (IN/CYCLE)	Log-Normal
	24	FCG-EXPNT	FATIGUE CRACK GROWTH EXPONENT	Constant
	25	FCG-THOLD	FCG THRESHOLD IN KSI-SQRT(INCH)	Constant
	26	LIMIT-DSL	LIMIT CRACK DEPTH FOR SMALL LEAK (INCH)	Normal
2nd	27	LIMIT-PBS	LIMITING STRESS FOR PIPE BREAK (KSI)	Normal
14th	28	STRESS-DL	DESIGN LIMITING AXIAL STRESS (KSI)	Log-Normal
	29	FREQ-DLTR	FREQUENCY OF DL TRANSIENT PER YEAR	Constant

*Order of corresponding variable for the simplified input of Table D-7.

Table D-9

DEFAULT MEDIAN VALUES* FOR PIPING STRUCTURAL RELIABILITY MODEL

NO.	VARIABLE	CARBON STEEL	304 SS	316 SS
1	PIPE-DIA	5.000E+00		
2	WALL/DIA	1.400E-01		
3	SRESIDUAL	6.835E+00	6.178E+00	6.178E+00
4	INT%DEPTH	1.100E+01		
5	L/D-RATIO	6.000E+00		
6	PROB/VOL	1.000E-04		
7	FIRST-ISI	5.000E+00		
8	FREQ-ISI	1.000E+01		
9	EPST-PND	5.000E-03	1.000E-03	1.000E-03
10	ASTAR-PND	-2.400E-01	-4.800E-01	-4.800E-01
11	ANUU-PND	3.000E+00	1.600E+00	1.600E+00
12	HOURS/CY	7.447E+03		
13	PRESSURE	1.300E+00		
14	STRESS-SS	7.518E+00	6.796E+00	6.796E+00
15	SCC-COEFF	1.795E-10	1.795E-08	1.616E-09
16	SCC-EXPNT	2.161E+00		
17	SCC-TIMEI	1.000E+00		
18	ECW-RATE	6.370E-07	6.370E-09	6.370E-09
19	NOFTRS/HR	6.000E+01		
20	STRESS-FT	5.468E+00	4.943E+00	4.943E+00
21	NOSTRS/CY	5.000E+00		
22	STRESS-ST	1.504E+01	1.359E+01	1.359E+01
23	FCG-COEFF	1.202E-11	9.140E-12	9.140E-12
24	FCG-EXPNT	3.700E+00	4.000E+00	4.000E+00
25	FCG-THOLD	3.500E+00	4.600E+00	4.600E+00
26	LIMIT-DSL	-9.700E-01		
27	LIMIT-PBS	6.835E+01	6.178E+01	6.178E+01
28	STRESS-DL	1.777E+01	1.606E+01	1.606E+01
29	FREQ-DLTR	1.000E-03		

* Note: A negative number in this table indicates that the absolute value is the specified fraction of the pipe wall thickness.

TABLE D-10

SAMPLE OUTPUT FROM THE SPFMPROF PROGRAM

```
STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE          PROBABILITY OF FAILURE PROGRAM SPFMPROF          ESBU-NTD
================================================================================
      INPUT VARIABLES FOR CASE  1: CARBON STEEL PIPE WELD SMALL LEAK

      NCYCLE =     40            NFAILS =  1000           NTRIAL =  5000
      NOVARS =     29            NUMSET =     6           NUMISI =     5
      NUMSSC =      7            NUMTRC =     7           NUMFMD =     4

    VARIABLE       DISTRIBUTION      MEDIAN       DEVIATION    SHIFT    USAGE
   NO.   NAME      TYPE      LOG      VALUE       OR FACTOR    MV/SD   NO. SUB

    1  PIPE-DIA    NORMAL    NO    5.0000D+00   2.5000D-02      .00     1  SET
    2  WALL/DIA    NORMAL    NO    1.4000D-01   4.2000D-03      .00     2  SET
    3  SRESIDUAL   NORMAL    YES   6.8349D+00   1.4125D+00      .00     3  SET
    4  INT%DEPTH   NORMAL    YES   1.1000D+01   1.4125D+00     1.00     4  SET
    5  L/D-RATIO   NORMAL    YES   6.0000D+00   1.4125D+00     1.00     5  SET
    6  PROB/VOL    - CONSTANT -    1.0000D-04                           6  SET
    7  FIRST-ISI   - CONSTANT -    5.0000D+00                           1  ISI
    8  FREQ-ISI    - CONSTANT -    1.0000D+01                           2  ISI
    9  EPST-PND    - CONSTANT -    5.0000D-03                           3  ISI
   10  ASTAR-PND   - CONSTANT -   -1.6000D-01                           4  ISI
   11  ANUU-PND    - CONSTANT -    3.0000D+00                           5  ISI
   12  HOURS/CY    NORMAL    YES   7.4473D+03   1.0500D+00     1.00     1  SSC
   13  PRESSURE    NORMAL    NO    1.3000D+00   1.5000D-02      .00     2  SSC
   14  STRESS-SS   NORMAL    YES   7.5184D+00   1.2589D+00      .00     3  SSC
   15  SCC-COEFF   NORMAL    YES   1.7950D-10   2.3714D+00      .00     4  SSC
   16  SCC-EXPNT   - CONSTANT -    2.1610D+00                           5  SSC
   17  SCC-TIMEI   - CONSTANT -    1.0000D+00                           6  SSC
   18  ECW-RATE    NORMAL    YES   6.3700D-07   2.3714D+00      .00     7  SSC
   19  NOFTRS/HR   - CONSTANT -    6.0000D+01                           1  TRC
   20  STRESS-FT   NORMAL    YES   5.4679D+00   1.4125D+00      .00     2  TRC
   21  NOSTRS/CY   - CONSTANT -    5.0000D+00                           3  TRC
   22  STRESS-ST   NORMAL    YES   1.5037D+01   1.2589D+00     1.00     4  TRC
   23  FCG-COEFF   NORMAL    YES   1.2017D-11   2.8508D+00     1.00     5  TRC
   24  FCG-EXPNT   - CONSTANT -    3.7000D+00                           6  TRC
   25  FCG-THOLD   - CONSTANT -    3.5000D+00                           7  TRC
   26  LIMIT-DSL   NORMAL    NO   -9.7000D-01   1.0000D-02      .00     1  FMD
   27  LIMIT-PBS   - CONSTANT -    0.0000D+00                           2  FMD
   28  STRESS-DL   - CONSTANT -    0.0000D+00                           3  FMD
   29  FREQ-DLTR   - CONSTANT -    0.0000D+00                           4  FMD
```

TABLE D-10

SAMPLE OUTPUT FROM THE SPFMPROF PROGRAM
(Continued)

PROBABILITIES OF FAILURE MODE: EXCEED LIMITING DEPTH FOR SMALL LEAK

NUMBER FAILED = 1000 NUMBER OF TRIALS = 1850

END OF CYCLE	FAILURE PROBABILITY WITHOUT FOR PERIOD	CUM. TOTAL	AND WITH IN-SERVICE INSPECTION FOR PERIOD	CUM. TOTAL
1.0	1.36468D-10	1.36468D-10	1.36468D-10	1.36468D-10
2.0	1.71093D-07	1.71230D-07	1.71093D-07	1.71230D-07
3.0	2.34797D-07	4.06027D-07	2.34797D-07	4.06027D-07
4.0	3.28531D-07	7.34558D-07	3.28531D-07	7.34558D-07
5.0	1.47684D-06	2.21140D-06	1.47684D-06	2.21140D-06
6.0	4.37640D-07	2.64904D-06	2.18820D-09	2.21359D-06
7.0	3.76057D-06	6.40961D-06	1.88261D-08	2.23241D-06
8.0	1.60713D-06	8.01674D-06	8.08027D-09	2.24049D-06
9.0	1.27135D-05	2.07302D-05	6.38155D-08	2.30431D-06
10.0	6.91949D-07	2.14222D-05	3.64978D-09	2.30796D-06
11.0	2.67330D-06	2.40955D-05	2.83224D-08	2.33628D-06
12.0	2.41523D-06	2.65107D-05	1.38335D-08	2.35012D-06
13.0	2.43428D-06	2.89450D-05	4.40257D-08	2.39414D-06
14.0	3.00890D-05	5.90340D-05	4.49872D-07	2.84401D-06
15.0	6.13420D-06	6.51682D-05	2.88792D-07	3.13281D-06
16.0	1.20157D-05	7.71839D-05	7.30622D-09	3.14011D-06
17.0	2.36884D-06	7.95528D-05	1.72898D-09	3.14184D-06
18.0	2.89801D-06	8.24508D-05	8.73480D-10	3.14271D-06
19.0	2.83002D-06	8.52808D-05	3.06338D-09	3.14578D-06
20.0	9.09847D-06	9.43793D-05	5.08191D-09	3.15086D-06
21.0	1.25743D-05	1.06954D-04	1.03947D-08	3.16125D-06
22.0	1.23907D-06	1.08193D-04	2.54459D-09	3.16380D-06
23.0	7.60720D-06	1.15800D-04	4.33213D-09	3.16813D-06
24.0	1.80315D-05	1.33831D-04	2.95939D-08	3.19773D-06
25.0	5.36309D-06	1.39194D-04	4.21174D-09	3.20194D-06
26.0	1.19744D-05	1.51169D-04	1.56101D-10	3.20209D-06
27.0	2.01290D-06	1.53182D-04	3.44819D-11	3.20213D-06
28.0	2.90411D-06	1.56086D-04	6.68258D-12	3.20213D-06
29.0	5.02873D-06	1.61115D-04	2.20601D-10	3.20235D-06
30.0	4.64305D-06	1.65758D-04	7.03606D-11	3.20243D-06
31.0	3.06310D-05	1.96389D-04	6.15526D-10	3.20304D-06
32.0	5.97345D-07	1.96986D-04	1.99758D-12	3.20304D-06
33.0	1.61848D-05	2.13171D-04	5.52295D-10	3.20359D-06
34.0	2.48078D-06	2.15652D-04	5.45726D-11	3.20365D-06
35.0	1.27331D-05	2.28385D-04	4.20968D-10	3.20407D-06
36.0	4.30501D-05	2.71435D-04	1.55216D-11	3.20409D-06
37.0	7.10732D-06	2.78542D-04	3.04749D-12	3.20409D-06
38.0	4.21054D-06	2.82753D-04	1.64209D-12	3.20409D-06
39.0	4.81422D-06	2.87567D-04	5.95713D-12	3.20410D-06
40.0	7.28623D-06	2.94853D-04	1.93065D-12	3.20410D-06

DEVIATION ON CUMULATIVE TOTALS = 6.32189D-06 9.69380D-07

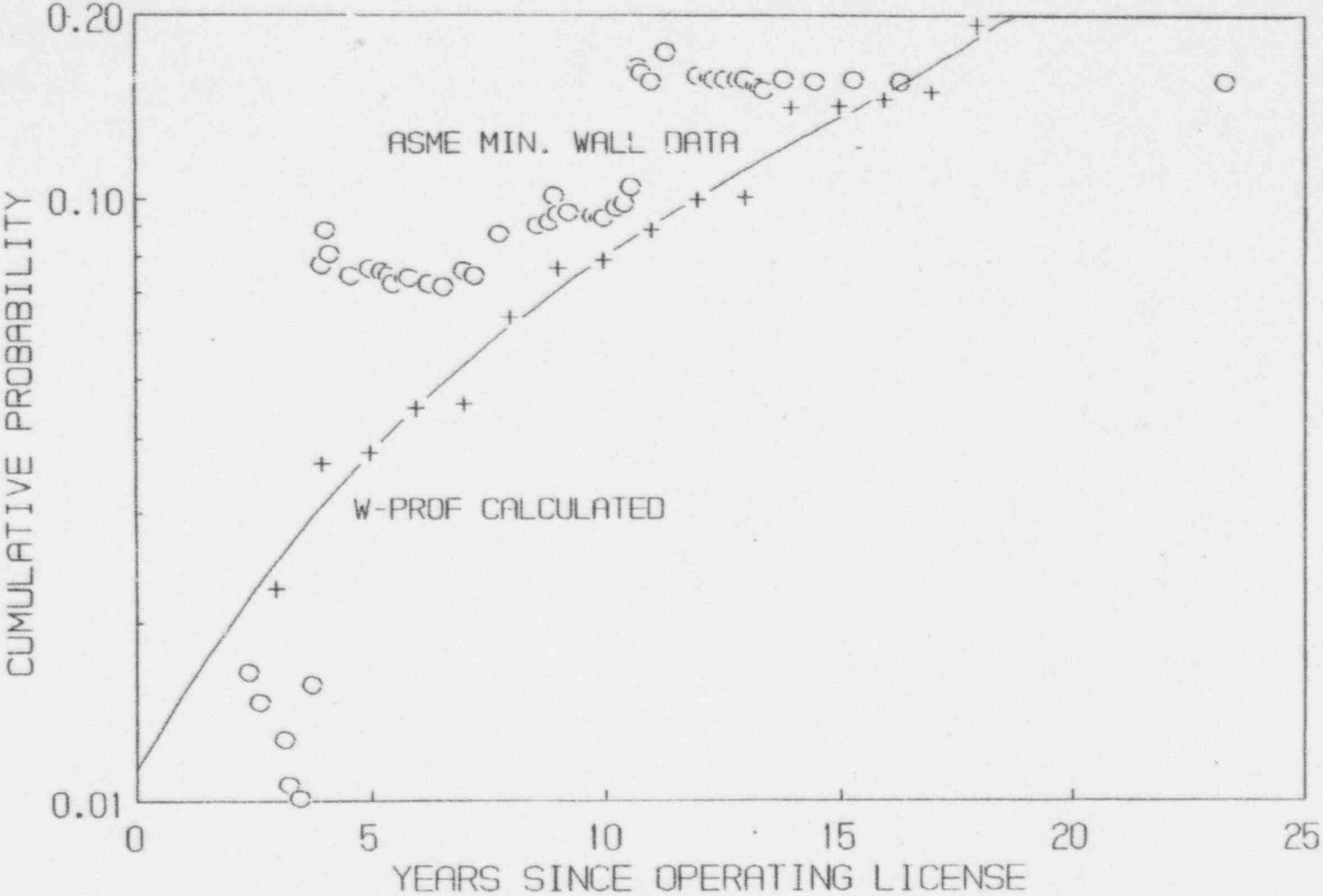


Figure D-1. PWR Wear Probability Comparison

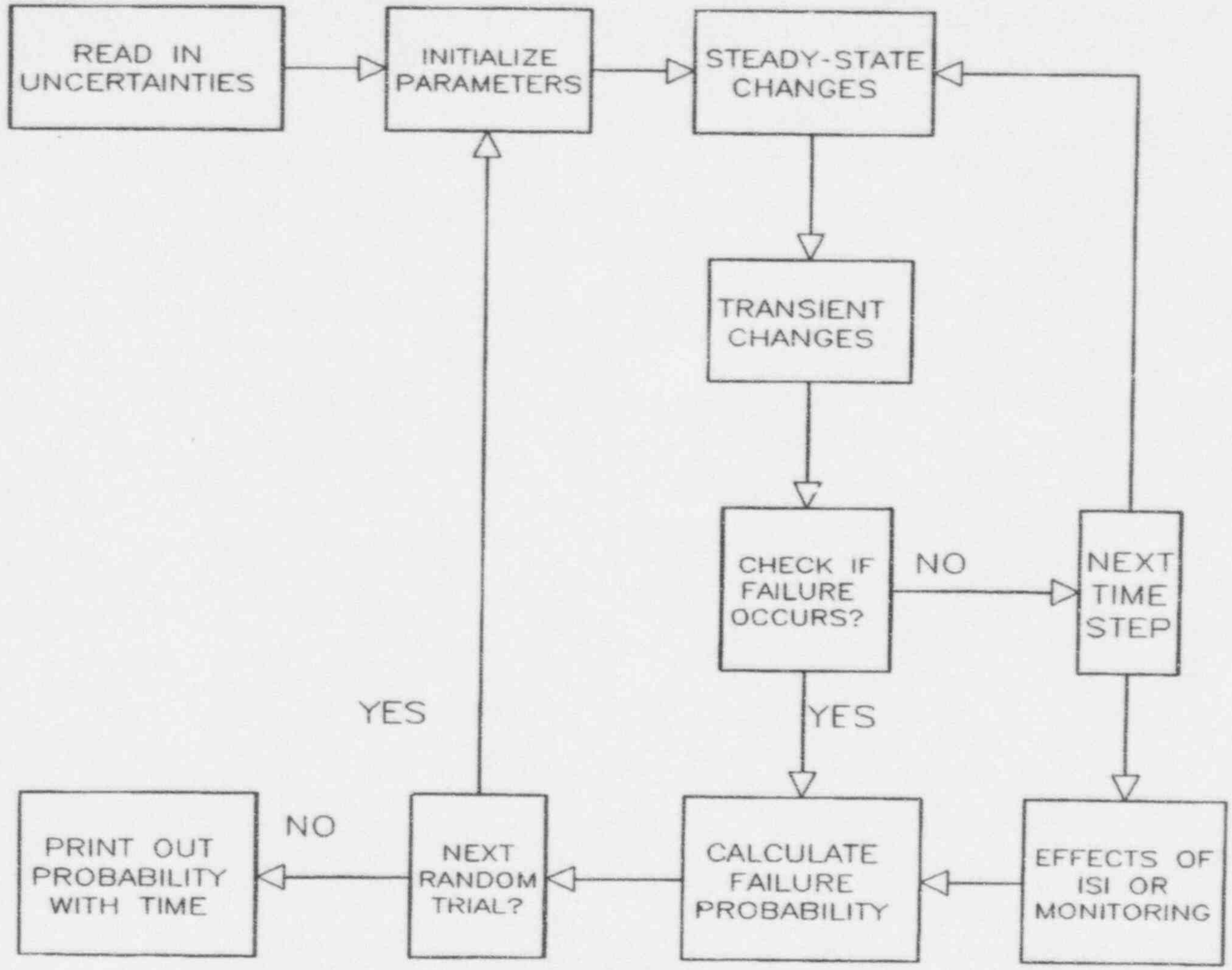


Figure D-2. Westinghouse PROF Program Flow Chart for Calculation of Failure Probabilities

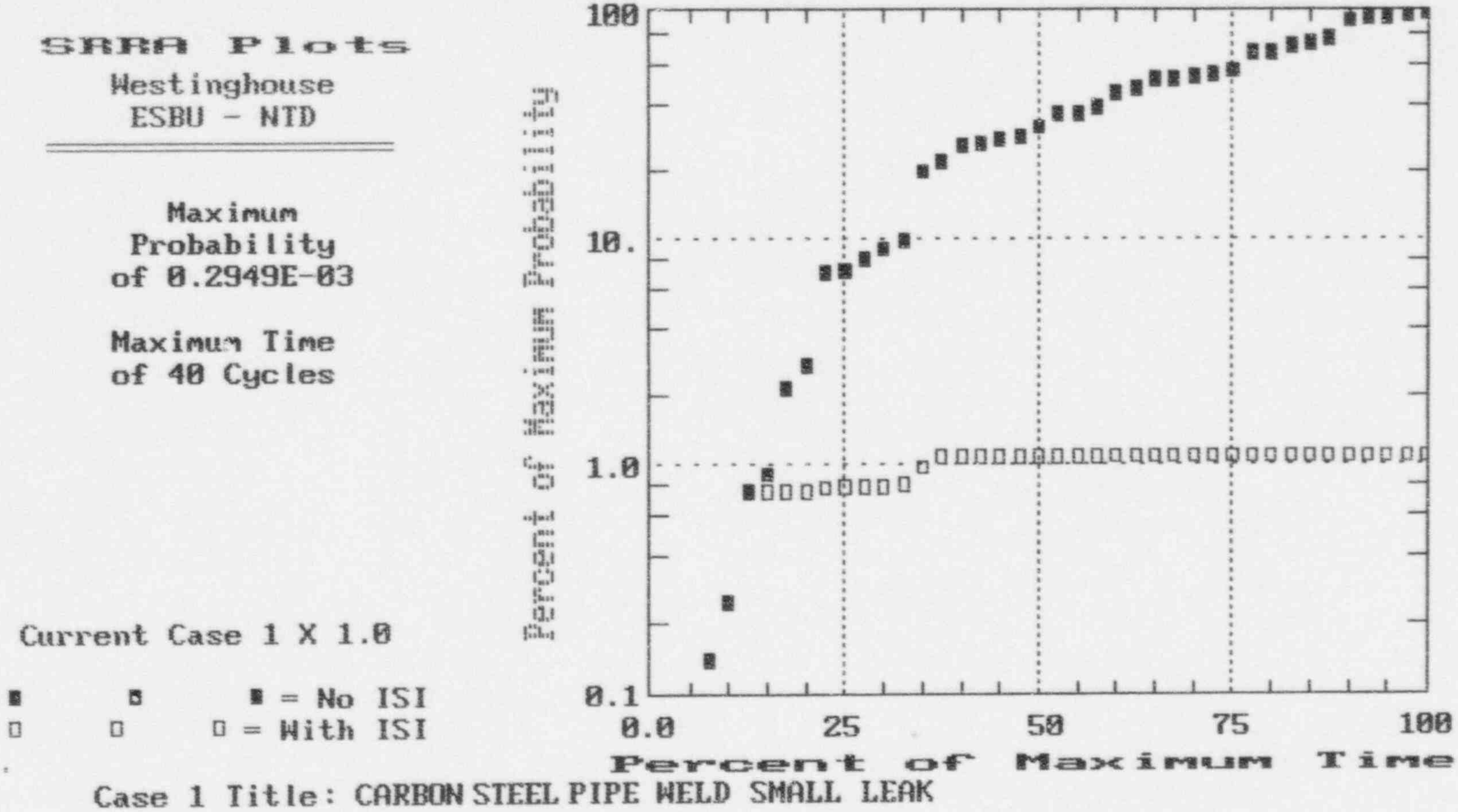


Figure D-3. Example Probability Plot From SPFMPROF and SRRAPLOT Programs

APPENDIX E
BENCHMARKING OF SRRA CODE

The probability of failure of the carbon and stainless steel piping as a function of operating time t, $Pr(t \leq t_f)$, is calculated directly for each set of input values using Monte-Carlo simulation with importance sampling. The simulation does not force the calculated output probability distribution to be of a fixed type (e.g. Weibull, Log-normal or Extreme Value). The actual output distribution is estimated based upon the distributions of the uncertainties in the key structural reliability model parameters and plant specific input parameters.

To apply the simulation method to the Millstone Unit 3 Risk-Based Inspection (RBI) Program, the existing Westinghouse PROF (probability of failure) Software System (object library) was combined with the piping structural reliability models developed for Idaho National Laboratory's RBI Programs for NRC and DoE (Bishop, 1993). The PROF library provides standard input and output, including plotting, and probabilistic analysis capabilities (e.g., random number generation, importance sampling). The result is the executable program SPFMPROF.EXE for calculation of piping failure probability with time.

The Westinghouse PROF Software Library, which was used to generate the SPFMPROF program used in the RBI program, has been verified and benchmarked in a number of ways. Table E-1 provides a comparison of probabilities from hand calculation with those from simple SRRA verification models, where the only input random variables are the initial and limiting crack depths. The crack growth due to the two independent mechanisms is deterministic (variables are constant). As can be seen the W-PROF calculated values agree very well (less than 4% error) for a number of different distributions and with the effects of importance sampling.

The calculation of failure probability using the W-PROF methods and importance sampling was also compared to that calculated by an alternative methods for more complex models. The more complex SRRA verification model also included the uncertainties in growth rate, where the rate was a function of the crack depth. The alternative method was the @RISK add-in for Lotus 1-2-3 spreadsheets (Palisade, 1992). As seen in Figure E-1, the comparison of calculated probabilities is excellent at the low probability values, where importance sampling is normally used.

In the verification of the simplified piping fracture mechanic (SPFM) structural reliability programs for risk-based inspection, the calculated probabilities for thermal transient-induced fatigue-crack growth were compared with results from the pc-PRAISE program (Harris and Dedhia, 1992). PRAISE, which was developed by Lawrence Livermore National Laboratory for the NRC, is the nuclear industry's standard for calculating the structural reliability of piping. As shown in Figure E-2, the comparison of calculated leak probabilities with the number of operating cycles, without the effects of inspection, is excellent for both the SPFMPROF and SPFMSRRA programs. The SPFMSRRA program uses Westinghouse developed approximations to estimate the changes in probability with time due to changes in the input variables relative to a reference case. The reference case is initially calculated using the SPFMPROF Program, which is the type of program used for the Millstone Unit 3 risk-based inspection.

When the same inservice inspection frequency and accuracy is used, Figure E-3 shows that essentially the same failure probabilities are calculated by pc-PRAISE, SPFMPROF and SPFMSRRA. Therefore, it is concluded that the Westinghouse methods employed in calculating probabilities with the SPFMPROF.EXE program have been sufficiently verified and benchmarked for application to the Millstone Unit 3 Risk Based Inspection Program.

Table E-1

SIMPLE VERIFICATION OF RESULTS FOR WESTINGHOUSE PROF METHODS

Type of Distribution on Crack Depths (1)	Import. Sampling Shift (2)	Hand Calculated Prob. (3)	W-PROF Calculated Probability	Percent Error
Normal	0.0	0.1003	0.10004	-0.26
Normal	± 1.0	0.1003	0.09889	-1.41
Log-Normal	0.0	0.1003	0.09880	-1.50
Log-Normal	± 1.0	0.1003	0.09652	-3.77
Uniform	0.0	0.1003	0.10393	+3.62
Log-Uniform	0.0	0.1003	0.10018	-0.12
Weibull	0.0	0.0950	0.0934	-1.68

(1) Same type of distribution on random values of initial and limiting crack depths.

(2) Median value of initial depth shifted +1 standard deviation and median value of limiting depth shifted -1 standard deviation when importance sampling is used with less than half the number of trials.

(3) Calculated using stress-strength overlap techniques on crack depth.

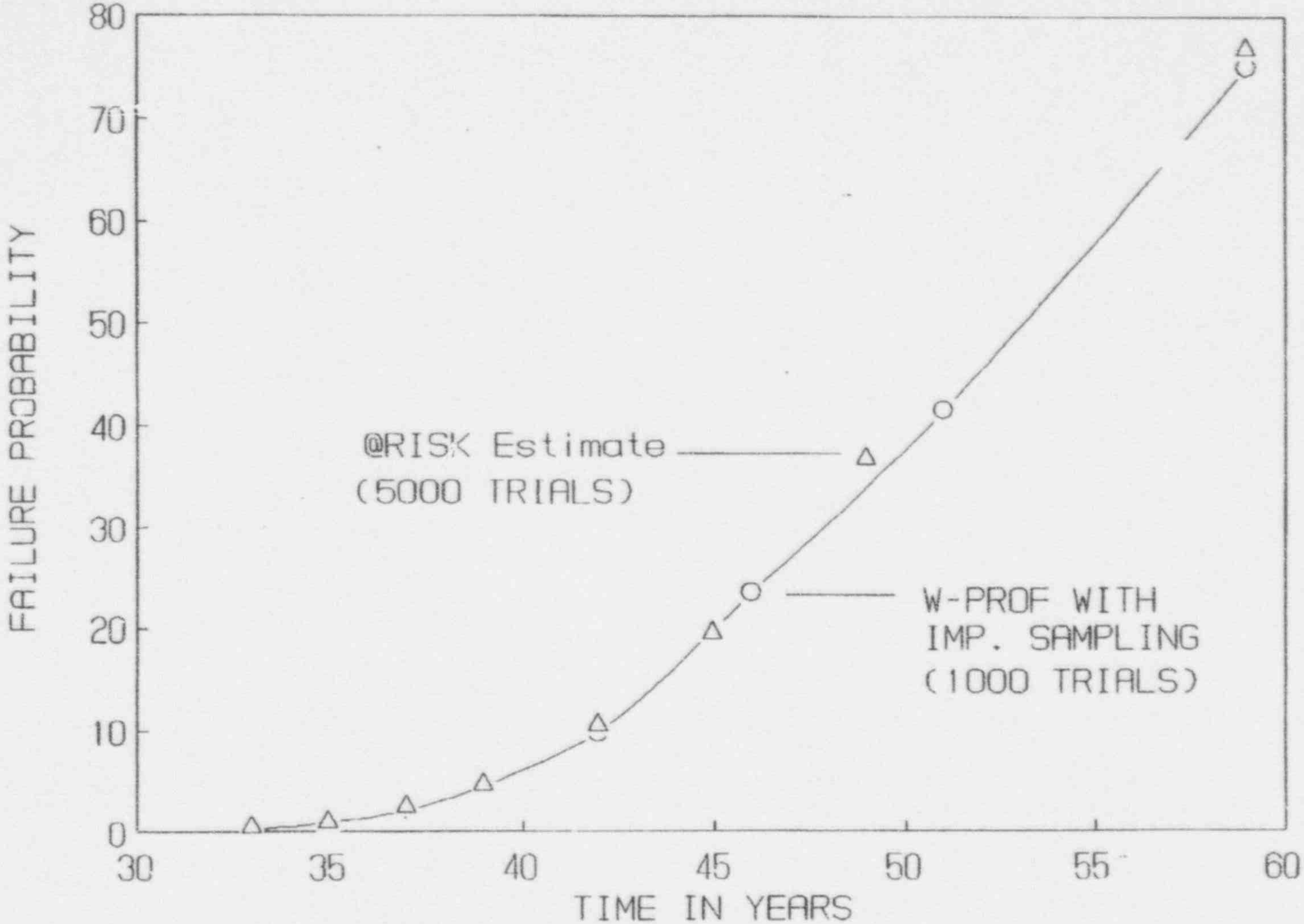


Figure E-1. Importance Sampling Check of Westinghouse PROF Methods

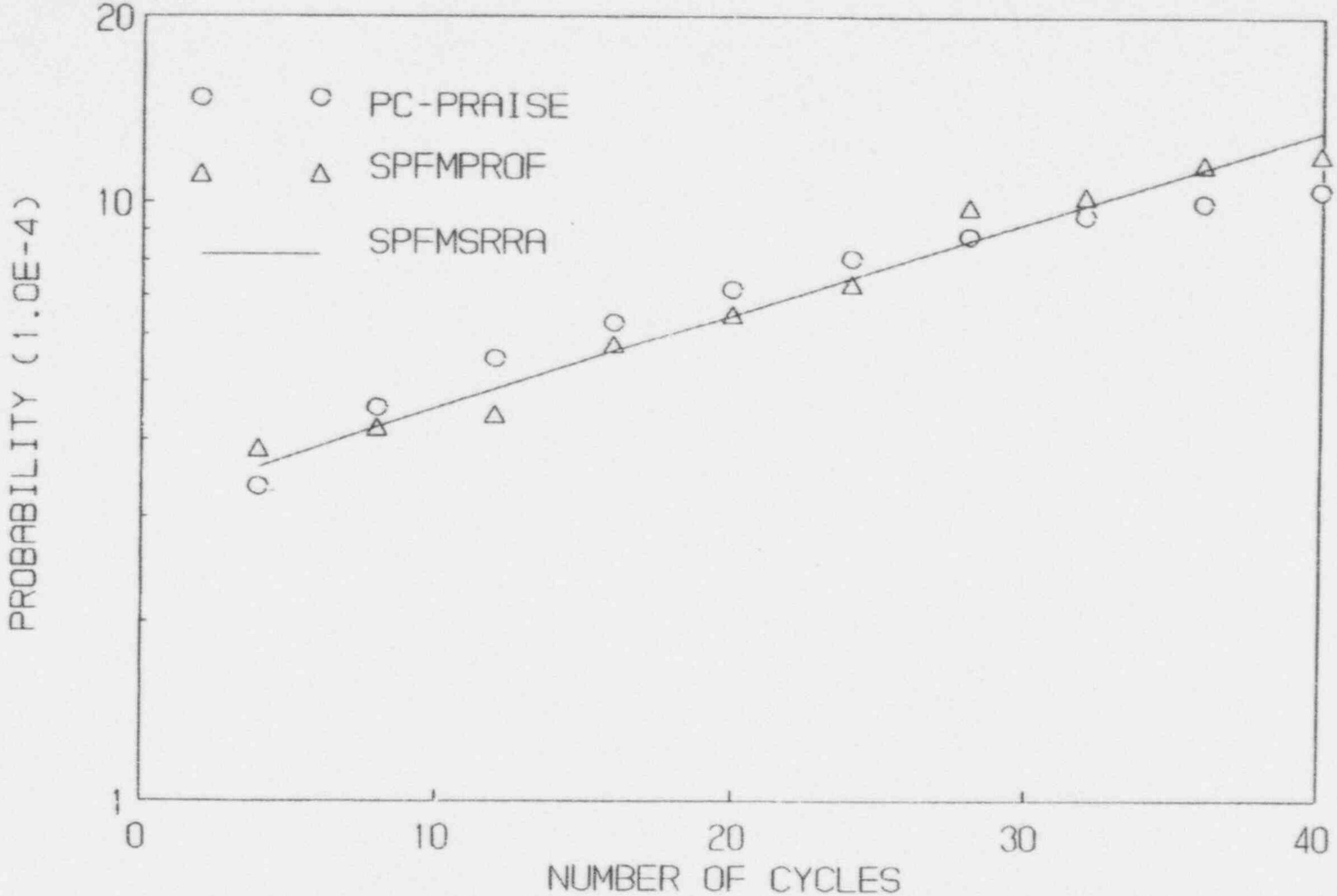


Figure E-2. Comparison of Leak Probabilities Without Inspection

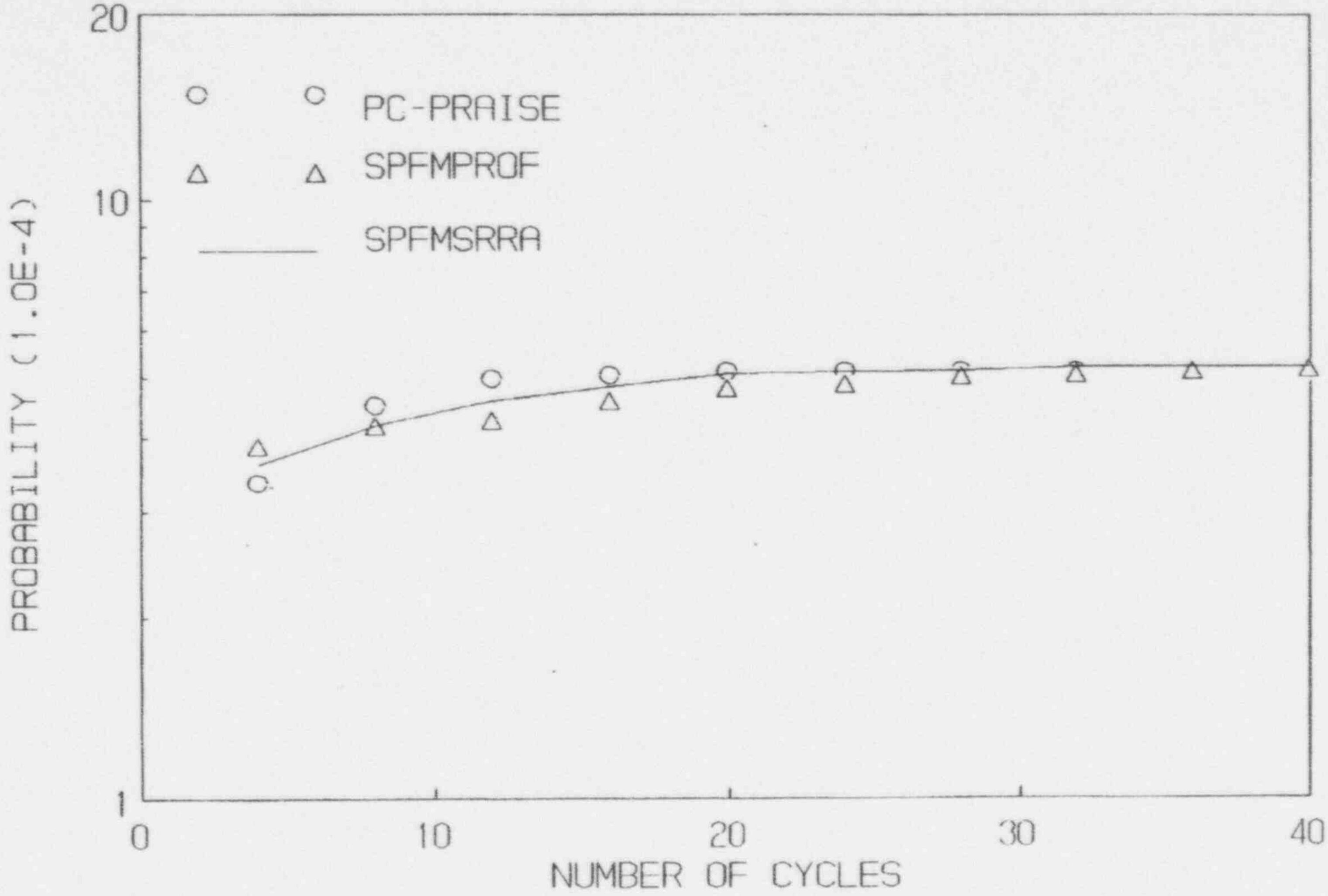


Figure E-3. Comparison of Leak Probabilities with Inservice Inspection

SUPPLEMENTAL INFORMATION

FROM REPRESENTATIVE WOG PLANT

MILLSTONE UNIT 3 RISK-BASED ISI SUPPLEMENTAL INFORMATION

Introduction

This supplemental information summarizes an implementation of the Westinghouse Owners Group Risk-Based Inservice Inspection (ISI) application for nuclear plant piping systems. Millstone Unit 3 (MP3) was the reference plant for the effort, which took place in a one year period from February 1995 to March 1996. The following provides a brief overview of the WOG process as applied to Millstone Unit 3, with enclosures providing additional detail.

Summary of Results

The Risk-Based Inservice Inspection Project at Millstone Unit 3 has been completed using the methodology described in WCAP-14572. Although Millstone will not request a NRC exemption at this time, the more safety-significant structural elements identified by the new program will be incorporated into the second 10-year ISI program as augmented examinations. A total of 119 elements/ locations have been selected for some type of examination under the Risk-Based ISI program as compared to 753 welds now scheduled under the current ASME Section XI program. Enclosure 1 provides the essence of the Risk-Based ISI program plan. Each of the identified elements will be scheduled for examination in accordance with the requirements of the ASME Code Section XI, Table IWB-2412-1 - Inspection Program B. Simplified P&IDs show all the segments and potential break locations identified by the process, but only those that were selected for examination need to be identified in an Owner's Risk-Based ISI program.

Compliance with PSA Application Guide

The proposed risk-based process is generally consistent with the Electric Power Research Institute (EPRI) PSA Applications Guide, TR-105396, dated August 1995. Enclosure 2 provides a table in which the Guide's Appendix B checklist for technical consistency in a probabilistic safety assessment (PSA) model was addressed for Millstone Unit 3. The MP3 PSA model utilized in this risk-based application is an updated version to the PSA submittal (Reference 1) to the NRC in response to Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities (IPE). The NRC Staff evaluation of that submittal was provided to Northeast Nuclear Energy Company (NNECo) in May of 1992 (Reference 2).

Documentation

The documentation to support the WOG/MP3 application of risk-based methods to piping inservice inspection includes:

- Northeast Utilities Service Company (NUSCo) Calculation File PRA96YQA-1198-S3, Rev. 0, "MP3 Risk-based Inspection CDF Calculations"
- MP3 RBI Expert Panel Meeting Minutes from 11/1995 to 3/1996
- MP3 RBI Sub-panel Meeting Minutes from 1/1996 to 3/1996
- Simplified Piping Segment drawings

- Expert Panel Worksheets for each system, and
- Failure Probability Data Collection Worksheets for each systems' piping segments (these are also included in Westinghouse Calc CN-RAS-96-39).

Westinghouse calculation file:
- CN-RAS-96-39, "WOG Risk-Based ISI: Piping Structural Reliability Estimates for Millstone 3"
- CN-RAS-96-40, "WOG Risk-Based ISI: Millstone 3 Walkdown"
- CN-RAS-96-41, "WOG Risk-Based ISI: Expert Panel Segment Ranking Worksheets for Millstone 3"
- CN-RAS-96-42, "WOG Risk-Based ISI: Piping CDF Calculation and Sensitivity Studies for Millstone 3"

Scope of Program

The structural elements considered for the Millstone 3 application included the examination items presently included under Examination Categories B-F, B-J, C-F-1, C-F-2 and D-A only as it relates to Class 3 piping systems that would be included under this category in the 1992 and later editions of ASME Section XI. The process also included evaluation of additional areas and volumes of base material and examination zones such as weld counterbore areas and fitting material with consideration to all piping welds to nozzles, valves and fittings such as tees, elbows, branch connections and safe ends. Welded attachments and piping supports were not included in the program. However, possible snubber degradation was given consideration as a factor which may increase piping fatigue effects.

Overview of the Risk-Based ISI Process

Piping systems were chosen for evaluation based on the criteria provided in Section 3.2 of the WCAP-14572 Topical Report. This required effort in part by personnel from the PSA Section, Stress Analysis and the unit's ISI Group to determine the applicable systems. The chosen systems were then reviewed by the Expert Panel for consistency and completeness. Twenty-one systems were chosen to be evaluated in more detail as shown in Section 3.2-1 of the Topical. To determine the direct and indirect consequences, PSA insights as well as plant design and operations information were used. To determine the indirect effects, the Millstone Unit 3 Hazards Evaluation and the Internal Flooding Analysis performed for the IPE were reviewed. A walkdown was also performed with both Millstone and Westinghouse personnel to address any areas of question.

A total of 259 piping segments were identified through the consequence determination process. Refer to Table 3.6-1 in the Topical for the breakdown of piping segments among systems. The consequences with and without operator action were identified and provided the necessary input to determine the conditional core damage frequency/probability contribution for each piping segment as shown in Section 3.4 and 3.6 of the Topical. In parallel with the consequence determination effort, the Stress Analysis area provided the required input to the structural reliability/risk assessment (SRRA) model to determine the failure probabilities for each piping segment as discussed in Section 3.5.

The risk ranking process involved many sensitivity calculations of the existing PSA model for Millstone Unit 3. These calculations could be broken down into three different types: 1) initiating event consequence only, 2) mitigating system(s) consequence, and 3) initiating event and mitigating system(s) consequence. Existing PSA information provided within the plant IPE submittal was available to determine only the initiating event consequence. As shown in Equation 3-1 of the Topical Report, the required values are the specific initiating event frequency and the associated core damage frequency contribution. The actual PSA model had to be manipulated for the other two types of calculations discussed in Section 3.6. Surrogate basic event(s) representing the same consequence as the piping segment failure were set equal to 1.0 (or failed) within the model to perform the calculations. The total model was recalculated to ensure no sequences were deleted as a result of the original model truncation. For the initiating event and mitigating system(s) consequence calculations, both the specific initiating event as well as the mitigating system basic events were set equal to 1.0 within the database and recalculated. The Millstone Unit 3 PSA model will generally execute the calculations within 20 minutes which made these sensitivity calculations achievable. Table 3.6-1 of the Topical Report provides the core damage frequency contributions for each of the systems addressed. Other considerations, such as external events (seismic, fire, tornado etc.), shutdown and containment performance, were supplied as qualitative information to the Expert Panel in the form of Expert Panel Worksheets (see Table 3.6-3 of the Topical).

<u>Expert Panel</u>

The Expert Panel used for this application is the same panel which is used for the Maintenance Rule and includes personnel from the following disciplines:

Plant Operations
Plant Maintenance
Plant Engineering
Probabilistic Safety Analysis
Safety Analysis
Maintenance Rule Coordinator
Plant Work Planning and Control

In addition to these traditional panel members, personnel with the following expertise in this application were added:

Stress Analysis
Plant ISI
Welding and Test Engineering
Nondestructive Examination.

This additional set of experts also served as a sub-panel for the structural element selections.

The initial meeting of the Expert Panel was a training session on the specific application, PSA and the use of importance measures, and the role of the Expert Panel in the process. To aid the Expert Panel in determining the more safety significant piping

segments, piping segment worksheets as well as associated simplified piping drawings were provided. The Expert Panel's responsibilities included the acceptance of the system list for this application, review of piping segment boundaries and consequence, providing additional information on the worksheets, and the final determination of safety significance. The Expert Panel provided significant input in the area of consequences which resulted in changes to the originally postulated consequences and in a few cases, changes to the piping segment boundaries. Operations was critical in determining whether operator recovery action was possible given a specific pipe rupture. Safety Analysis also provided input on the time available to take certain operator actions if necessary. A total of eight expert panel meetings, each taking about 1-2 hours, were held to evaluate the safety significance of the piping segments.

Enclosure 3 provides an example of the MP3 RBI Expert Panel Meeting Minutes (w/o Attachments). Table 3.6-6 of the Topical lists the number of piping segments determined to be more safety-significant by the Expert Panel. A total number of 96 or 37% of the piping segments were determined to be more safety-significant. The Expert Panel included all the piping segments with a Risk Reduction Worth (RRW) of 1.005 as well as several others based on other considerations such as shutdown risk.

Structural Element Selection

Once the more safety-significant segments were identified, a sub-panel consisting of the members with special expertise in this area met to evaluate which structural elements should be examined and the examination method to be employed. Four sub-panel meetings were held to address the 96 more safety-significant segments. The sub-panel reviewed each proposed inspection location within those segments and verified that these were the potential failure locations within each respective segment. In some cases, more than one element was selected. For each element, the sub-panel determined the examination method to be used. The panel considered available technology which would best detect any flaws. The final list of structural elements was consolidated into a single list and input to a modified version of the existing Section XI database program.

Risk / Safety Evaluation

Figure 4.3-1 of the Topical shows a comparison of the core damage frequency being addressed by examination of the 119 structural elements in the Risk-Based ISI program versus the 753 weld locations that are examined per current ASME Section XI requirements. Examination of the current ASME Code weld locations addresses a total CDF of 1.0E-08/yr (44% of total) while examination of the Risk-Based ISI structural elements addresses a total CDF of 2.25E-08/yr (98% of total) for pressure boundary piping failures. Thus, safety is enhanced with far less locations being inspected.

The total piping core damage frequency is a small fraction of the total plant core damage frequency of 5.87E-05/yr. Examination of the plant piping at the risk-based locations, however, will verify that the risk of piping pressure boundary failure remains a small contributor to total risk as the unit ages over its licensed life.

Economic Evaluation

The economic evaluation addresses the current Section XI piping examination costs and the savings to be realized in the reduction of these examinations. The basis of this comparison includes the direct costs incurred in performing current Section XI required examinations during MP3's fifth refueling outage. The direct costs are as follows

Refueling Outage 5 Examination Costs	
Examination Costs	$167,000
Insulation Removal/Reinstallation	$123,000
Required Scaffolding	$ 96,000
Total	$386,000

The Risk-Based ISI application resulted in a reduction of approximately 86% of the piping examinations required in comparison with the current program. Therefore, on a strictly direct cost basis, the ISI program savings associated with implementing the risk-based program would be approximately $332,000 per outage. MP3 has three more 10 year inspection intervals remaining within the current operating license. Given that MP3 averages 5 outages per interval, this $332,000 savings is expected to be gained 15 times over the licensed life of the unit.

During Refueling Outage 5, it is estimated that 17 Person-Rem were expended in performing the examination of current Section XI required locations. It is estimated that only 2 Person-Rem would be expended to examine the risk-based locations, resulting in a 15 Person-Rem savings each outage.

In addition, other indirect cost savings are expected from items such as reduction in costs associated with evaluating flaw indications, which may not really exist (i.e., false call), in less safety-significant piping systems.

Utility's Perspective

A significant amount of time was spent in developing this process; however, the actual implementation effort would be greatly reduced for the other Northeast Utilities units. Other nuclear plants could implement the process with similar efficiency. In addition, the knowledge gained from this application is able to be applied to other risk-based applications. Hence this is a worthwhile application for any utility. With moderate resources and team effort, the program will be a success in terms of both safety and economic benefits.

References:

1) E. J. Mroczka Letter to U.S. Nuclear Regulatory Commission, " Millstone Nuclear Power Station, Unit No. 3 Response to Generic Letter 88-20 Individual Plant Examination for Severe Accident Vulnerabilities Summary Report Submittal", dated August 31, 1990.

2) U.S. NRC Letter to J. F. Opeka, " Staff Evaluation of Millstone 3 Individual Plant Examination (IPE) - Internal Events, GL 88-20 (TAC No. M74434)", dated May 8, 1992.

ENCLOSURE 1

MILLSTONE UNIT 3 RISK-BASED

ISI PROGRAM PLAN

Millstone Unit 3
Risk-Based ISI Plan Element Selections

Reactor Component Coo ing System

CCP-1
Failure Mode: Cracking - External Loads — FP 1.0E-08*
Not in ISI Program. (CCP-010-492-RBI-1-3) — VT-2, MT
10" Pipe to Flange Fillet Weld — Class 3 C/S — Piping shown in Zone 157
DWG# S&W 12179-CI-CCP-264 SH.1 of 3

CCP-2
Failure Mode: Cracking - External Loads — FP 1.0E-08*
Not in ISI Program. (CCP-010-28-RBI-1-3) — VT-2, MT
10" Pipe to Flange Fillet Weld — Class 3 C/S — Piping shown in Zone 167
DWG# S&W 12179-CI-CCP-6 SH.3 of 4

CCP-4
Failure Mode: Cracking - Vibration Fatigue — FP 1.7E-08
Not in ISI Program. (CCP-018-1-RBI-1-3) — VT-2, RT
14" Expander to Flange Weld — Class 3 C/S — Piping shown in Zone 165
Exam volume to extend 1" on expander side of weld.
DWG# S&W 12179-CI-CCP-22 SH.3 of 3

CCP-5
Failure Mode: Cracking - Vibration Fatigue — FP 1.7E-08
Not in ISI Program. (CCP-018-2-RBI-1-3) — VT-2, RT
Not in ISI Program. (CCP-018-3-RBI-1-3) — VT-2, RT
14" Expander to Flange Weld — Class 3 C/S — Piping shown in Zone 159
14" Expander to Flange Weld — Class 3 C/S — Piping shown in Zone 163
Exam volume to extend 1" on expander side of weld.
DWG# S&W 12179-CI-CCP-23 SH.3 of 3 / 12179-CI-CCP-24 SH.3 of 3

Chemical & Volume Control System

CHS-3
Failure Mode: Cracking - Vibration Fatigue — FP 1.0E-08*
Not in ISI Program. (CHS-002-565-RBI-1-2) — VT-2, RT
Not in ISI Program. (CHS-002-564-RBI-1-2) — VT-2, RT
Not in ISI Program. (CHS-002-566-RBI-1-2) — VT-2, RT
Three 2" Pipe to Reducer Welds — Class 2 S/S
Schedule exams following pump test.
DWG# S&W 12179-CP-374002 SH.3 of 3 / 12179-CP-374508 SH.3 of 3 / 12179-CP-374509 SH.3 of 3

*Failure Probability < 1.0E-08

CHS-5
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
ISI Zone: 128 (SIH-3-5-SW-D) 4" Tee to Pipe Weld VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20990 Class 2 S/S

CHS-7
Failure Mode: Cracking - External Loads FP 1.0E-08*
ISI Zone: 133 (CHS-35-2-SW-4) 3" Pipe to Tee Weld VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20354 Class 2 S/S

CHS-23
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
ISI Zone: 128 (SIH-3-FW-10) 3" Pipe to Penetration Weld VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20990 Class 2 S/S

Emergency Core Cooling System

ECCS-0
Failure Mode: Cracking - OD Corrosion External Loads FP 1.0E-08*
ISI Zone: 119 (SIL-508-RBI-1) Class 2 S/S VT-2, ET
Base metal of 24" pipe at ground interface. Exam area 3" above and below interface.
DWG# 25212-20896

Auxiliary Feedwater System

FWA-7
Failure Mode: Cracking - Vibration Fatigue FP 1.0E-08*
2" Reducer to Pipe Weld. Class 3 C/S
Not in ISI Program. (FWA-002-142-RBI-1-3) VT-2, RT
Schedule exams following pump test or inservice operation.
DWG# S&W 12179-CI-FWA-9 SHT.1 of 6

FWA-12
Failure Mode: Cracking - External Loads FP 1.0E-08*
4" Tee to Pipe Weld.
ISI Zone: 110 (FWA-8-FW-36-1) VT-2, MT, RT
Exam volume and area to extend 1" on each side of weld.
DWG# 25212-20887 Class 2 C/S

FWA-13 Less-Safety-Significant Segment & Element Chosen For Plant Reliability
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
16" Reducing Elbow to SG Nozzle Weld on Main Feedwater Line Loop D.
ISI Zone: 68 (FWS-25-FW-71) VT-2, RT
Exam volume to extend 2" on each side of weld. Leak probability w/o ISI 9.5E-04.
DWG# 25212-20966 Class 2 C/S

FWA-14
Failure Mode: Cracking - External Loads FP 1.0E-08*
4" Tee to Pipe Weld.
ISI Zone: 109 (FWA-19-4-SW-F) VT-2, MT, RT
Exam volume and area to extend 1" on each side of weld.
DWG# 25212-20859 Class 2 C/S

FWA-15 Less-Safety-Significant Segment & Element Chosen For Plant Reliability
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
16" Pipe to SG Nozzle Weld on Main Feedwater Line Loop C.
ISI Zone: 67 (FWS-22A-FW-38) VT-2, UT
Exam volume to extend 2" on each side of weld. Leak probability w/o ISI 9.9E-04.
DWG# 25212-20965 Class 2 C/S

FWA-16
Failure Mode: Cracking - External Loads FP 1.0E-08*
4" Tee to Pipe Weld.
ISI Zone: 108 (FWA-5-4-SW-E) VT-2, MT, RT
Exam volume and area to extend 1" on each side of weld.
DWG# 25212-20858 Class 2 C/S

FWA-17 Less-Safety-Significant Segment & Element Chosen For Plant Reliability
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
16" Pipe to SG Nozzle Weld on Main Feedwater Line Loop B.
ISI Zone: 66 (FWS-21-FW-50) VT-2, UT
Exam volume to extend 2" on each side of weld. Leak probability w/o ISI 9.8E-04.
DWG# 25212-20964 Class 2 C/S

FWA-18
Failure Mode: Cracking - External Loads FP 1.0E-08*
4" Tee to Pipe Weld.
ISI Zone: 107 (FWA-6-1-SW-C) VT-2, MT, RT
Exam volume and area to extend 1" on each side of weld.
DWG# 25212-20857 Class 2 C/S

FWA-19 Less-Safety-Significant Segment & Element Chosen For Plant Reliability
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
16" Reducing Elbow to SG Nozzle Weld on Main Feedwater Line Loop A.
ISI Zone: 65 (FWS-23A-FW-125) VT-2, RT

Exam volume to extend 2" on each side of weld. Leak probability w/o ISI 9.5E-04.
DWG# 25212-20963 Class 2 C/S

Quench Spray System

QSS-2
Failure Mode: Cracking - Vibration Fatigue FP 1.0E-08*
ISI Zone: 98 (QSS-1-1-SW-D) 12" Pipe to Elbow Weld VT-2, UT
ISI Zone: 97 (QSS-3-1-SW-D) 12" Pipe to Elbow Weld VT-2, UT
Schedule exams following pump test. Class 2 S/S
DWG# 25212-20874 & 20873

Reactor Coolant System

RCS-1
Failure Mode: Cracking - Thermal Fatigue FP 4.1E-07
12" Branch Connection Weld to 29" Run Pipe.
ISI Zone: 12 (RCS-LP1-FW-HL1-CMR) VT-2, UT
UT exam limited to branch connection side of weld and the exam volume will be extended 1" on the branch side of the weld.
DWG# 25212-20910 Class 1 S/S to Cast S/S

RCS-2
Failure Mode: Cracking - Vibration Fatigue FP 4.1E-07
29" Dissimilar Metal Weld Elbow to SG.
ISI Zone: 12 (RCS-LP1-FW-4) VT-2, RT or UT
Special UT exam would be performed from the ID of the SG channel head nozzle.
DWG# 25212-20910 Class 1 Cast S/S to C/S

RCS-3
Failure Mode: Cracking - Vibration Fatigue FP 1.0E-07
2" Branch Connection Weld to 31" Run Pipe.
ISI Zone: 12 (RCS-LP1-EC2-SW-F) VT-2
The selected break location is on the 31"pipe. Selection of this exam method is justified by the leak before analysis described in WCAP-10587 dated June 1984.
DWG# 25212-20910 Class 1 S/S to Cast S/S

RCS-4
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
2" Pipe to Branch Connection Weld and Two 2" Pipe to Elbow Welds.
ISI Zone: 47 (RCS-374076-FW-1) VT-2, RT
ISI Zone: 47 (RCS-374076-FW-5) VT-2, RT
ISI Zone: 47 (RCS-374076-FW-6) VT-2, RT
DWG# 25212-20949 Class 1 S/S

RCS-5
Failure Mode: Cracking - External Loads FP 3.5E-07
27.5" Pipe to Valve Weld.
ISI Zone: 12 (RCS-5-FW-8) VT-2
Leak before break analysis has been considered in the selection of this exam method in accordance with WCAP-10587 dated June 1984.
DWG# 25212-20910 Class 1 Cast S/S

RCS-6
Failure Mode: Cracking - Thermal Fatigue FP 3.4E-07
27.5" Dissimilar Metal Weld Safe End to RPV Nozzle.
ISI Zone: 12 (RCS-301-121-C) VT-2, UT
Full volume UT exam performed from the ID of the nozzle during RPV weld exams.
DWG# 25212-20910 Class 1 S/S to C/S

RCS-7
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
10" Pipe to Branch Connection Weld.
ISI Zone: 22 (SIL-4-FW-11) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20924 Class 1 S/S

RCS-8
Failure Mode: Cracking - Thermal Fatigue FP 5.5E-07
29" Dissimilar Metal Weld Safe End to RPV Nozzle.
ISI Zone: 13 (RCS-302-121-D) VT-2, UT
Full volume UT exam performed from the ID of the nozzle during RPV weld exams.
DWG# 25212-20911 Class 1 S/S to C/S

RCS-9
Failure Mode: Cracking - Vibration Fatigue FP 4.1E-07
29" Pipe Weld to Valve Cast S/S.
ISI Zone: 13 (RCS-LP2-FW-2) VT-2
Leak before break analysis has been considered in the selection of this exam method in accordance with WCAP-10587 dated June 1984.
DWG# 25212-20911 Class 1 Cast S/S

RCS-10
Failure Mode: Cracking - Vibration Fatigue FP 4.4E-07
29" Dissimilar Metal Weld Elbow to SG.
ISI Zone: 13 (RCS-LP2-FW-4) VT-2, RT or UT
Special UT exam would be performed from the ID of the SG channel head nozzle.
DWG# 25212-20911 Class 1 Cast S/S to C/S

RCS-11
Failure Mode: Cracking - Vibration Fatigue FP 6.4E-07
31" Pipe to Elbow Weld.
ISI Zone: 13 (RCS-LP2-EC2-SW-B) VT-2
Leak before break analysis has been considered in the selection of this exam method in accordance with WCAP-10587 dated June 1984.
DWG# 25212-20911 Class 1 Cast S/S

RCS-12
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
2" Pipe to Branch Connection Weld and Two 2" Pipe to Elbow Welds.
ISI Zone: 49 (RCS-374078-FW-11) VT-2, RT
ISI Zone: 49 (RCS-374078-FW-5) VT-2, RT
ISI Zone: 49 (RCS-374078-FW-7) VT-2, RT
DWG# 25212-20951 Class 1 S/S

RCS-13
Failure Mode: Cracking - External Loads FP 3.8E-07
27.5" Pipe to Valve Weld.
ISI Zone: 13 (RCS-10-FW-18) VT-2
Leak before break analysis has been considered in the selection of this exam method in accordance with WCAP-10587 dated June 1984.
DWG# 25212-20911 Class 1 Cast S/S

RCS-14
Failure Mode: Cracking - Thermal Fatigue FP 4.8E-07
27.5" Dissimilar Metal Weld Safe End to RPV Nozzle.
ISI Zone: 13 (RCS-301-121-D) VT-2, UT
Full volume UT exam performed from the ID of the Nozzle during RPV weld exams.
DWG# 25212-20911 Class 1 S/S to C/S

RCS-15
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
1.5" Pipe to Valve Weld.
ISI Zone: 38 (RCS-408045-FW-4) VT-2, RT
DWG# 25212-20940 Class 1 S/S

RCS-16
Failure Mode: Cracking - Thermal Fatigue FP 5.5E-07
29" Dissimilar Metal Weld Safe End to RPV Nozzle.
ISI Zone: 14 (RCS-302-121-A) VT-2, UT
Full volume UT exam performed from the ID of the nozzle during RPV weld exams.
DWG# 25212-20912 Class 1 S/S to C/S

RCS-17
Failure Mode: Cracking - Vibration Fatigue FP 4.1E-07
29" Dissimilar Metal Weld Elbow to SG.
ISI Zone: 14 (RCS-LP3-FW-4) VT-2, RT or UT
Special UT exam would be performed from the ID of the SG channel head nozzle.
DWG# 25212-20912 Class 1 Cast S/S to C/S

RCS-18
Failure Mode: Cracking - Vibration Fatigue FP 4.1E-06
2" Branch Connection Weld to 31" Run Pipe.
ISI Zone: 14 (RCS-LP3-EC2-SW-F) VT-2
Break location is on the 31" pipe. Leak before break analysis has been considered in the selection of this exam method in accordance with WCAP-10587 dated June 1984.
DWG# 25212-20912 Class 1 S/S to Cast S/S

RCS-19
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
2" Pipe to Branch Connection Weld and One 2" Elbow to Pipe Weld.
ISI Zone: 50 (RCS-374079-FW-1) VT-2, RT
ISI Zone: 50 (RCS-374079-FW-7) VT-2, RT
DWG# 25212-20952 Class 1 S/S

RCS-20
Failure Mode: Cracking - External Loads FP 3.5E-07
27.5" Pipe to Valve Weld.
ISI Zone: 14 (RCS-15-FW-28) VT-2
Leak before break analysis has been considered in the selection of this exam method in accordance with WCAP-10587 dated June 1984.
DWG# 25212-20912 Class 1 Cast S/S

RCS-21
Failure Mode: Cracking - Thermal Fatigue FP 4.7E-07
27.5" Dissimilar Metal Weld Safe End to RPV Nozzle.
ISI Zone: 14 (RCS-301-121-A) VT-2, UT
Full volume UT exam performed from the ID of the nozzle during RPV weld exams.
DWG# 25212-20912 Class 1 S/S to C/S

RCS-22
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
10" Pipe to Branch Connection Weld.
ISI Zone: 25 (SIL-6-FW-11) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20927 Class 1 S/S

RCS-23
Failure Mode: Cracking - Thermal Fatigue FP 4.1E-07
12" Branch Connection Weld to 29" Run Pipe.
ISI Zone: 15 (RCS-LP4-FW-HL1-CMR) VT-2, UT
UT exam limited to branch connection side of weld and the exam volume will be extended 1" on the branch side of the weld.
DWG# 25212-20913 Class 1 S/S to Cast S/S

RCS-24
Failure Mode: Cracking - Vibration Fatigue FP 4.4E-07
29" Dissimilar Metal Weld Elbow to SG.
ISI Zone: 15 (RCS-LP4-FW-4) VT-2, RT or UT
Special UT exam would be performed from the ID of the SG channel head nozzle.
DWG# 25212-20913 Class 1 Cast S/S to C/S

RCS-25
Failure Mode: Cracking - Vibration Fatigue FP 6.4E-07
31" Pipe to Elbow Weld.
ISI Zone: 15 (RCS-LP4-EC2-SW-B) VT-2, RT or Vibration Monitoring
This weld may have a relatively higher vibration level than other welds in the segment. If vibration monitoring is used and the results are negligible, only a VT-2 exam of this weld will be performed. This exam method is then justified by the leak before break analysis described in WCAP-10587 dated June 1984.
DWG# 25212-20913 Class 1 Cast S/S

RCS-26
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
2" Pipe to Branch Connection Weld and 2" Elbow to Pipe Weld.
ISI Zone: 52 (RCS-374077-FW-11) VT-2, RT
ISI Zone: 52 (RCS-374077-FW-7) VT-2, RT
DWG# 25212-20954 Class 1 S/S

RCS-27
Failure Mode: Cracking - External Loads FP 3.8E-07
27.5" Pipe to Valve Weld.
ISI Zone: 15 (RCS-20-FW-38) VT-2
Leak before break analysis has been considered in the selection of this exam method in accordance with WCAP-10587 dated June 1984.
DWG# 25212-20913 Class 1 Cast S/S

RCS-28
Failure Mode: Cracking - Thermal Fatigue FP 4.8E-07
27.5" Dissimilar Metal Weld Safe End to RPV Nozzle.
ISI Zone: 15 (RCS-301-121-B) VT-2, UT
Full volume UT exam performed from the ID of the nozzle during RPV weld exams.
DWG# 25212-20913 Class 1 S/S to C/S

RCS-29
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
10" Pipe to Branch Connection Weld.
ISI Zone: 26 (SIL-7-FW-11) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20928 Class 1 S/S

Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
14" Elbow to Pipe Weld.
ISI Zone: 16 (RCS-SL-FW-3) VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20918 Class 1 S/S

RCS-31
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
6" Pipe to Elbow Weld.
ISI Zone: 20 (RCS-516-1-SW-5) VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20922 Class 1 S/S

RCS-32
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
6" Elbow to pipe Weld.
ISI Zone: 20 (RCS-516-FW-13-1) VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20922 Class 1 S/S

RCS-33
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
6" Elbow to pipe Weld.
ISI Zone: 20 (RCS-516-6-SW-2) VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20922 Class 1 S/S

RCS-34
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
3" Tee to Pipe Weld.
ISI Zone: 21 (RCS-513-1-SW-7) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20923 Class 1 S/S

RCS-35
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
3" Pipe to Flange Weld.
ISI Zone: 21 (RCS-513-3-SW-2) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20923 Class 1 S/S

RCS-36
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
3" Pipe to Flange Weld.
ISI Zone: 21 (RCS-513-4-SW-2) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20923 Class 1 S/S

RCS-38
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
2" Pipe to Elbow Weld.
ISI Zone: 43 (RCS-408005-FW-1) VT-2, RT
DWG# 25212-20945 Class 1 S/S

RCS-40
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
6" Dissimilar Metal Weld Pressurizer Spray Nozzle to Safe End.
ISI Zone: 7 (RCS-03-X-5641-E-T) VT-2, RT
DWG# 25212-20905 Class 1 C/S to S/S

RCS-42
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
12" Pipe to Tee Weld.
ISI Zone: 27 (RCS-501-1-SW-5) VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20929 Class 1 S/S

RCS-43
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-07
8" Valve to Pipe Weld.
ISI Zone: 29 (RCS-504A-FW-2) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20931 Class 1 S/S

RCS-45
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
4" Branch Connection to Pipe Weld.
ISI Zone: 17 (RCS-518-FW-1) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20919 Class 1 S/S

RCS-47
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
3" Elbow to Branch Connection Weld and 3" Pipe to Elbow Weld.
ISI Zone: 48 (3-CHS-14-FW-12) VT-2, UT
ISI Zone: 48 (3-CHS-14-FW-20) VT-2, UT
Exam volume to extend 1" on the pipe side of the pipe to elbow weld.
DWG# 25212-20950 Class 1 S/S

RCS-49
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
Two 1.5" Tee to Pipe Welds and One Tee to Reducer Weld.
ISI Zone: 37 (RCS-408046-FW-6) VT-2, RT
ISI Zone: 37 (RCS-408046-FW-7) VT-2, RT
ISI Zone: 37 (RCS-408046-FW-9) VT-2, RT
DWG# 25212-20939 Class 1 S/S

RCS-50
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
6" Pipe to Elbow Weld.
ISI Zone: 24 (SIL-13-4-SW-C) VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20926 Class 1 S/S

RCS-51
Failure Mode: Cracking - Thermal & Vibration Fatigue FP 1.0E-08*
8" Pipe to Valve Nozzle Weld.
ISI Zone: 30 (RCS-504B-FW-4) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20932 Class 1 S/S to Cast S/S

RCS-53
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
4" Pipe to Branch Connection Weld.
ISI Zone: 18 (RCS-517-FW-1) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20920 Class 1 S/S

RCS-54
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
10" Pipe to Elbow Weld.
ISI Zone: 23 (SIL-5-6-SW-B) VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20925 Class 1 S/S

RCS-55
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
6" Pipe to Valve Weld.
ISI Zone: 39 (RCS-LP3-FW-27) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20941 Class 1 S/S

RCS-56
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-07
8" Pipe to Valve Weld.
ISI Zone: 31 (RCS-504C-FW-2) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20933 Class 1 S/S

RCS-58
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
3" Pipe to Elbow Weld.
ISI Zone: 51 (RCS-507-1-SW-2) VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20953 Class 1 S/S

RCS-60
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
1.5" Pipe to Valve Weld.
ISI Zone: 40 (RCS-408044-FW-10-1) VT-2, RT
DWG# 25212-20942 Class 1 S/S

RCS-61
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
12" Pipe to Branch Connection Weld.
ISI Zone: 28 (RHS-502-FW-1) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20930 Class 1 S/S

RCS-62
Failure Mode: Cracking - Thermal & Vibration Fatigue FP 1.0E-08*
8" Pipe to Valve Nozzle Weld.
ISI Zone: 32 (RCS-504D-FW-4) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20934 Class 1 S/S to Cast S/S

RCS-64
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
3" Elbow to Branch Connection Weld and 3" Pipe to Elbow Weld.
ISI Zone: 48 (3-CHS-15-FW-15) VT-2, UT
ISI Zone: 48 (3-CHS-15-FW-26) VT-2, UT

Exam volume to extend 1" on pipe side of the pipe to elbow weld.
DWG# 25212-20950 Class 1 S/S

RCS-66
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
Two 1.5" Pipe to Tee Welds and One 1.5" Tee to Reducer Weld.
ISI Zone: 41 (RCS-408043-FW-14) VT-2, RT
ISI Zone: 41 (RCS-408043-FW-8) VT-2, RT
ISI Zone: 41 (RCS-408043-FW-9) VT-2, RT
DWG# 25212-20943 Class 1 S/S

Containment Recirculation System

RSS-11
Failure Mode: Cracking - Vibration Fatigue FP 1.0E-08*
ISI Zone: 90 (SIH-12-FW-8) 6" Pipe to Valve Weld VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20990 Class 2 S/S

High Pressure Safety Injection System

SIH-1
Failure Mode: Cracking - External Loads FP 1.0E-08*
ISI Zone: 90 (SIH-13-3-SW-E) 8" Tee to Pipe Weld VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20867 Class 2 S/S

SIH-2
Failure Mode: Cracking - Vibration Fatigue FP 1.8E-08
Not in ISI Program. (SIH-9-RBI-1) 3" Flange to Reducer Weld VT-2, RT
Schedule exam following pump test. Exam volume to extend 1" on reducer side.
DWG# 25212-20403 SH.25 / S&W 12179-CI-SIH-9 SHT.1 of 4 Class 2 S/S

SIH-3
Failure Mode: Cracking - Vibration Fatigue FP 1.8E-08
Not in ISI Program. (SIH-7-RBI-1) 3" Flange to Reducer Weld VT-2, RT
Schedule exam following pump test. Exam volume to extend 1" on reducer side.
DWG# 25212-20403 SH.17 / S&W 12179-CI-SIH-7 SHT.1 of 5 Class 2 S/S

SIH-4
Failure Mode: Cracking - External Loads FP 1.0E-08*
ISI Zone: 127 (SIH-8-FW-6) 4" Pipe to Valve Weld VT-2, UT
Exam volume to extend 1" on pipe side of weld. Class 2 S/S
DWG# 25212-20289

Low Pressure Safety Injection System

SIL-1
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
3" Pipe to Branch Connection Weld. Class 2 S/S
Not in ISI Program. (RHS-003-16-RBI-1-2) VT-2, RT
Exam volume to extend 1" on pipe side of weld. Piping shown in Zone 118
DWG# S&W 12179-CI-RHS-9 SHT.1 of 4

SIL-2
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
12" Pipe to Tee Weld.
ISI Zone: 113 (SIL-9T-FW-10) VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20890 Class 2 S/S

SIL-3
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
8" Pipe to 3" Long Transition Piece Weld and 3" Transition Piece to Valve Weld.
ISI Zone: 89 (SIL-11-1-SW-M) VT-2, RT
ISI Zone: 89 (SIL-11-FW-2) VT-2, RT
Exam volume to extend 1" on each side of first weld and 1" on transition piece of second weld including the valve side counterbore region of the valve body.
DWG# 25212-20866 Class 2 S/S

SIL-4
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
6" Tee to Pipe Weld.
ISI Zone: 79 (SIL-501-1-SW-5) VT-2, UT
Exam volume to extend 1" on each side of weld.
DWG# 25212-20971 Class 2 S/S

SIL-5
Failure Mode: Cracking - Thermal Fatigue FP 1.0E-08*
6" Reducer to Pipe Weld.
ISI Zone: 80 (SIL-504-1-SW-7) VT-2, UT
Exam volume to extend 1" on pipe side of weld.
DWG# 25212-20972 Class 2 S/S

Service Water System

SWP-1
Failure Mode: Cracking - Vibration Fatigue & Erosion FP 2.6E-08
30" Elbow to Pipe Weld.
Not in ISI Program. (SWP-030-7-RBI-1-3) VT-2, VT-3, UT

Exam volume to extend 1" on each side of weld. Internal VT-3 of ARCOR coating to be performed at time of check valve dissassembly once per interval.
Piping shown in Zone 181 Class 3 C/S/CUNi Clad
DWG# S&W 12179-CI-SWP-18 SHT.1 of 6

SWP-2
Failure Mode: Cracking - Vibration Fatigue & Erosion FP 2.6E-08
30" Elbow to Pipe Weld.
Not in ISI Program. (SWP-030-2-RBI-1-3) VT-2, VT-3, UT
Exam volume to extend 1" on each side of weld. Internal VT-3 of ARCOR coating to be performed at time of check valve dissassembly once per interval.
Piping shown in Zone 181 Class 3 C/S/CUNi Clad
DWG# S&W 12179-CI-SWP-18 SHT.1 of 6

SWP-3
Failure Mode: Cracking - Vibration Fatigue & Erosion FP 2.6E-08
30" Elbow to Pipe Weld.
Not in ISI Program. (SWP-030-415-RBI-1-3) VT-2, VT-3, UT
Exam volume to extend 1" on each side of weld. Internal VT-3 of ARCOR coating to be performed at time of check valve dissassembly once per interval.
Piping shown in Zone 182 Class 3 C/S/CUNi Clad
DWG# S&W 12179-CI-SWP-19 SHT.1 of 6

SWP-4
Failure Mode: Cracking - Vibration Fatigue & Erosion FP 2.6E-08
30" Pipe to Chek Valve Flange Weld.
Not in ISI Program. (SWP-030-18-RBI-1-3) VT-2, VT-3, UT
Exam volume to extend 1" on pipe side of weld. Internal VT-3 of ARCOR coating to be performed at time of check valve dissassembly once per interval.
Piping shown in Zone 182 Class 3 C/S/CUNi Clad
DWG# S&W 12179-CI-SWP-19 SHT.1 of 6

SWP-5
Failure Mode: Cracking - External Loads & Erosion FP 1.0E-08*
8" Pipe Branch Tee With a Reinforcing Collar Welded to the 8" & 10" Pipe.
Not in ISI Program. (SWP-008-74-RBI-1-3) VT-2, PT
Not in ISI Program. (SWP-008-74-RBI-2-3) VT-2, PT
Exam area includes the inside collar weld RBI-1 and the outside collar weld RBI-2.
Piping shown in Zone 178 Class 3 CUNi
DWG# S&W 12179-CI-SWP-33 SHT.2 of 8

SWP-6
Failure Mode: Cracking - External Loads & Erosion FP 1.0E-08*
6" Pipe to Pipe Bimetallic Weld.
Not in ISI Program. (SWP-006-48-RBI-1-3) VT-2, UT
Exam volume to include 2" of base metal on the CUNi side of the weld for wall thinning.

Piping shown in Zone 175 Class 3 CUNi to Monel
DWG# S&W 12179-CI-SWP-30S SHT.1 of 6

SWP-7
Failure Mode: Cracking - External Loads & Erosion FP 1.0E-08*
8" Pipe Branch Tee With a Reinforcing Collar Welded to the 8" & 10" Pipe.
Not in ISI Program. (SWP-008-73-RBI-1-3) VT-2, PT
Not in ISI Program. (SWP-008-73-RBI-2-3) VT-2, PT
Exam area includes the inside collar weld RBI-1 and the outside collar weld RBI-2.
Piping shown in Zone 179 Class 3 CUNi
DWG# S&W 12179-CI-SWP-33 SHT.1 of 8

SWP-8
Failure Mode: Cracking - External Loads & Erosion FP 1.0E-08*
6" Pipe to Pipe Bimetallic Weld.
Not in ISI Program. (SWP-006-33-RBI-1-3) VT-2, UT
Exam volume to include 2" of base metal on the CUNi side of the weld for wall thinning.
Piping shown in Zone 175 Class 3 CUNi to Monel
DWG# S&W 12179-CI-SWP-30S SHT.2 of 6

SWP-15
Failure Mode: Cracking - External Loads & Erosion FP 1.0E-08*
1.5" Elbow to Pipe Brazed Joint. Class 3 CUNi
Not in ISI Program. (SWP-150-106-RBI-1-3) VT-2, UT
Exam volume to cover under bell of fitting from pipe side with special UT. Leak concern due to proximity of pump and failure probability for leak w/o ISI is 7.4E-06.
DWG# S&W 12179-CP-319716 SHT.1 of 4

SWP-22
Failure Mode: Erosion FP 1.0E-08*
1.5" Elbow to Pipe Brazed Joint. Class 3 CUNi
Not in ISI Program. (SWP-150-104-RBI-1-3) VT-2, UT
Exam volume to cover under bell of fitting from pipe side with special UT. Leak concern due to proximity of pump and failure probability for leak w/o ISI is 7.4E-06.
DWG# S&W 12179-CP-319735 SHT.1 of 3

SWP-23
Failure Mode: Cracking - Thermal Fatigue & Erosion FP 1.0E-08*
24" Welded Pipe Branch Tee With a Reinforcing Saddle Welded to the 24" & 30" Pipe.
Not in ISI Program. (SWP-024-91-RBI-1-3) VT-2, VT-3, UT
Exam volume to extend 1" from the inside saddle weld on the 24" pipe. This weld will have a UT exam for ID cracking from the 24" pipe side of the weld. Internal VT-3 of ARCOR coating to be performed once per interval.
Piping shown in Zone 183 Class 3 C/S/CUNi Clad
DWG# S&W 12179-CI-SWP-23 SHT.1 of 7

SWP-25
Failure Mode: Cracking - Thermal Fatigue & Erosion FP 1.0E-08*
24" Welded Pipe Branch Tee to 30" Pipe.
Not in ISI Program. (SWP-024-93-RBI-1-3) VT-2, VT-3, UT
Exam volume to extend 1" on each side of weld. Internal VT-3 of ARCOR coating to be performed once per interval.
Piping shown in Zone 183 Class 3 C/S/CUNi Clad
DWG# S&W 12179-CI-SWP-23 SHT.1 of 7

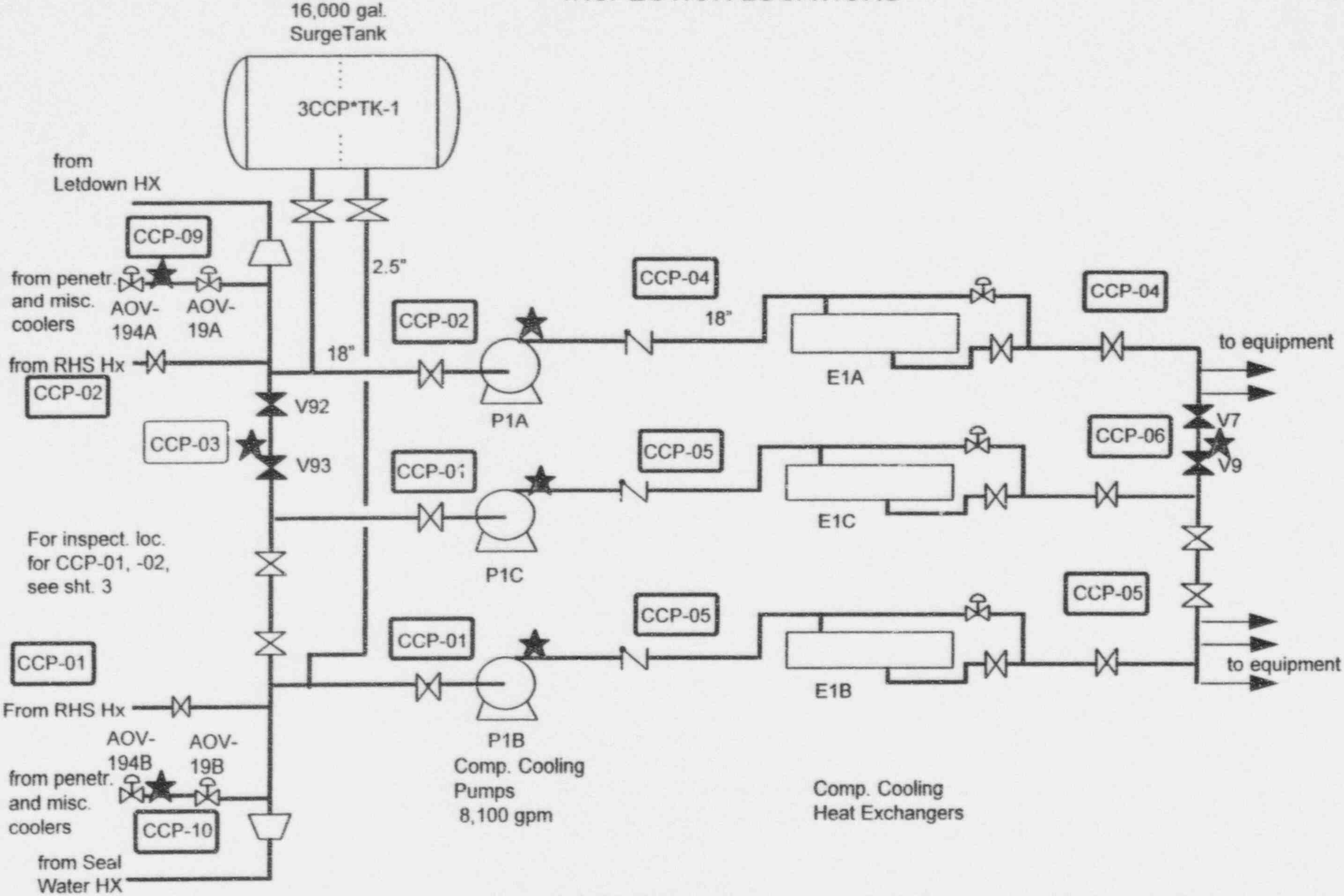
SWP-26
Failure Mode: Cracking - Vibration Fatigue & Erosion FP 2.6E-08
No piping exists in this segment only equipment that is flanged and bolted together.
Service Water Pump P1D to Check Valve V1.
(SWP-030-D-26-3) VT-2
DWG# P&ID 25212-26933 SH.1 of 4 or S&W 12179-EM-133A-16

SWP-27
Failure Mode: Cracking - Vibration Fatigue & Erosion FP 2.6E-08
No piping exists in this segment only equipment that is flanged and bolted together.
Service Water Pump P1B to Check Valve V3.
(SWP-030-B-27-3) VT-2
DWG# P&ID 25212-26933 SH.1 of 4 or S&W 12179-EM-133A-16

SWP-28
Failure Mode: Cracking - Vibration Fatigue & Erosion FP 2.6E-08
No piping exists in this segment only equipment that is flanged and bolted together.
Service Water Pump P1C to Check Valve V5.
(SWP-030-C-28-3) VT-2
DWG# P&ID 25212-26933 SH.1 of 4 or S&W 12179-EM-133A-16

SWP-29
Failure Mode: Cracking - Vibration Fatigue & Erosion FP 2.6E-08
No piping exists in this segment only equipment that is flanged and bolted together.
Service Water Pump P1A to Check Valve V7.
(SWP-030-A-29-3) VT-2
DWG# P&ID 25212-26933 SH.1 of 4 or S&W 12179-EM-133A-16

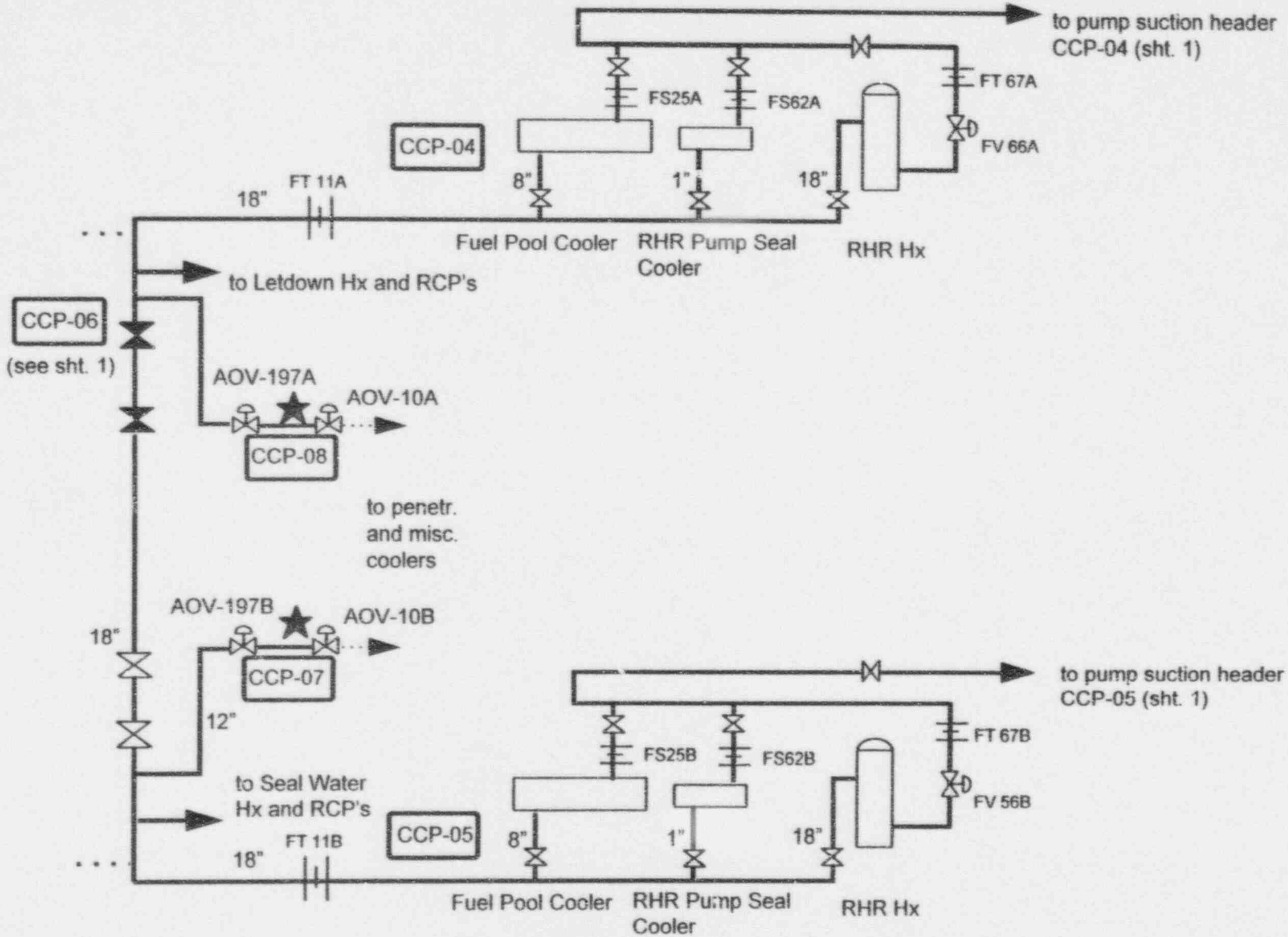
REACTOR PLANT COMPONENT COOLING : CCP-1
INSPECTION LOCATIONS



3/27/96

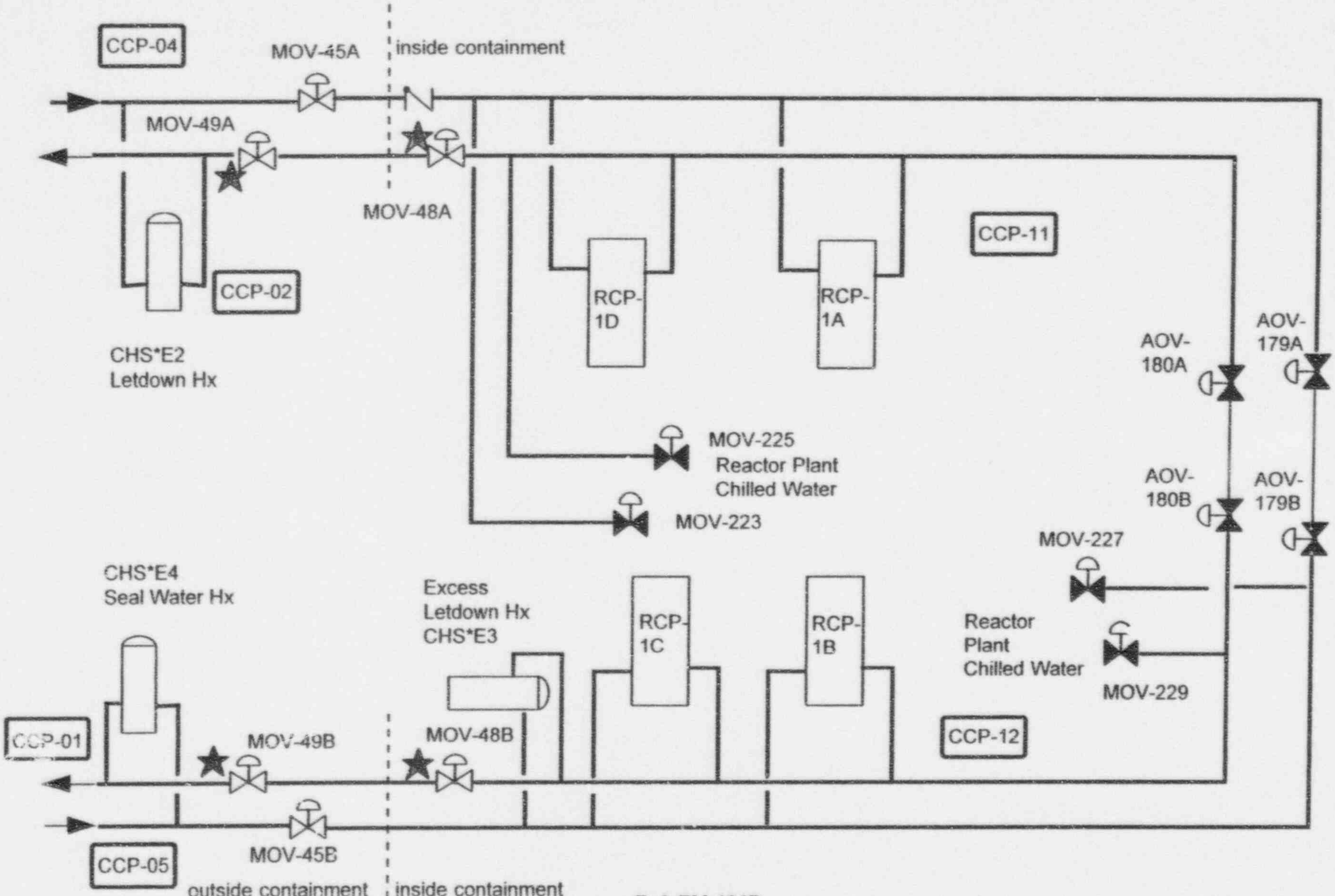
Ref. EM-121A

REACTOR PLANT COMPONENT COOLING : CCP-2
INSPECTION LOCATIONS



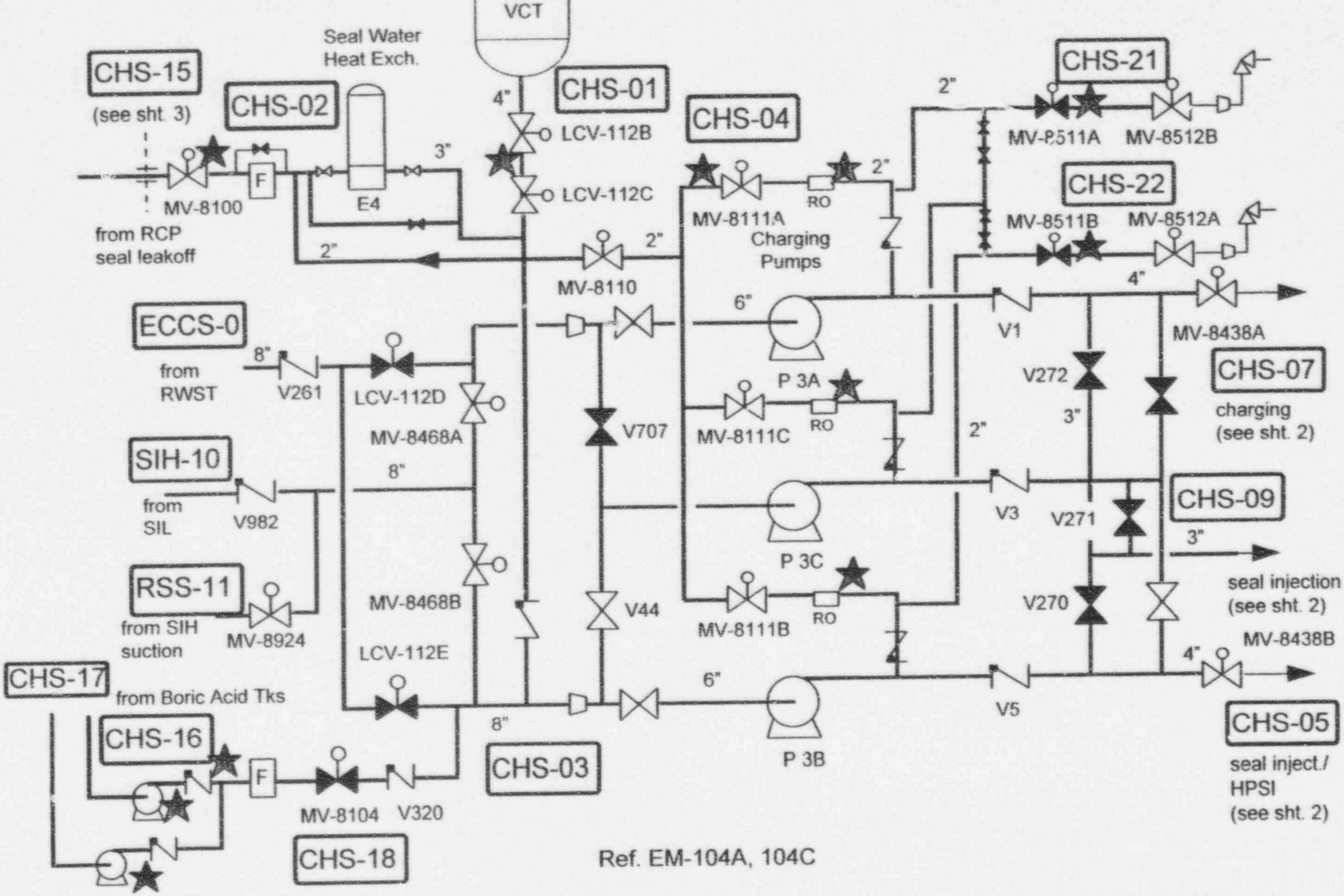
Ref. EM-121A

REACTOR PLANT COMPONENT COOLING : CCP-3
INSPECTION LOCATIONS



Ref. EM-121B

CHEMICAL AND VOLUME CONTROL : CHS-1
INSPECTION LOCATIONS

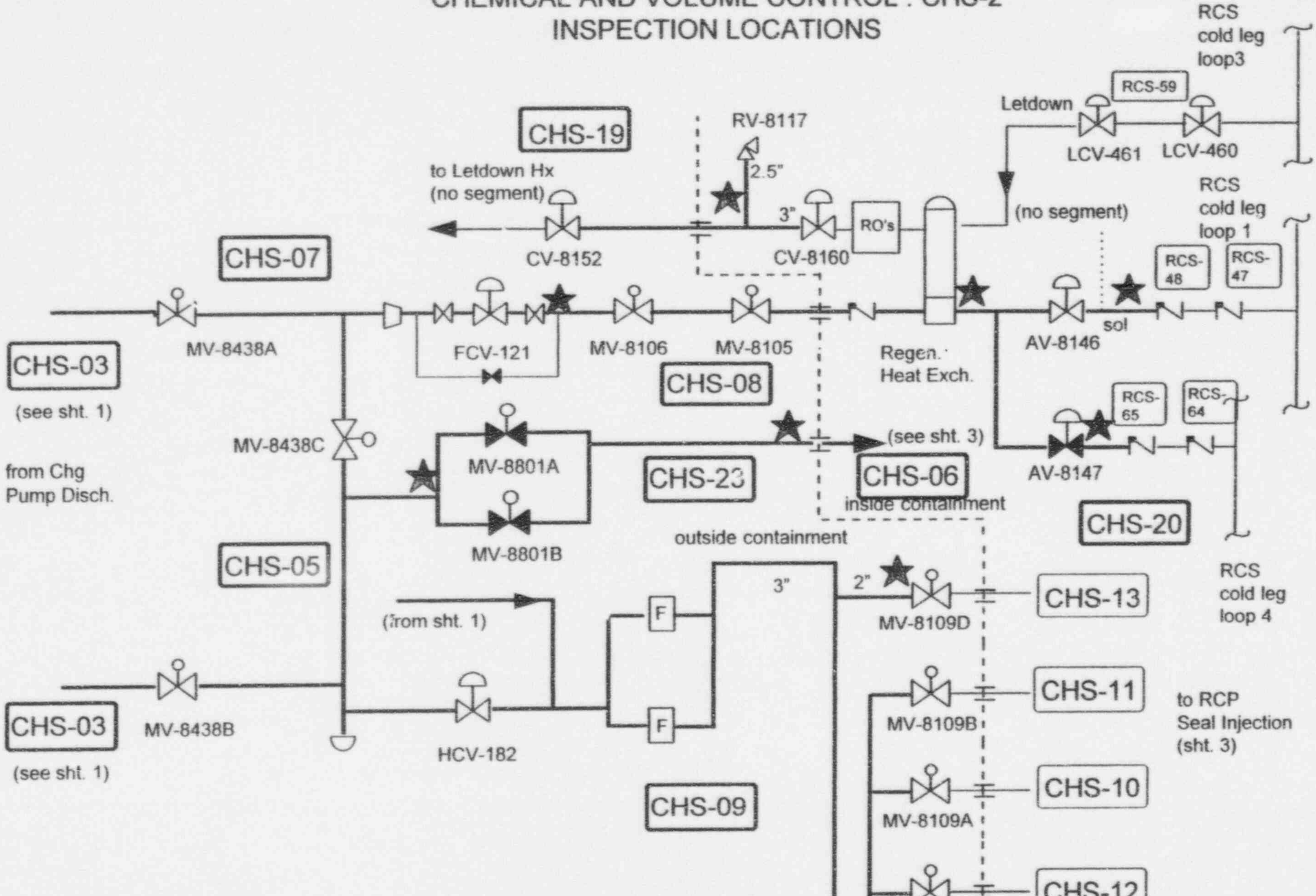


Ref. EM-104A, 104C

3/27/96

RBI P&ID - 21

CHEMICAL AND VOLUME CONTROL : CHS-2
INSPECTION LOCATIONS

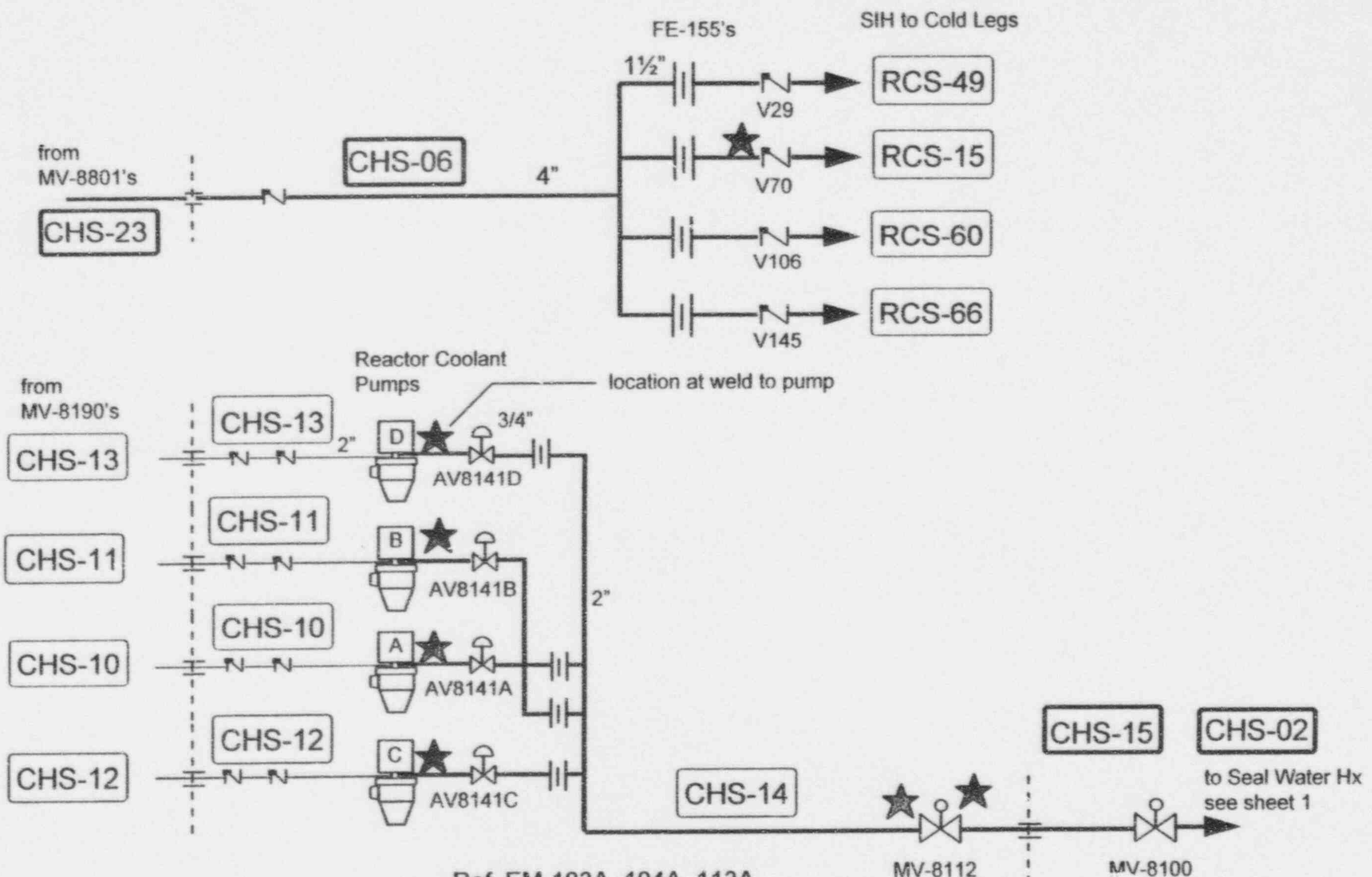


Ref. EM-104A, 104C, 104D

3/27/96

RBI P&ID - 22

CHEMICAL AND VOLUME CONTROL : CHS-3
INSPECTION LOCATIONS

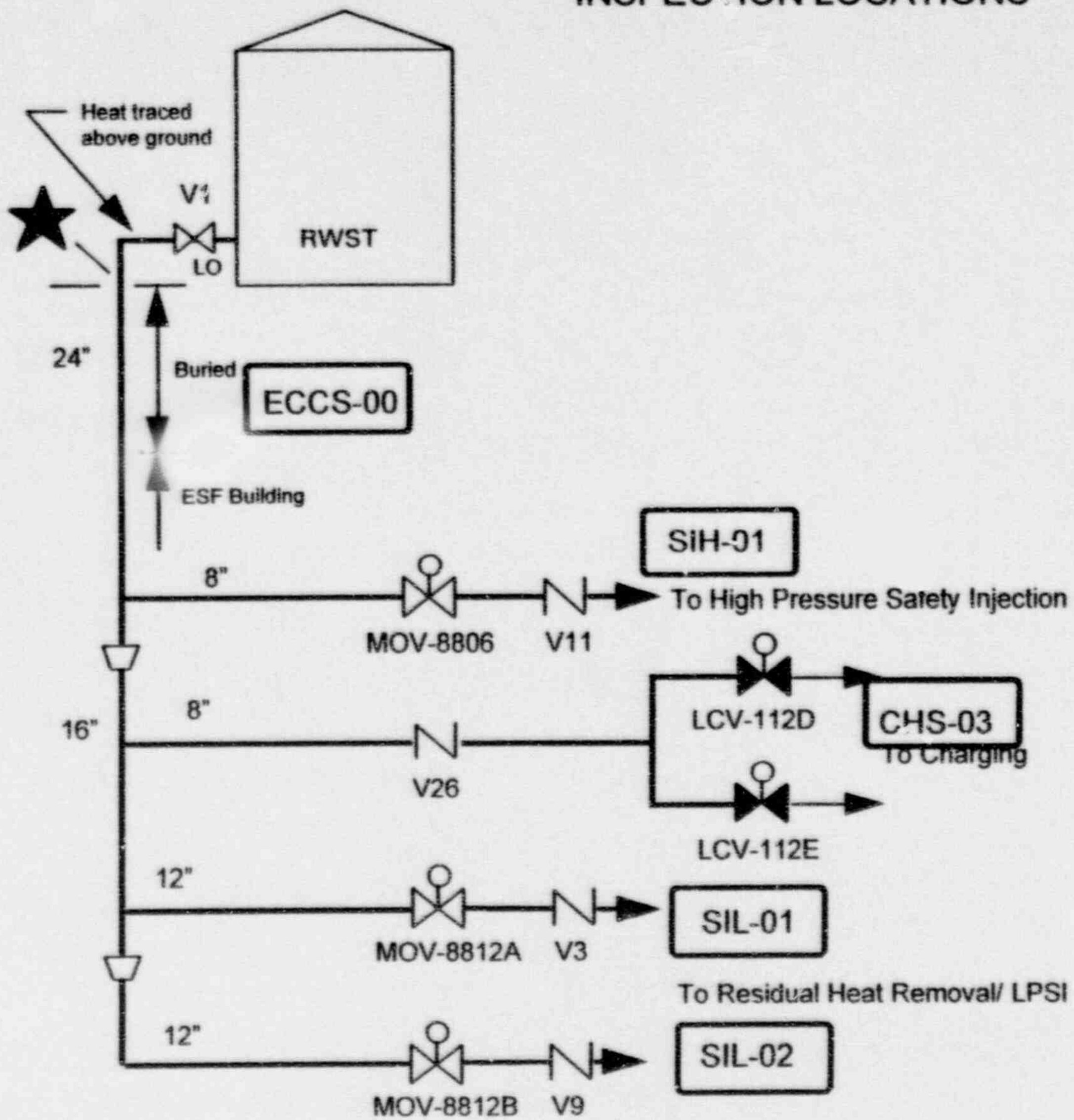


Ref. EM-103A, 104A, 113A

3/27/96

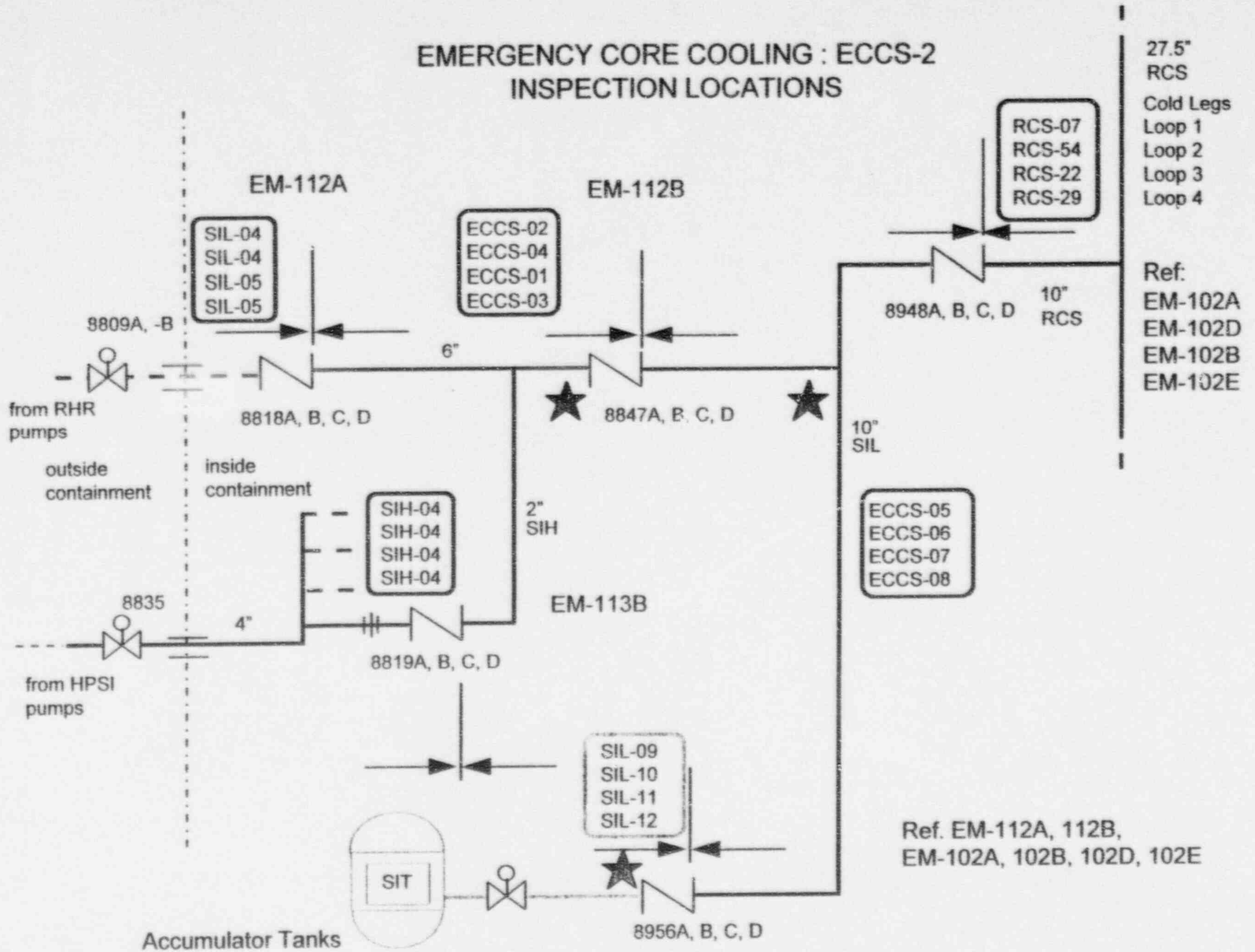
RBI P&ID - 23

EMERGENCY CORE COOLING : ECCS-1
INSPECTION LOCATIONS



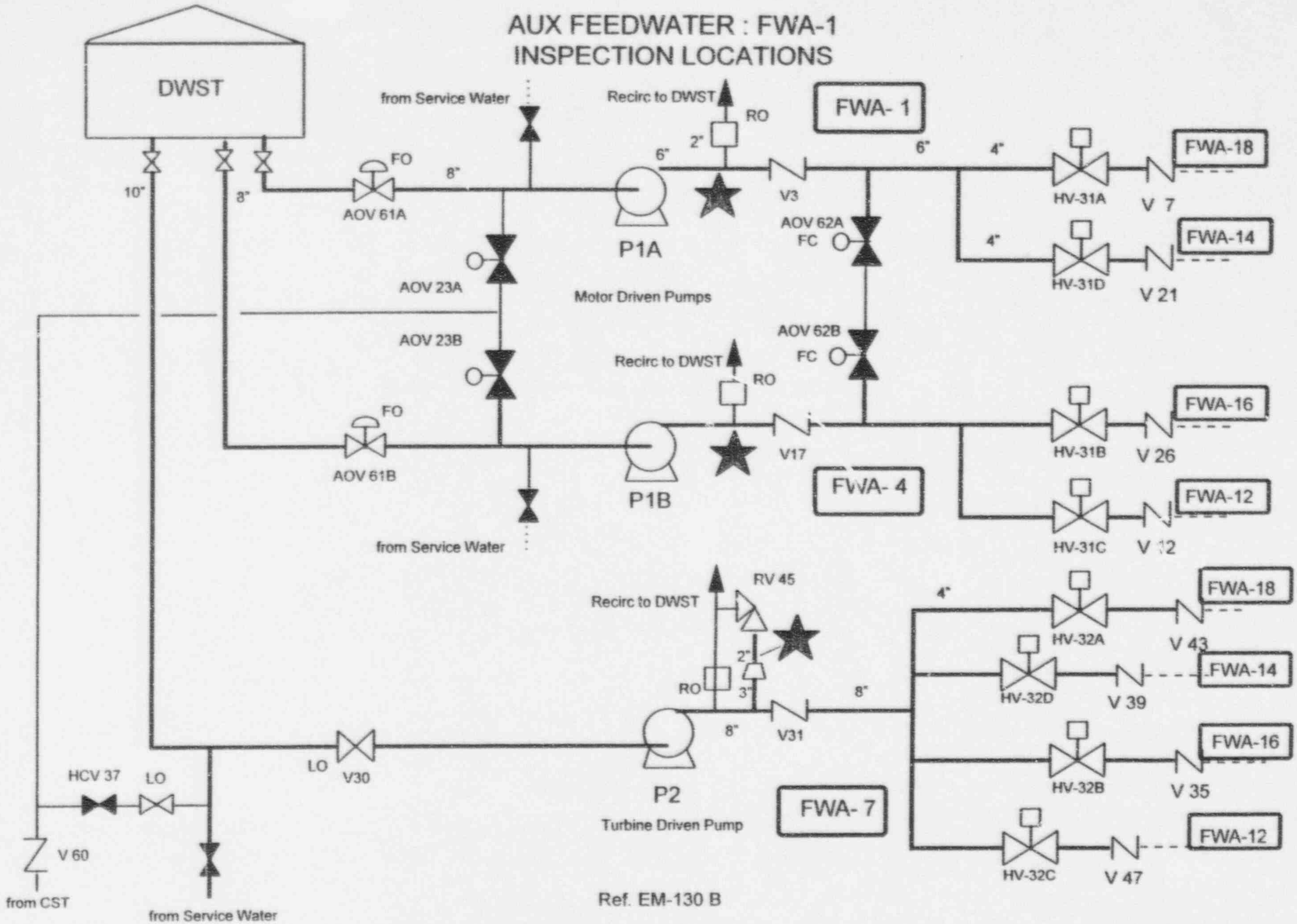
Ref. EM-112A, 113A, 113B, 104A

EMERGENCY CORE COOLING : ECCS-2
INSPECTION LOCATIONS



Ref. EM-112A, 112B,
EM-102A, 102B, 102D, 102E

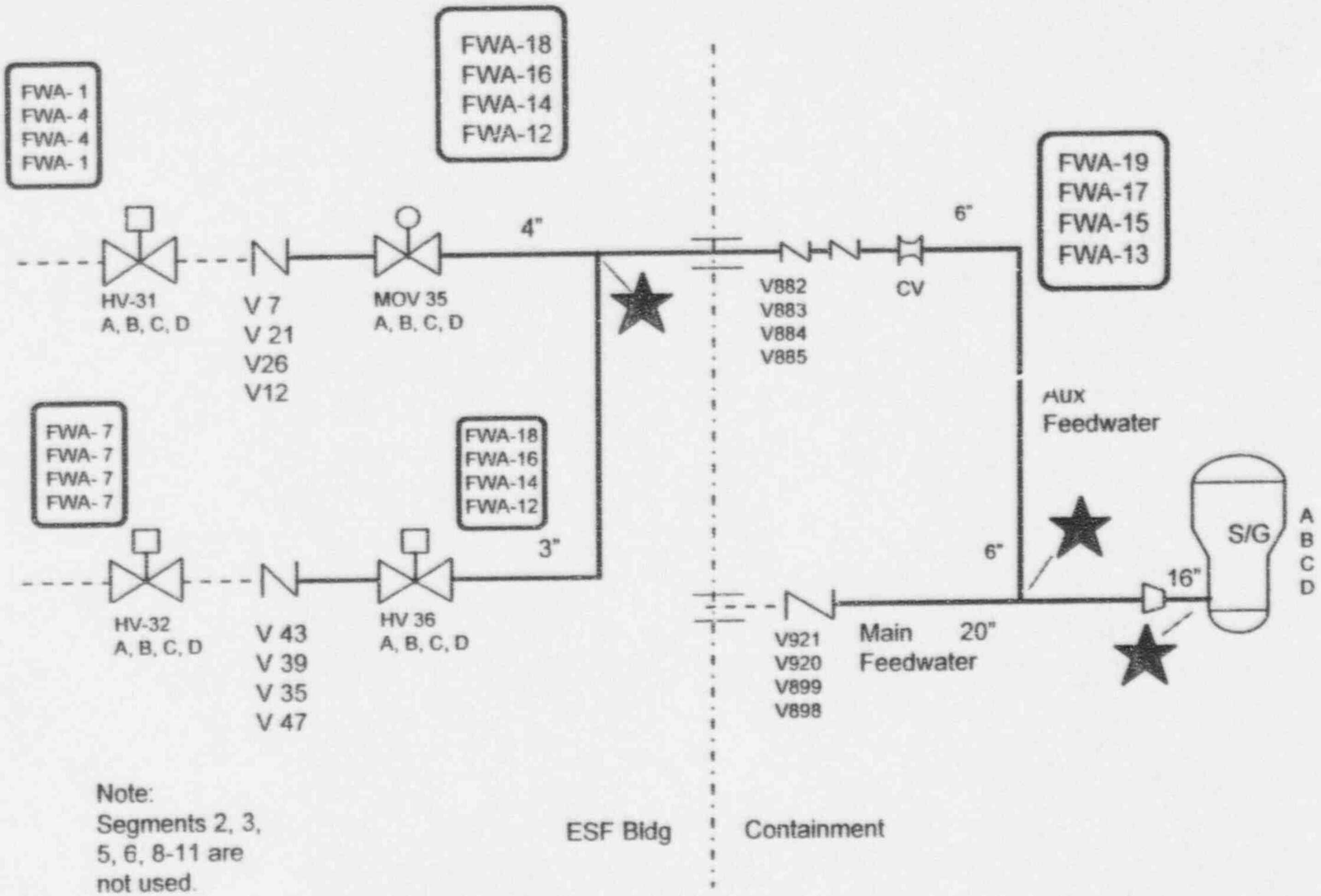
AUX FEEDWATER : FWA-1
INSPECTION LOCATIONS



Ref. EM-130 B

3/27/96

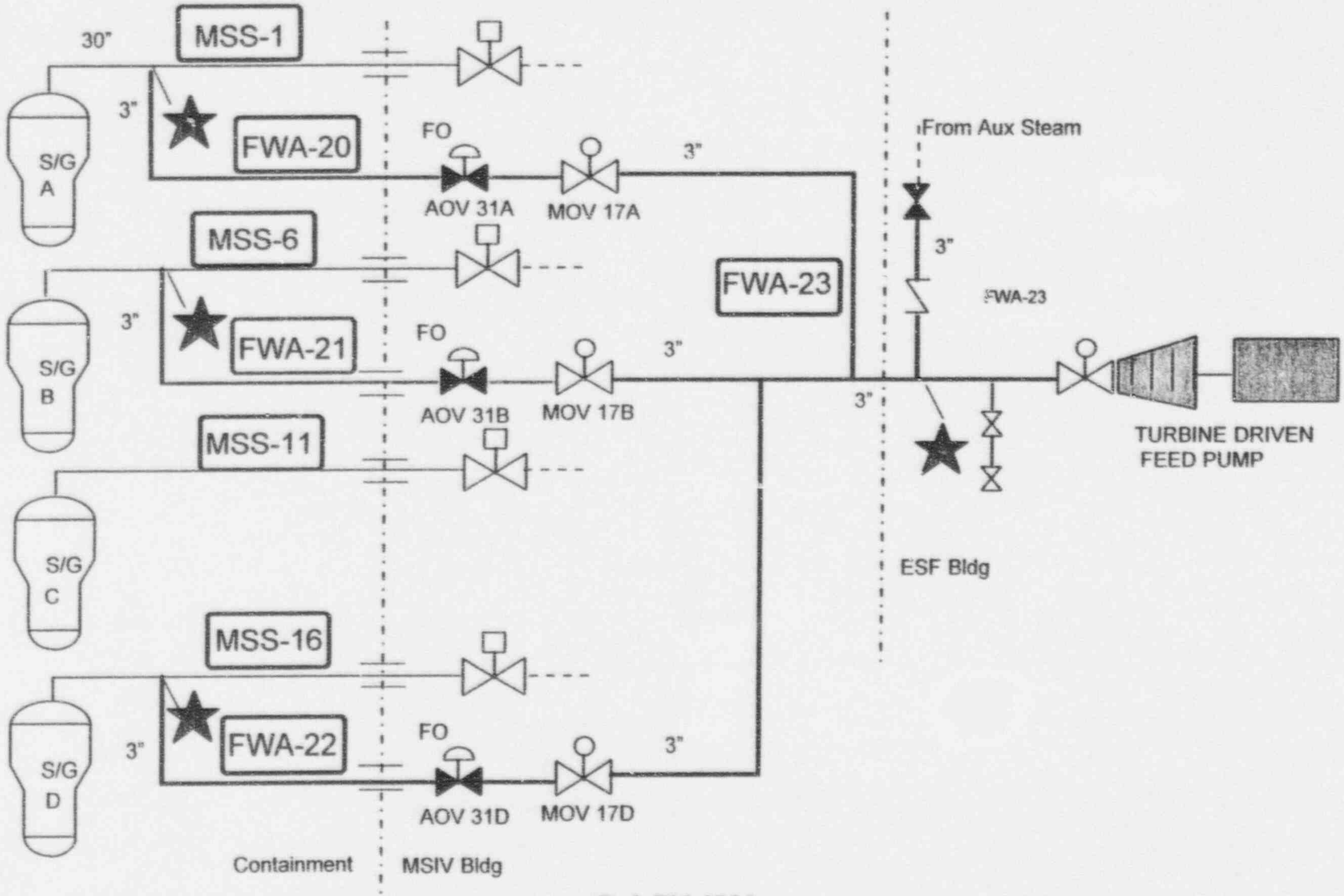
AUX FEEDWATER : FWA-2
INSPECTION LOCATIONS



Note:
Segments 2, 3,
5, 6, 8-11 are
not used.

Ref. EM-130 A, B, C, D

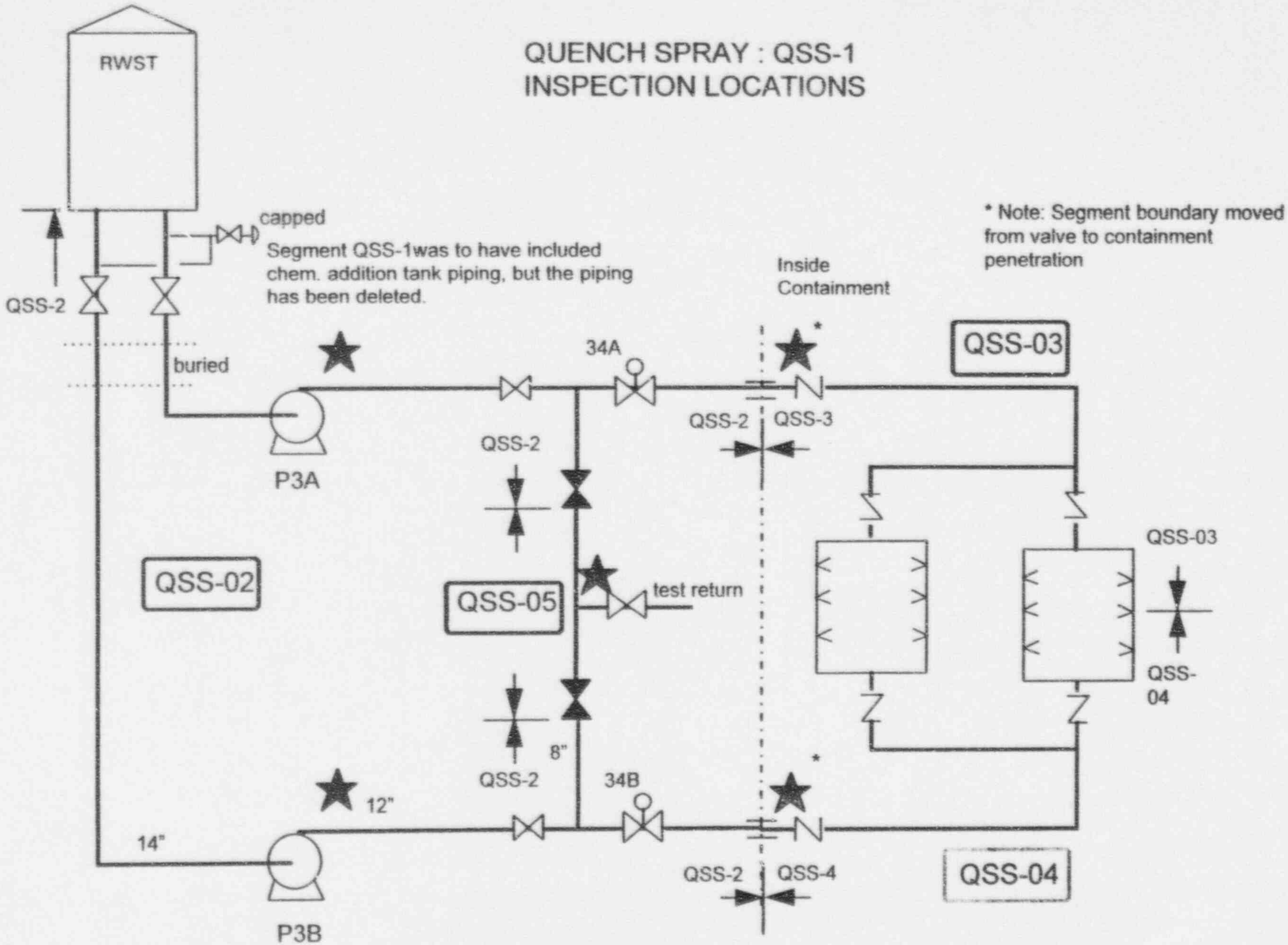
AUX FEEDWATER : FWA-3
INSPECTION LOCATIONS



Ref. EM-123A

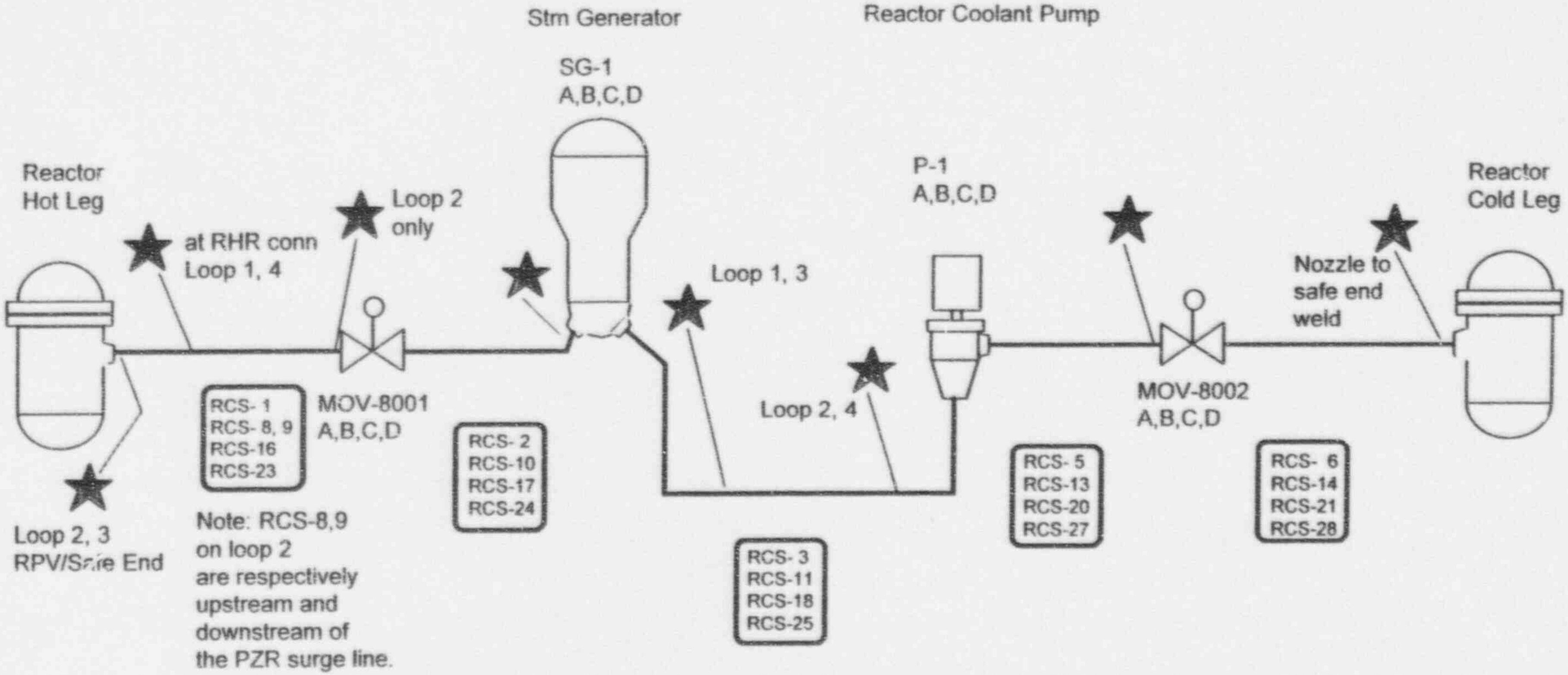
3/27/96

QUENCH SPRAY : QSS-1
INSPECTION LOCATIONS



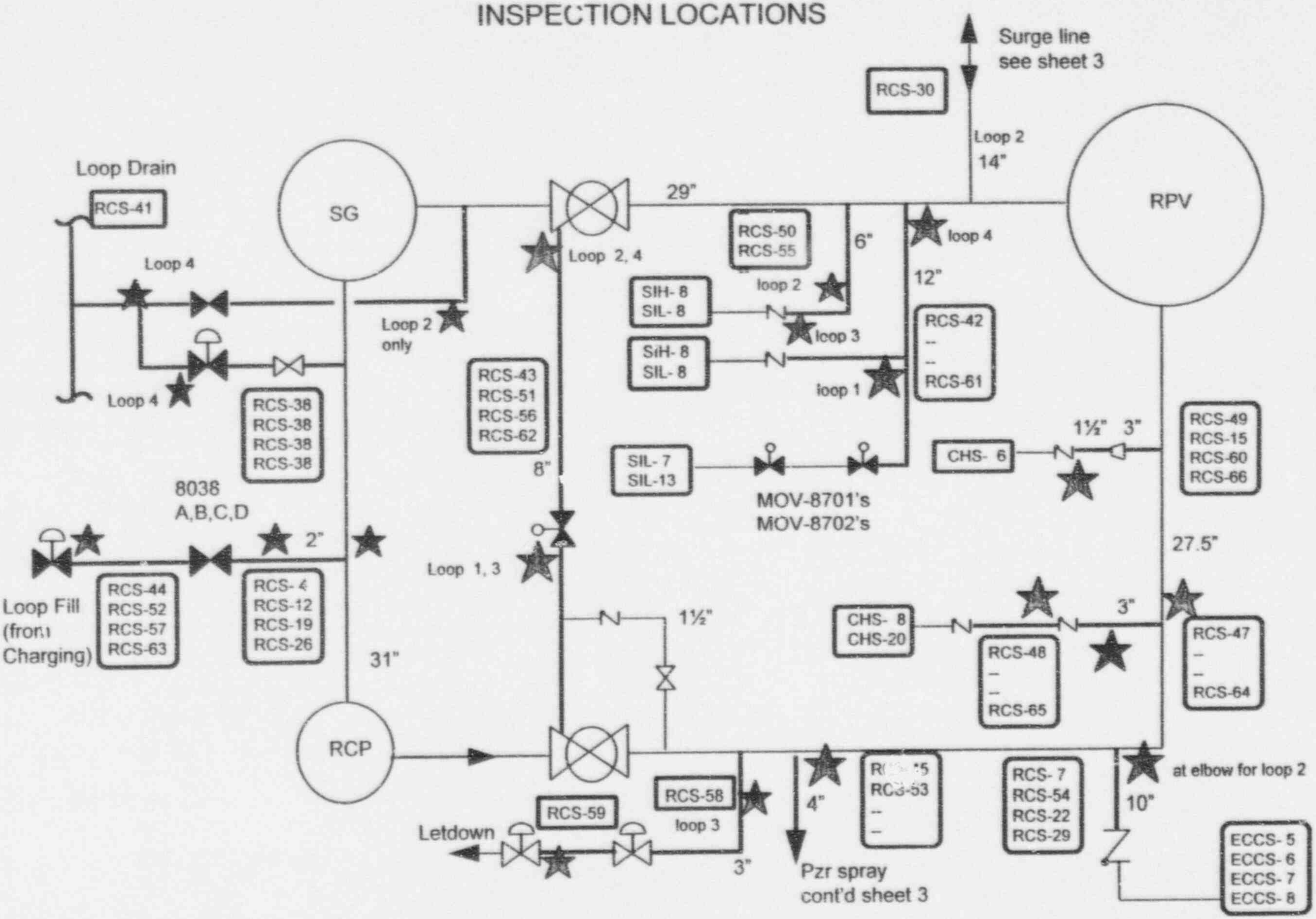
Quench Spray System Ref. EM-115A

REACTOR COOLANT SYSTEM : RCS-1
INSPECTION LOCATIONS



Ref. EM-102A, 102B, 102D, 102E

REACTOR COOLANT SYSTEM : RCS-2
INSPECTION LOCATIONS



Ref. EM-102A, 102B, 102E, 102F

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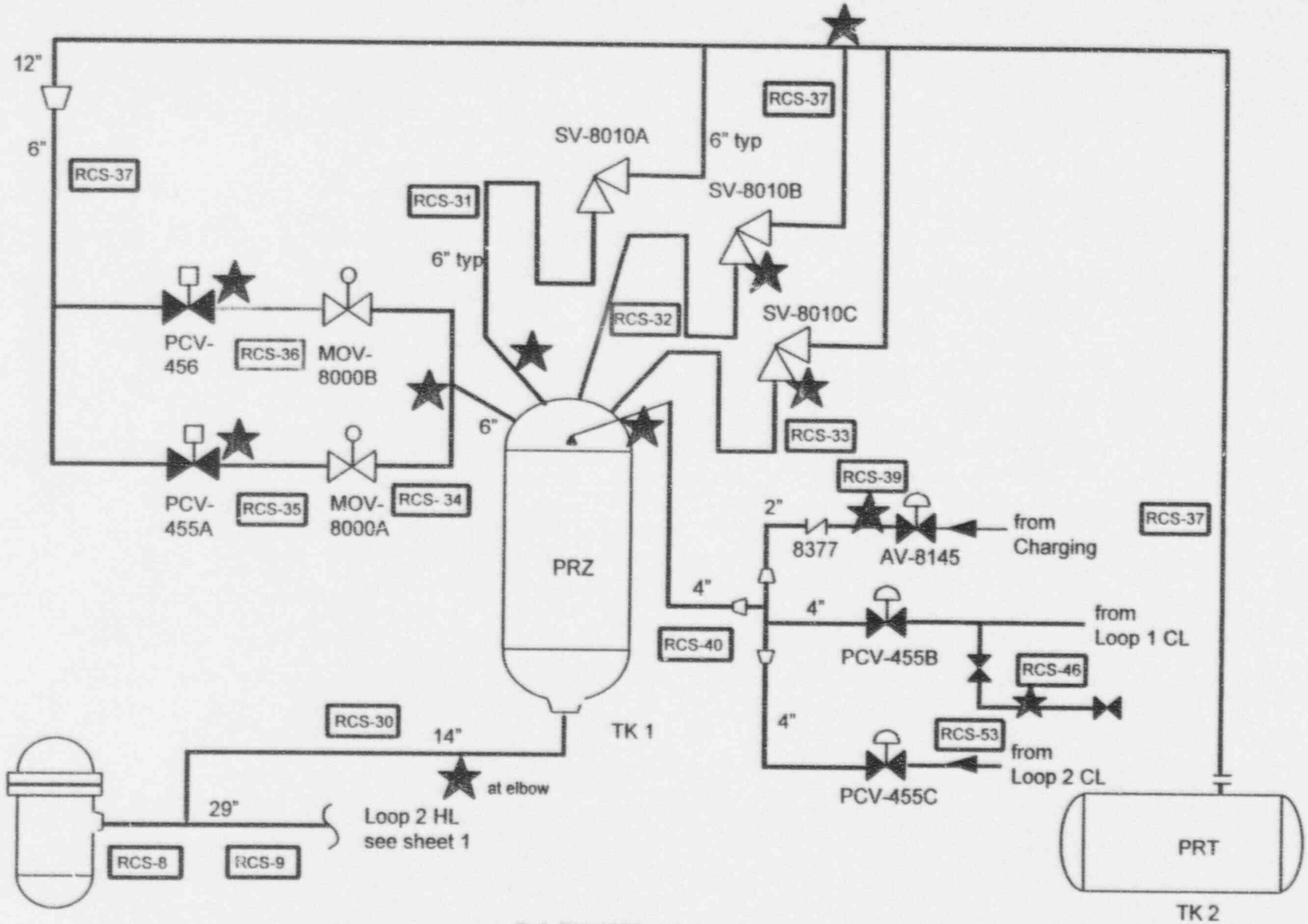
RBI P&ID - 25

REACTOR COOLANT SYSTEM : RCS-3
INSPECTION LOCATIONS



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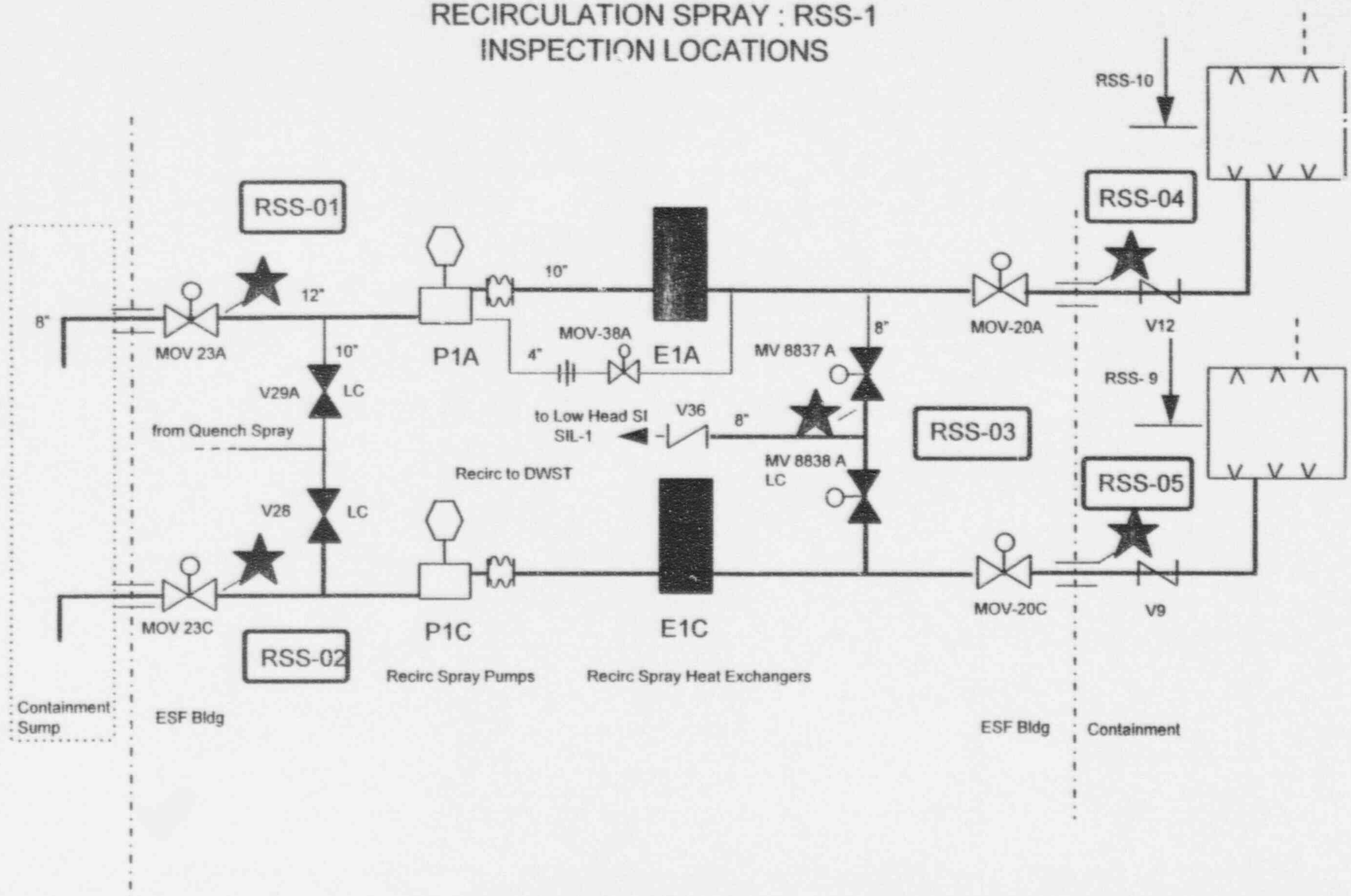
Ref. EM-102C

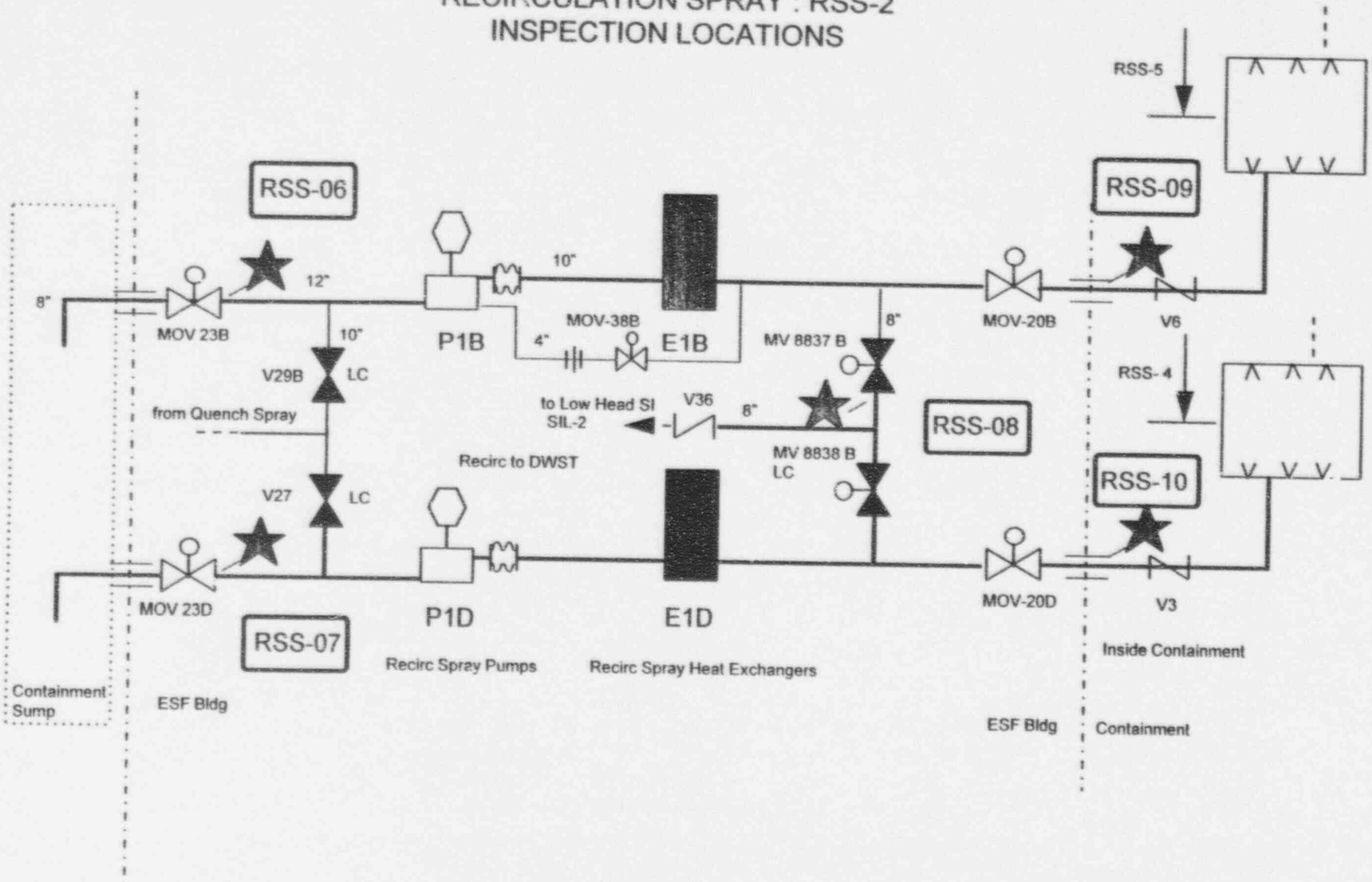
RBI P&ID - 26

RECIRCULATION SPRAY : RSS-1
INSPECTION LOCATIONS



Ref. EM-112C

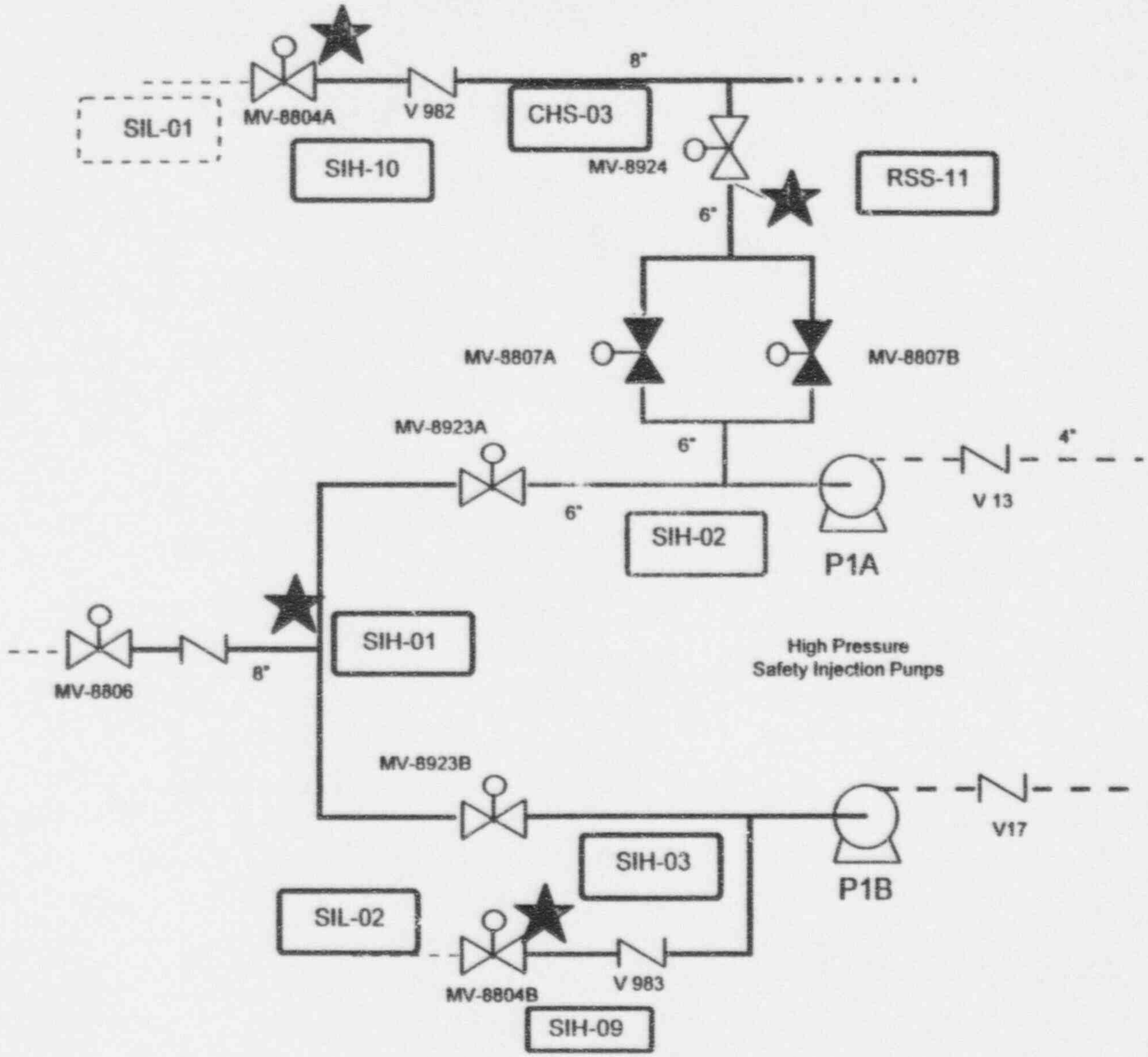
RECIRCULATION SPRAY : RSS-2
INSPECTION LOCATIONS



Ref. EM-112C

3/27/96

RECIRCULATION SPRAY : RSS-3
INSPECTION LOCATIONS



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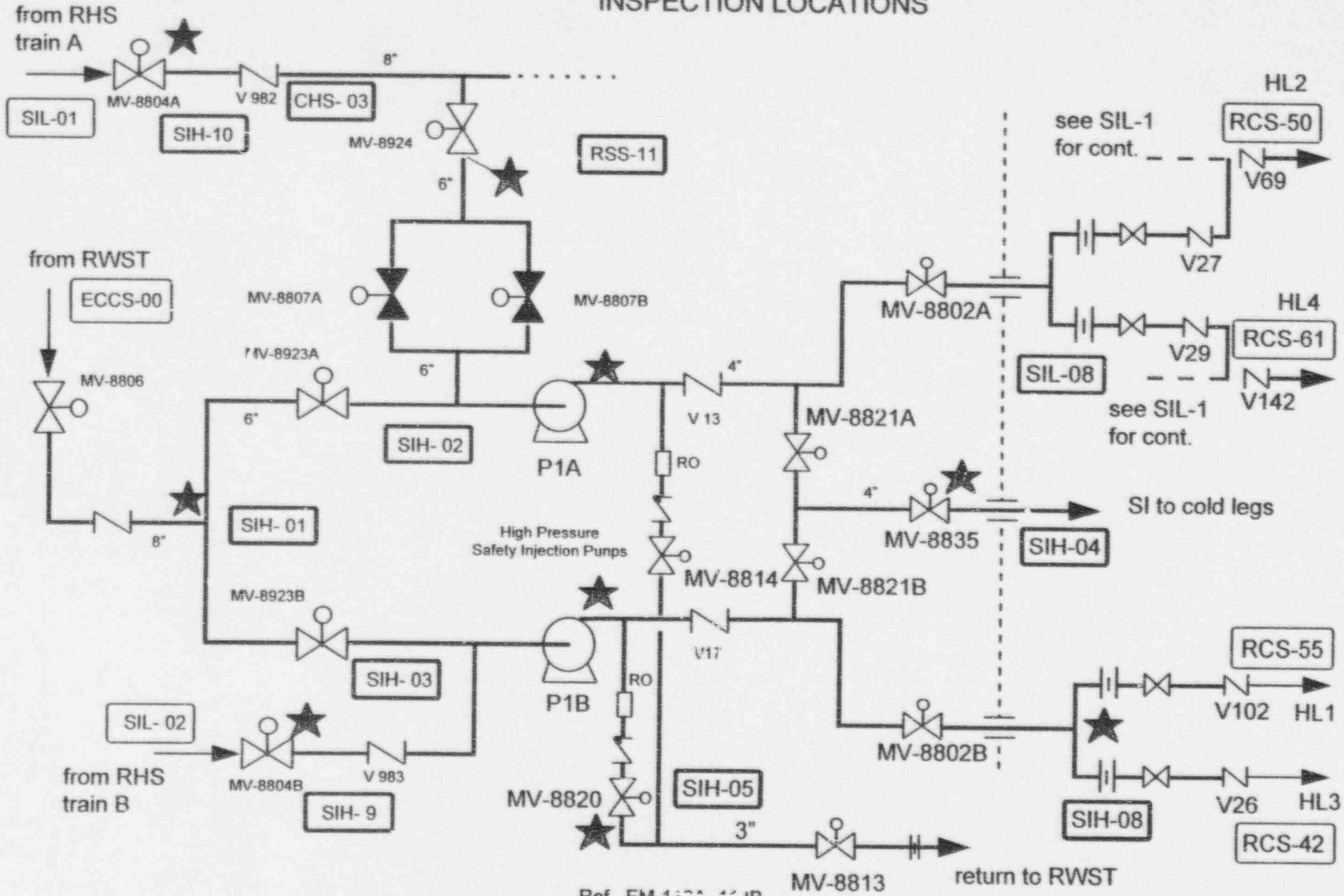
Ref. EM-113B

HIGH PRESSURE SAFETY INJECTION SYSTEM : SIH-1
INSPECTION LOCATIONS



Ref. EM-112A, 113B

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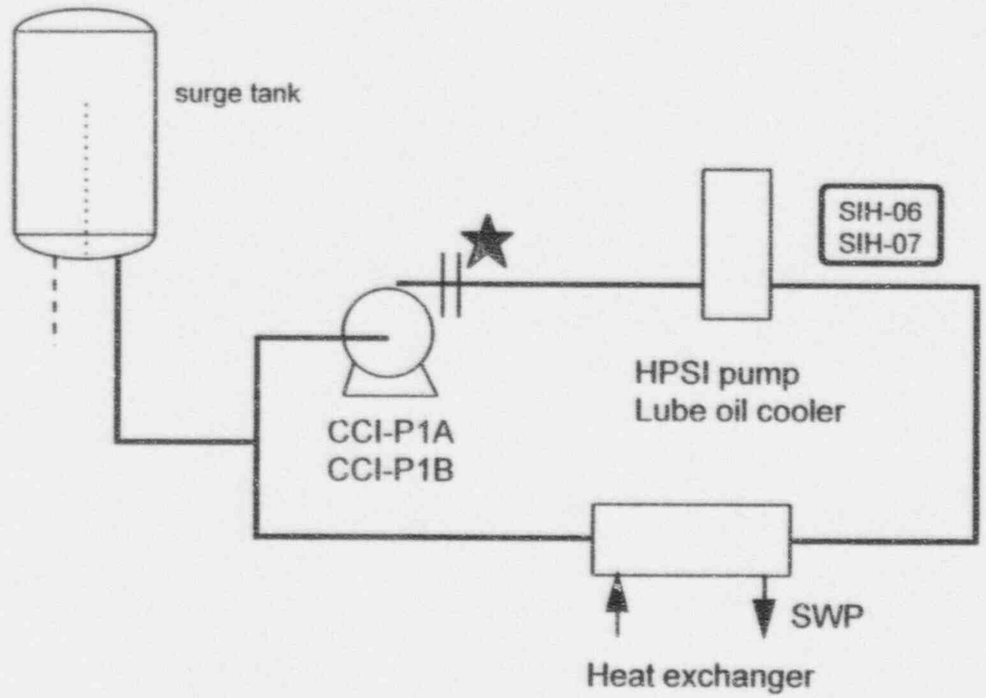
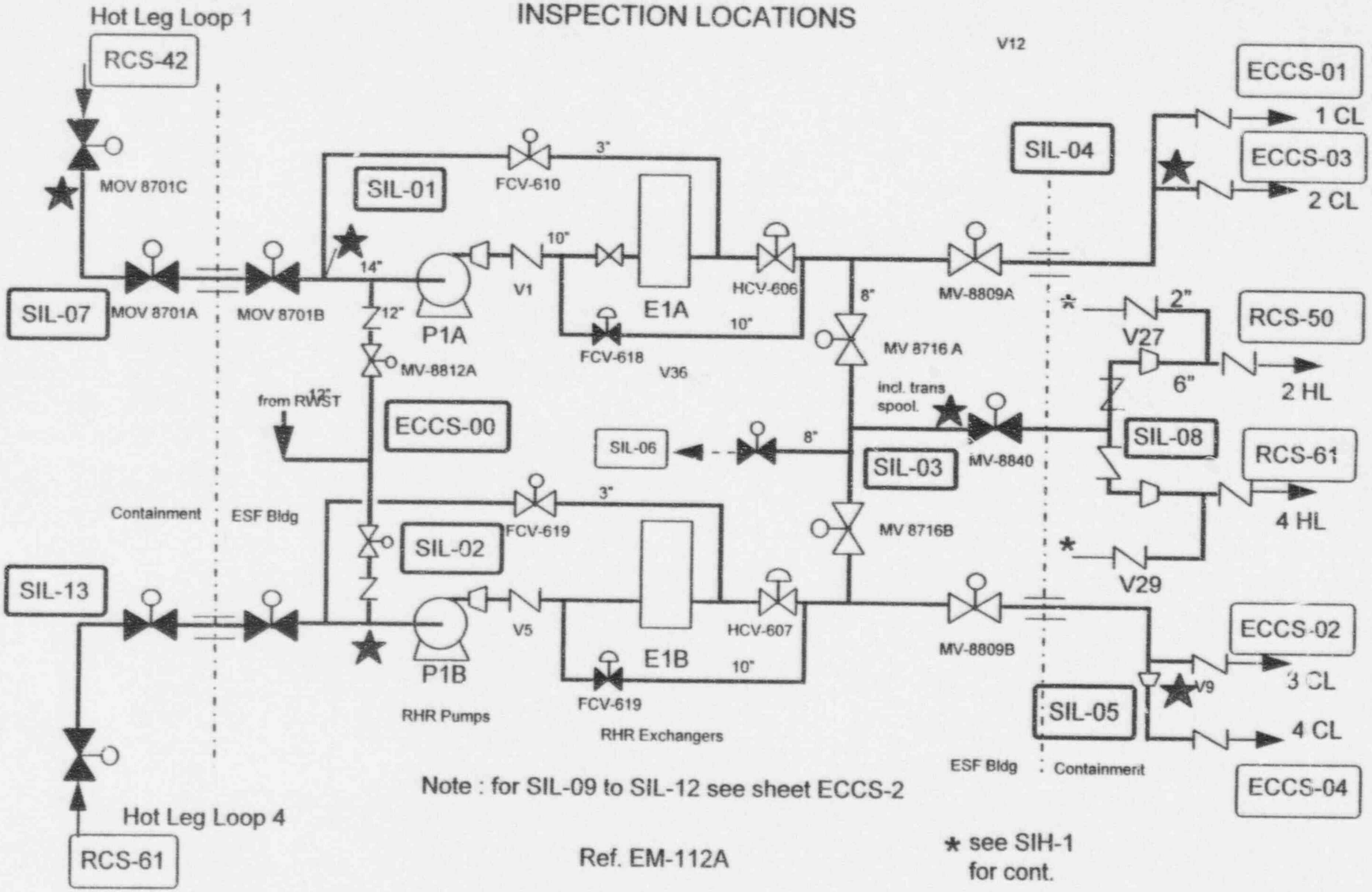
RBI P&ID - 19

HIGH PRESSURE SAFETY INJECTION SYSTEM : SIH-2
INSPECTION LOCATIONS
(CCI SUBSYSTEM)



LOW PRESSURE SAFETY INJECTION : SIL-1
INSPECTION LOCATIONS



Note : for SIL-09 to SIL-12 see sheet ECCS-2

Ref. EM-112A

* see SIH-1 for cont.

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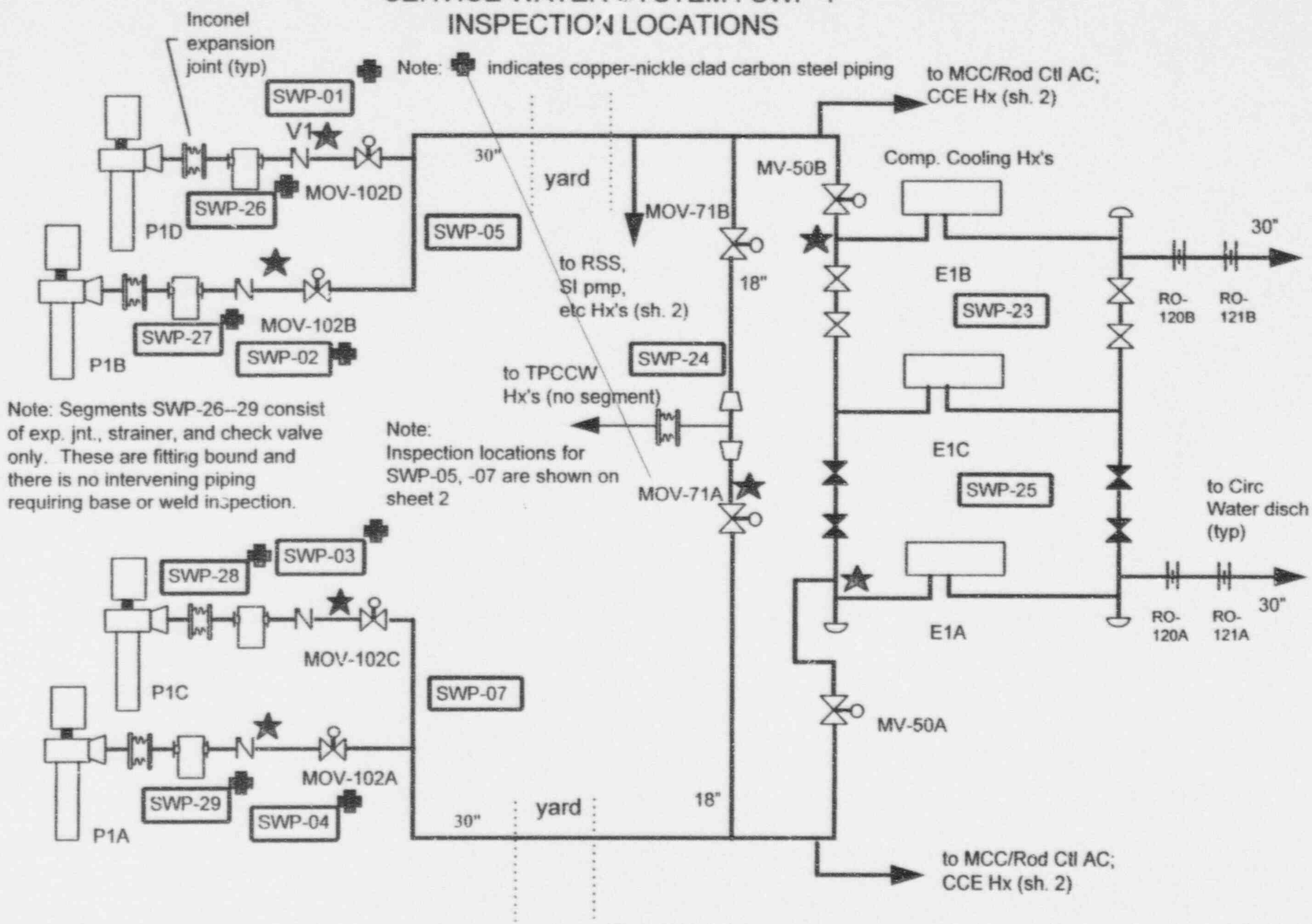
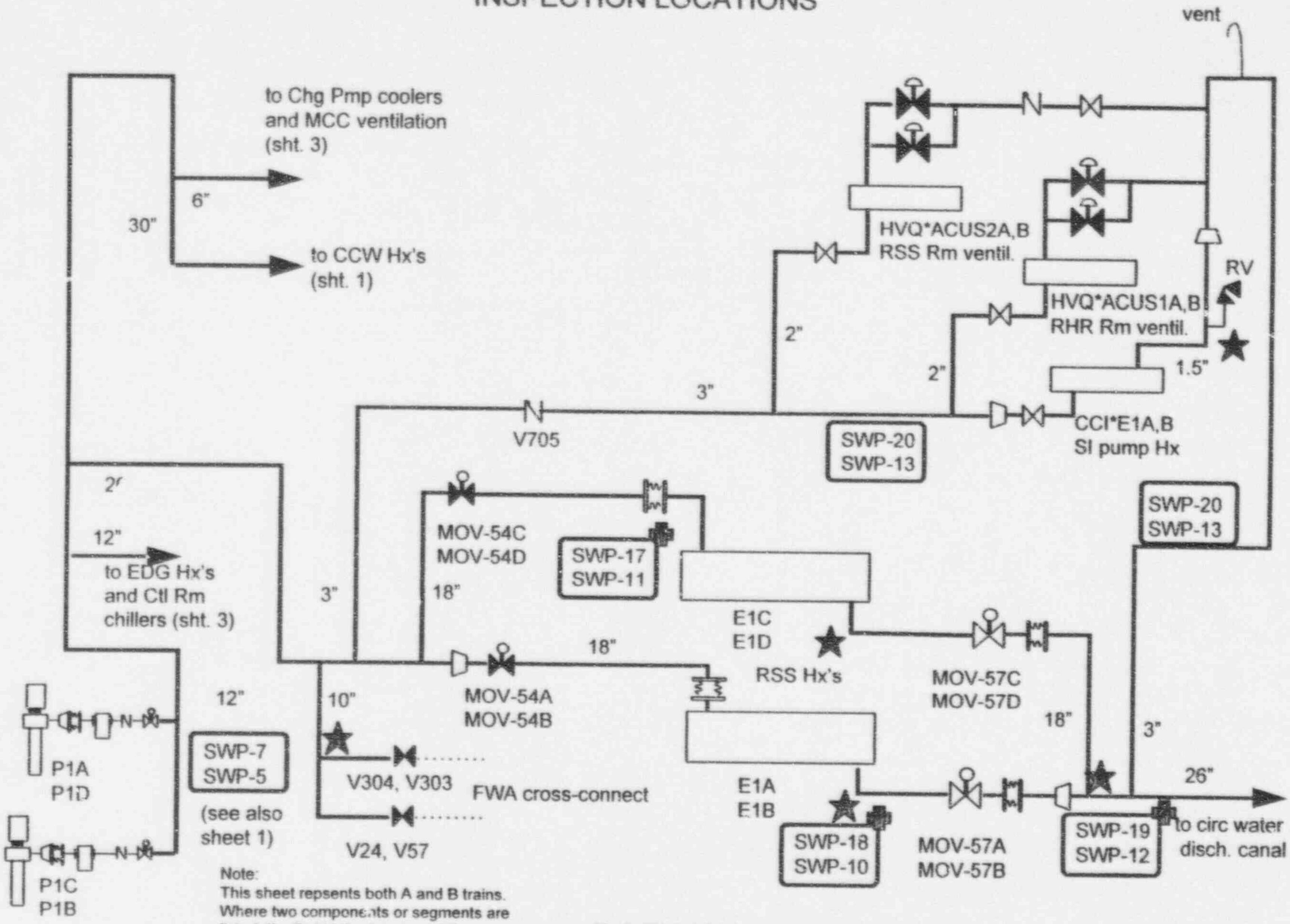
SERVICE WATER SYSTEM : SWP-1
INSPECTION LOCATIONS



Ref. EM-133A

3/27/96

SERVICE WATER SYSTEM : SWP-2
INSPECTION LOCATIONS



Note:
This sheet repsents both A and B trains. Where two components or segments are listed, the first is the A train and the second is the B train.

Ref. EM-133A, 133B

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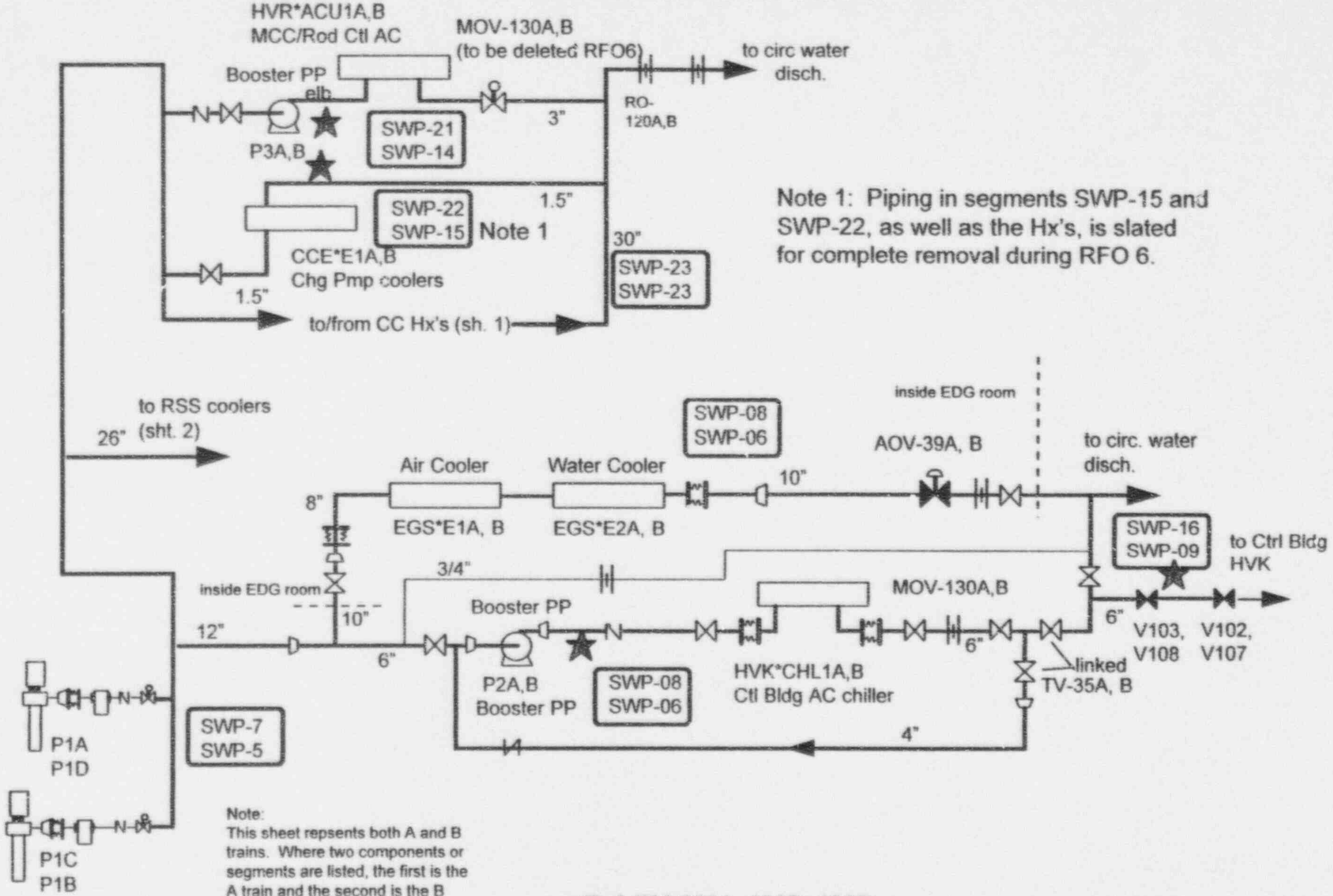
SERVICE WATER SYSTEM : SWP-3
INSPECTION LOCATIONS



Note 1: Piping in segments SWP-15 and SWP-22, as well as the Hx's, is slated for complete removal during RFO 6.

Note:
This sheet repsents both A and B trains. Where two components or segments are listed, the first is the A train and the second is the B train.

Ref. EM-133A, 133B, 133D

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ENCLOSURE 2

MILLSTONE UNIT 3

CHECKLIST FOR TECHNICAL CONSISTENCY IN PSA MODEL

(BASED ON EPRI PSA APPLICATIONS GUIDE APPENDIX B)

ISI PROGRAM PLAN

Checklist For Technical Consistency in a PSA Model

Appendix B of the EPRI document entitled "PSA Applications Guide" discusses several issues that have been found, in various PSAs, to be significant in determining the risk profile, but that have also either been neglected or treated superficially in the PSA models. The PSA report classifies the issues of concern in three major categories:

1. Issues related to whether the values of the PSA model parameters are within nominal ranges.

2. Issues concerned with whether the PSA model assumptions are justifiable.

3. Issues dealing with the proper documentation that support modeling decisions.

The following table provides a sanity check to confirm that the Millstone Unit 3 PSA model conforms to the industry state-of-the-art with respect to completeness of coverage of potential scenarios.

Issue [EPRI PSA Applications Guide, Appendix B]	Issues Description [EPRI PSA Applications Guide, Appendix B]	Related Millstone Unit 3 (MP3) PSA Model Assumptions
I. Initiating Events	1. If the Support Systems (such as CCW, SW, AC Power, DC Power, HVAC, and Instrument/Station Air) have not been identified as being significant, they should have screened out on the basis of one of the following reasons: 1.1 Not causing a reactor trip. 1.2 Not required for shutdown.	1. The following initiating events are considered in the Millstone Unit 3 PSA model: Loss of service water (train A or B), total loss of service water, loss of one vital DC Bus (A or B), total loss of vital DC Power, loss of vital AC Bus 1 or 2, loss of vital AC Bus 3 or 4.

Issue [EPRI PSA Applications Guide, Appendix B]	Issues Description [EPRI PSA Applications Guide, Appendix B]	Related Millstone Unit 3 (MP3) PSA Model Assumptions
	1.3 The frequencies of loss of these support systems (as initiating events) are low. A critical review is required if the frequency is < 10^{-2}/RY for a single train system, or 10^{-4}/RY for a redundant system.	The following initiating events are screened out (not modeled in MP3 PSA): Loss of Instrument Air, loss of Reactor Plant CCW, loss of Turbine Plant CCW, loss of charging pump and component cooling pump area ventilation.
	However, if such frequency is bounded in probability and consequence by another initiator, it could be screened out.	The reasons for screening out these support systems (special initiating events) are documented in MP3 Event Tree Analysis Calculation File # W3-517-1084-RE, Rev. 0, Pages 5 and 6.
	2. While Interfacing System LOCAs may have low frequencies, they are risk significant from the stand point of public risk irrespective to their frequencies of such sequences are assessed to be low, then these calculations must be well documented in the PSA.	2. In the Millstone Unit 3 PSA, Interfacing System LOCAs were assumed to lead directly to core melt and containment bypass, and, therefore, required no event tree analysis. The initiating event frequency of IS LOCAs is 2.21E-7/RY.

Issue [EPRI PSA Applications Guide, Appendix B]	Issues Description [EPRI PSA Applications Guide, Appendix B]	Related Millstone Unit 3 (MP3) PSA Model Assumptions
	3. SGTR sequences are also risk significant from the public risk stand point. If the frequencies of SGTR events are assessed to be low, then the supporting calculations have to be well documented.	3. The initiating event frequency of SGTR is 2.84E-2/RY and this event is considered a risk significant from a public risk stand point.
II. <u>Event Sequence Development</u>	1. Transient-Induced LOCAs such as a stuck-open PORV and RCP Seal LOCA after a loss of offsite power (or loss of seal cooling) are important sequences and should be properly addressed in the PSA event sequence development.	1. Millstone Unit 3 PSA models the following consequential failures: 1.1 Consequential Small LOCA due to a pressurizer PORV being challenged and failing to reset resulting in a small LOCA 1.2 Systems responsible for maintaining RCP seal cooling fail leading to a seal LOCA 1.3 A secondary side relief or safety valve is opened and fails to reset (i.e., a steamline break).
	2. The PSA event sequence models should address the time available for operator actions. Especially important are the sequences where the time available to complete the actions may be short compared to the time necessary to complete the task.	2. Operator actions such as "Operator Fail to Establish Bleed and Feed", "Operator Fail to Depresurize SGs", "Operator Fail to Isolate Faulty SG", are modeled in the PSA. Operators fail to establish sump recirculation is also modeled.

Issue [EPRI PSA Applications Guide, Appendix B]	Issues Description [EPRI PSA Applications Guide, Appendix B]	Related Millstone Unit 3 (MP3) PSA Model Assumptions
	3. The PSA event sequence models should include all relevant EOPs.	3. In the development of each sequence model, relevant EOPs are taken into consideration such as "Establishing Bleed and Feed" and "Isolating Faulty SG".
III. Systems Analysis	1. As a minimum, the impact of room cooling on the control room and switchgear (or relay) rooms should be addressed in the PSA.	1. MP3 PSA models HVAC including: ESF (train A and train B ventilation fails), charging pump, and CCW pump ventilation, AFW, Mechanical Room ventilation, service water, screen house ventilation, control building chilled water, Switchgear and DG room cooling.
	2. Equipment operability under harsh environment during some sequences such as conditions inside the containment after a LOCA. The PSA should document whether or not, equipment are qualified to perform under degraded conditions.	2. Equipment qualification and operability under harsh environments are not addressed in MP3 Level-I PSA. A very limited consideration of these issues is provided in the Level-II portion.
	3. Batteries are standby components whose useful life is limited usually to a few hours and should be realistically credited in the PSA models. This issue is of concern for plants that have no "*backup*" to the preferred DC supply.	3. Millstone Unit 3 has dedicated a SBO Diesel generator.

Issue [EPRI PSA Applications Guide, Appendix B]	Issues Description [EPRI PSA Applications Guide, Appendix B]	Related Millstone Unit 3 (MP3) PSA Model Assumptions
	4. In some PSAs, failure of passive components are assessed to be important and, therefore, are included in the PSA models. In such cases, there is a concern that modeling failures of passive components may obscure other more important contributions. Therefore, the PSA should document why failures of passive components are important.	4. RWST rupture and failure to open of check valves are modeled in the MP3 PSA. However, back-leak failure of check valves is not modeled. Passive failures of components were only modeled if failure resulted in loss of multiple trains and/or systems (i.e., RWST isolation valve).
IV. <u>Parameter Estimation</u>	1. Recommended ranges for unscheduled maintenance unavailability's: **Turbine-Driven Pump Train:** 0.01 - 0.05 **Motor-Driven Pump Train:** 0.001 - 0.01 **Valves (MOVs):** 0.0001 - 0.005 **Diesel Generators:** 0.005 - 0.05 **Buses:** 0.0001 - 0.001 <u>Values outside these ranges should be noted and the reasons identified in the PSA</u>.	1. In MP3 PSA, the lower bound values of unavailabiities are above the recommended lower bound values. In general, all unavailability's are within the recommended ranges.

Issue [EPRI PSA Applications Guide, Appendix B]	Issues Description [EPRI PSA Applications Guide, Appendix B]	Related Milistone Unit 3 (MP3) PSA Model Assumptions
	2. Whether "power breakers" are included in the component boundaries in the fault trees.	2. Circuit breakers are modeled in the fault trees (especially those for pumps). More significantly, the control and motive power sources to breakers were modeled.
	3. Component failure rates and unavailability's should reflect plant experience to the extent possible.	3. To the extent possible, all component failure rates and unavailability's are MP3 plant-specific data. Only, in those cases where no plant-specific data are available, generic data are employed.
V. <u>Dependent Failures</u>	1. As a minimum, CCF should include:	
	1.1 Redundant standby pumps.	1.1 MP3 PSA model considers CCFs of redundant standby pumps, redundant MOVs, AOVs, and Cvs.
	1.2 Redundant MOVs/AOVs that change state.	1.2 Circuit breakers are not modeled for CCFs. These are bounded by the CCFs of the associated pumps or diesel generators MOVs (for example: CCF of a braker is an order of magniture less than that of pumps and MOVs).

Issue [EPRI PSA Applications Guide, Appendix B]	Issues Description [EPRI PSA Applications Guide, Appendix B]	Related Millstone Unit 3 (MP3) PSA Model Assumptions
	1.3 Redundant breakers, diesel generators. 1.4 Redundant check valves. 1.5 Any other components that change state.	
	If they are not included, the reasons for not including CCFs between normally operating components should be explicitly given. 2. Any PSA which has a CCF event probability which is less than 1/100 of the single component failure probability is somewhat out of line, and the reasons for this justifications should be carefully reviewed and documented.	2.1 In the MP3 PSA model, the beta factors for CCFs are $\geq$ 0.01. 2.2 Only for 4/4 component CCF case, the beta factor is 8.0E-3. 2.3 Most of MP3 PSA model CCFs eta factors are based on the following reference: *EPRI, "Advanced Light Water Reactor Requirements Document", Appendix A, Rev. 0.0, June 1989.*

Issue [EPRI PSA Applications Guide, Appendix B]	Issues Description [EPRI PSA Applications Guide, Appendix B]	Related Millstone Unit 3 (MP3) PSA Model Assumptions
VI. Human Reliability Analysis	1. If pre-initiating event human errors (such as failure to restore a system to its correct configuration following testing or maintenance, or miscalibration) are screened out, the basis for that decision should be documented.	1. Pre-initiating event human errors (also called latent errors) are exclusively modeled the fault trees. Furthermore, only those latent errors that affect more than one system or multiple trains of the same system are modeled. The two pre-initiating human errors considered are: manual valve 3SIL*V1 (from RWST) misaligned closed and manual valve 3RHS*V43 misaligned open.
	2. Post-initiating event human errors (i.e., failures to perform appropriate actions as directed by the EOPs or AOPs) should be plant and scenario specific. Any human error probability (HEP) that lies outside the following recommended ranges should be reviewed:	2. Millstone Unit 3 PSA models post-initiating event human errors (these errors are called type C errors) they are divided into two subgroups: C_p which are those actions dictated by operating procedures and C_R which represents recovery actions that may not be covered by procedure. Type C operator action (OA) errors are quantified by assigning a value based on whether it is a skill, rule, or knowledge-based response:

Issue [EPRI PSA Applications Guide, Appendix B]	Issues Description [EPRI PSA Applications Guide, Appendix B]	Related Millstone Unit 3 (MP3) PSA Model Assumptions
	2.1 Failure to initiate primary feed and bleed: 10^{-1} - 10^{-3} 2.2 Failure to isolate ruptured SG: 10^{-2} - 10^{-3}	Skill-Based: 5E-5 to 5E-3 Rule-Based: 5E-3 to 5E-1 2.1 Failure to initiate primary feed and bleed: 10^{-2} 2.2 Failure to isolate ruptured SG: $10^{-[illegible]}$ - 10^{-3}
	2.3 Failure to initiate depressurization and cooldown (LOCAs and SGTR): 10^{-3} - 10^{-5} 2.4 Failure to switch over to recirculation: 10^{-1} - 10^{-3}	2.3 Failure to initiate depressurization and cooldown (LOCAs and SGTR): 10^{-2} 2.4 Failure to switch over to recirculation: 10^{-3}

Issue [EPRI PSA Applications Guide, Appendix B]	Issues Description [EPRI PSA Applications Guide, Appendix B]	Related Millstone Unit 3 (MP3) PSA Model Assumptions
	3. Failure to address dependencies between the individual HEPs that may occur in the same sequence cutset is a non-conservative approach.	3. Two rule files are developed for the recovery analysis. The first rule "RULE18.TXT" prohibit application of multiple operator actions within the same cut set. The second rule file "RULE18-2.TXT" allows multiple OAs to be applied within a cut set. This is necessary to allow some cut sets, which are otherwise too conservative, to have multiple recoveries.
	4. Use of multiple recovery actions (i.e., equipment repairs) in a single cut set is not recommended and may unrealistically drive down the core damage frequency. In such cases, the PSA should be carefully reviewed.	4. Operator recovery actions not placed within an event tree are added by a post-quantification cut set manipulator called *"Recovery Expert"*. This algorithm ensures that multiple operator actions are not added to the same cut set.
VII. <u>Quantification</u>	1. System mission times should be consistent with the accident scenario. This is, mission times could be less than 24 hours (which is normally assumed in most PSAs).	1. Millstone Unit 3 PSA model considers a 24 hours mission time except for sequences such loss of offsite power and station blackouts where less than 24 hours mission time is used (i.e., based on time-dependent calculations).

Issue [EPRI PSA Applications Guide, Appendix B]	Issues Description [EPRI PSA Applications Guide, Appendix B]	Related Millstone Unit 3 (MP3) PSA Model Assumptions
	2. The methodology of fault tree linking is based on: 2.1 ANDing the solutions of the fault trees. 2.2 Constructing and solving a large fault tree created by ANDing the contributing system fault trees.	2. MP3 PSA follows the fault tree linking methodology as described under 2.2, i.e., constructing and solving a large fault tree created by ANDing the contributing system fault trees.
VIII. Quantification of Plant Damage States	In the fault tree linking approach, the use of the delete term approach to accounting for the successes in event sequences is necessary to assure that the correct cut sets are generated. This is of more concern for the sequences associated with plant damage states than it is for quantification of overall core damage frequency. Not performing the deletion can give conservative results.	Millstone Unit 3 PSA model accounts for the deletion term as described in the EPRI PSA Applications Guide. For example, in a given plant damage state, if LPSI was successful but other equipment failed. Then, we delete any cut set that represents LPSI failed from the fault tree of that PDS. The latter fault tree is formed by ANDing all the fault trees that represent equipment failures.
IX. Analysis of Results	Truncation limit has an impact on the importance evaluation such that events with high RAWs may be deleted. The recommended truncation limit is 10^{-4} below the baseline CDF.	MP3 core damage frequency is 5.87E-5/RY and, therefore, the truncation limit would be 5.87E-9/RY. The current truncation limit in the MP3 PSA model is 1.0E-8/RY.

Issue [EPRI PSA Applications Guide, Appendix B]	Issues Description [EPRI PSA Applications Guide, Appendix B]	Related Millstone Unit 3 (MP3) PSA Model Assumptions
		<u>Sensitivity Study</u>. The following sensitivity study documents the impact of lowering the truncation limit from 1.0E-8 to 1.0E-9 on the CDF and the number of generated cut sets. The results of this sensitivity study shows that lowering the truncation limit from 1.0E-8 to 1.0E-9 has resulted in increasing the CDF by about 15% and the number of cut sets increased from 571 to about 2600 cut sets.
X. <u>ATWS</u>	Conservative success criteria with respect to pressure based on moderator temperature coefficient can make ATWS more important than it should be.	MP3 PSA model update follows WCAP-11993 report entitled *"Assessment of Compliance with ATWS Rule Basis for Westinghouse PWRs"*, Dec. 1988.
	If RRW of 1.005 or F-V of 0.005, respectively, is used as a measure of risk significance, it could lead to some additional SSCs being identified as risk significant.	For ATWS events, the MP3 PSA model conservatively assumes that fuel is at the beginning of cycle. Based on the WCAP-11993 discussion, there are times during core cycle for which an ATWS will exceed 3200 psig, despite the success of full AFW flow and full PRZ valve response. This would not be the case had manual rod insertion (MRI) been credited as an additional source of negative reactivity. This action was omitted for simplicity and conservatism.

Issue [EPRI PSA Applications Guide, Appendix B]	Issues Description [EPRI PSA Applications Guide, Appendix B]	Related Millstone Unit 3 (MP3) PSA Model Assumptions
XI. <u>Loss of Offsite Power and Station Blackout</u>	RCP Seal LOCA model assumptions: 1. Should be clearly stated. 2. If it is not a contributor, the reasons why should be determined. 3. Are the reasons based on plant specific design features rather than the adoption of an optimistic model?	RCP Seal LOCA is a major contributor to the MP3 core damage frequency. The MP3 Seal LOCA model assumptions are conservative - *that is,* we assume RCP Seal failure at time t = zero, given failure of thermal barrier cooling and failure of seal injection (i.e., no time delay is assumed).

ENCLOSURE 3

MILLSTONE UNIT 3

EXAMPLE EXPERT PANEL MEETING MINUTES

Northeast
Utilities System

REVIEWED BY	
D.A.D	DAD
M.S.K.	
S.D.W	SDW
M.V.	
M.V.B.	
E.A.D.	

Memo

January 2, 1996
NE-96-SAB-003

To: Distribution

From: E. A. Oswald EAO.

Subject: MP3 Risk-Based Inspection Expert Panel Meeting Minutes- 12/20/95

The Millstone Unit 3 RBI Expert Panel meeting was called to order on December 20, 1995, at 2:00 PM, in Conference Room C-101 of Bldg. 475. A quorum was present. Meeting attendees were:

R. Enoch	M.Gharakhanian	R. Schonenberg
R. West	H. Covin	P. Parulis
G. Gardner	E. Oswald	M. Smith

Elizabeth A. York, a Hartford Steam Boiler Authorized Nuclear Inservice Inspector was also present. The meeting minutes from the December 13 meeting were reviewed and accepted with no major comments. The purpose of this meeting was to have the Expert Panel review the Risk-Based Piping Inspection System List and concur that all the systems which are being evaluated for this program are being evaluated, and to review the safety significance of piping segments within the Emergency Generator Fuel Oil System (EGF). Two piping segments from the High Pressure Safety Injection System were also reviewed with the additional information which had been requested in a previous meeting.

RBI System Identification

The Expert Panel reviewed the list provided in Attachment 1.0 to determine its completeness to this application. The systems had been selected based on three criteria: 1) all Class 1, 2 and 3 systems currently within the ASME Section XI Program, 2) piping systems modeled within the PRA, and 3) various balance of plant (non-nuclear code class) fluid systems determined to be of importance. Twenty-one systems have been selected to be evaluated in more detail throughout this process. In Attachment 1.0 Table titled "Evaluation of Piping Systems for Exclusion in RBI Program," this system list was originally provided by H. Covin of MP3 Operations as systems which would result in a reactor trip. The Panel reviewed and discussed the reasons for exclusion from the

Program and accepted the final system list with further investigation of one system, Auxiliary Steam. M. Gharakhanian was concerned with the possible pipe rupture within the Auxiliary Steam System. Millstone Unit 2 has a problem with this system, in that an Auxiliary Steam pipe rupture will result in impacting the Control Room HVAC (habitability problem) and some MCCs. Further discussions with the System Engineer, Tom LaFauci, determined that there was no auxiliary steam piping in the vicinity of the Service Building which houses the Control Room. Hence, this system does not need to be addressed. Therefore, the Piping System List has been finalized.

Emergency Diesel Fuel System

The Emergency Diesel Fuel System (EGF) was divided into 4 segments (see Attachment 2.0). All of the piping segments have a failure probability of less than 1.0E-08. During panel discussions, it was noted that the cross-connect was credited, given a pipe rupture for EGF-1 and EGF-3. However, once the cross-connect valve V13 (V14) was opened and if the break was downstream of check valves V1(V7) and V3 (V9), the fuel oil would flow out the break. Therefore, the consequences of EGF-1 and EGF-3 were changed to be the same as EGF-2 and EGF-4 which is a loss of one Diesel Generator. It was also noted that there is an external-events impact associated with these piping segments. If the pipe ruptures in the vicinity of the operating Diesel Generator, there is a potential for a fire to result. However, based on room separation of the DGs, the consequence would be the loss of the operating Diesel. Based on the relatively low consequence, the importance measures are low - RRW of 1.0 and RAW of 176 (192). The Panel concurred that these piping segments were less safety-significant.

HPSI Piping Segments SIH-2 and SIH-3

The High Pressure Safety System (SIH) was reviewed by the Panel on November 15, 1995. A request for more information was made at that time concerning piping Segments SIH-2 and SIH-3. The consequence for these piping segments was a loss of the RWST; however, the pipe rupture size was relatively small (4" diam. pipe) and impacted only one HPSI train. The Panel wanted to evaluate the time available to take possible operator action, given there would be a flooding alarm and possible pump runout on high amps. It was thought that the switchover would be made earlier due to lower RWST level. M. Gharakhanian from Safety Analysis performed a simplified calculation which indicated that the RWST would be emptied in 8 hours, given this break (1000 gpm).

Discussions within the group came to the following conclusions:

- The flooding alarm would be silenced after the initiation of a LOCA, so it would be of little value.

- The RWST inventory which was lost out of the break would be significant. This would result in a loss of sump recirculation due to limited sump inventory.

- No operator action would be credited in injection phase of the LOCA.

Therefore, based on the high consequence - loss of the RWST, these piping segments were determined to be more safety-significant. Refer to Attachment 3.0 for the HPSI System Summary.

The next RBI Expert Panel meeting will be held on January 3, at 2:00 p.m., in Bldg. 475, C101. We will be reviewing the Reactor Coolant System. Please bring the packages already sent.

EAO:cms

Distribution:

R. Enoch
N. Closkey (Westinghouse)

Distribution (w/o Attachments):

K. Covin	T. Kulterman	T. Hamlin	M. Gharakhanian
D. Beachy	R. Rothgeb	M. Smith	D. McDaniel
P. Parulis	F. Cietek	B. Roy	R. Flanagan
J. Wilson	M. Powers	G. Miemiec	A. Silvia
R. Schonenberg	S. Sikorski	Y. Khalil	R. Rothgeb
R. West	G. Gardner	D. MacNeill	

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K. Hastings			