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Risk-based maintenance engineering and management

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RISK-BASED MAINTENANCE ENGINEERING AND MANAGEMENT

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ABSTRACT

RISK-BASED MAINTENANCE ENGINEERING AND MANAGEMENT

by Douglas O. Henry

This study develops an approach for the application of risk evaluation to maintenance engineering and maintenance management concerns such as costs and scheduling. The approach is based on approaches such as the Hazard and Operability (HAZOP) study, MIL-STD-882 risk assessment, and Probabilistic Risk Assessment (PRA). Recent work by the American Society of Mechanical Engineers (ASME) Research Task Force on Inspection Guidelines was also used. Systems engineering methods for analysis of reliability and maintenance were integrated with risk evaluation methods for the maintenance applications. Fitness-for-service evaluation was used for maintenance needs analysis. In addition to consequence of component failure, risk evaluation was applied for management considerations, such as scheduling, using a combination of the Program Evaluation and Review Technique (PERT) and the Critical Path Method (CPM).

This study includes detailed hypothetical examples and four case studies, predominantly related to material performance issues.

ACKNOWLEGEMENT

The author wishes to acknowledge Mr. Kenneth R. Balkey, P. E., and the other members of the ASME Research Task Force on Inspection Guidelines for their contribution to the field of risk evaluation and engineering. The Task Force reports demonstrated the application of probabilistic methods of risk evaluation beyond safety analysis and both inspired and enabled the author to prepare this study.

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Chapter 1

The Problem and Its Setting

1.1 Introduction

Modern risk assessment methods have been evolving since World War II to address safety and reliability concerns in defense and other industries. Beginning about 1975, comprehensive Probabilistic Risk Assessment methods suited to large complex systems were applied to nuclear power plants. More recently,(1) risk assessment technology has been applied directly to the inspection of systems and components. Since the publication in 1991 of the first volume of *Risk-based Inspection - Development of Guidelines*(1) by the American Society of Mechanical Engineers (ASME), a significant body of work has been developed for deploying risk assessment methods to determine inspection requirements for large-scale mechanical systems, such as power generating and chemical process plants. Benefits realized from using risk assessment methods and related technologies include improved reliability through predictive and preventive maintenance, reduced inspection frequencies for lower risk components, and reduced repair and replacement costs resulting from fitness-for-service approaches to evaluation of degraded components. The approach to risk-based maintenance engineering and management developed in this study is applied in the hypothetical examples and four actual case studies presented in Chapter 4.

1.2 The Statement of the Problem

The primary purpose of this study is to describe the application of risk assessment methods to aspects of maintenance engineering and management and to develop the methodology for such application. That purpose was approached using the body of work available describing risk assessment methodology, industry standards for fitness-for-service and flaw evaluation, standard systems engineering methods for maintenance analysis, and structured management methods.

1.3 Subproblems

Secondary purposes for this study are related to maintenance application. It will be necessary to adapt the methods described for risk assessment in support of safety, reliability, and inspection directly to the definition and support of maintenance objectives. A systems approach to maintenance analysis must be described in the context of risk-assessment. Also, management tools must be identified and adapted for maintenance applications using risk assessment data.

1.4 Research Objectives

The research objectives of this study were to (1) develop an approach using risk evaluation for maintenance engineering and management concerns such as costs, scheduling and decision making, (2) demonstrate how various analytical methods, or tools, for risk evaluation can be utilized in a maintenance context, and (3) show how fitness for service evaluations can be used to support a risk-based maintenance evaluation.

1.5 Significance

This study is unique in that it combines risk assessment methods with systems engineering and management methods to address maintenance issues, such as prioritizing the most risk-sensitive components in a system, decisions to repair or replace components, and the scheduling of maintenance. While this area has been given some treatment in the literature, it has mainly consisted of relatively brief segments in works related to other or broader topics. This study provides the reader with directly useful methods, information, and references on the topic of risk-based maintenance engineering and management.

Chapter 2

Review of the Literature and Background

This review encompasses a number of works related to the topic of risk-based maintenance engineering and management. A substantial number of recent studies and available standards relate to the inspection of components. Since inspection is an important part of an overall preventive maintenance strategy, and since these works deal effectively with the analysis of risk generally in a way that can also be applied to maintenance, considerable attention has been given to this area of the literature. As stated in Chapter 1, little has been published on the topic of risk-based maintenance per se; therefore, related works dealing with important aspects of risk that can be applied to maintenance engineering and management have been included in this study. Finally, this review includes standards which deal with the evaluation of flaws in the materials from which components are constructed as they relate to decisions regarding repairs or replacements.

2.1 Risk Assessment

Structured approaches to the assessment of risk in industry have been developed to quantify risk factors as a basis for comparative ranking and to determine the level of acceptability of the risk. These approaches include Probabilistic Risk Assessment (PRA) and the Hazard and Operability (HAZOP) study, which are discussed in sections 2.2 and 2.6, respectively. The stages of quantitative risk assessment have been generally described by Andrews and Moss(2) as including identification of hazards, estimation of consequences, estimation of the probability that the hazardous event will occur, and a determination of acceptability. The U. S. Department of Defense has developed a standard approach to risk assessment in MIL-STD-882, which is useful for reference and comparison because it includes defined consequence (mishap severity) and probability levels.(3)

The consequence of an event is based on the perspective of the organization evaluating the risk. It is often defined in economic terms and may include or be defined by other factors, such as safety or system failure. MIL-STD-882 defines four consequence levels, I through IV, as Catastrophic, Critical, Marginal, and Negligible, respectively. It distinguishes a hazard as a condition that poses the risk, and a mishap as the actual event that results in death, injury, loss of equipment or property, etc. As an example of defining magnitudes of consequence, MIL-STD-882 defines a Catastrophic mishap as one which could result in a loss exceeding one million dollars, a death, permanent disability, or some environmental damage that would violate applicable laws and regulations. Alternative definitions can be used based on context, such as a catastrophic event defined from a business perspective as one that would result in bankruptcy. A Negligible mishap as defined by the MIL-STD-882 exists when the potential loss does not exceed $10,000, does not result in a lost workday, or if the resulting environmental damage is insufficient to violate the applicable laws or regulations. From a business perspective, a negligible loss might be defined as one not exceeding the maximum insurance deductible.

Five probability categories, A through E, are defined by MIL-STD-882 as Frequent, Probable, Occasional, Remote, and Improbable, respectively. A Frequent event is one that has a probability of occurrence greater than 10^{-1} in the life of the system and an Improbable event is one with a probability less than 10^{-6}. In other words, an event is frequent if it is expected to happen more than once out of every 10 opportunities and improbable if it is expected to happen less than once in a million opportunities. Similar probability definitions are used by other organizations. For example, Bott and Eisenhawer[(4)] of the Los Alamos National Laboratory Probabilistic Risk and Hazard Analysis Group define seven probability intervals from Very Likely, 50% probability of occurrence, to Nearly Impossible at less than 10^{-6}, as shown in Table 1.

Table 1. Accident Progression Event Conditional Probability Intervals

Linguistic Descriptor	**Definition**	**Probability**
Very Likely	1 in 2 or greater chance of occurrence given the occurrence of the conditioning event	> 1/2
Likely	Between 1 in 10 and 1 in 2 chance of occurrence given the occurrence of the conditioning event	10^{-1} - 1/2
Unlikely	Between 1 in 100 and 1 in 10 chance of occurrence given the occurrence of the conditioning event	10^{-2} - 10^{-1}
Quite Unlikely	Between 1 in 1000 and 1 in 100 chance of occurrence given the occurrence of the conditioning event	10^{-3} - 10^{-2}
Very Unlikely	Between 1 in 10,000 and 1 in 1000 chance of occurrence given the occurrence of the conditioning event	10^{-4} - 10^{-3}
Highly Unlikely	Between 1 in 1,000,000 and 1 in 10,000 chance of occurrence given the occurrence of the conditioning event	10^{-6} - 10^{-4}
Nearly Impossible	Less than 1 in 1,000,000 chance of occurrence given the occurrence of the conditioning event	< 10^{-6}

2.2 Probabilistic Risk Assessment

According to Andrews and Moss,[2] one of the early milestones in such structured approaches, usually described as Probabilistic Risk Assessment (PRA) or Quantitative Risk Assessment (QRA), was the development by F. R. Farmer of an approach based on the concept that systems are always subject to some risk and that as consequences of hazardous events increase, the probability should decrease. The concept was founded on Farmer's convictions that the risk could not be entirely eliminated and that catastrophic accidents were often composed of a number of more common events. The original Farmer frequency vs. number of curies (*F-N)* curve, relating the frequency of a potential iodine-131 release to its consequences as measured in the number of curies, has been adapted as a widely used format for describing the probability and consequences of risk and distinguishing acceptable from unacceptable probability and consequence combinations. A structured, comprehensive approach to PRA was developed by the Nuclear Regulatory Commission (NRC) and implemented for nuclear power plants. That approach, described in NUREG/CR-2300, includes the use of detailed event trees or fault trees to quantify the risk factors represented by various subsystems and components to potential core damage or release of radiation.[5] The American Society of Mechanical Engineers recently issued its own standard for nuclear plant PRAs.[6] A nuclear plant PRA characterizes risk in terms of probabilities of occurrence of events related to safety, core damage frequency (CDF) and Large Early (radiation) Release Frequency (LERF), rather than as a risk related to a specific dollar-value of economic consequence.

The major milestone in PRA for nuclear plants was the 1975 WASH-1400 report,[7] a quantitative risk evaluation of two light water-reactor plants, a Pressurized Water Reactor (PWR) at the Surry site in Virginia, and a Boiling Water Reactor (BWR) at the Peach Bottom site in Pennsylvania. Other plants followed by performing PRAs in support of safety changes and to assess the risk to populations near plant sites. In 1988, the NRC issued Generic Letter 88-20,[8] which required Individual Plant Examinations (IPEs) to be performed for all nuclear plants to identify plant-specific vulnerabilities to severe accidents that could be remedied at a reasonable cost. In response, 74 PRAs were developed in support of 106 plants (more than one plant of the same design may occupy a single site). The PRAs have been used in support of system improvements such as system cross-ties to improve reliability and optimizing surveillance programs such as inservice inspection[9] as discussed further in Section 2.4. The NRC has encouraged these industry efforts by issuing Regulatory Guide 1.174 that described an acceptable approach for using PRAs in making risk informed decisions.[10]

2.3 Risk and Inspection

While risk assessment in various forms has been used for inspection planning, the field has been explored extensively under a research project sponsored by the American Society of Mechanical Engineers (ASME) and published in multiple volumes in the early 1990s. In its general document,[1] the ASME Research Task Force on Inspection Guidelines defines the overall risk-based inspection process as consisting of four parts: (1) Definition of the structural system to be analyzed, (2) A qualitative risk assessment process for incorporating actual experience with failure modes, causes, and consequences of failure for ranking of components, (3) A quantitative risk analysis method, including enhanced failure modes, effects, and criticality analysis (FMECA), considering uncertainties, to determine a highest calculated risk to optimize the inspection effort, and (4) A program of inspection that updates an initial strategy based on the results of inspections performed.

The enhanced FMECA defines numerical values for probabilities and failure consequences based on design, operating experience, expert opinion, known degradation mechanisms for the structural materials, prior inspection results, and any previous PRA studies, if applicable. Uncertainties are estimated and treated as intervals when constructing risk-ranking diagrams of failure probabilities against consequences. For that purpose an interval is a range of numbers with upper and lower limits assigned to parameters, one example being the dollar amount of the economic consequence.

To develop an initial inspection strategy, candidate strategies are evaluated for their practicality and effectiveness considering the damage potential and inspection method reliability, and the risks involved in performing the inspection itself. Once the inspection is performed, the effect on potential damage mechanisms and failure probability can be assessed and sensitivity analysis performed to optimize and update the inspection strategy.

2.4 Inspection of Nuclear Power Plant Components

Inspection requirements for commercial light water reactor plants in the United States, both the pressurized water reactor (PWR) and boiling water reactor (BWR), are described in Section XI of the ASME Boiler and Pressure Vessel Code (ASME Section XI).[11] ASME Section XI describes specific selection criteria and inspection methods required for various components.

In their study of light water reactor inspection,[12, 13] the ASME Research Task Force on Inspection Guidelines states that the Section XI approach, originally developed in 1970 when little operating experience was available, is based on an implied, qualitative risk assessment derived from the safety classification of components used in design and construction. That safety classification related to the potential for damage to the reactor core coupled with a perceived notion that likelihood of failure at welded joints is higher than at other locations. The study further states that current qualitative input from industry is that the most likely causes of component failures are material damage mechanisms not well addressed in their design, such as intergranular stress corrosion cracking (IGSCC) in the heat-affected zones of austenitic stainless steel piping welds and erosion-corrosion of plain carbon steel piping. Both of those damage mechanisms resulted in unforeseen piping failures involving unplanned repairs and replacements, lost operating time, and, in the case of erosion-corrosion, loss of life.[14, 15] For the IGSCC, the inspection procedures were modified and inspection capability demonstrations required because previous inspections were insufficiently sensitive to detect the damage.[16] For erosion-corrosion, an inspection strategy and program had to be developed since they were needed to determine the extent to which inspections for such damage had been performed previously and what inspection was necessary to determine the full extent of the degradation.[17]

Efforts to optimize inspection through risk-based methods have produced several Code Cases (alternatives to ASME Section XI requirements). The alternative requirements described by the Code Cases utilize risk assessment methods to rank components based on failure consequences and probabilities.[18, 19, 20] While two somewhat different approaches to risk assessment are described by each of the Code Cases, both approaches consider prior PRAs, available for many plants, and the results of Individual Plant Examinations (IPEs)[8] required by the regulatory authority, to determine the consequences of failure. Both Code Case approaches also consider the likely material damage mechanisms based on industry and individual plant operating experience to more effectively identify the probability of failure. One Code Case approach involves a structured risk analysis process that defines the known damage mechanisms and defines the associated failure potential of each, and the other relies heavily on the use of an expert panel similar to those used for Hazard and Operability (HAZOP) studies used in the process industries (discussed in Section 2.6). In parallel with development of the Code Cases, pilot applications of risk-based piping inspection programs were undertaken at several nuclear plants including Surry-1 in Virginia and Millstone-3 in Connecticut.[13]

2.5 Risk Assessment Methodology for Fossil Power Plants

The experience gained over the long-term fossil power industry operating history has formed the basis for inspection strategies currently practiced in fossil fuel fired power plants. While no nationwide regulatory mandate exists comparable to that imposed on nuclear power plants, the ASME Research Task Force on Inspection Guidelines explains that the motivations for inspection still exist in support of reducing occupational hazards to plant personnel and to reducing operating and maintenance costs by minimizing unplanned downtime. To that end, it is the industry practice to share operating experience information between utility companies through organizations like the Edison Electric Institute and the Electric Power Research Institute.

Information regarding failures, such as those due to stress-rupture in boiler tubes and welding flaws in reheat piping, are communicated throughout the industry prompting inspection activity.(1) In practical terms, some of the inspections carried out in fossil fuel fired power plants in reaction to such failures can be described qualitatively as risk-based. More formal qualitative and quantitative approaches were developed by the ASME Research Task Force on Inspection Guidelines.(21) In their study, the general methods of qualitative and quantitative risk assessment were presented using industry data and plant-specific examples for fossil fuel-fired power plants.

In the Task Force study, industry failure rate data (most derived from the North American Electric Reliability Council) for various plant systems was presented, and variations in those failure rates were plotted against consequences in terms of plant downtime. As can be seen in the resulting graph, shown in Figure 1, coal-fired boilers had a higher failure rate than other systems, such as gas turbines and oil boilers, and together with its consequence of failure, an average of over one million lost megawatt-hours, was the highest risk system. That risk ranking clearly indicated coal boilers as a system where expenditures on inspection and preventive maintenance had greater potential for cost savings than for other system types owned by the same generating company.

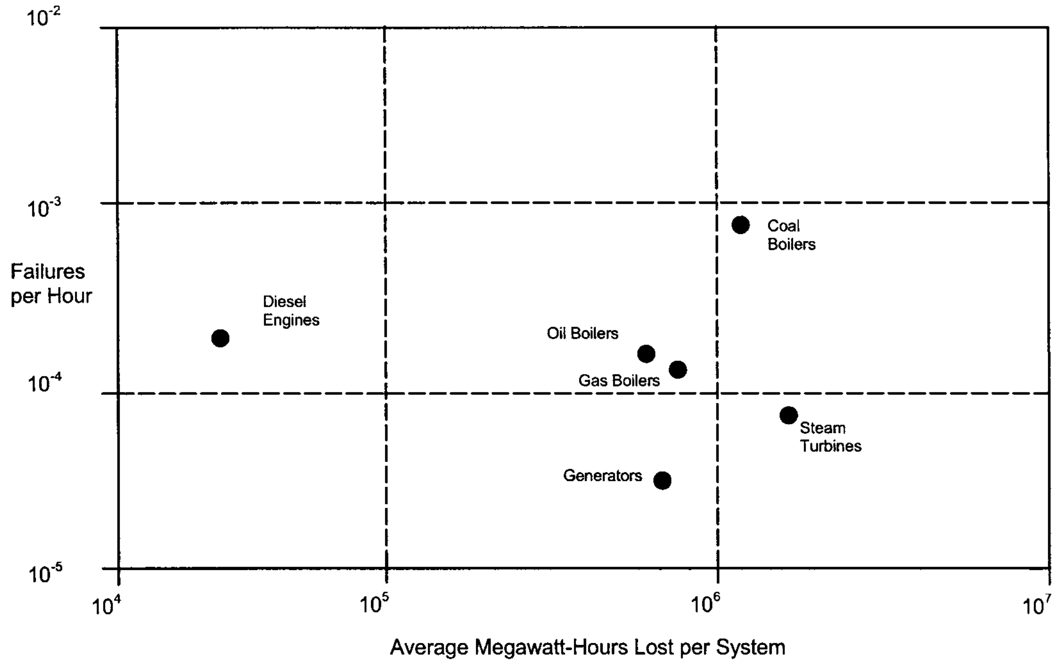


Figure 1. Fossil Power System Failure Rate vs. Lost Power Consequence

Further breakdown of the subsystems and components, using a coal-fired boiler, plant-specific system example, was performed with fault-tree logic diagrams (preceded in some cases by event tree diagrams). Those diagrams were directly effective for qualitative analysis using minimal cutsets, i.e., combinations of events which needed to cause the ultimate consequence. The qualitative analysis alone determined where system failures could be caused by failures of either single components or combinations of specific components. That qualitative analysis showed that the failure of a single sootblower in a specific coal fired boiler system could cause an entire system failure.

With the same plant-specific example, the fault-tree approach was modified for quantitative analysis. Factors such as failure probabilities, derived from failure rate information, and the unavailability, i.e., the probability that the component would not be operational, were incorporated for the quantitative model. The resulting quantitative analysis of all the minimal cutsets enabled the percentage contribution of each minimal cutset to the ultimate system failure to be determined and the top cutset identified. Through the quantitative analysis of the example system, the loss of boiler waterwall tubes was identified as the top contributor in the system in the coal boiler system analyzed. Finally, a form of sensitivity analysis was constructed using several importance measures, namely the Fussell-Vesely (FV) Importance, the Risk Reduction Ratio, and the Risk Achievement Ratio for each of the minimal cutsets. The FV is the fractional probability of total system failure represented by events involving specific components. The Risk Reduction Ratio represents the degree to which the overall risk to the system would be reduced if a particular failure event never occurred. The Risk Achievement Ratio represents the degree to which the system risk would be increased if a particular failure event always occurred.[22]

2.6 Risk Assessment in the Chemical and Petroleum Process Industry

Hazard and Operability (HAZOP) studies are a qualitative risk-assessment tool widely used in the process industry to identify major hazards and often provide information useful in improving plant operability. As described by Andrews and Moss,[2] the HAZOP study is a structured system review, which involves a team of system experts from various backgrounds. The team completes a report tabulating deviations, potential causes, consequences, and hazards. The HAZOP study uses guide words, such as "no/not," "more," "less," and "reverse" to categorize the deviations as they relate to the process. For example, the guide words "no/not," "more," and "less" as applied to flow might represent no-flow, low-flow, and high-flow conditions, respectively.

Important considerations for the process industry are events that might result in fire or explosion. For example, the size of a possible leak of flammable material is evaluated since it affects the consequence of failure. The length of time required before the flammable material ignites, and the quantity of material that accumulates before the ignition, must both be considered in order to evaluate the consequences.

2.7 Considering Risk in Building Construction

Due to increased concerns regarding safety and economic risk resulting from seismic events, the Federal Emergency Management Agency (FEMA) contracted the SAC Joint Venture (the Structural Engineers Association of California, the Applied Technology Council, and the California Universities for Research in Earthquake Engineering) to prepare a report describing recommended criteria for design and construction of new buildings with moment-resisting steel frame construction. The report, FEMA-350,[23] was prepared largely in response to the Northridge, California earthquake of 1994 in which structural members and joints in moment-frame buildings, previously believed to be subject only to damage in the form of ductile yielding, were found to have experienced brittle fractures. Frequently, the fractures were found in full-penetration welded connections.

The FEMA-350 report states that it supplements FEMA-302[24] and building code requirements to provide a "high level of confidence" that structures will resist collapse in a major seismic event. The measures required to support that high level of confidence include conservative Charpy V-notch (CVN) toughness requirements for base metal and weld metal. For example, where steel frames are enclosed and not expected to experience temperatures lower than 50° F, weld filler metals for full penetration groove welds of beam webs to column flanges are required to provide a minimum CVN toughness of 20 ft-lbs at -20° F and 40 ft-lbs at 70° F. FEMA-353[25] is imposed by FEMA-350 and provides a risk-based categorization for inspection of welds in building structures. Welds are categorized for inspection by qualitative consequence and demand matrix, where the demand represents a probability of failure. The Weld Demand Categories are high, medium, and low, where the high category is for welds expected to see loads at or beyond the yield strength and to experience some inelastic strain and the low category is for welds not expected to exceed the design stress or to be in compression. As shown in Table 2, welds subject to the greatest inspection, category 1, are those with the highest risk, i.e., considering consequence and demand (probability).

Table 2: FEMA-353 Risk-based Inspection Matrix

Consequence	Demand		
	A (high)	B (medium)	C (low)
High	1 (most inspection)	1 (most inspection)	2 (medium inspection)
Medium	1 (most inspection)	2 (medium inspection)	3 (least inspection)
Low	2 (medium inspection)	3 (least inspection)	3 (least inspection)

2.8 Fitness for Service

Essential to the assessment of systems during their operating life is an acknowledgment that equipment degrades during its period of service. Evaluating material degradation such as general corrosion, pitting, and even cracks, which may develop in components during service, is necessary to determine if the equipment is satisfactory for continued service or should be repaired or replaced. American Petroleum Institute (API) Recommended Practice 597 (API 597)[26] describes a methodology for performing such assessments under the name "Fitness for Service" or "FFS." Although API 597 was developed for use in the refining and petrochemical industry, it includes methodologies for assessing both crack-like and non-crack-like flaws in carbon and low-alloy steel components that are suitable for general use in evaluating equipment for continued service. ASME Section XI,[11] mentioned in Section 2.4, is another source of recognized flaw evaluation methodology, which includes evaluation techniques for planar flaws in both ferritic and austenitic steel components.

API 579 describes methods for determining fitness for service, not only for meeting the original operating requirements, but also for service under reduced operating requirements, such as determining reduced maximum allowable working pressures for pressure vessels and piping. It describes assessment methodology for susceptibility to brittle fracture along with mitigation and monitoring recommendations for susceptible components. The damage mechanisms addressed by API 597 include general and localized material loss, pitting, laminations and blisters, and creep, as well as crack-like flaws.

2.9 Maintenance Systems Engineering Considerations

As described by Blanchard and Fabrycky,[27] maintenance is classified as either corrective or preventive. Corrective maintenance is unplanned maintenance, such as resulting from a component failure, whereas preventive maintenance is the planned maintenance scheduled in such a way as to maintain the system or component in a manner satisfactory for performing its function. For the purposes of analysis, quantifying measures have been developed to describe the maintenance process. The mean time between maintenance (MTBM), scheduled, unscheduled or both, is related to frequency of required maintenance tasks. The mean corrective maintenance time (Mct) or mean time to repair (MTTR) is the time required to restore the system or component to operability. The mean preventive maintenance time (Mpt) is the time required for those tasks to maintain the component or system at a satisfactory level of performance, as opposed to the MTTR, which is the time required to fix the component or system after it fails. Delay times are defined by their cause, logistics delay time (LDT), the time required to obtain support such as spare parts or repair-related equipment, administrative delay time (ADT), such as for scheduling personnel, and maintenance down time (MDT) which is the total time during which a system or component is not operational, including the delay times (ADT and LDT) and the required maintenance time (MTTR or Mpt).

From a risk analysis standpoint, Andrews and Moss[2] describe how the various maintenance measures are used to determine the ultimate effects on component or system availability. The mean time to failure (MTTF) is a measure of the useful service interval of a system or component before some maintenance action, e.g., repair or replacement, is required. The failure rate is the reciprocal of the MTTF if the failure rate is a constant. Since there are some inconsistencies in terms and their use between the primary authorities on this topic, Blanchard and Fabrycky and Andrews and Moss, for purposes of this study, the MTTF will be defined as synonymous to the Mean Up Time (MUT). The mean time between failures (MTBF) is distinguished from MTTF, by Andrews and Moss, and defined as the sum of the MUT and MTTR; however, since that definition does not address Mpt and does not include ADT and LDT, for this study, the MTBF is better described as the sum of the MUT and MDT as shown in Figure 2. When the MUT and MDT for individual components or sub-systems are determined, a form of availability diagram can be constructed for a system, as shown in Figure 3. Availability is defined as the ratio of MUT to the sum of MUT and MDT, or MUT to MTBF. For an overall plant or a higher-level system, the MTTF in combination with the MDT can also be used to construct an F-N type curve to show the probability of failure (as MTTF) versus the consequences (in terms of MDT) in terms of the system or plant downtime.

Reliability measures, such as the MTBF and MTTR, can be determined as statistical measures if sufficient data are available. Statistical methods can be used to determine the probabilities related to those measures, and to estimate the associated confidence in the measure.[28, 29]

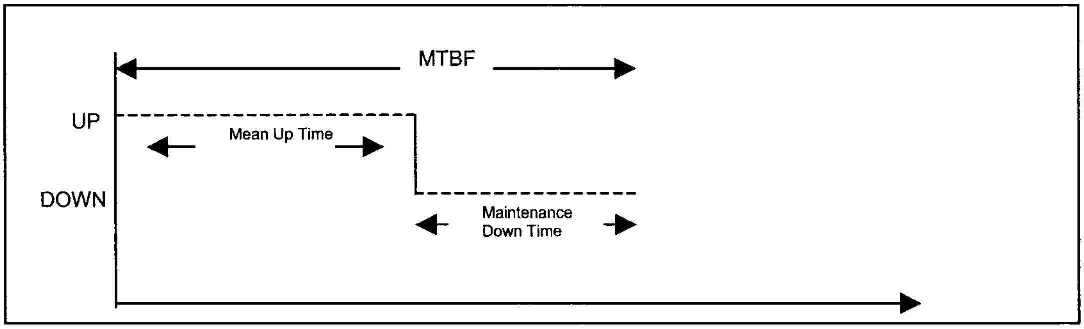


Figure 2. Mean Time Between Failures

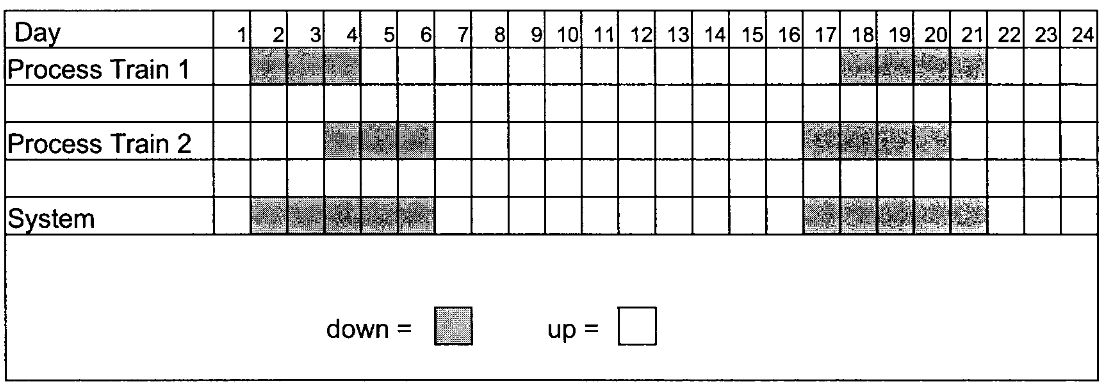


Figure 3. Availability Diagram for a System with Two Trains in Series

A maintenance optimization method that applied risk evaluation methods to structural components in nuclear power plants was developed by Electricite de France.[30] The method, called OMF Structures, optimizes maintenance by selecting high-risk elements (components) for preventive maintenance. The OMF Structures method consists of a multi-step process involving (1) system decomposition into segments, (2) functional analysis (3) failure modes and effects analysis (FMEA) performed at the segment level to identify severe failure modes (4) decomposition of the segments into structural elements representing the severe failure modes (5) a failure modes, effects, and criticality analysis (FMECA) performed at the element level, and (6) definition of preventive maintenance for elements associated with the critical failure modes. The safety consequence considers the results of the probabilistic risk analysis. The economic consequence is estimated based on operating experience and expert judgment. Probability of failure is determined using reliability models and degradation models, supplemented by operating experience and expert judgment when less quantitative data are available.

2.10 Assessing Benefits of Inspection and Preventive Maintenance

The probability of failure is related to the probability of a component being in a damaged or flawed condition during service. That probability is greatly affected by the effect of inspections and preventive maintenance in eliminating those conditions by repair, replacement, or other corrective actions, such as operation at a reduced load. The Structural Reliability and Risk Assessment (SRRA) process can be used to account for the benefits of inspection and preventive maintenance in a quantitative manner.[1] The objective of the SRRA is to develop a best estimate of the probability of component failure. Beginning with an assumed initial damage condition, inputs for the SRRA include the reliability of the inspection method, the inspection schedule, likely degradation mechanisms, loading history, and planned or design life of the component.

The SRRA process was programmed as the PRAISE Code, which has been used to address issues such as the conditional probability of a double-ended break in nuclear power plant primary coolant piping, based on assumed crack depths and inspection schedules. SRRA evaluations were performed in the early 1980s for nuclear reactor vessels, and more recently have been performed for nuclear power plant piping for degradation mechanisms including fatigue, IGSCC, and erosion-corrosion.[13]

2.11 Management Considerations

Decision-making, scheduling, and budgeting are examples of maintenance management considerations that can involve, utilize, or affect risk assessments.

Structured, systematic approaches to making management decisions have been developed by Kempner and Tregoe[31] to evaluate objectives, alternatives, and potential risks. Their straightforward decision analysis method involves identifying the relative importance of objectives, and distinguishing between the necessary, desirable, unnecessary, and sometimes undesirable aspects of various alternatives. The various aspects of alternatives are rated in terms of their relative importance to each of the identified objectives. The ratings of objectives and the aspects of various alternatives are assigned as numeric values, and the result is a weighted value for each alternative, as shown in Figure 4.

Objective	Priority Weight	Alternative 1	Score	Weighted Score	Alternative 2	Score	Weighted Score
Objective 1	10	Reason 1	6	60	Reason 1	2	20
Objective 2	5	Reason 2	3	15	Reason 2	9	45
Weighted Score				75			65

Figure 4. Kempner-Tregoe Decision Analysis Process

Network scheduling techniques are a quantitative tool to describe workflow, duration, and task interdependence. Kertzner[32] describes two widely used techniques, the Program Evaluation and Review Technique (PERT) and the Critical Path Method (CPM). The PERT technique involves deriving task completion times from several estimates assuming a normal distribution which Kertzner explains is an inherently probabilistic approach and provides a means of determining risks related to maintenance project completion. The related CPM is primarily resource dependent and is used when more accurate time estimates are available. Either technique or a combination of the two can be used to develop diagrams that define the workflow and identify the critical path and greatest risks to maintenance project completion. These structured techniques can also be used to determine the effects of variations in capital and labor resources on project completion.

2.12 Background for Reliability and Statistics

A discussion of important reliability and statistical evaluation methods that may be useful in performing risk-based maintenance and management evaluations are described in this section. While these methods may not be used in every evaluation, a number of these methods were included in this study since each maintenance project will involve unique factors requiring knowledge of the available evaluation tools. Although this section has been included for background, the presentation of these methods and the examples used were developed specifically for this study by the author.

2.12.1 Reliability Networks

The probability of failure is simply the opposite of reliability. The sum of the reliability (R) and the probability of failure (Q) for a component (C_i) or a system is 1. For an example component C_A:

$$RC_A + QC_A = 1 \qquad [1]$$

From Equation 1, the probability of failure of component C_A is:

$$QC_A = 1 - RC_A \qquad [2]$$

Therefore, reliability network models can be used for system failure probability determination. Series networks are those in which all the components must operate reliably for the system to operate reliably. The reliability of a series network, such as shown in Figure 5, is the product of the reliabilities of its components or

$$R_{SYS} = RC_A{*}RC_B{*}RC_C. \qquad [3]$$

Conversely, the probability of the network failure is

$$Q_{SYS} = 1 - (1 - RC_A){*}(1 - RC_B){*}(1 - RC_C). \qquad [4]$$

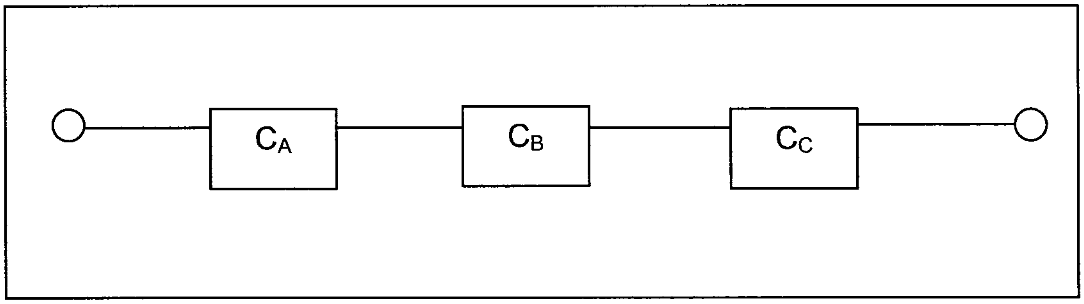


Figure 5. Series Reliability Network

Systems designed to operate effectively if at least one of its subsystems operate effectively are parallel networks. A simple two component parallel network is shown in Figure 6.

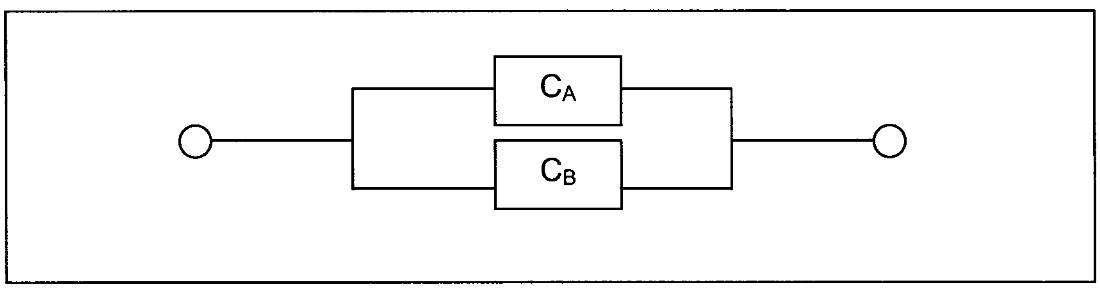


Figure 6. Parallel Reliability Network

The reliability of a two-component parallel network is one less the product of the failure probabilities,

$$R_{SYS} = 1 - (1 - RC_A)*(1 - RC_B) \qquad [5]$$

and the failure probability of the network failure is

$$Q_{SYS} = QC_A*QC_B\,. \qquad [6]$$

More complex systems can be represented by variations of series-parallel networks, which are analyzed by reduction to the series and parallel elements.

2.12.2 Application of Statistical Methods

Basic statistical methods are used in a number of ways to support risk evaluation. A few statistical methods are presented in this section to support the overall topic of risk applied to engineering and management. For detailed treatment of various probability distributions and statistical methods, the cited references for this section[28, 29] can be consulted. One of the most challenging aspects of risk evaluation is estimating probabilities and treatment of the statistical data. Sometimes, as often is the case with failure probabilities, only anecdotal information or expert opinion is available. However, if sufficient data concerning maintenance histories or the service life for a given type of component are available, basic statistical methods can be used.

2.12.2.1 Binomial Distribution

When reviewing the service history data for a type of component in terms of the number of failures or, alternatively, successes, i.e., those operating satisfactorily, in order to assess the risk of current operation, the binomial distribution can be used. The binomial distribution is called a discrete distribution because the parameter measured can only take on certain values, in this case the number of successes or failures. The binomial distribution requires *Bernoulli trial* conditions, i.e., there are only two outcomes (success or failure), an equivalent probability of success for each trial (e.g., components and the service conditions are the same or similar), the number of trials is a constant, n, and the trials are independent (i.e., the outcome of one does not influence the outcome of another). Under these conditions, just as in the reliability discussion of Section 4.2.1, if the probability of success is p, the probability of failure, q, is $(1-p)$, such that $p + q = 1$. The probability of S successes in n trials defined by the binomial distribution is described by the following equation.

$$B(S;\, n, p) = [n!/S!(n - S)!]p^S(1-p)^{n-S} \qquad [7]$$

Such a distribution could be used, for example, to evaluate a replacement option. If the service history indicates that 60 percent of a specific valve type fails after 10 years of service, and the plant has 5 such valves, what is the probability that more than one of the 5 will require replacement after 10 years of service? The probability that 4 out of 5 valves will continue to operate successfully is $B(4;5,0.6) = [5!/(4!(1 - 0.6)]0.6^4(1 - 0.6)^{5-4} = 0.26$. So the probability that more than one valve will fail is $1 - 0.26 = 0.74$ or 74%. If the consequence to plant operation of valve failure is \$1000 per day and 3 days are required to obtain a replacement valve, the risk of not having a second replacement valve in stock is \$3000(0.74) = \$2220. Once the risk is determined, the decision whether to stock an additional replacement valve can be made in a risk-informed manner.

2.12.2.2 Normal Distribution

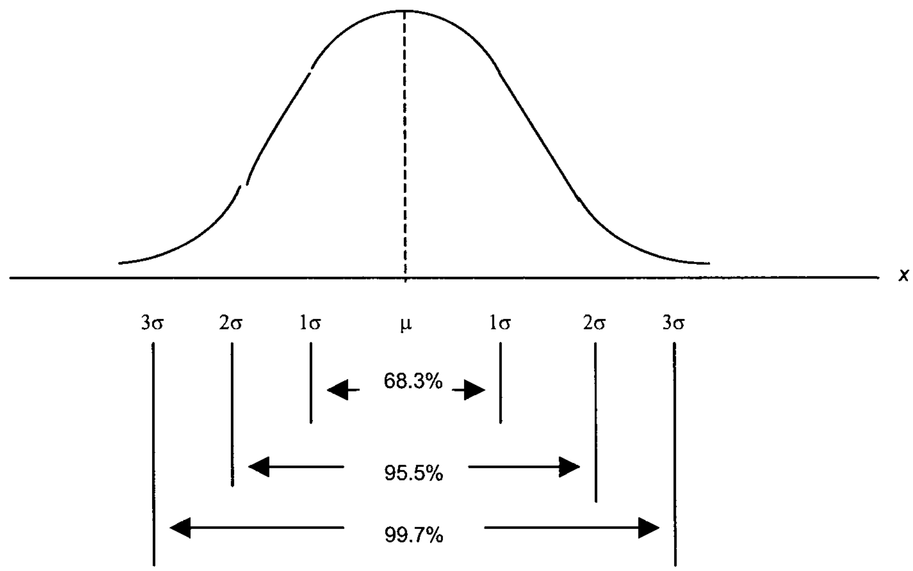
When the variable of concern is a continuous random variable, such as the time to repair, it is analyzed as a continuous distribution. One of the most commonly used continuous distributions is the *normal distribution*. (Note that if the number of trials is large, the normal distribution can be used with a mean of np and variance of $np(1-p)$ as an approximation of the binomial. Terms associated with continuous distributions are: 1) the mean (μ), defined as the arithmetical average, 2) the median, defined as the fiftieth percentile of the distribution, 3) the variance (σ^2), defined as the scatter or variation of the data, and 4) the standard deviation (σ), the square root of the variance. Those terms and the normal distribution function curve are further described in Figure 7. If service records indicate that the mean time to repair (MTTR) is 4 hours and the standard deviation is 1 hour, according to Figure 7, the probability that the repair will take between 3 and 5 hours is 68.3%. The probability that the repair will take between 2 and 6 hours (i.e., 2σ) is a more certain 95.5%. To find the probability that the repair would take less than three hours, the standardized random variable (z) is determined to be (3 - 4)/1 = -1. Values of the standardized normal distribution function $F(z)$ are tabulated in numerous textbooks on statistics, including the references for this section. Since $F(-1)$ = (1-0.8413) = 0.1587 [28, 29], the probability that the repair would take less than 3 hours is 15.9%.

To analyze the data obtained by a number of observations or samples, methods of determining the confidence associated with the mean are used. If the mean of a random sample of n values, $\bar{x}$, is taken from a population with a mean of μ and a standard deviation of σ, then the following statement of the central limit theorem can be made

$$z = \frac{(\bar{x} - \mu)}{\sigma / \sqrt{n}} \qquad [8]$$

The term $(\bar{x} - \mu)$ is the error of the mean. The confidence is the probability that the value $(\bar{x} - \mu)$ is, in fact correct. A confidence interval concerning values of the mean can be determined as follows:

$$\bar{x} - Z_{\alpha/2} \frac{\sigma}{\sqrt{n}} \le \mu \le \bar{x} + Z_{\alpha/2} \frac{\sigma}{\sqrt{n}} \qquad [9]$$



Normal Distribution and Areas Under the Curve

Normal Probability Density Function:

$$f(x;\mu,\sigma^2)=\frac{1}{\sqrt{2\pi\sigma}}e^{-\frac{1}{2}\left(\frac{x-\mu}{\sigma}\right)^2} \qquad [10]$$

Standardized Normal Distribution:

$$F(z)=\frac{1}{\sqrt{2\pi}}\int_{-\infty}^{z}e^{-\frac{1}{2}t^2}\,dt \qquad [11]$$

Variance:

$$\sigma^2=\frac{\sum_{i=1}^{N}(x_i-\mu)^2}{N} \qquad [12]$$

Standard Deviation:

$$\sigma=\sqrt{\frac{\sum_{i=1}^{N}(x_i-\mu)^2}{N}} \qquad [13]$$

Standardized Random Variable (z):

$$z=\frac{(x-\mu)}{\sigma} \qquad [14]$$

Sample Standard Deviation:

$$s=\sqrt{\frac{\sum_{i=1}^{n}(x_i-\bar{x})^2}{n-1}} \qquad [15]$$

Figure 7. The Normal Distribution and Statistical Terms

For example, if the service history of 30 components indicates an average service life of 40 months and the standard deviation is 5, it can be said with 95% confidence ($Z_{\alpha/2} = 1.96$)[28, 29] that 40-(1.96*0.91) $\leq \mu \leq$ 40+(1.96*0.91) or 38.2 $\leq \mu \leq$ 41.8.

In other words, the mean is between 38.2 and 41.8 (inclusive) in 95% of the cases, but in 5% of the cases it may be outside that range. For a single sided confidence interval, i.e., to determine value less than or greater than the mean, Z_α is used instead of $Z_{\alpha/2}$.

2.12.2.3 Small Sample Sizes and the *t* Distribution

Where the data consist of a small number of observations, generally fewer than 30, as is often true in industry, the actual standard deviation is unknown and the sample standard deviation (s) is calculated using the sample mean ($\bar{x}$), and the *t* distribution, as shown in Figure 8, is used.

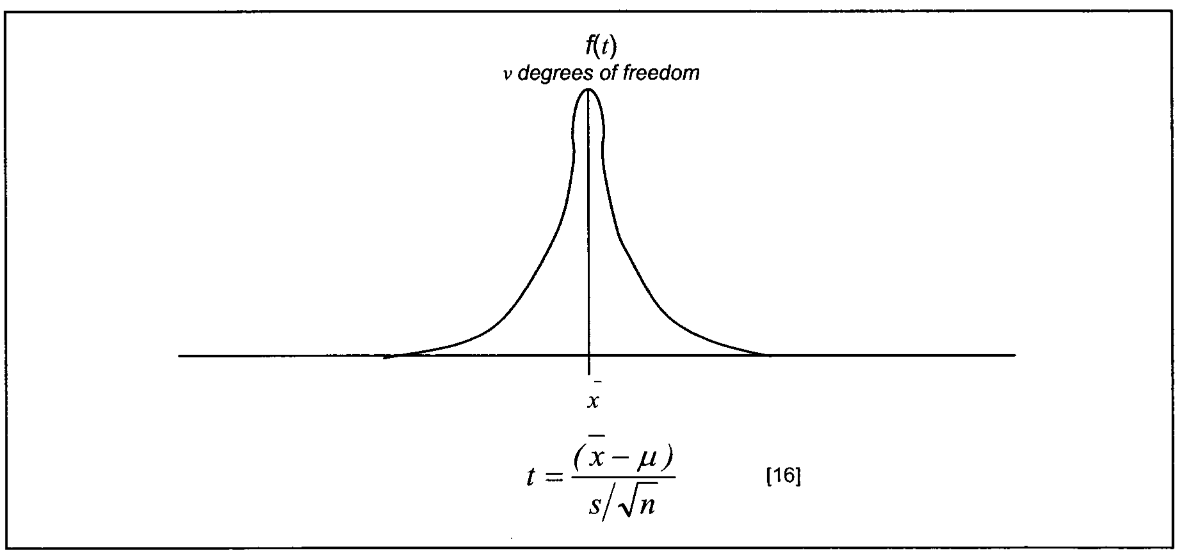


Figure 8. The *t* Distribution

The t distribution assumes that the actual population is normal and for each incremental increase in the degree of freedom v (equal to n-1), the *t* distribution more closely approaches the normal. In a manner similar to that of the normal distribution, a confidence interval for the *t* distribution can be determined as follows:

$$\bar{x} - t_{\alpha/2}\frac{s}{\sqrt{n}} \le \mu \le \bar{x} + t_{\alpha/2}\frac{s}{\sqrt{n}} \qquad [17]$$

Values of the t_α for various degrees of freedom are tabulated in numerous textbooks, including the references for this section. [29, 30] As an example using the *t* distribution, if a service history of 10 components indicates an average service life of 40 months, and the sample standard deviation is 5, $v = 9$ and it can be said with 95% confidence ($t_{\alpha/2} = 2.13$)[28, 29] that 40-(2.13*1.58) ≤ μ ≤ 40+(2.13*1.58) or 36.6 ≤ μ ≤ 43.4. For a single sided confidence interval, i.e., to determine value less than or greater than the mean, t_α is used instead of $t_{\alpha/2}$.

2.12.2.4 Other Distributions

There are a number of other continuous distributions used in statistical analysis that are not explained in detail, but are useful representations of certain types of data. The Exponential distribution, expressed as:

$$f(x) = \lambda e^{-\lambda x} \qquad [18]$$

is often used to represent reliability. The scale parameter, λ, is used to represent the failure rate and the mean $\mu = \frac{1}{\lambda}$ then represents the mean time to failure (MTTF). For the exponential distribution, the mean is $\mu = \dfrac{1}{\lambda}$ and the variance is $\sigma^2 = \dfrac{1}{\lambda^2}$. The multi-parameter Weibull distribution is useful because the shape of the distribution can be changed to model the population or events being studied. The Weibull distribution is expressed by

$$f(t) = \alpha\beta t^{\beta-1} e^{-\alpha t^{\beta}} \qquad [19]$$

where $t > 0$, $\alpha > 0$, and $\beta > 0$. The shape of the distribution is changed by changing the values of parameters α and β. A Weibull-type distribution has even been used to model the classical bathtub-shaped reliability curve that is used to model the reliability characteristics of a component over its service lifetime.(33) The bathtub curve describes failure rates that decrease with time during early life, and increase with time during later life. Early failure rates are high during the burn-in period due to causes such as inherent and manufacturing flaws but decrease relatively rapidly as the defective population is eliminated. This high failure rate period is followed by an intermediate period of a relatively low and constant failure rate, which is then followed by an increasing failure rate in the "wear-out" period.(2) More recently, the Weibull distribution has been used to model uncertainty for fracture toughness and flaw size in support of Partial Safety Factors for probabilistic fracture mechanics.(34) Further information on the exponential distribution and other useful distributions, such as the gamma distribution and the Wiebull distribution, can be found in the references to this section.(28, 29)

Chapter 3

Methodology

Several research methods have been employed to develop this study. Actual case studies, some involving interviews with subject matter experts, were used along with hypothetical example applications, and a review of the literature.

3.1 Maintenance Case Studies

Cases of actual maintenance operations and industry events have been researched to present scenarios that feature the various techniques and methods of risk-based maintenance and management. The background and circumstances of each case is described to develop the context for application of the risk aspects. The case studies are based on the author's past projects and projects undertaken by the author specifically for this study. All of these are presented in detail in Chapter 4. The cases of erosion-corrosion of piping and of stress corrosion cracking of jet pump beams were both industry issues in which the author has had direct experience. In those cases, risk aspects of the project have been developed specifically for the purposes of this study that were not part of the original project. The cases involving remote manipulator cable failure and dissimilar metal weld failures in boiler tubes were undertaken specifically for this study.

Where specific details for part of a case study were not available, such as for a fitness for service evaluation of piping damage due to erosion-corrosion, representative examples were used to further develop the application of the risk methodologies to the case. Where proprietary information was involved, some of the names or facts were omitted or changed in such a way as was sufficient to protect the information without compromising the technical relevance of the material.

3.2 Interviews

Interviews were an essential research tool because they were the only source for much of the background information and essential facts related to some of the maintenance case studies. In particular, nearly all of the information on the remote manipulators and much of the information on the boiler tube case histories was obtained by interviews with the cognizant individuals. The interviews consisted of discussions related to the project, sometimes conducted over days or weeks, and transcripts were not prepared. The cognizant individuals did review the written case studies to confirm that the information they provided was accurately rendered in the case study. Where information from interviews was used, the background information concerning the interview, e.g., the job title of the interviewee and the relationship of the interviewee relative to the case study have been presented. In addition to basic facts concerning the specific maintenance scenario, the opinions of the interviewees relative to the various scenarios were also considered in the studies since the interviewees were selected for their expertise relative to the case study.

3.3 Examples

Although the case histories were extensive, they did not provide the opportunity to fully describe and develop all of the tools which can be used in the approach to risk-based engineering and management. Therefore, examples in the form of maintenance scenarios were used to provide a format for some of the tools and risk assessment techniques. While not as extensive as the case histories, the examples were presented in a practical maintenance context so as not to dilute their relevance.

3.4 Review of the Literature

In addition to the literature reviewed to develop the process of risk-based engineering and management discussed in Chapter 2, specific information, including risk analysis in support of inspection programs, published failure analyses, material property and reliability data have been used as a method of developing the specific example risk-analyses discussed in Chapter 4. Recent risk evaluations of nuclear plant piping systems described by the ASME Research Task Force on Risk-based Inspection Guidelines [12, 13] have been reviewed and similar methods applied to related maintenance implications. Failure analyses related to components analyzed in this study have been used as a resource for information on material damage mechanisms. Handbooks, standards, and specifications have been used to obtain material property and reliability data essential to understanding the performance of materials under specific service conditions. In addition, literature related to the development of the background and basic facts of the case study, such as various historical reports on events related to industrial failures, have been used and cited. In some case studies, related engineering evaluations and reports have also been referenced and cited. Where material from other sources is used as a basis for graphs and tables, the information has been adapted specifically for the intended example and is not a direct reproduction of the original. Readers interested in utilizing the material from those other sources are encouraged to obtain a copy of the reference or contact the referenced organization. Information from the literature has also been used to support the examples, such as system descriptions and other information specific to an individual case study or example. Thus additional literature sources have been cited in Chapter 4 where they were specifically required to develop a topic or to support a case study.

Chapter 4

Research Approach, Results, and Discussion

In Chapter 2, a number of evaluation tools from the literature were described to support the analysis of risk, decision-making, and management in order to form a foundation for an integrated approach for risk-based maintenance engineering management. An approach to using risk-analysis for maintenance engineering and management was developed for this study and uses those tools in the various steps of the process. The various tools are explained in the context of maintenance activities and developed together as a process through examples and application to specific, actual maintenance scenarios. Most often the use of risk evaluation in maintenance is directed toward specific projects, usually of high economic consequence or safety significance, but can be applied to projects involving lower consequence impact as well. Both hypothetical examples, such as those in Section 4.2, and case studies, as presented in Section 4.3 are used to demonstrate practical application of risk evaluation techniques involving various levels and types of consequence.

4.1 Overview of the Approach

A risk-based approach to maintenance engineering and management may be applied at any level. When considering a complete plant, many systems can be evaluated, ranked, prioritized, and scheduled, but the approach can also be used at the sub-system or component level. The term component as used in this chapter will generally describe a functional entity, such as a pump or a valve, which is usually purchased or considered as a unit.

A generalized approach to risk-based maintenance engineering and management has been developed in this study. The basic stages in the approach can be described as follows:

(1) Definition of the system or component
(2) Qualitative risk assessment
(3) Quantitative risk assessment
(4) Maintenance needs analysis
(5) Maintenance scheduling analysis

The extent to which each of the five steps is used, and whether all steps are applied, will vary widely on the scope and circumstances of the project. Risk evaluation can be applied not only to system failures, but also to maintenance and scheduling, and as the complexity of the analysis increases more detailed information on the system or component may be required. Therefore, the approach to risk-based engineering and management is an iterative process, as illustrated in Figure 9.

The application of risk-based maintenance engineering and management process should be appropriate to the needs of the particular project. Preventive maintenance planning at the plant level may consider multiple systems and components ranked using qualitative and quantitative risk assessment, but the maintenance needs analysis might be limited to do-nothing, repair, or periodic replacement options. Where the scope is limited to the component and sub-component level, the maintenance needs and scheduling analysis might include fitness for service evaluations, the extent of required repair, parts

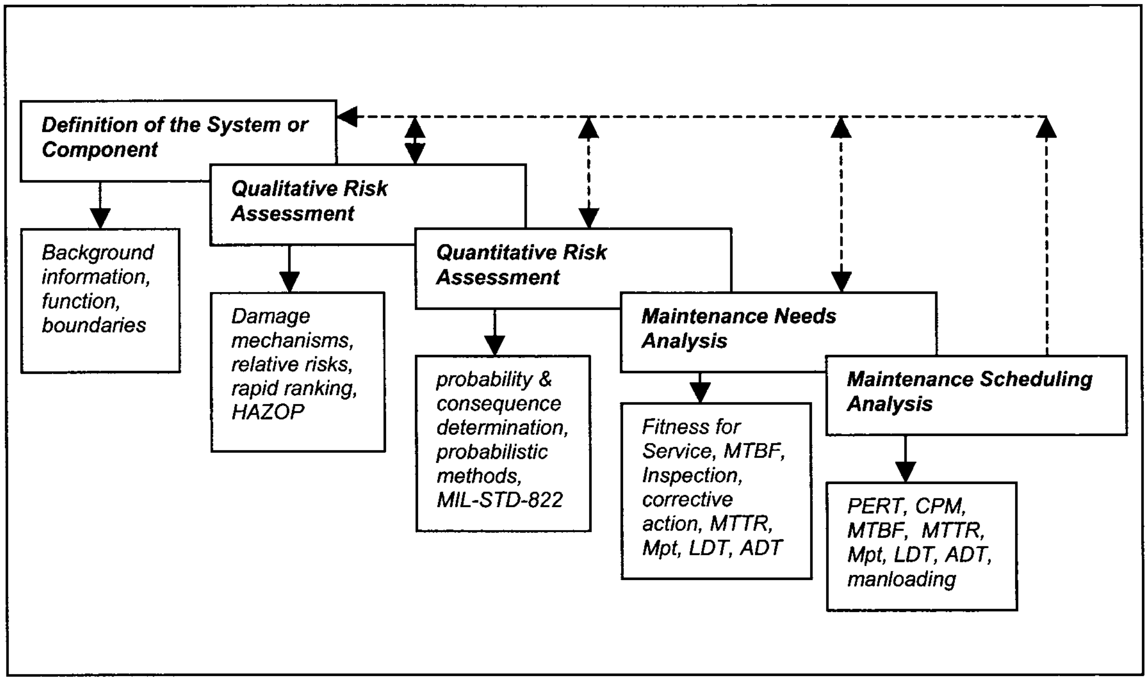


Figure 9. The Risk-based Maintenance Engineering and Management Process

replacement options, manloading, and detailed schedule. For larger-scale maintenance projects, tools such as the Program Evaluation and Review Technique (PERT) and the Critical Path Method (CPM) can be effectively used for the scheduling needs analysis to quantify risks associated with scheduling options. The extent to which specific risk-analysis tools and processes are applied depends primarily on the magnitude of the economic risk. A preventive maintenance decision may be an obvious choice where the economic consequences of a single failure are so high relative to the cost of the maintenance that forgoing the maintenance would be unacceptable even for low failure probabilities. In such cases, no further detailed analysis is necessary. Similarly, where the economic consequences of failure are very low, corrective maintenance may be the obvious choice. For many applications, MIL-STD-882 is a suitable and practical guide for a basic quantitative risk evaluation, but for projects involving potentially higher economic or safety consequences, such as those involving nuclear safety, more rigorous Probabilistic Risk Assessment (PRA) methods may be justified or required. In the case of nuclear power, use of PRA methodology is driven by regulatory requirements related to public safety, but economic risk analysis was applied in determining liability in support of the Price-Anderson legislation, the legislation that established financial coverage for a nuclear incident associated with a commercial nuclear power plant. The unique aspects of each project will determine the type and the extent of the evaluation.

4.2 Examples

The hypothetical examples presented in this section were created specifically for this study to demonstrate Fitness for Service (FFS) evaluation, Failure Modes and Effects Analysis (FMEA), risk ranking, and graphic methods of representing risk. Power plant piping flaws are used to demonstrate fitness for service evaluation in Section 4.2.1. A nuclear plant redundant safety system is used to demonstrate FMEA and risk ranking in Section 4.2.2. Purely hypothetical and simple values were selected to demonstrate the constant-risk graph and illustrate the relationship between risk, consequence, and probability in 4.2.3. The technique of graphic representation was further developed for this study in Section 4.2.3 by representing the entire risk classification method of MIL-STD-882 graphically for the first time. That example not only demonstrates graphic representation, but also enables MIL-STD-882 methodology to be readily understood and accessible to the reader.

4.2.1 Fitness for Service Evaluation

The term Fitness for Service refers to the assessment of equipment for various forms of degradation due to various mechanisms to determine the suitability of equipment for continued operation. Fitness for Service evaluations can be used directly to determine the need for immediate repair or replacement and determine the time that equipment can remain in operation prior to failure. Damage types subject to Fitness for Service evaluation methods include general and localized material loss, pitting, laminations and blisters, creep, and cracking. The examples in this section are hypothetical examples created for this study using piping types, pressures, and temperatures representative of actual power plants. The flaw dimensions were created for these examples by the author to demonstrate the evaluation methods and are not from actual plant data. The evaluation methods of API Recommended Practice 579 (API 597)(35), and ASME Section XI(11) were used for the examples in this section. ASME Section XI Code Case N-513-1(36) was also applied. Excerpts from API 579, showing equations and figures referenced in this study, are included in Appendix D.

4.2.1.1 Flaw Tolerance and Material Condition

The following example illustrates the use of a Failure Assessment Diagram (FAD) for evaluating crack-like flaws. A Level 1 assessment using Section 9 of API 597 was performed.

Consider a crack-like flaw in the heat affected zone of a circumferential butt weld on the inside surface of an SA-333 Grade 6, pipe 22 NPS, schedule 100 (wall thickness = 1.375 inch). The flaw is parallel to the weld and its dimensions were reported as 0.25 inches in depth (a) by 2.8 inches in length (2c or *l*) as determined by ultrasonic examination. The pipe is in an ASME Section III, Class 3 portion of a nuclear power plant feedwater system with a design pressure of 1200 psi and a design temperature of 575 °F. The weld was subject to post weld heat treatment (PWHT). The distance to the nearest structural discontinuity is 47 inches.

API 579 paragraph 9.2.2.1(e)2(b)c states that this method is limited to carbon steel materials with a fracture toughness equal to or greater than the lower bound K_{IC} (as ksi √in, °F) determined from API 579, Appendix F, section F4.4.1 c, as follows:

$$K_{IC} = 36.5 + 3.084e^{0.036(T - T_{ref} + 56)} \qquad [20]$$

where T_{ref} corresponds to a 15 ft-lb impact energy for carbon steels. Paragraph 9.2.2.1(e)2(b)c further states that carbon steel that has not had its toughness degraded by environmental conditions, such as overheating, will meet that requirement. That assumption is based on the fact that API 579 considers that only materials acceptable under the ASME Code, API, or ANSI standards were used for construction.

The value for T_{ref} was determined as 40 °F from API 579 Figure 3.3 using Curve B and a wall thickness of 1.375 inch. Using an evaluation temperature of T = 30 °F, the K_{IC} value determined by Equation 20 is $K_{IC} = 36.5 + 3.084e^{0.036(30-40+56)} = 41.7 ksi\sqrt{in}$. As illustrated by Figure 10, for a temperature of T = 30 °F and T_{ref} = 40 °F, Curve B of

API 579 Figure 9.15 shows that an infinite flaw length is acceptable for such a flaw for material in the PWHT condition. If the material had not been in the PWHT condition, Curve C shows little flaw tolerance for the same evaluation temperature and, in fact, up to the evaluation temperature of 80 °F the flaw would have been unacceptable based on length. Since such a flaw may see loading below 80 °F, such temperatures cannot be used for evaluation and the flaw remains unacceptable. When performing risk assessment, identifying flaw tolerant material conditions, such as PWHT in welding, or

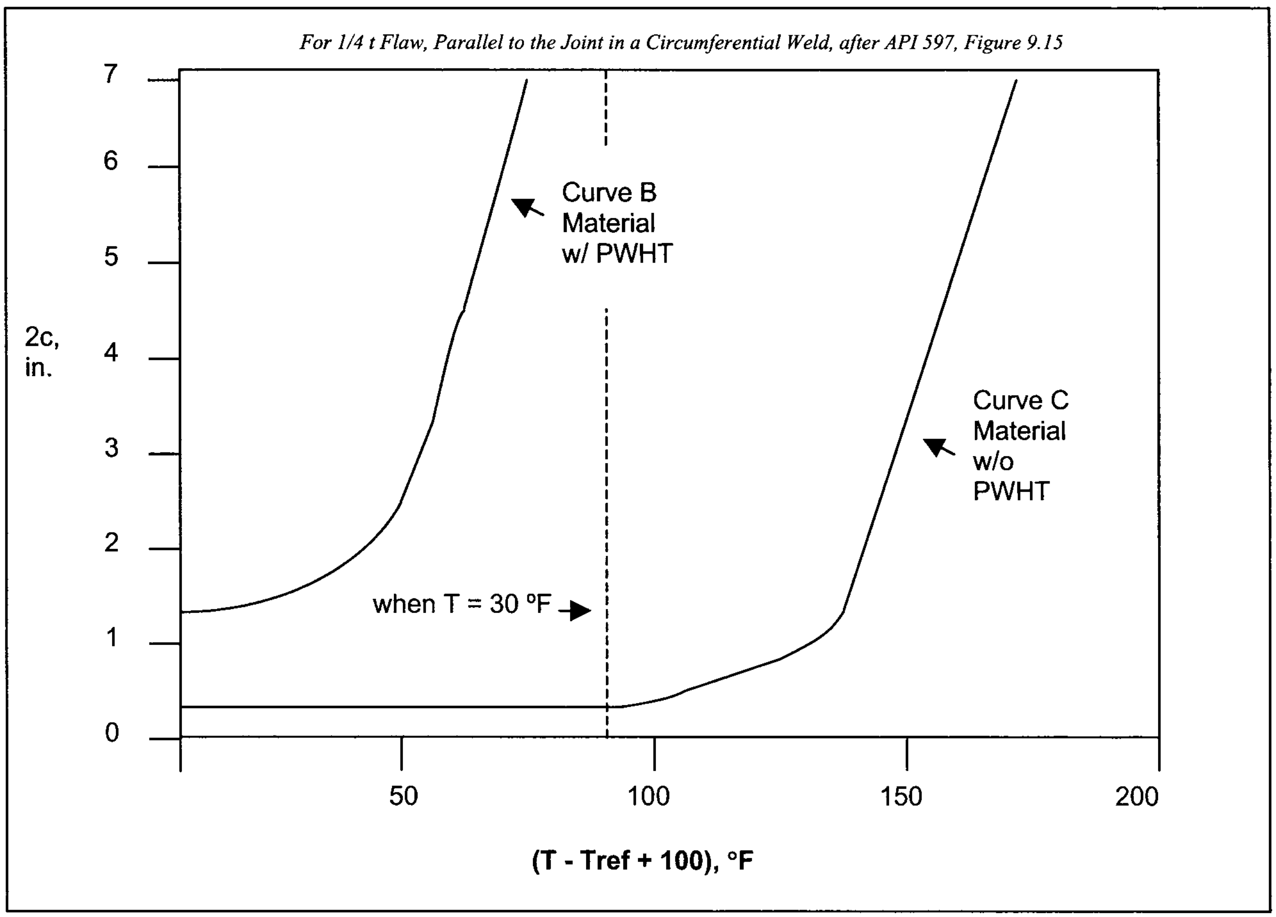


Figure 10. Allowable Flaw Length SA-333 Grade 6

toughness-increasing heat treatments for base metal (e.g., normalize and tempered vs. as-rolled) can be used to reduce the relative failure probability for components so inspection schedules and maintenance planning for those components can be modified accordingly.

4.2.1.2 Leak-Before-Break and Flaw Growth

Consider a crack-like flaw in the heat affected zone of a circumferential butt weld initiating on the inside surface of an SA-312 Type 304, pipe 14 NPS, schedule 20 (wall thickness = 0.312 inch). The ultimate tensile strength of the material is 75 ksi and the yield strength (σ_y) is 30 ksi. The flaw is parallel to the weld and its dimensions were reported as 0.15 inches in depth (a) by 1.2 inches in length (2c or l) as determined by ultrasonic examination. This is in an ASME Section III, Class 3 portion of a piping system in a nuclear power plant with a design pressure of 275 psi and a design temperature of 200 °F (moderate energy). The active damage mechanism is determined to be intergranular stress corrosion cracking. The flaw is found during the last of a series of routine inspections, and repairing the flaw would extend the shutdown several days with a consequence in lost generation of $0.5M/day. Therefore, it is desirable to determine if the repair can be deferred to a planned shutdown. Because austenitic stainless steel is a ductile material, the effect of a through wall crack will be evaluated using ASME Code Case N-513-1 and ASME Section XI, Appendix C (Section XI, Appendix C).

For through wall flaw evaluation Code Case N-513-1 references the Section XI, Appendix C procedure using the flaw penetration a/t = 1. The stress, P'_b, for a wrought base material, assuming a gas metal arc weld (GTAW), from Section XI, Appendix C, C-3320, is

$$P'_b = SF(P_m + P_b) - P_m \qquad [21]$$

The value P'_b for failure is determined from Section XI, Appendix C, C-3320, as follows:

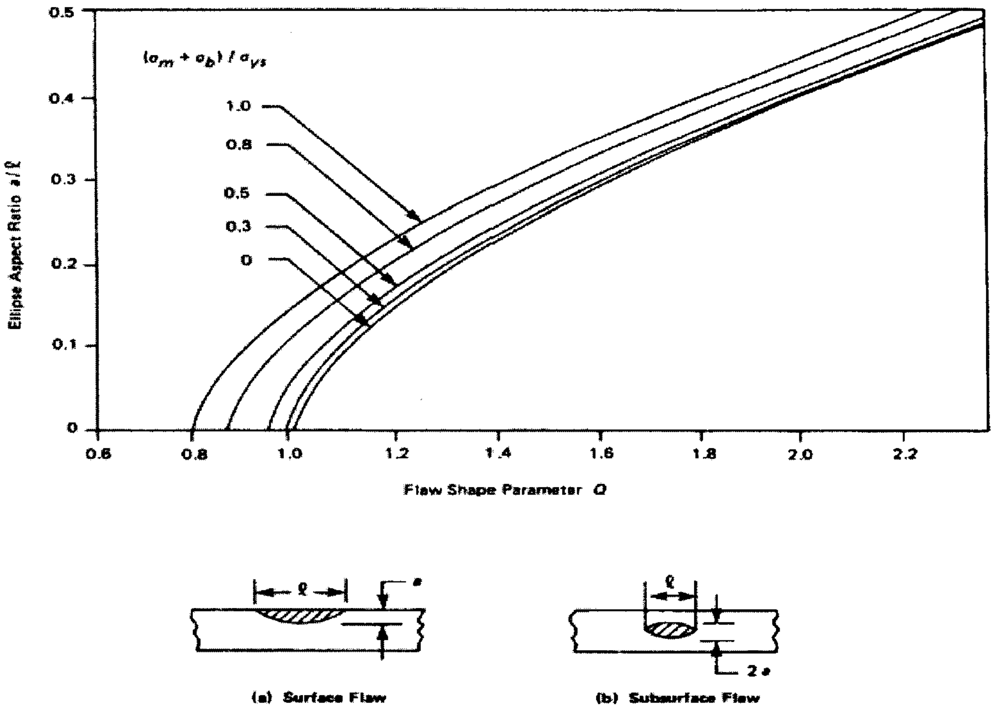
$$P'_b = \frac{6S_m}{\pi}(2\sin\beta - \frac{a}{t}\sin\theta) \qquad [22]$$

where $\beta = \frac{1}{2}(\pi - \frac{a}{t}\theta - \pi\frac{P_m}{3S_m})$ and θ = the half flaw angle [23]

The nominal radius of the 14 NPS schedule 20 pipe is 6.844; therefore the angle θ = 0.6/6.688 = 0.09 (approx). From ASME Section II, Appendix D, S_m = 20 ksi.[37] P_m is conservatively assumed as 0.5 S_m (10 ksi) per Section XI, Appendix C, and the P_b is 14 ksi for this example.Based on that input, the value for β = 0.5(π - (1)0.09 - π (10/60)) = 1.26. So the value P'_b, necessary for failure by incipient plastic collapse from Equation 22 is: $\frac{6(20{,}000psi)}{\pi}(2\sin 1.26 - (1)\sin(0.09)) = 69ksi$. For normal operating conditions, the safety factor, SF, from Section XI, Appendix C is 2.77; so the required stress from Equation 21 is $P'_{b\,(accept)} = SF(P_m + P_b) - P_m$ = (2.77)(24) - 10 = 56.5ksi. Since the stress required for failure with the example through-wall flaw is higher than the required stress for normal operating conditions, the flaw is acceptable structurally. However, since leakage is not acceptable, the item must be repaired before the flaw grows through the pipe wall. The flaw growth rate da/dt in inches per hour for stress corrosion cracking given by Code Case N-513-1, is $da/dt = S_T C K_{max}^n$, where K_{max} (in ksi √in.) is the maximum stress intensity factor for long-term steady state conditions. For intergranular stress corrosion cracking in austenitic stainless steel at temperatures $T \leq 200$ °F, the temperature correction factor, S_T, is 1, and the applicable material constants (for da/dt in in./hr and K in ksi $\sqrt{in.}$) are $n = 2.161$ and $C = 1.79 \times 10^{-8}$. For this example, the K_{max} value will be determined for an elliptical flaw with the simplified form[38] $K = \sigma\sqrt{\pi a/Q}$. Using Figure 11 for an $a/l = 0.125$ and by conservatively assuming $\sigma/\sigma_y = 1$, $Q = 1.0$. To determine the time required for the crack to grow through 80% of the wall thickness, and neglecting changes in crack growth rate due to crack extension, a conservative bounding value of a equal to 0.80(0.312) = 0.250 was used. Then $K = 30\sqrt{\pi(0.25)/1.0} = 26.6$ ksi $\sqrt{in}$. Therefore $da/dt = (1.79 \times 10^{-8})(26.6)^{2.161} = 2.1 \times 10^{-5}$ in/hour. At that rate, the time required for the crack to reach 80% through wall (0.25 inch) is $(0.25\text{-}0.15)/\ 2.1 \times 10^{-5}$ = 4650 hours or approximately 6.5 months. Based on the fitness for service approach, the weld could remain in service without undergoing repair for a time probably sufficient to schedule a maintenance shutdown, allowing the repair to proceed with other planned maintenance activities, and thus reducing the cost of the repair itself and removing it from the critical path shutdown schedule. In a similar manner, flaws can be postulated to exist in various systems, and those flaws evaluated, in order to determine the relative probability of failure for purposes of risk ranking to prioritize inspection and preventive maintenance activities.

4.2.2 FMEA and Fault Tree Methods for Risk Assessment and Ranking

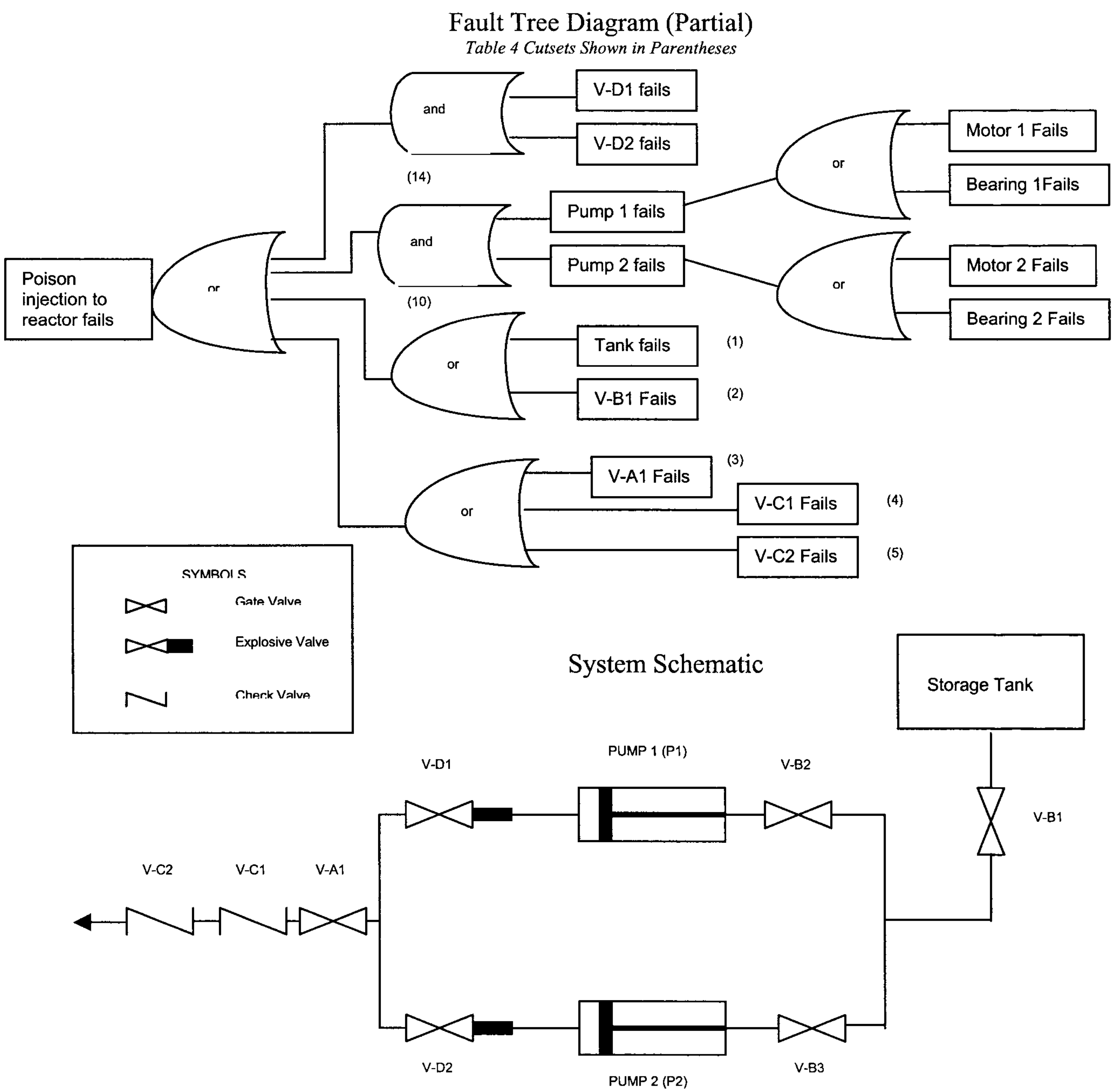
Failure Modes and Effects Analysis (FMEA) and fault trees are widely used tools



Often reproduced figure also previously published in Section XI of the ASME Code

Figure 11. Flaw Shape Parameter[38]

in quantitative risk assessment. An example risk of ranking using FMEA and fault tree analysis was developed for this study. A partial fault tree diagram, along with a system schematic, is shown in Figure 12 for the standby liquid control (SLC) system of a Boiling Water Reactor (BWR) power plant. The diagram is partial because it shows only four event chains, although more are possible through additional event combinations shown in Tables 3 and 4. The SLC system is a redundant safety system that injects boron, in the form of sodium pentaborate, into the nuclear reactor vessel. The sodium pentaborate is referred to as a "poison" because the boron slows the flow of neutrons and, in the



The system schematic is a slightly simplified version of an actual system piping and instrumentation diagram not showing test lines and an additional pair of redundant gate valves downstream of the pumps.

Figure 12. SLC System Fault Tree Example

Table 3. Failure Modes and Effects Analysis for SLC System

Component	Failure Mode (Event)	Effect	Criticality	Mitigation	Failure Mode Probability	Component Failure Probability	Component Reilability	Parallel Reduction - Reliability
Poison Storage Tank	Sudden failure due to rupture	Loss of poison supply	No poison supply	None	1.00E-06	1.00E-06	9.99999E-01	9.99999E-01
Tank Discharge Gate Valve (V-B1)	Wedge disconnect from stem	Flow blockage	No flow to header	None	1.00E-04	1.00E-04	9.99900E-01	9.99900E-01
Pump Suction Gate Valve (V-B2)	Wedge disconnect from stem	Flow blockage	No flow to pump	Redundant Train	1.00E-04	1.00E-04	9.99900E-01	
Pump Suction Gate Valve (V-B3)	Wedge disconnect from stem	Flow blockage	No flow to pump	Redundant Train	1.00E-04	1.00E-04	9.99900E-01	
Pump 1	Motor failure bearing, short	No pumping	No flow to explosive valves	Redundant Train	1.50E-03			
Pump 1	Pump failure bearing seize	No pumping	No flow to explosive valves	Redundant Train	1.00E-03	2.50E-03	9.97502E-01	
Pump 2	Motor failure bearing, short	No pumping	No flow to explosive valves	Redundant Train	1.50E-03			
Pump 2	Pump failure bearing seize	No pumping	No flow to explosive valves	Redundant Train	1.00E-03	2.50E-03	9.97502E-01	
Explosive Valve 1 (V-D1)	Actuator failure	Flow blockage	No flow to isolation valves	Redundant Train	1.00E-02	1.00E-02	9.90000E-01	
Explosive Valve 2 (V-D2)	Actuator failure	Flow blockage	No flow to isolation valves	Redundant Train	1.00E-02	1.00E-02	9.90000E-01	9.99842E-01
Isolation Valve 1 (V-A1)	Wedge disconnect from stem	Flow blockage	No flow to check valves	None	1.00E-04	1.00E-04	9.99900E-01	9.99900E-01
Check Valve 1 (V-C1)	Hinge failure	Partial Flow Blockage	No flow to Check Valve 2	None	1.00E-05	1.00E-05	9.99990E-01	9.99990E-01
Check Valve 2 (V-C2)	Hinge failure	Partial Flow Blockage	No flow to Vessel	None	1.00E-05	1.00E-05	9.99990E-01	9.99990E-01
System Reliability								9.90E-01
System Failure Probability								1.02E-02

(Refer to Figure 12)

quantity injected by the SLC system, stops the reaction altogether. The SLC system is manually actuated and could be used to shutdown the reactor in an emergency if control rods could not be inserted.[39] The SLC system was chosen for this example because it is a simple system with relatively few components. Referring to Figure 12, the top level event for this example is the failure of the system to perform its function, i.e., injection of the sodium pentaborate poison. Simple inspection of the cutsets in a fault tree allows a qualitative assessment of risk, i.e., an assessment where the failure probability (or failure frequency data) is not considered. There are five minimum cutsets shown in the fault tree of Figure 12, where in this simple example a single failure of the storage tank or its release valve, discharge line valve, or the isolation valve would cause the system failure. The cutset type involving Pump 1 and Pump 2 is called second-order, i.e., both pumps must fail to cause a system failure. The cutset involving explosive valves V-D1 and V-D2 is also second order, while the cutset involving and check valves V-C1 and V-C2 is first-order. A qualitative analysis ranks those components whose single failure would cause a system failure higher than those in the second order cutsets. For most systems, the fault trees are sufficiently complex that software programs, such as GADSRAM and UNIRAM, are usually recommended to perform a quantitative analysis.[21] For quantitative risk analysis in this example, the FMEA shown in Table 3 was used. The component failure modes, probabilities, and impact on the system are shown in the FMEA. Note that the failure mode events and probabilities in Table 3 are entirely hypothetical, created for the purpose of illustration, and are completely unrelated to the actual failure modes or reliability of these components. Where consequences are considered as an economic value, they can be included, enabling an economic risk value to be determined. In this example, the system consequence is simply a system failure to inject the boron poison rather than a specific economic consequence value, so a column for consequence (severity) is not used. When more than one failure mode event is considered for a single component, such as the pumps, the component failure probability must be determined based on the combined event probabilities. A further reliability breakdown for the system, including reduction of the parallel portion of the network to support the quantitative evaluation, is provided by the FMEA. Since probability of component failure is considered in the FMEA, risks not apparent from the qualitative analysis of the fault tree become more apparent. The higher failure probability of the explosively actuated valves, V-D1 and V-D2, apparent from inspection of Table 3, shows that those components are the most risk significant as individual components. A more rigorous treatment that considers all the possible events that can lead to the ultimate system failure can identify other components as being highly risk-significant.

Quantitative analysis of the fault tree yields the risk significance of combinations of component failure events. For this example, the minimum cutsets at the component failure event level have been listed in Table 4, along with the individual component event probabilities (failure or success) and the event (failure) probability for the cutset. This simplified example involves 14 minimum cutsets for this system, whereas an actual PRA may involve more than 40 minimum cutsets. Cutsets 1 through 5 involve the failure of a single component, whereas cutsets 6 through 14 (highlighted) involve the failure of two components since those components are in the parallel portion of the network. Table 4

Table 4 Minimum Cutset Probability Model (Component Level)

No.	Tank	P	VB-1	P	VB-2	P	P1	P	V-D1	P	VB-3	P	P2	P	V-D2	P	V-A1	P	V-C1	P	V-C2	P	SYS	P
1	0	1.0E-06	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	F	1.000E-06
2	W	1.0E+00	0	1.0E-04	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	F	1.000E-04
3	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	0	1.00E-04	W	1.0E+00	W	1.0E+00	F	1.000E-04
4	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	0	1.00E-05	W	1.0E+00	F	1.000E-05
5	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	0	1.00E-05	F	1.000E-05
6	W	1.0E+00	W	1.0E+00	0	1.00E-04	W	1.0E+00	W	1.0E+00	0	1.00E-04	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	F	1.000E-08
7	W	1.0E+00	W	1.0E+00	0	1.00E-04	W	1.0E+00	W	1.0E+00	W	1.0E+00	0	2.50E-03	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	F	2.500E-07
8	W	1.0E+00	W	1.0E+00	0	1.00E-04	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	0	1.00E-02	W	1.0E+00	W	1.0E+00	W	1.0E+00	F	1.000E-06
9	W	1.0E+00	W	1.0E+00	W	1.0E+00	0	2.50E-03	W	1.0E+00	0	1.00E-04	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	F	2.500E-07
10	W	1.0E+00	W	1.0E+00	W	1.0E+00	0	2.50E-03	W	1.0E+00	W	1.0E+00	0	2.50E-03	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	F	6.250E-06
11	W	1.0E+00	W	1.0E+00	W	1.0E+00	0	2.50E-03	W	1.0E+00	W	1.0E+00	W	1.0E+00	0	1.00E-02	W	1.0E+00	W	1.0E+00	W	1.0E+00	F	2.500E-05
12	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	0	1.00E-02	0	1.00E-04	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	F	1.000E-06
13	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	0	1.00E-02	W	1.0E+00	0	2.50E-03	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	F	2.500E-05
14	W	1.0E+00	W	1.0E+00	W	1.0E+00	W	1.0E+00	0	1.00E-02	W	1.0E+00	W	1.0E+00	0	1.00E-02	W	1.0E+00	W	1.0E+00	W	1.0E+00	F	1.000E-04

Notes: 0 = component failure, W = component working properly, F = system failure P = Probability of Failure or enter "1" if component is working properly (W) shows the assumed condition of each component in the system, i.e., working properly (W) or failed (F), and the probability of failure for those components assumed to be in the failed condition. The minimum cutsets consist of the components assumed to be failed, and the probability of the minimum cutset is calculated from the failure probabilities of those components. While the probability of an explosive valve failure, either VD-1 or VD-2, is 1 x 10^{-2}, the probability of the minimum cutset is less when the failure probability is calculated by Equation 6 from Section 2.12.1. The minimum cutsets put the probability of component failure events in perspective. Cutset 14, involving the failure of both valves VD-1 and VD-2, has the same probability as cutsets 2 and 3; however, that probability is orders of magnitude lower than the individual component failure probability. On the other hand, the failure events for VD-1 and VD-2 are involved in several other cutsets, all of which must be considered to evaluate the risk of a specific component failure relative to that of other components.

Quantitative risk assessment for complex systems frequently uses standard risk importance measures. As mentioned in Section 2.5, the Fussell-Vessely (FV) Importance measure considers all the minimum cutsets in which a specific contributor is a part as a fraction of the total number of cutsets. The standard formula for FV[22] is shown below:

$$FV = \frac{Rc}{R} = \frac{\sum_{c's} Mi}{\sum Mi} \qquad [24]$$

where R is the total risk measure, $\sum Mi$ is the sum of the evaluated minimum cutsets, Rc is the contribution involving a specific contributor c, and $\sum_{c's} Mi$ is the sum of the system failure probabilities for the minimal c's, cutsets with contributor c. Calculation of the FV enables risk ranking that represents the significance of an individual component failure when it occurs in multiple scenarios represented in the cutsets of the fault tree diagram. Another measure often used is Risk Reduction Worth (RRW), a relative measure of benefit for guaranteed reliability of the component. For FV values less than 0.1, the RRW value is equal to 1/(1-FV).[21] Both the FV and RRW can be used to identify the components having the greatest impact on the system performance and can be used for sensitivity analysis by observing the effect of changing the failure probabilities for various components. The FV Importances for the failures of individual components in the SLC system are shown in Table 5. While the failure probability of cutset 14 involving explosive valves VD-1 and VD-2 is equivalent to cutsets 2 and 3 involving VB-1 and VA-1, valves VD-1 and VD-2 are more risk-significant because they are in the parallel network and involved in more minimum cutsets. The relative risk-significance is quantified by the value of the FV Importance.

Actions to improve the reliability of those components, such as modifications or periodic refurbishment (assuming the value for the probability of failure was related to the age or operating time for the component) would have the greatest impact on the system reliability.

If the risk to the system exceeds an acceptable value for corrective maintenance, a preventive maintenance strategy is developed. Although the risk significance can be used to establish the initial preventive maintenance priority, further knowledge of the basis for the failure probability is needed in order to determine the maintenance strategy.

Table 5. Fussell-Vesely Importance Measures For Component Failures

Component Failure (contributor)	Cutsets Containing Component	$\sum Mi$	$\sum_{c's} Mi$	*FV*
Tank	1	3.80E-04	1.00E-06	2.63E-03
VB-1	2	3.80E-04	1.00E-04	2.63E-01
VB-2	6, 7, 8	3.80E-04	1.26E-06	3.32E-03
P1	9, 10, 11	3.80E-04	3.15E-05	8.29E-02
V-D1	12, 13, 14	3.80E-04	1.26E-04	3.32E-01
VB-3	6, 9, 12	3.80E-04	1.26E-06	3.32E-03
P2	7, 10, 13	3.80E-04	3.15E-05	8.29E-02
V-D2	8, 11, 14	3.80E-04	1.26E-04	3.32E-01
V-A1	3	3.80E-04	1.00E-04	2.63E-01
V-C1	4	3.80E-04	1.00E-05	2.63E-02
V-C2	5	3.80E-04	1.00E-05	2.63E-02

For example, if the failure probability is time dependent, periodic replacement or refurbishment of the component can be implemented. In some cases, the periodic maintenance strategy may include a preliminary inspection of the component to determine the need for further action. A revised failure probability can be calculated based on increased component reliability resulting from preventive maintenance, using methods such as the Structural Reliability and Risk Assessment (SRRA) process discussed in Section 2.10. The FMEA and FV calculations can then be revised in an iterative manner to incorporate failure probability changes as preventive maintenance measures are developed for various components.

The SLC system risk evaluation illustrates the methodology applied on the system level. More complex systems and multiple systems may be evaluated in a similar manner to determine their contribution to a consequence such as plant unavailability or specific economic loss related to the system or multiple system failure.

4.2.3 Representing Risk Graphically

Another useful tool in risk evaluation is to develop a risk plot, similar to the F-N curve mentioned in Section 2.2. An example of a basic plot using the constant risk technique, illustrating economic risk, is shown in Figure 13. The example was created for using constant risks of $1,000 per year and $10,000 per year. The risk plot can be used not only to graphically illustrate the economic risk of specific events or component failures, but it also allows the user to determine an acceptable risk level (shown as the constant risk lines) and group the events or component failures according to their risk. In the example, if a risk of $10,000 per year is acceptable, that means that for an event with a probability of 1 in 1,000, i.e, 10^{-3}, an acceptable consequence for that event would be ten million (10^{7}) dollars. Uncertainty can be shown in the risk plot by bracketing values, as represented by the rectangles in Figure 13. The ranking method of MIL-STD-882 also provides a means of grouping events quantitatively by risk categories. MIL-STD-882 does not provide a constant-risk categorization, but instead groups the risks by economic threshold category as high, serious, medium, and low.

For this study, a graphic representation of the risk categories described in MIL-STD-882 was created as shown in Figure 14. Refering to Figure 14, an event with a "critical" consequence of $800,000 and an "occasional" probability between 1 in 100 and 1 in 1000, is represented by a "serious" risk category of 6 in MIL-STD-882.

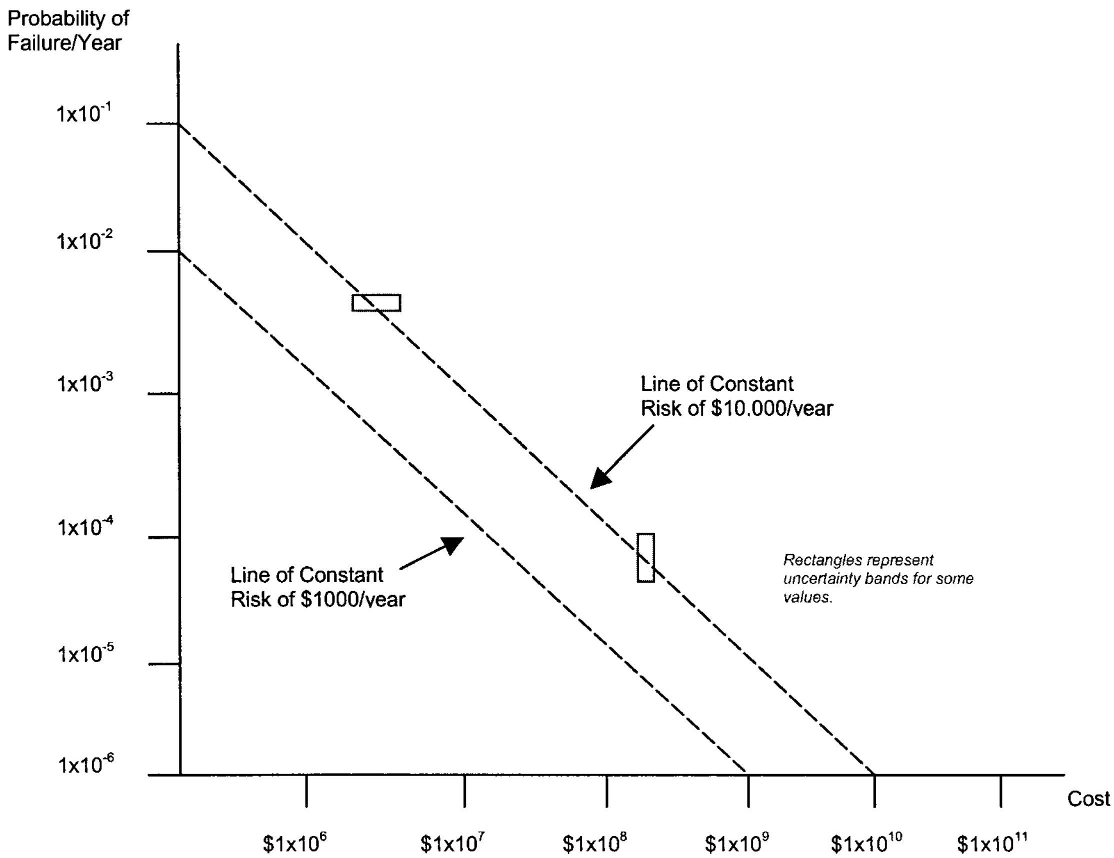
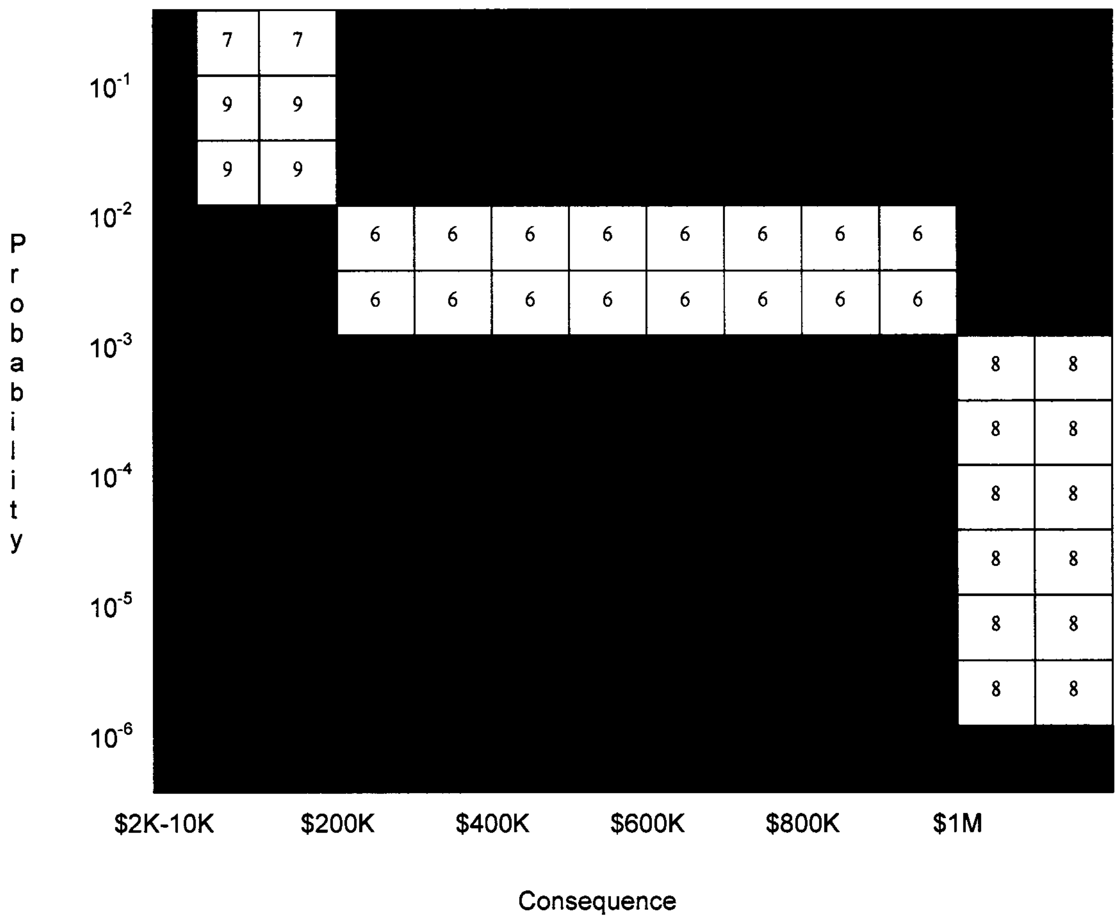


Figure 13. Basic Risk Plot



MIL-STD-882 Risk Categories

Consequence:	Catastrophic (> $1M)	Critical ($1M>c>$200K)	Marginal ($200K>c>$10K)	Negligible ($10K>c>$2K)
Probability:				
Frequent ($p> 10^{-1}$)			7	
Probable ($10^{-1}>p>10^{-2}$)			9	
Occasional ($10^{-2}>p>10^{-3}$)		6		
Remote ($10^{-3}>p>10^{-6}$)	8			
Improbable ($p<10^{-6}$)				

	SERIOUS	

Figure 14. Risk Plot with MIL-STD-882 Risk Categories

4.3 Case Studies

The case studies presented in this section describe actual maintenance scenarios involving ongoing and past projects, some involving industry-wide maintenance issues as well as facility-specific problems. Where case history data have been obtained from sources other than interviews with the participants or the author's own experience, the sources have been cited and are included in the references. The case histories are presented to illustrate applications of risk-based evaluation methods to maintenance engineering and management issues.

4.3.1 Single-Phase Erosion-Corrosion of Feedwater System Piping

4.3.1.1 Introduction and Background

This case study examines a high-consequence failure scenario and preventive maintenance program utilizing an inspection program based on risk-ranking. The case study also includes a practical example of evaluation of material degradation using a fitness for service approach.

On December 9, 1986, a feedwater piping break at the Surry Nuclear Power Station, a Pressurized Water Reactor (PWR) plant in Virginia, resulted in the death of four maintenance workers and the serious injury of four others.[15] The cause of this catastrophe was erosion-corrosion of an 18 inch, schedule XS elbow, nominal wall thickness of 0.500 in., about one foot downstream of a 24 inch main feedwater suction header. When the gradual material loss from erosion-corrosion finally opened a pin-hole in the pipe, the energy from the 374 °F water flashing to steam blew out a two by four foot section of the wall of the piping.[15, 40, 41] Since the feedwater is a secondary system in a PWR, and the failure occurred outside of the primary containment, there was no release of radioactivity to the environment or adverse effect on the reactor coolant system. Nevertheless, it was the only accident in the history of commercial U. S. nuclear power generation resulting in a loss of life and resulted in swift action by both the Institute of Nuclear Power Operations (INPO), an industry self-regulating body, and the Nuclear Regulatory Commission (NRC).

The mechanism for this failure was fairly well understood and previous failures, albeit of lower consequence, had occurred at other domestic nuclear plants.[42] Although the means existed to evaluate the risk of this type of failure, the focus of the Probabilistic Risk Assessment for a nuclear power plant is on potential damage to the reactor core, and the more conventional risks posed by erosion-corrosion in a secondary system were not addressed. Since the consequences of the Surry failure were clearly catastrophic, inspections of susceptible piping were made mandatory, first by INPO and then by the NRC, within approximately 6 months of the event.

4.3.1.2 Risk Assessment and Inspections in Support of Preventive Maintenance

The Surry feedwater elbow piping material was reported as ASTM A234, Class 301 carbon steel. A234 has been a typical material used for both steam and water piping systems for temperatures up to 750° F.[43] The operating parameters for the system were as shown in Table 6.

Table 6: Surry Unit 2 Feedwater System Operating Parameters[40]

Temperature	374 °F
Pressure	376 psig
Bulk Fluid Velocity (average)	17.6 ft/sec
pH	8.8 - 9.2
Oxygen (average)	4 ppb

Despite its widespread use in piping systems, plain, low carbon steel piping is known to have only fair corrosion resistance, at best, for single and dual phase water systems. Because such piping is low-cost, the piping engineer compensates for the lack of corrosion resistance by adding a corrosion allowance. However, corrosion allowances provided by the manufacturers are based on general corrosion rates and are insufficient when severe erosion-corrosion conditions are prevalent. Low alloy steels containing small amounts of chromium dramatically improve resistance to erosion-corrosion. A commercial low alloy steel containing 2-1/4% Chromium and 1% Molybdenum (2-1/4 Cr- 1 Mo) has a resistance 4 times greater than plain carbon steel.[42] However, due to the lack of awareness and data concerning the susceptibility of materials to erosion-corrosion in the 1960s and early 1970s, when most of these plants were designed, and the higher cost of low alloy steels, they were not generally used. In addition to material, water chemistry, temperature, and fluid velocity (as well as turbulent flow conditions) are the key factors in predicting susceptibility to erosion-corrosion.

Two types of oxides may be formed on the surface of plain carbon steel in the presence of water, magnetite (Fe_3O_4) and hematite (Fe_2O_3). When water is deoxygenated (oxygen < 5 ppb) magnetite (Fe_3O_4) formation is favored over hematite (Fe_2O_3). Under general corrosion conditions, either oxide can provide a measure of protection for the substrate and reduce the corrosion rate over time, but when significant water flow conditions exist, magnetite, which is much more soluble in water than hematite, does not adhere sufficiently to protect the substrate from further corrosion. That is, the magnetite layer is eroded by the water flow and the exposed metal continues to corrode. When the oxygen levels are higher (oxygen $\geq$ 5 ppb) a positive electrochemical potential exists which favors hematite formation.[44] Under alkaline conditions (pH > 7) typical of powerplant water systems, the Iron-Iron Oxide Pourbaix diagram shows that the oxide (hematite) is stable when the electrochemical potential remains above 0.3 volts.[45] Further, research has shown that corrosion rates are reduced as the pH rises, at least to 9.7.[42, 44]

Water temperature and flow rate profoundly affect susceptibility to erosion-corrosion. Research by I. S. Woolsley and others,[44] as shown in Figure 15, indicates significantly increased material loss rates occur between about 220 °F and 320 °F with a maximum in the 260 °F to 290 °F range. The general trend toward increasing material loss with increases in flow is also illustrated in Figure 15. An even greater predictor of damage is turbulence, controlled by the local piping and component geometry. Fluid dynamics equations have been developed with empirical correlations to account for turbulent flow, and are now incorporated in CHECKMATE™, a widely used erosion-corrosion damage prediction program available from the Electric Power Research Institute. For purposes of this case study, a less rigorous semi-quantitative approach, adapted from a method developed by Technicon Enterprises as an initial response to the

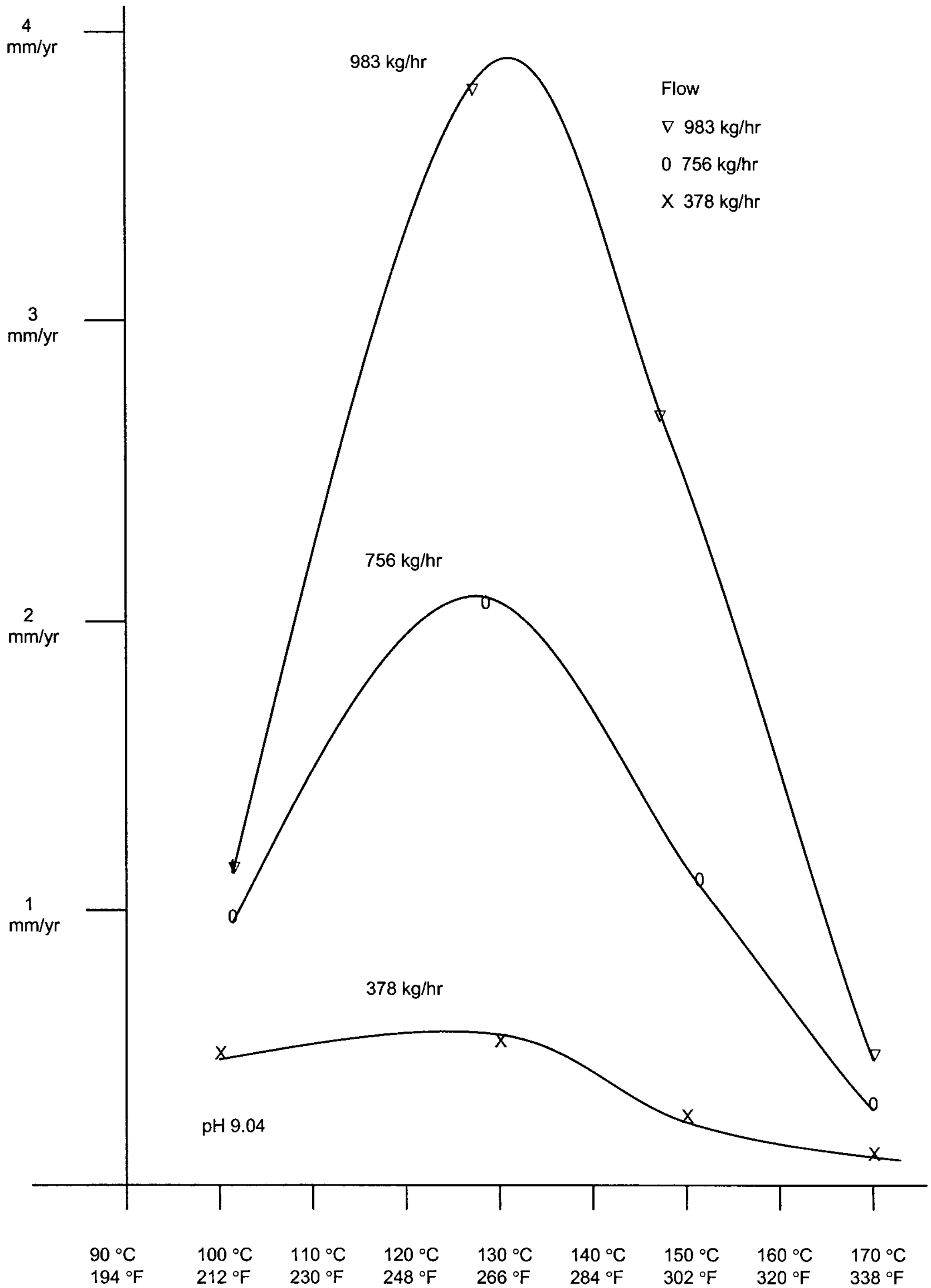


Figure 15. Effects of Flow and Temperature on Erosion-Corrosion (After Woosley)[44]

events at Surry, was used to determine the relative damage probability.

The plain carbon steel feedwater piping in a PWR power plant supplies water to a steam-generator heat exchanger pressure vessel with a low alloy steel shell and containing nickel alloy (Alloy 600) tubes inside. Reactor water at 660 °F flowing on the inside of the tubes causes the water on the outside of the tubes, supplied by the feedwater system, to boil and produce steam, which then flows out of the vessel into the main steam lines to the turbine. The feedwater is called a secondary system because it is isolated from the closed-loop primary reactor water that carries radioactive contaminants. As a practical matter, while water chemistry does have a profound effect on erosion-corrosion, the PWR secondary water chemistry is maintained alkaline (nominal pH of 9) with very low dissolved oxygen (< 5 ppb) in a manner developed to protect the Alloy 600 tubes, containing primary reactor coolant water, from corrosion.[46] A leak due to corrosion of the steam generator tubes can result in release of small amounts of primary water containing radioactive contamination. Such a leak usually requires an immediate plant shutdown. The plant must remain shutdown until testing each of the approximately 3,000 tubes in a typical steam generator heat exchanger reveals the leaking tube, which then must be taken out of service (plugged) since individual tubes cannot be replaced. Unfortunately, the pH and oxygen concentration values maintained to protect the tubes render the plain carbon steel feedwater piping susceptible to erosion-corrosion. Since the economic consequences and public safety concerns are so high for the failure of steam generator heat-exchanger tubes, the erosion-corrosion problems in the feedwater system must be handled as an ongoing maintenance issue.

Erosion-corrosion is required by regulation to be handled as preventive maintenance that relies on an inspection program, utilizing ultrasonic thickness measurement techniques, to identify pipe and fittings with wall thinning for replacement prior to failure. The degraded components are then replaced during planned shutdowns, sometimes using components made of more resistant low-alloy steel, while reactor refueling operations are performed. Not only the feedwater system, but many other plant systems are evaluated for areas with a reasonably high probability of experiencing significant erosion-corrosion. In operating plants today the CHECKMATE™ computer program is used and all the possible factors, material, water chemistry, temperature, flow, and turbulence are input. The initial evaluations in response to the Surry failure were much simpler. Since the material was usually plain carbon steel and the water chemistry was maintained within a specific range to protect the Alloy 600 steam generator heat exchanger tubes, those factors were considered constants and were not used in the relative probability ranking of systems and areas of the piping for inspection. Illustrated in Table 7 is a relative risk-ranking scheme used for early evaluations performed in late 1986 and early 1987 . Only the relative contribution of each factor to the probability of failure is reflected in Table 7. In other words, the relative consequence of failure of all components is assumed to be equal and have a value of one because the examinations were mandated by regulation and examinations could not be waived or reduced by economic consequence analysis. While simple and expedient, the scheme was effective in identifying a number of areas at San Onofre Unit 1 that did have erosion-corrosion induced wall thinning requiring corrective action during the life of the plant. For other reasons, that plant is now being decommissioned. In the ranking system of Table 7,

Table 7. Relative Probability Ranking for Secondary System Piping

Factor	Relative Risk	Factor	Relative Risk
Fluid Velocity		Piping Configuration	
20-30 ft/sec	5	Control valve	10
20-25 ft/sec	4	Splitting tee	10
15-20 ft/sec	3	180° bends	10
10-15 ft/sec	2	Check valve	8
5-10 ft/sec	1	Globe valve	8
Temperature		Flow orifice	8
260 - 320 °F	5	Closely spaced less than 10 pipe diameters apart	8
245 - 260 °F	4		
320 - 350 °F	4	90° bend	6
230 - 245 °F	3	Elbow	6
350 - 390 °F	3	Butterfly valve	2
210 - 230 °F	2	Reducer	4
390 - 410 °F	2	Gate valve (open/closed)	2
190 - 210 °F	1	Butt weld in straight pipe	2
410 - 450 °F	1		
Below 190 °F	0		
Above 450 °F	0		
Note: This table is provided as an example only, may be not be accurate and is not intended for use.			

since turbulent flow was associated with the configuration of the piping system, various components, fittings, and connections were given a relative ranking related to their associated flow characteristics. Also noteworthy was the risk factor of closely spaced components. Under this ranking system, a globe valve connected to a reducer would not only receive a rating of 12 for the combination of the reducer and the globe valve, but would also receive an additional rating of 8 for the close spacing between the components, resulting in a total rating of 20 for the configuration. At San Onofre Unit 1, wall thinning of reducers associated with the valve to reducer combination resulted in the greatest number of corrective actions, i.e., repairs or replacements. The sum of the ratings of temperature, flow velocity, and configuration resulted in the relative ranking. The areas with the highest ranking were examined first in order to identify potential failures before they occurred. The example shows that in the practical application of risk evaluation, rigorous determination of factors such as specific material loss rates may not be necessary to achieve the desired result. That principle is also applied to ranking on the basis of expert panel recommendations using processes such as the HAZOP process described in Section 2.6.

4.3.1.3 Maintenance Needs and Fitness for Service Evaluation

When local pipe wall thinning due to erosion-corrosion is detected , the area must be evaluated to determine the appropriate maintenance strategies. Strategies include a number of options, (1) leave the item in service and monitor for further degradation, (2) leave the item in service for some designated period prior to repair or replacement, (3) perform immediate repair, or (4) perform immediate replacement. The simplest evaluation of such wall thinning is to compare the remaining wall thickness with the minimum design wall thickness; however, that approach is very conservative for localized wall thinning and is likely to result in untimely repairs or replacements. For the risk-based maintenance approach, a fitness for service evaluation should be performed in most cases to determine if the item is suitable for continued service. Most inspections are performed during planned system shutdowns. If the item can remain in service until the next planned system shutdown, there will be no schedule impact on the current maintenance shutdown and no unplanned costs, i.e., the economic consequence of the event is lowered. As described in Section 2.8, standardized procedures for fitness for service evaluations have been developed for degraded material conditions including localized wall thinning.

The following example was developed for this study to illustrate a fitness for service evaluation for erosion-corrosion. Piping similar to the failed feedwater pipe of the Surry plant was used, but the dimensions of wall thinning were hypothetical, though realistic, values selected for the example. The objective of this example is to determine if an elbow, thinned by erosion-corrosion, must be replaced immediately or has sufficient remaining wall thickness to be periodically monitored and replaced at a later time. As was often the case, the thickness of a susceptible pipe or fitting was measured for the first time after five or ten years of service, and if the pipe or fitting was acceptable for service, periodic measurements were used to monitor the rate of wall thinning for maintenance planning. Consider a long radius elbow with localized material loss, evaluated in accordance with the Level 1 Assessment methodology of API 579,[26] and assuming that the elbow is subject only to pressure loading design conditions similar to the Surry power plant piping. The API 579 equations used in this example are shown in Appendix D. The configuration is an elbow, NPS 12, outside diameter (D_o) of 12.75 inch, Schedule 40, nominal wall thickness of 0.406 inch, fabricated from A234 Grade WCB material, selected because Class 301, reported as the Surry material is not a current ASTM designation. The design pressure is 350 psig and the design temperature is 400 °F. A thinned wall area measuring 7 inches in the longitudinal direction and 3 inches in the circumferential direction is located in approximately the middle third of the elbow, as shown in Figure 16, and has an ultrasonically measured minimum thickness reading (t_{mm}) of 0.15 inch. The centerline bend radius (R_b) is 18 inches and the mean radius of the pipe (R_m) is 5.97 inches. A flaw in the center section of the elbow has a position angle of 90°. So the Lorenz factor (L_f), which relates the stress magnitude in an elbow to that of a straight pipe, as determined by API 579 Equation A.198 is 0.875. From the Construction Code, American National Standards Institute ANSI/ASME B31.1, the allowable stress is 15,000 psi, and the weld joint efficiency is 1.0, assuming seamless pipe.[47] The nearest structural discontinuity is a pipe support located 20 inches away from the edge of the thinned area.

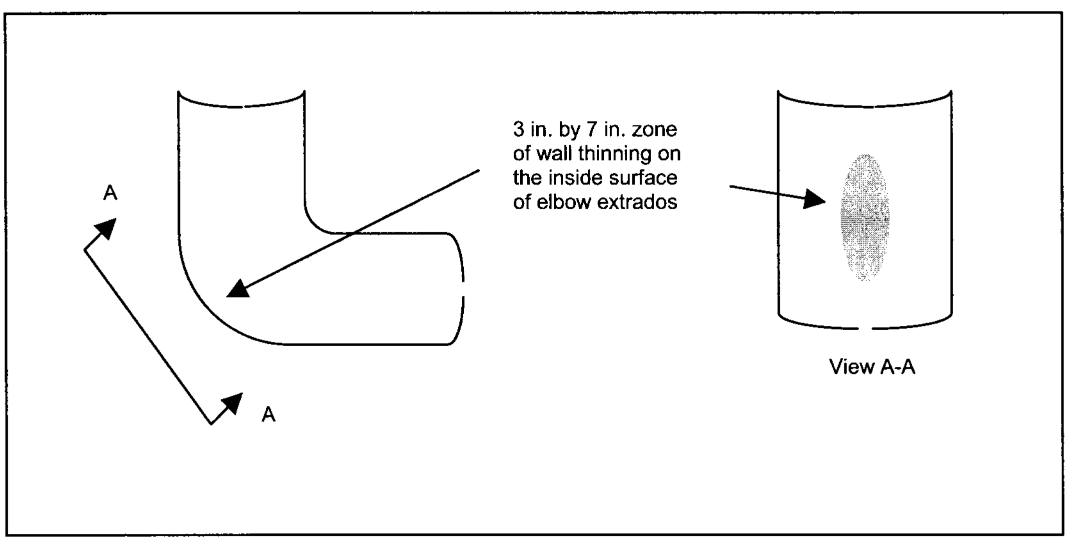


Figure 16. Example Location of Erosion-Corrosion Induced Wall Thinning

The minimum wall thickness based on the circumferential stress by API 579 Equation A.193 is:

$$t^{c}_{min} = \frac{PD_o}{2\left(\frac{SE}{L_f} + PY\right)} + MA \qquad [25]$$

where S = the allowable stress, E = joint efficiency (assumed to be 1 for a radiographed weld), P = pressure, Y = coefficient for temperatures, and D_o = outside diameter. The term MA is a mechanical allowance for threaded components and is zero for welded components. The Y coefficient for temperatures less than 900 °F is 0.4 from B31.1, Table 102.4.6(B.1.1)[47]. The t^C min for the example case is:

$$t^{C}_{min} = \frac{(350)(12.75)}{2((15000/0.872)(1.0) + (350)(0.4))} = 0.129 \text{ in.}$$

The minimum wall thickness based on the longitudinal stress (t^L_{min}) by API 579 Equation A.187 is:

$$t^{L}_{min} = \frac{PD_o}{4(SE + PY)} + t_{sl} + MA \qquad [26]$$

Since there are no supplemental loads, i.e., only pressure loading is considered, the supplemental thickness (t^{sl}) is zero so the t^L_{min} for the example case is:

$$t^{L} \min = \frac{(350)(12.75)}{4((15000)(1.0) + (350)(0.4))} = 0.074 \text{ in.}$$

The larger of t^L_{min} or t^C_{min} is the minimum wall thickness for evaluation (t_{min}) by API 579 Equation A.190, so t_{min} = 0.129.

The Level 1 flaw evaluation procedure of API 597 for localized wall thinning utilizes two parameters, the remaining thickness ratio (R_t), and the shell parameter (λ). Given the t_{mm} of 0.15 inch from above, a future corrosion allowance (FCA) of 0.01 inches, a value determined conservatively to allow for additional material loss prior to a future scheduled repair/replacement, a metal loss area extending 7 in. in the longitudinal direction (the value of *s*) as shown in Figure 16, and an inside diameter (D) of 11.938, the parameters R_t, and λ and are calculated as follows from API 579 Equations 5.3, and 5.4, respectively:

$$R_t = \frac{t_{mm} - FCA}{t_{min}} = \frac{0.15 - 0.01}{0.129} = 1.109 in. \qquad [27]$$

$$\lambda = \frac{1.285s}{\sqrt{Dt_{\min}}} = \frac{1.285(7)}{\sqrt{11.938(0.129)}} = 7.25 in. \qquad [28]$$

The limiting flaw size criteria of API 597 are satisfied if $R_t \geq 0.2$ and the distance to the nearest structural discontinuity (L_{msd}) is greater than the value of the expression:

$$L_{msd} = 1.8\sqrt{Dt_{min}} = 1.8\sqrt{11.938(0.127)} = 2.23 \qquad [29]$$

and t_{min} - FCA $\geq$ 0.1 inch. Since Rt = 1.109 in., L_{msd} = 20 inches, and t_{min} - FCA = 0.129 - 0.01 = 0.119, all three limiting flaw size criteria are satisfied.

Since the limiting flaw size criteria are satisfied, the longitudinal and circumferential extents of the flaw are evaluated using the screening criteria curves of API 579 Figures 5.6 and 5.7, respectively, which are shown in Appendix D. The longitudinal screening criteria curve is based on a remaining strength factor of 0.90.

From $R_t = 1.109$ and $\lambda = 7.25$, the extent of metal loss in the longitudinal direction satisfies the Figure 5.6 screening criteria. The circumferential screening criteria are described in API 579 Equations 5.64 and 5.65. Since the circumferential extent of the metal loss (c) is 3 inches, the c/D value is 3/11.938 = 0.251. Given the acceptable R_t = 0.2, as determined by Equation 5.64, the extent of metal loss in the circumferential direction also satisfies the screening criteria of Figure 5.7.

In the example situation, the elbow was determined to be fit for service and the repair/replacement can be postponed and the elbow periodically monitored. The postponement reduces the economic consequence of the repair/replacement since the life of the elbow has been extended, the costs are postponed, and any schedule delays for the current shutdown are avoided. In the API 597 method, the assumptions made for the future corrosion allowance are a key factor, but if they are based on sound engineering, including ultrasonic thickness monitoring data compiled over several shutdown periods, the corrosion allowance for a single shutdown cycle can be projected with confidence.

4.3.2 Maintenance of Manipulators for Remote Handling

4.3.2.1 Introduction and Background

The manipulator is presented as an example of risk-based maintenance engineering and management at the component level. This case study is useful since both technical detail and expert opinion were available. The remote manipulator is a tool used as a replacement for the technician's hands when hazardous conditions of parts or materials prevent handling the material directly. Typical manipulators, as shown in

Figure 17, may be used to handle parts and materials when chemical, biological, radiological, or other hazards exist. While a number of other risk-based issues could be explored with respect to such hazards, this discussion is focused on the manipulators themselves and their maintenance.

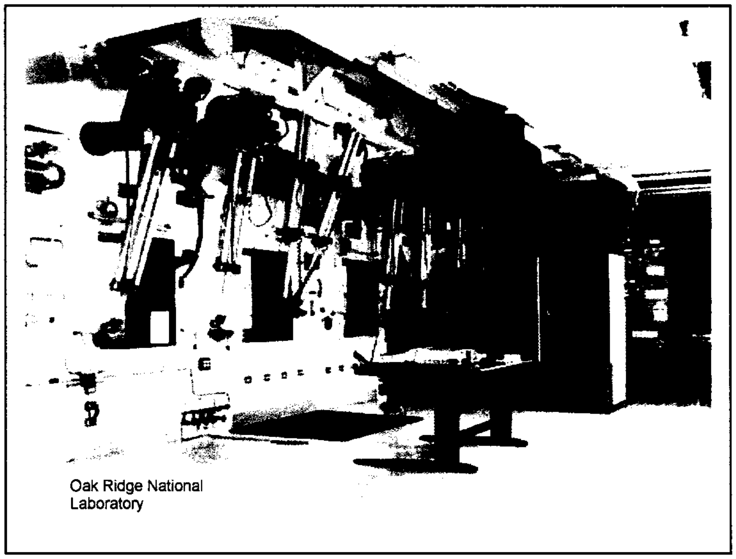


Figure 17. Typical Manipulators for Remote Handling of Hazardous Materials

Most of the operating experience data and other input for this case history was taken from an expert, the Chief Facilities Technologist at a radioactive materials handling facility. At this facility, radioactive materials are handled in closed compartments called hot cells. The cells are shielded to protect the technicians from radiation, primarily gamma, resulting from activation of some alloying elements, such as cobalt, present in some of the parts handled at the facility. Those elements have been activated by exposure to neutron radiation over long periods of service of a part in a nuclear reactor.

Each cell contains at least two manipulators, operated manually by the technicians. At the end of long tubular arms, the manipulators have clamps connected by a joint that permits complex motion similar to that of a wrist. The motion of the technician's hand on the manipulator-handling unit is translated to the joint and clamp by means of a metal cable and tape assembly operating through multiple pulleys. Those metal cables and tapes are subject to failure and must be replaced periodically. The primary damage mechanism is wear, but overload is reported to be a contributing factor, and for one cell in this case study, corrosion appears to be an active damage mechanism as well. A failed cable and mechanism are shown in Figure 18. At the time the study was conducted, an explicit preventive maintenance was not a scheduled activity, although sometimes the technicians request maintenance prior to failure because they detect a change in the "feel" of the manipulator. A pair of manipulators is generally required to perform work operations in the cell, and there are two pairs of manipulators in each cell, one pair positioned for work at each of two windows in the cell. If one manipulator fails, work operations could continue using the other pair of manipulators and only one window of the cell. However, depending on the complexity of the project that mode of operation may cause a slowdown in the work and could impact a project schedule. Note that the work performed in the cells is not line production, but individual projects of varying duration and complexity.

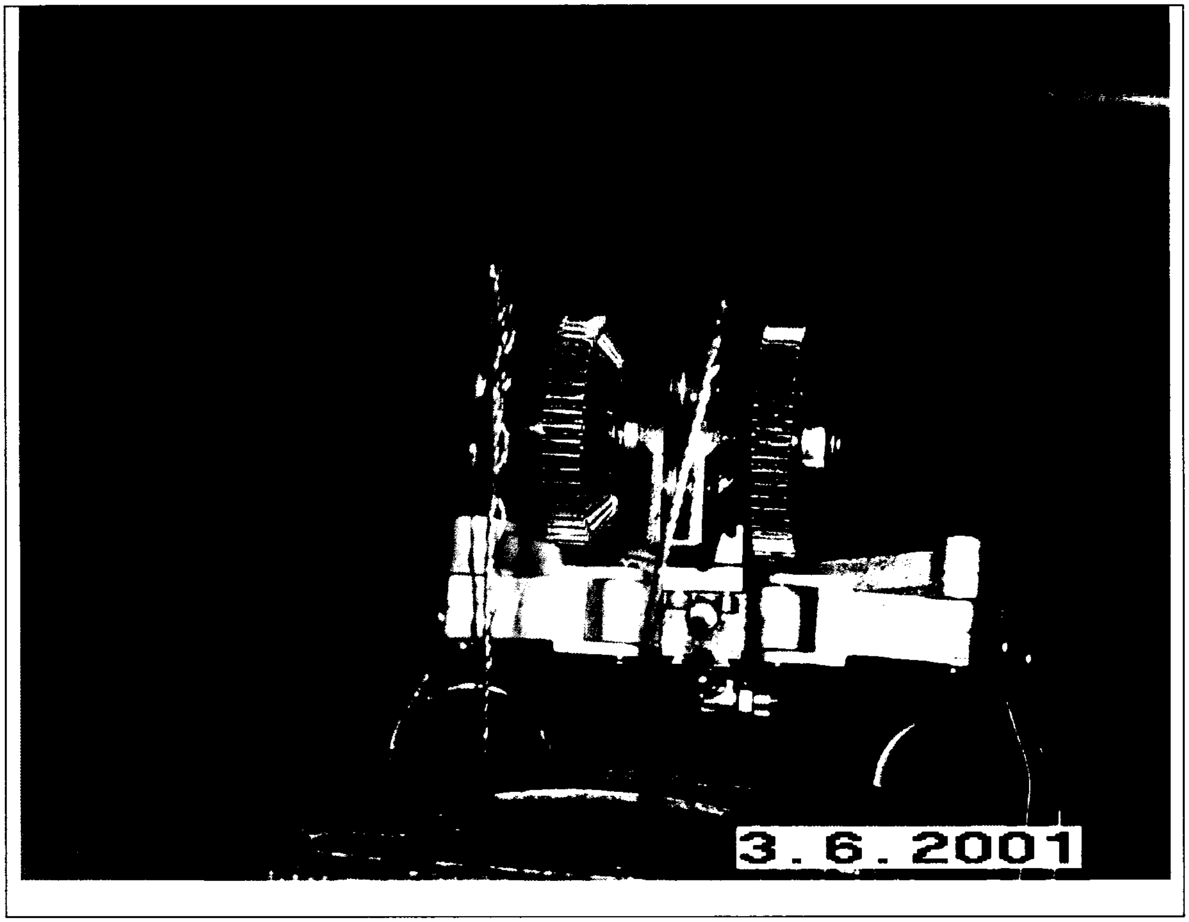


Figure 18. Failed Manipulator Cable & Mechanism

4.3.2.2 Manipulator Maintenance Risk Assessment

For a relatively small system, such as the manipulator, a qualitative risk analysis, which would rank various components of the system, is unnecessary. Similarly, the quantitative risks associated with the failure of this relatively small system may be easily assessed. To determine consequence a number of factors must be considered, but secondary economic effects, such as the likelihood of injury or death, are not possible consequences so the determination is somewhat simplified. Since each cell is equipped with two manipulators, the failure of a single manipulator would not automatically stop work; however, because work is slowed significantly, an assumption can be made that it is desirable to repair the failed manipulator to service at the earliest possible time.The corrective maintenance process involves three stages: preparation, removal or "pulling" of the failed manipulator, and replacement with a functional manipulator.

Since several failures occur every year, the economic consequence of manipulator failures in a year is determined as slightly over $10,000 as shown in Table 8. The cost of production time used for the MDT is determined based on a commercial billing rate, i.e., an assumed loss, and the cost of the actual maintenance is based solely on labor cost since the cost of repair parts is very low and unaffected by variables such as staffing.

The total maintenance down time (MDT) includes one-half hour of cell production time completely lost in the maintenance process to remove the manipulator and replace it with a spare, i.e, the logistics delay time, (LDT). As shown in Figure 17, each cell window has two manipulators; when one is out of service some production can continue with a single manipulator, but at a reduced level. The maintenance operation requires two trained technicians who must be reassigned from other ongoing work. Since maintenance technicians must be scheduled to do the replacement, it is estimated that there are sixteen hours of reduced production due to the availability of only a single manipulator; this is the administrative down time (ADT). So, as shown in Table 8, the half-hour of LDT is valued at $150 per occurrence, and the 16 hours of ADT is valued at $2400. The combined LDT and ADT is the MDT, valued at $2550.

Table 8. Manipulator Maintenance

Activity		Man-Hours	Value/Hr	Cost per occurrence	Frequency Failures per year	Annual Costs (indexed)		Remaining Cell Life (years)	Present Worth of Lifetime Costs*	
Production Stopped	LDT	0.5	$300	$150	3 per year	LDT+ ADT	$7650	10	LDT+ ADT	$65640
Production Slowed	ADT	16	$150	$2400						
Restore Manipulator to Cell (2 repair technicians)	MTTR	4	$50	$200		MTTR	$3000		MTTR	$25740
Rebuild Manipulator		16	$50	$800						
Combined MDT (LDT+ADT)				$2550		Total: $10650			$91380	

* Present worth calculated by discounting using an interest rate of 7% and an inflation rate of 4%[48]

The mean time to repair (MTTR), involves both replacement and repair of the defective manipulator. The repair is done in a separate staging area and does not directly impact production, but its cost is a consequence. As shown in Table 8, the MTTR consequence for each occurrence consists of four hours of restoration time valued at \$200 and 16 hours of rebuilding time valued at \$800. Since there are approximately three manipulator failures per year, the MDT (LDT + ADT) cost is \$7,650 per year and the MTTR cost is \$3,000 per year, for a total annual manipulator maintenance cost of \$10,650. The economic consequence together with the known failure probability, based on operating experience, enable the relative risk to be determined. Although an amount of just over \$10,000 only translates to the lower end of marginal severity per MIL-STD-882, its frequent probability, i.e., a probability of occurrence greater than 10^{-1} in the item's life, results in a relative risk level of seven ("serious") on a scale of one to twenty. The risk ranking allows the maintenance item to be considered along with others to establish priorities among the maintenance items. As shown in Table 8, a specific value of the economic risk can be expressed as the present worth of the corrective maintenance costs over a projected remaining cell life of 10 years. The present worth of the 10 years of MDT (LDT + ADT) costs is \$65,640 and the present worth of 10 years of MTTR cost is \$25,740, for a total present worth of \$91,380. That value of \$91,000 provides a basis upon which decisions regarding specific maintenance options may be made.

For purposes of this case study, it was assumed that the "serious" risk-level drives development of a maintenance strategy to reduce costs. The maintenance strategy options are considered as a function of the economic consequence of the manipulator failure.

4.3.2.3 Manipulator Maintenance Needs and Scheduling Analysis

The maintenance needs and scheduling analysis considers both corrective and preventive maintenance approaches. The preventive maintenance reduces the frequency component of risk, MTTF. However, even if no preventive maintenance is exercised, any maintenance optimization reducing MDT or MTTR will still reduce the consequence component of risk. So the available options are considered within the context of both the corrective and preventive approach.

A formal Failure Modes and Effects Analysis (FMEA) was not required for the manipulator since the available operating experience information, as supplied by the expert, cited only 2 failure modes, failure of the tape and failure of the cable. A preventive measure to reduce tape failures was actually developed by the Chief Maintenance Technologist. The primary damage mechanism was abrasive wear caused by the tape, a type 304 stainless steel, rubbing against a pin of unknown composition, but apparently much harder than the type 304 tape. As reinforcement, an additional layer of tape was added at the pin joint; that design change did not eliminate the wear problem, but it did reduce the MTTF. The cable failures are also initiated by wear from the cable sliding against the pulley as shown in Figure 18. The cables are comprised of wire rope, which is typically a 1074 plain, high-carbon steel. According to the literature, the abrasion resistance of wire rope is improved by using strands composed of larger wires; therefore, replacing the existing cables with cables fabricated from larger wire strands should improve cable performance[49]. Since the overall risk, expressed in cell lifetime cost, was substantial but not great enough to warrant a large-scale redesign, only low-cost preventive maintenance efforts are indicated. In the risk assessment, the high failure rate of manipulators was the primary contributor to the elevated risk, so any decrease in the MTTF will have a substantial impact on the risk associated with the manipulator failure. Both the tape reinforcement and the cable change were simple, yet effective, preventive maintenance measures that reduced the MTTF and were employed at very low cost. Since the tapes and cables are not readily available for inspection, a more aggressive preventive maintenance effort could still be developed for planned replacement during cell non-use periods based on a projected MTTF. The projected MTTF for the planned replacements should address reliability for improvements in tape and cable reliability by the design changes discussed previously.

Apart from preventive maintenance considerations, the economic consequences (costs) are directly affected by staffing, i.e., the greatest component of cost is the ADT, which is based on the availability of technicians trained to perform manipulator repairs and replacements. The LDT is not affected by staffing, since LDT represents only the cell shut-down associated with the entry for replacement. The MTTR is only moderately affected by staffing since the optimum maintenance team is two technicians. The staffing sensitivity analysis for the corrective maintenance of the manipulator is shown in Figure 19. The sensitivity analysis assumes a reduction in the ADT of 8 hours per trained technician available until a pool of four technicians is available, where a steady ADT of four hours is assumed. The MTTR is shown only to illustrate that there would be some impact on cost if only one technician were available to assist in the repair of the defective manipulator. Even acknowledging that the assumed reduction in ADT is likely to vary, the sensitivity analysis shows the immediate impact that additional trained personnel would have on the consequence of a manipulator failure.

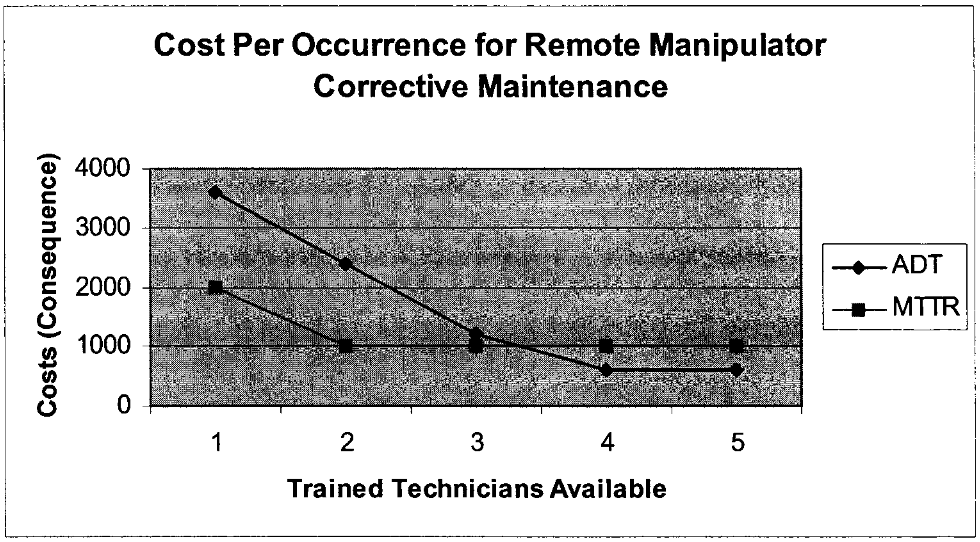


Figure 19. Sensitivity Analysis

4.3.2.4 Manipulator Case History Summary

The case study of the manipulator shows how a risk-based approach to maintenance engineering and management can be used effectively even on small systems. Because of the size of the system, the access to reliable expert operating experience and maintenance data, the failure modes and costs were determined in a timely fashion. Preventive maintenance strategies were facilitated by the identification of the failure modes, and for preventive periodic replacement, the current MTBF was available through the known maintenance history. Since there were few activities involved in the maintenance schedule, PERT and CPM techniques were unnecessary. The scheduling consideration of primary concern in this maintenance activity was the effect of the availability of trained technicians on the cost associated with ADT. The sensitivity analysis expressed the quantitative effect of technician availability on the ADT cost. The recommendations resulting from the case study were (1) continue to use tape reinforcement, (2) use cables with larger diameter wire strands, (3) consider performing planned periodic maintenance during periods of cell non-use, and (4) train two additional technicians to perform manipulator repairs.

4.3.3 Intergranular Stress Corrosion Cracking of Jet Pump Beams in a Nuclear Reactor

4.3.3.1 Introduction and Background

This case study involves replacement of jet pump beams in at a nuclear power plant. The replacement is discussed in Section 4.3.3.4. This section summarizes the extensive history of problems with this component leading up to the situation described in the case study. Section 4.3.3.2, contains safety considerations and the damage mechanism. Preventive maintenance inspection is discussed generally in Section 4.3.3.3 to give the reader a context for the plant shutdown case study discussed in the final Section 4.3.3.4. A number of Boiling Water Reactor (BWR) internal components as shown in Figure 20, have experienced intergranular stress corrosion cracking (IGSCC).

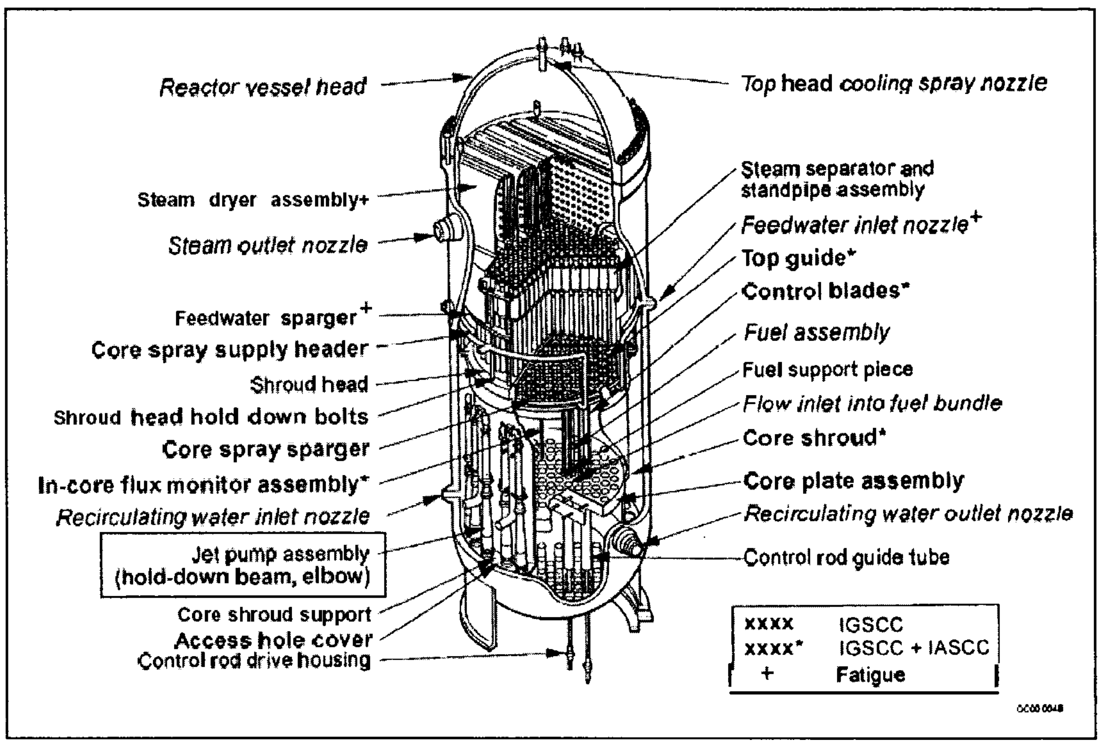


Figure 20. IGSCC Failure Areas in a BWR

The materials affected by IGSCC are Type 304 and Type 316 stainless steels and nickel Alloy X-750.[50] The resulting failures have had varying consequences, but failures of nickel Alloy X-750 jet pump hold-down beams (jet pump beams) have resulted in a number of unplanned plant shutdowns. The jet pump beams are part of the jet pump assembly as shown in Figure 21. BWR jet pumps force water flow through the reactor core yielding greater power than would be possible with only natural circulation.

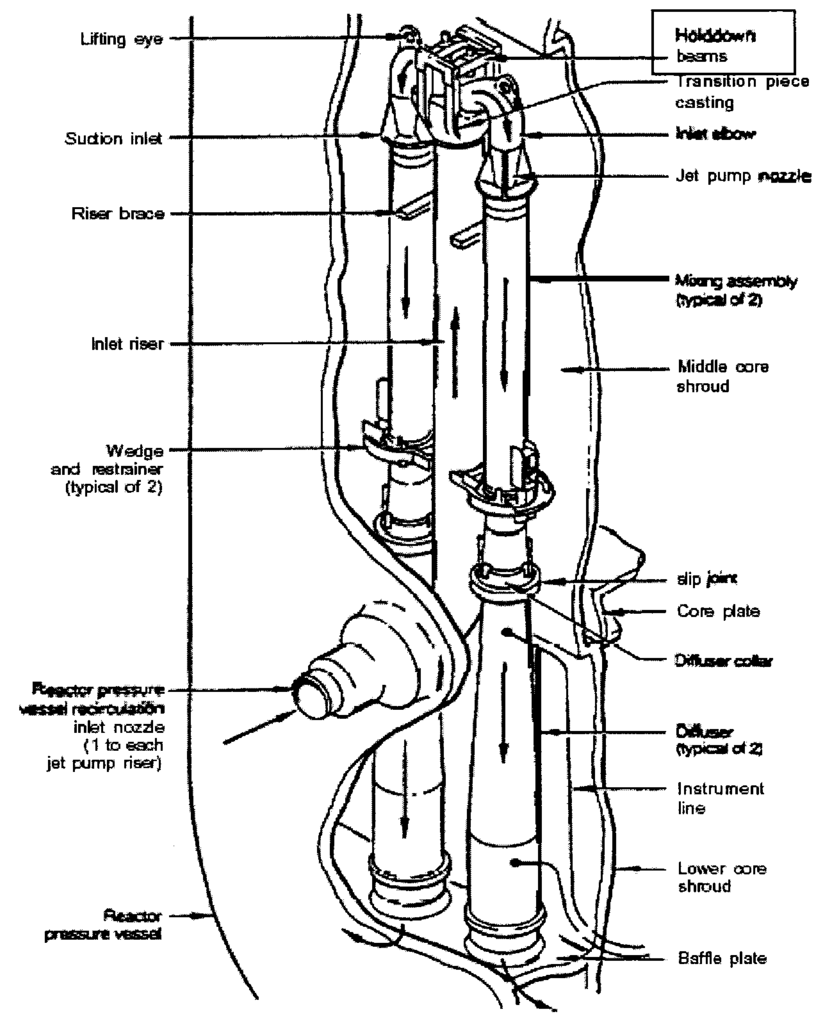


Figure 21. Jet Pump

As shown in Figure 21, two jet pump mixer assemblies are connected to a single riser by 180° elbows. The elbow is an integral part of each jet pump mixer assembly. Each mixer assembly is connected to a diffuser with a slip joint collar. The jet pump mixer assembly is mechanically secured to the riser and the slip joint by the jet pump beam. Inlet water is fed by an external pump through the recirculation inlet nozzle, shown in Figure 20, into the riser and forced through the jet pump nozzle drawing internal reactor water along with it into the mixer assembly. The combined streams join in the diffuser and are discharged below the core.

The first unplanned shutdown caused by a jet pump beam failure occurred on February 2, 1980, at the Dresden Unit 3 plant, and was caused by a complete fracture of a jet pump beam through the ligament area adjacent to the beam bolt.[51] As a result of the failure, the preventive maintenance recommended for other plants by the supplier was replacement of the original (BWR/3) beams with improved beams and reducing the preload on the beams. Visual and periodic ultrasonic examination were recommended for beams that were not replaced. The ultrasonic examination of the ligament area of the jet pump beams supported successful preventive maintenance of jet pump beams until September 13, 1993 when a jet pump beam at the Grand Gulf plant failed in the transition area between the body of the beam and the beam end.[52] A further consequence of the Grand Gulf beam failure was that the mixer assembly was completely dislodged from the slip joint and damaged. A typical jet pump beam assembly and both failure zones are shown in Figure 22. The jet pump configuration after the mixer assembly was dislodged as illustrated in Figure 23. As a result of the Dresden and Grand Gulf failures, operators of most BWR plants replaced their jet pump beams, while some continued operation with

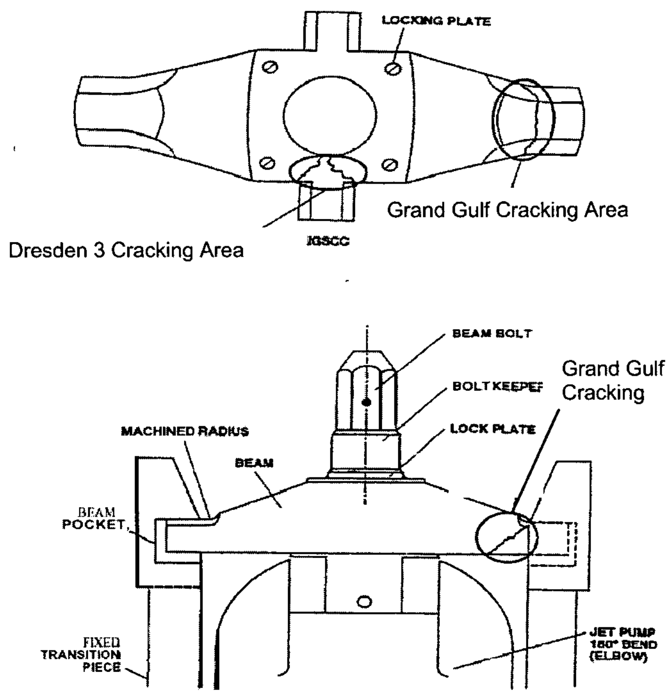


Figure 22. Jet Pump Beam Assembly

the original beams and periodic ultrasonic examinations augmented to cover both of the known failure zones. On January 9, 2002 a jet pump beam failure at the Quad Cities Unit 1 plant caused an unplanned shutdown.[53] That failure occurred in the tapered region of the beam between the two failure areas shown in Figure 22. The plant had seen approximately 20 years of full power operation and was one of the last with the original beams. As a result of this failure the beams were replaced at Quad Cites Unit 1 as well as the other two remaining plants with original beams in the U.S., which happened to be owned by the same operating company.

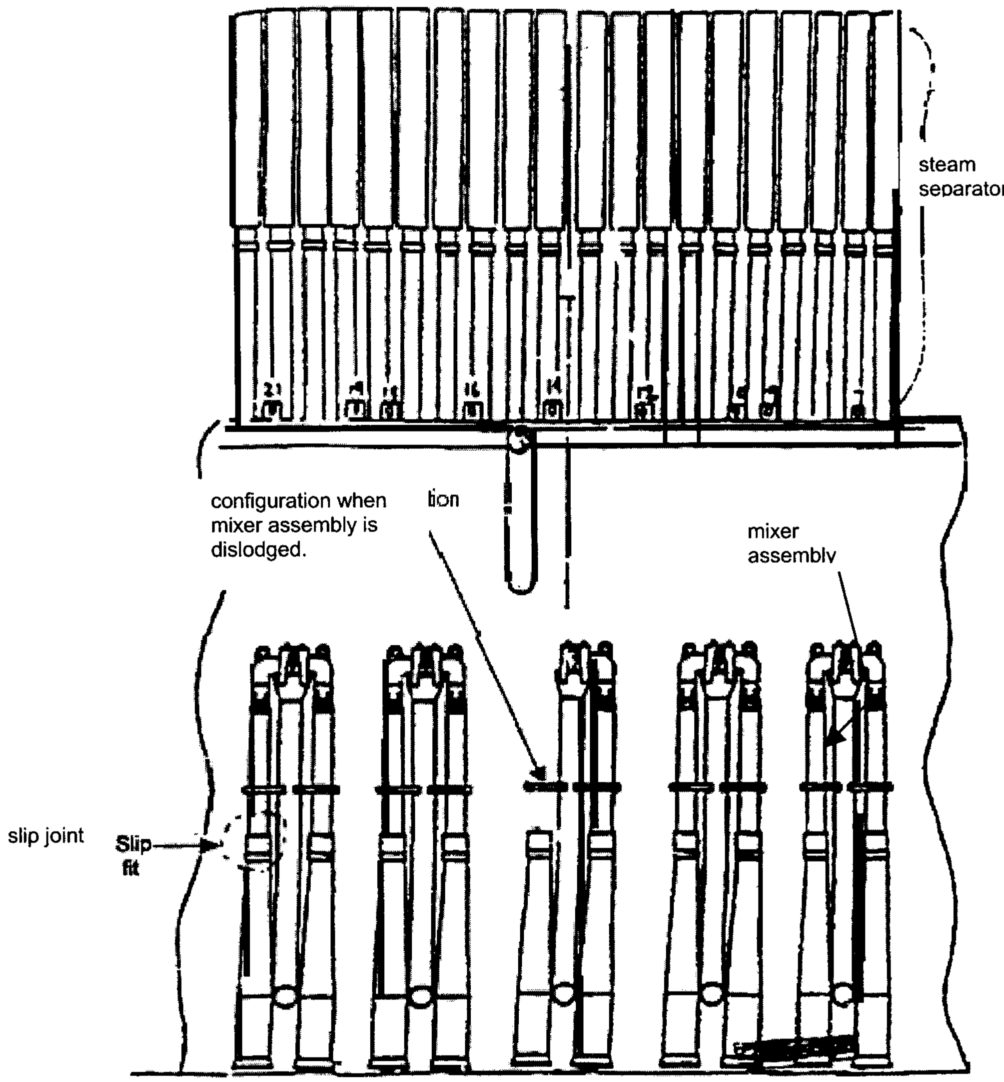


Figure 23. Jet Pump Mixer Dislodged as a Result of Hold-Down Beam Failure[52]

4.3.3.2 Risk Assessment for IGSCC in Alloy X-750 Jet Pump Beams

4.3.3.2.1 Safety Considerations for the Jet Pump

The jet pump itself does not perform a safety-related function in the BWR and its failure is not considered in the plant Probabilistic Risk Assessment (PRA). There is also virtually no concern that parts from a fractured beam could enter and damage the core, and such parts have usually been recovered from the baffle plate at the bottom of the annulus, shown in Figure 21, which is outside the core. However, the complete failure of a jet pump beam will cause the mixer assembly to lift at the slip joint, which can cause the mixer to be completely dislodged as shown in Figure 23. That could cause a lower core flood elevation in the event of a loss of coolant accident (LOCA) and could potentially interfere with the effectiveness of emergency core cooling systems. For those reasons, continued operation of a plant with a known jet pump beam failure is not permitted.

4.3.3.2.2 The IGSCC Damage Mechanism and Mitigation Efforts in Jet Pump Beams

As with any type of stress corrosion cracking, three factors, shown in typical fashion in Figure 24, are required for intergranular stress corrosion cracking: tensile

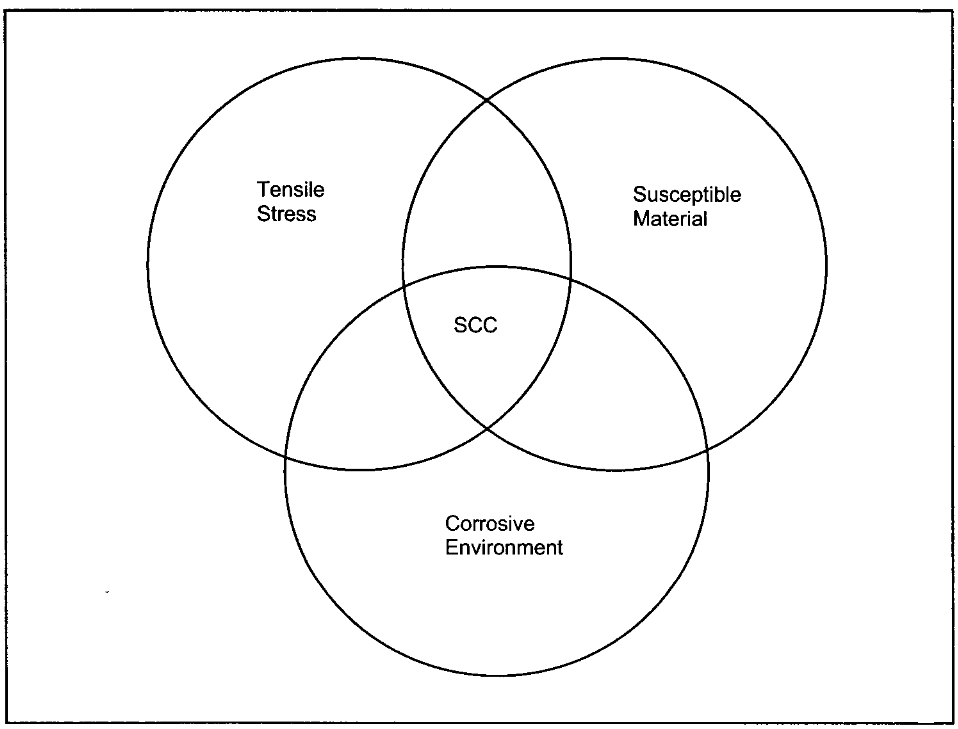
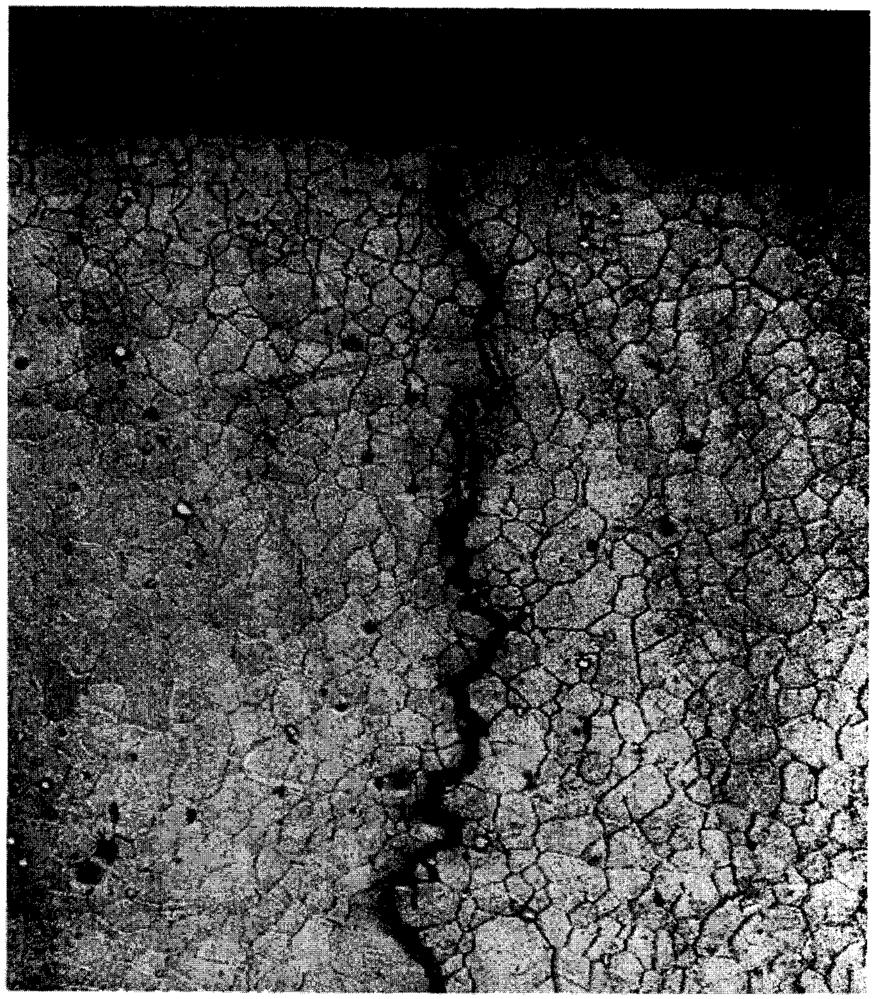


Figure 24. Factors Required for Stress Corrosion Cracking (SCC)

stress, susceptible material, and a corrosive environment. It is called intergranular because the cracking follows the grain boundaries rather than cracking through the grains. IGSCC typical of that seen in Alloy X-750 is shown in Figure 25.



Cracking in Alloy X-750 at 100X showing propagation along grain boundaries

Figure 25. IGSCC in Alloy X-750

The susceptibility of the original jet pump beam material was attributed to its heat treatment. The original beams were subjected to a heat treatment called equalized and aged, heating to 1625 °F and holding for 24 hours followed by an aging treatment at 1300 °F for another 20 hours. The replacement beams were annealed at 2000 °F for 1 hour instead of the equalizing treatment, followed by the same aging step at 1300 °F for 20 hours as was used for the original beams. Testing has demonstrated that the higher temperature anneal renders the alloy X-750 much less susceptible to IGSCC; however, more than one explanation has been offered for that effect. While NUREG/CR-6677[50] reported that chromium carbides developed at the grain boundaries are responsible for increased IGSCC resistance, the ASM Handbook[54] states that it is believed that new grain boundaries resulting from recrystallization are free of precipitates and impurities. Nevertheless both sources agree that the temperature of the annealing step is a key metallurgical factor in developing resistance to IGSCC. The aging step is unchanged and is sufficient to develop the gamma prime precipitate in the matrix that in turn enables the alloy to develop a high ultimate tensile strength (160,000 psi) regardless of the annealing temperature. The improved replacement beams utilized the high temperature annealing heat treatment.

The stress factor required for IGSCC of jet pump beams includes both the preload developed by the tightening of the beam bolt and the stresses developed by thermal expansion during operation. The original beams were installed with a preload of 30 kips. To reduce the stress on the beams, it was recommended that preload be reduced to 25 kips.[55]

With respect to the environment, the BWR uses high purity demineralized water (< 0.3 S/cm) with nominally 0.2 ppm dissolved oxygen and neutral pH . Industry efforts to address water chemistry were undertaken, but for reasons unrelated to the jet pump beam. Water chemistry control was implemented to mitigate IGSCC of 300 series stainless steel in piping and other components that have been sensitized by welding. Since laboratory tests showed that at a low oxygen level of 0.015 ppm, electrochemical potentials were reduced and initiation of IGSCC was very difficult in 300 series stainless steel, hydrogen injection was recommended to reduce oxygen levels.[54] Because the performance of jet pump beams was already addressed by improved heat treatment and preload reduction, no water chemistry recommendations were specifically made for jet pump beams.

The risk assessment applied by the industry, although qualitative, was sufficient to address the issue. The equalized and aged beams have a much higher probability of failure. That probability could be slightly reduced, by reducing the preload on the beams, but ultimately, reducing the probability of failure significantly depended on installing replacement beams with solution annealed and aged material.

4.3.3.3 Maintenance Needs Inspection Strategy to Support Preventive Maintenance

Access to jet pump beams for planned inspections or maintenance is limited to refueling outages, normally scheduled every 18 to 24 months. When the reactor vessel is opened for refueling, inspections and maintenance of the jet pump beams must be performed remotely due to radiation and the location of the beams in the beltline region of the reactor vessel, the region of the vessel that is approximately the middle third of the vessel adjacent to the fuel. Both the reactor vessel and the refueling cavity above the reactor are filled with water during refueling and the remote cameras, and other examination and maintenance equipment must be lowered more than 60 feet below the surface of the water. The beams are examined visually using the remote cameras and ultrasonically using a specially designed fixture containing transducers positioned to monitor the more susceptible regions of the beam. While the environment and material aspects of SCC susceptibility are constant, the stress distribution is complex. The simple two-dimensional finite element model, in Figure 26, of the jet pump beam shows qualitatively the more highly stressed areas of the jet pump beam. While a more complex, three dimensional finite element model can be created to accurately quantify the stresses to support design and fracture mechanics evaluations, the simplified model in Figure 26 is sufficient to identify the areas of greatest interest. The top view in Figure 26, modeled for a tensile stress at the top of the beam, shows the most highly stressed areas are in the bolt-hole ligament of the beam at the location of the Dresden 3 failure. The side view of Figure 26, modeled considering the bending due to the load applied near the end of the beam, shows that the stress concentration at the transition to the beam end is also a highly stressed area, a result consistent with the Grand Gulf failure.

4.3.3.4 Jet Pump Beam Maintenance Scheduling and Decision Analysis

In October of 2002, during a planned shutdown of a non-U.S. BWR plant, visual examinations performed with a remote underwater camera revealed crack-like indications in some jet pump beams similar to those found at the Quad Cities Unit 1 plant. That plant and its sister unit at the same site were the last in the world that had the original

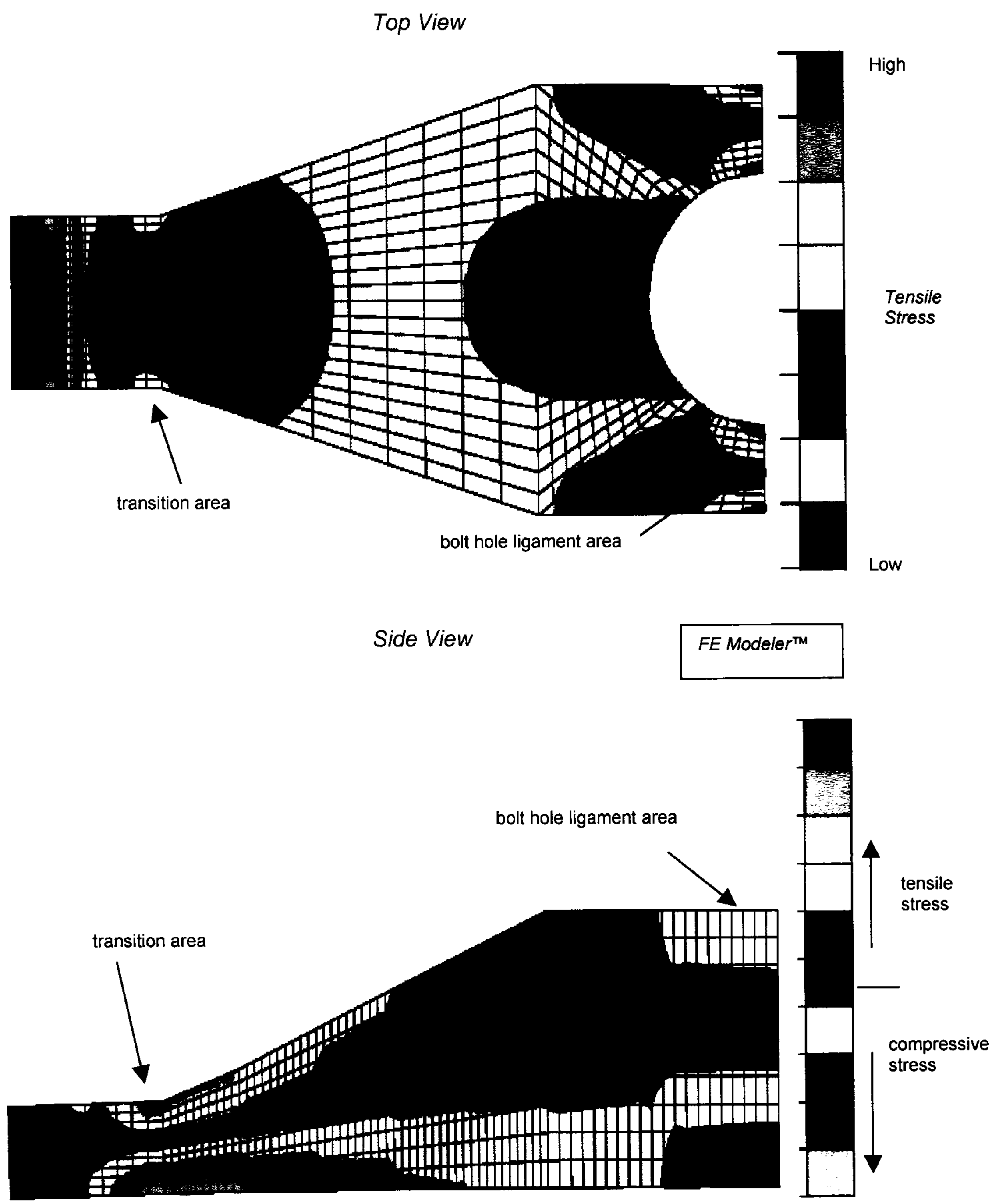
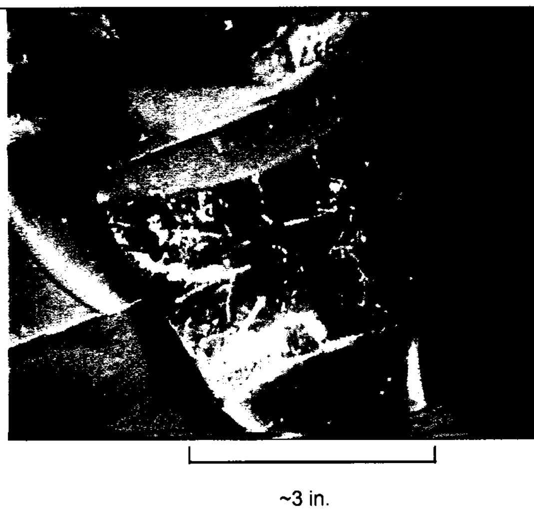


Figure 26. Finite Element Models (Orthogonal Views) of the Jet Pump Beam

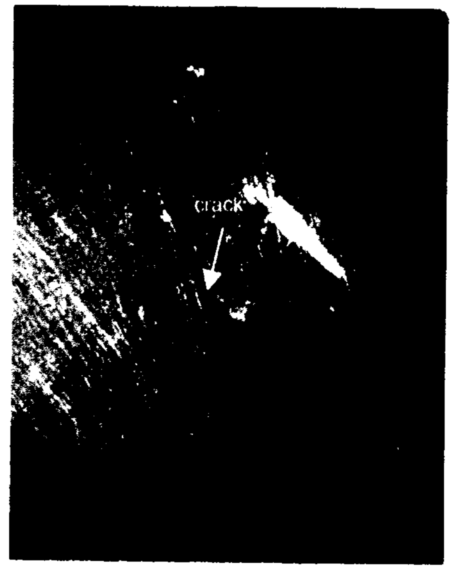
BWR/3 beams, equalized and aged material. The plant had been operating only for ten years, approximately half the operating life of Quad Cities Unit 1 (its sister plant had been operating only 4 years). Beam replacement had been planned for the next shutdown in another 18 months. Ultrasonic examinations had not been planned and, even if they had, the available ultrasonic systems were not designed to examine the tapered area of the beam. Removal of a jet pump beam from the reactor for further examination on site was not possible since the expected radiation dose rate was 50 Rem per hour on contact, far too high for any direct handling of the beams. The country in which the plant was located had no remote handling facilities suitable for parts with such high dose rates. Therefore, there was no available means to confirm or to further assess the severity of the cracking, and the decision to replace or allow the beams to remain in service had to be made only on the basis of the condition of the beams as observed with the remote camera.

Since IGSCC cracks in Alloy X-750 jet pump beams are fine and tight on the surface, determining their extent or even their presence is often difficult. Views of a cracked jet pump beam in the reactor using a remote camera and after its removal and cleaning in a remote handling facility are shown in Figure 27. Even after removal and cleaning, the crack was not prominent, and observed in the reactor it was even more difficult to discern. For that reason, conservative assumptions were made regarding the presence and extent of cracking in the jet pump beams.

There were essentially three maintenance options considered: (1) let all of the jet pump beams remain in service and continue with the existing plan to replace all of the beams at the next scheduled plant shutdown, and take the risk of an unplanned shutdown, (2) schedule a maintenance shutdown before the next refueling outage for the purpose of replacing the jet pump beams, and (3) replace all of the beams during the current shutdown, or at least the two beams in which crack-like indications had been observed by the remote camera.



Beam Surface Observed In-Vessel Using an Underwater Remote Camera

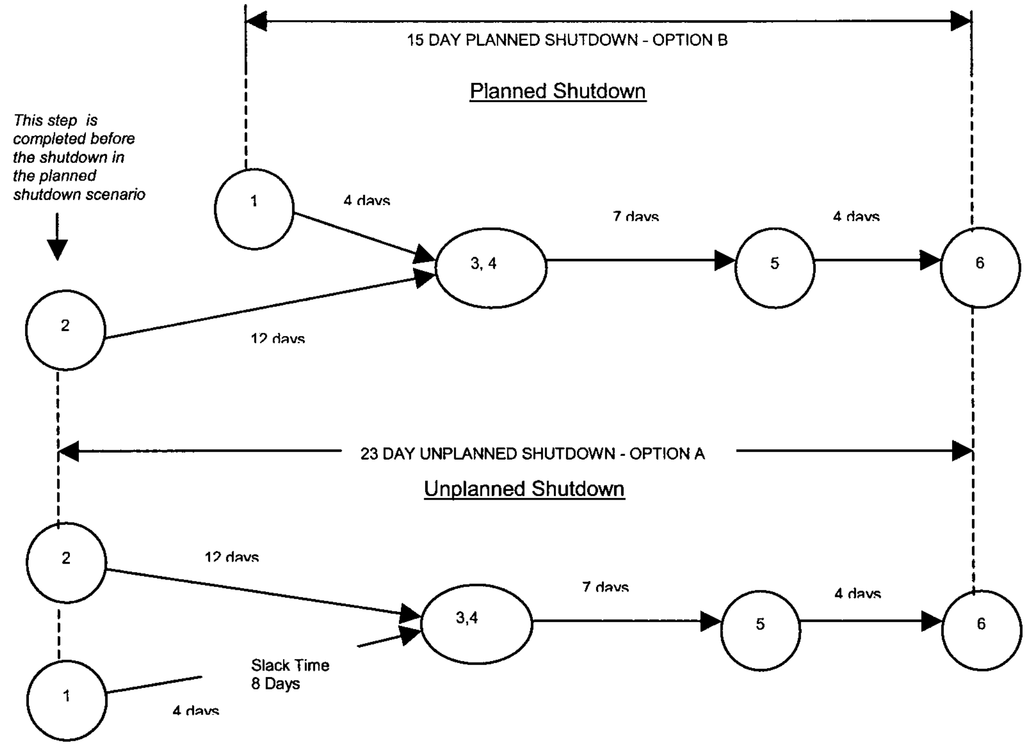


Beam Surface Observed with Viewer after Removal and Cleaning in a Remote Handling Facility

Figure 27. Jet Pump Beam Observations

In two of the beams where cracking was assumed in the thinner portion of the tapered area, there was little assurance that the beams would survive for an additional 18-month fuel cycle. From past experience and metallurgical examinations of removed beams from other plants, it was known that cracking could potentially propagate to a depth of 0.4 inch over a single fuel cycle of 18 months. If existing cracks in the thinner area of the tapered region could grow at the same rate, the crack could theoretically exceed one-half the beam thickness; however, the critical size would probably be less than that, and beam failure would result.

For perspective, the probability category for MIL-STD-882 from Figure 14 was considered as frequent in this case, i.e., greater than 10^{-1}, or 1 in 10, that failure would result before the next planned shutdown. The economic consequence of an unplanned shutdown in this case was less related to the costs associated with the replacement of damaged equipment than it was related to the cost of lost generation while the plant was not operating. That cost for the 625 MWe plant was $800,000 per day. The cost to perform the work, including the cost of replacement beams, specialized equipment, and a maintenance crew (all to be flown in from the US) to replace the beams was $950,000. The cost of replacing only 2 beams was $800,000 since the bulk of the cost was related to maintenance crew and equipment. The actual time for the replacement was estimated to be 10 days for all 20 beams or 7 days for only 2 beams, and it would require 16 days to mobilize a crew on the site. A total of 26 days would therefore be required for all the jet pump beams to be replaced. In MIL-STD-882 terms, the potential consequences were catastrophic and the situation posed the highest risk category. If the third option, of replacing beams prior to restarting the reactor was not selected, the other two alternatives were: (1) replacement during an unplanned shutdown (as a result of beam failure) and (2) a planned shutdown for replacement (after 6 months of operation), based on the qualitative perception that 6 months of operation would represent an acceptably low probability of any existing cracks growing to a critical size. An unplanned shutdown resulting from a beam failure would require a minimum of 23 days, i.e., 19 days for beam replacement (including mobilization time and assuming only two beams would be replaced), and an optimistic 4 days for reactor reassembly, leakage testing, and power ascension. A planned maintenance shutdown (in six months), before the next refueling outage, for the purpose of replacing beams would involve only 15 days, assuming replacement of only two beams and since the mobilization time would not be on the critical path of the schedule. A comparison of the schedules and risks for jet pump beam replacement during the planned and unplanned maintenance outage is shown in Figure 28, to illustrate the schedule evaluation. The evaluation of the option of extending the current outage to replace the beams did not involve the same tasks as the planned and unplanned special maintenance outages, so it is not shown graphically in Figure 28, but is discussed later. The simplified schedules are presented Figure 28 in a PERT/CPM format to illustrate the timeline differences between the planed and unplanned outage options. For evaluation of specific tasks, the schedule risk is shown in Figure 28 based on the probability that a task would require an additional day of critical path and extend the shutdown. The highest risk task being shown for the mobilization of crew and equipment for the unplanned shutdown in an unplanned shutdown as 0.5, compared with 0.01 for the same task when the shutdown is planned in advance. The overall risk shown for the alternatives at the bottom of Figure 28 was based on the probability that the event (the shutdown) would occur. The probability that a beam would fail and cause an unplanned outage was estimated to be 0.5, while the probability of a planned



TASK	DESCRIPTION	ESTIMATED TIME	PROBABILITY OF EXCEEDING TIME ESTIMATE BY 1 DAY		SCHEDULE CONSEQUENCE: COST OF ONE DAY OF SHUTDOWN	RISK	
			Unplanned	Planned		A	B
1	SHUTDOWN REACTOR	n/a*					
2	REQUEST JP BEAM REPLACEMENT CREW/EQUIP	n/a*					
3	COMPLETE DISASSEMBLY REACTOR	4 days	0.01	0.01	$800K	$8K	$8K
4	JP BEAM REPLACEMENT CREW/EQUIP MOBILIZED ON SITE	12 days	0.5	0.01	$800K	$400K	$8K
5	COMPLETE REPLACEMENT OF 2 JET PUMP BEAMS	7 days	0.1	0.1	$800K	$80K	$80K
6	REACTOR REASSEMBLED AND RESTARTED	4 days	0.01	0.01	$800K	$8K	$8K
SCHEDULE RISK						$496	$104K
RISK = PROBABILITY THAT A WILL OCCUR (0.5) x SCHEDULE RISK						$248	
RISK = PROBABILITY THAT B WILL OCCUR (1.0) x SCHEDULE RISK							$104

n/a - not significant to and/or not directly related to comparative shutdown schedules

Figure 28. Risk Evaluation of Alternative Jet Pump Beam Replacement Options

maintenance shutdown was 1.0. As shown in Figure 28, the schedule uncertainty for the task of mobilizing a maintenance crew and equipment for an unplanned outage resulted in a probability of 0.5 that a delay of one day in the schedule would occur. Although the probability of the unplanned outage itself was 0.5, the risk for the unplanned outage was nearly two and a half times that of the planned outage.

If the jet pump beam replacement was performed during the current outage before restarting the reactor, even though there were other parallel ongoing planned maintenance activities, a complete jet pump beam replacement would have still added 18 days to the critical path time of the shutdown. Replacing only 2 beams would have saved 3 days of critical path time, extending the shutdown only 15 days, equivalent to that of the planned shutdown, but performing the replacement during the current outage was considered qualitatively to involve less uncertainty with respect to task completion. That small level of schedule assurance and consideration of the relatively small probability that a jet pump beam failure during operation before the planned shutdown could cause a mixer assembly to become damaged, as occurred at Grand Gulf, resulted in a decision to replace before restart. Since mixer assembly replacement was not an anticipated maintenance need, no spare mixer assemblies available on site, and no vendors stocked or routinely produced replacement mixers. A replacement mixer would have been a special order and would have required a lead time of several months.

4.3.4 Dissimilar Metal Weld Failures in Boiler Tubes

4.3.4.1 Introduction and Background

In the higher temperature regions of a fossil-fuel-fired power boiler, austenitic stainless steel boiler tube material is used. In other areas of the boiler, low alloy ferritic steel is used for economic reasons. Welds joining the two materials are known as dissimilar-metal welds (DMWs). Those welds are subject to early failure due to two damage mechanisms. Creep damage, associated with time at high temperatures under static loads, is the primary cause of failure, and cracking associated with oxidation ("oxide-notch") causes failure in some cases. Thermal expansion differences between the two metals being joined, decreased resistance to creep at high temperatures, and corrosion all contribute to the degradation of dissimilar-metal welds.(56) This case study describes a typical power plant that had experienced several forced outages as a result of DMW failures. Since creep is considered to be the primary damage mechanism in DMWs, it was the focus of the technical effort for this case study. An evaluation was performed to understand the risks of operation with mature DMW welds near the end of their expected service life and to support development of inspection, repair, and replacement strategies. The equipment histories, including failure data, as well other essential information to support the risk evaluation, was obtained through interviews with the plant staff, in particular the Chief Maintenance Engineer and the cognizant Boiler Mechanical Engineer. The case study also relied on the extensive research published by the Electric Power Research Institute (EPRI), considered the definitive body of work on dissimilar metal welds in fossil-fuel-fired boilers.(57)

4.3.4.2 Qualitative Risk: Metallurgical Factors Associated with DMW Failure

ASME SA-213 is typically specified for the material grades used for boiler tubes. The ferritic alloys used for the boiler in this case study were Grade T-22, a P-5A, 2.25Cr-1Mo material, and Grade T-11, a P-4, 1.25Cr-0.5Mo-Si material. Both steels are rated at 60 ksi minimum tensile strength. The austenitic alloys used for the boiler in this case study were Grade 347H, a P-8, 18Cr-10Ni-Cb material, and Grade 304H, a P-8, 18Cr-8Ni material. Both austenitic alloys are rated at 75 ksi minimum tensile strength.(58) When dissimilar metal welds are made between the ferritic and austenitic alloys, chromium carbides form at or near the weld interface, depleting the ferritic material of carbon. The creep strength of the ferritic material is largely a function of its carbon content, so the carbide formation reduces the creep strength of the base metal adjacent to the weld. The base metal adjacent to the weld is also subject to the highest induced strains due to the difference in the thermal expansion coefficients of the two materials at the weld joint.(56)

Two different types of filler metal have been used for dissimilar metal weld joints, stainless steel, e.g., E309 and E310, and nickel-based alloys, e.g., Inconel™ 182 (ENiCrFe-3) and Inconel™ 132 (ENiCrFe-1). The filler metal determines the predominant characteristics of the carbide formation at the weld interface. Since the creep voids form and propagate adjacent to the carbides, ultimately resulting in crack formation, the characteristics of the carbide formation affect the crack propagation mode. When stainless steel filler metals are used, small carbides are diffused throughout the weld heat affected zone (HAZ). That morphology was identified as Type II carbide in the EPRI study. When nickel-based filler metals are used, larger carbides are formed, chiefly at the fusion line. That morphology was identified as Type I carbide in the EPRI study, and those Type I carbides comprise approximately 80% or more of the carbides in the nickel-based DMW. Approximately 20% or less of the carbides present when the nickel-based filler metals are used are Type II. In DMWs with stainless steel filler metal, the cracking is primarily intergranular along the prior austenite grain boundaries. In DMWs with nickel-based filler metal, the cracking is predominantly along the fusion line following the Type I carbides. As a crack propagates in a nickel-based filler metal DMW, if Type II carbides are encountered, cracking either stops or moves to the low alloy HAZ and continues as intergranular. The expected life of the nickel-based filler metal DMWs is approximately four times that of the austenitic filler metal DMWs. That is due in part to reduced stress concentration since the thermal expansion coefficient of the nickel-based filler metal is closer to that of the low-alloy steel than the austenitic filler metal. Also it may be due to the length of time required in the nickel-based filler metal DMWs for the precipitates to develop carbides of the size associated with formation of creep microvoid nucleation, 1.5 to 2.0 μm. Because of their inherently higher creep-resistance, the nickel based filler metal DMWs were known to be low priority and were not included in the detailed risk evaluation.

The microstructures of the low alloy steel in boiler tube DMWs are not grossly decarburized, and the carbides, chromium rich ($M_{23}C_6$) and molybdenum rich (M_6C), form by relatively slow diffusion at the service temperature, usually less than 1050 °F. Although the carbides are not formed during the welding process, as they are during sensitization of the heat affected zone in stainless steels, the welding process variables do influence the amount of Type I and Type II carbides developed. Therefore, the welding process variables should not be overlooked in assessing the DMW conditions, especially for those with nickel-based filler metal. A greater degree of dilution, resulting from process variables such as low travel speed and high heat input generally, favors the formation of Type II carbides. In the EPRI research, post-weld heat treatment (PWHT) at 1350 °F for 1/2 hour produced fine precipitates at the interface area which eventually developed into Type II carbides. However, since boiler tubes are fabricated from grade T-11 and T-22 materials, P-4 and P-5A respectively, most low alloy boiler tubes would be exempt from PWHT under ASME Code rules based on wall thickness. Further, PWHT is never required for the austenitic, P-8, materials.(59) Therefore, it would be unlikely to encounter installed boiler tube DMWs in the PWHT condition.

A number of the welds in this case study were friction welded without any filler metal. The EPRI study did not include any such welds, but since the autogenous weld interface is still between austenitic steel and low alloy steel, it is expected that Type II carbide formation would be favored, albeit within a much narrower HAZ region.

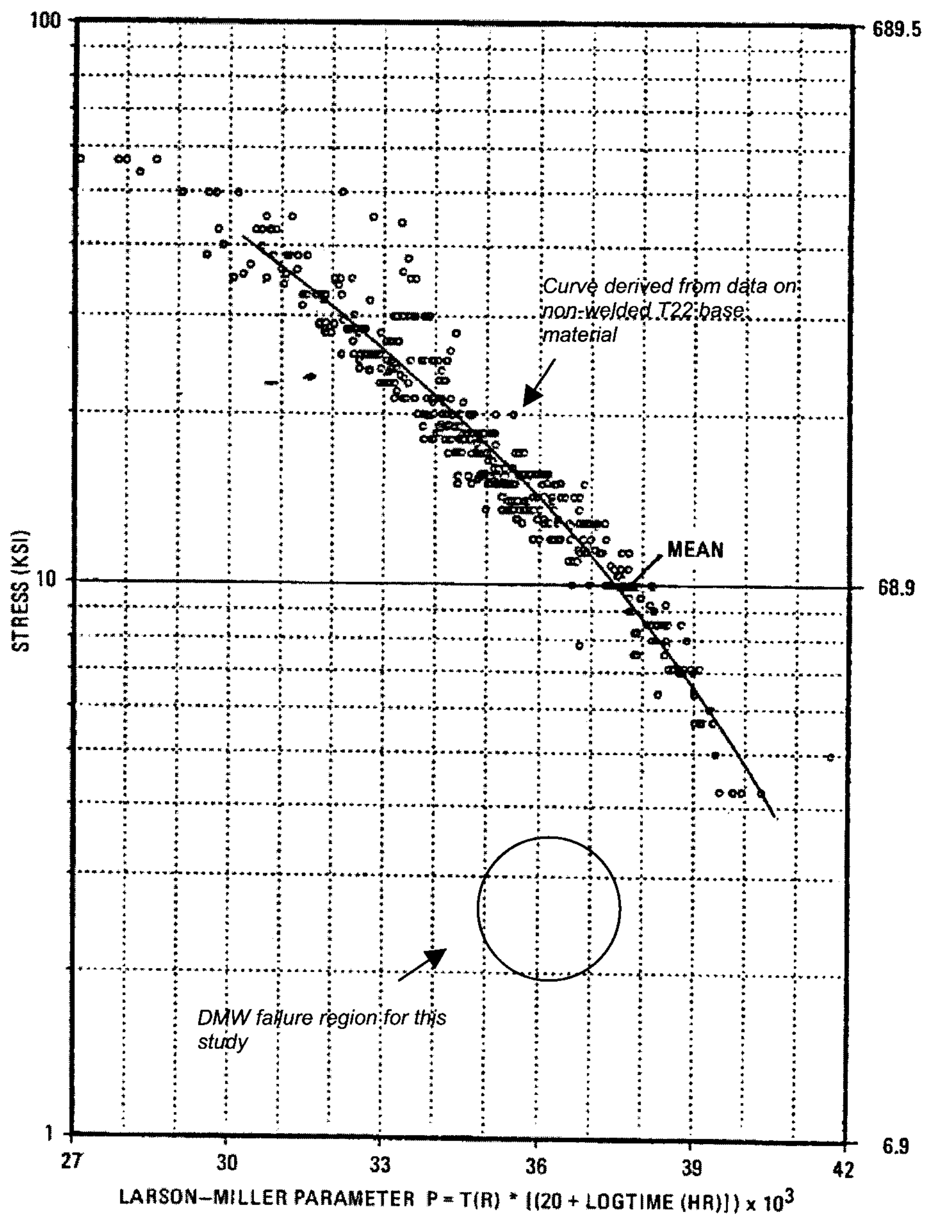
In addition to creep, another damage mechanism that occasionally results in DMW failure is cracking associated with an oxide notching. Oxide notching at the weld toe in the low alloy base metal is commonly observed by visual inspection in DMWs, but cracking associated with these locations is not usually the cause of DMW failure. These notches probably initiate as cracks due to stress concentration at the weld toe. Increases in crack opening displacement by rotation are required to continue to concentrate stress at the crack tip as the crack propagates, but because boiler tubes are relatively thick walled, such rotation is resisted and crack arrest usually occurs. The arrested crack then oxidizes forming the observable oxide notch. Failure due to cracking associated with such notches is more likely to occur in the thinner walled tubes, such as those in the reheater sections of the boiler, where the thickness of tube walls is often a quarter inch or less.

4.3.4.3 Stress and Temperature Factors Associated with DMW Failure

Creep is defined as the as the time dependent part of the strain resulting from stress and is associated with high temperatures relative to the melting point of the material.[60] Low alloy steels, such as Grades T-11 and T-22, are creep resistant up to about 800 °F, above which, like other thermal processes, the temperature compensated time to reach a given strain (or failure) can be expressed as an Arrhenius-type equation.[61] A related function that is based on that Arrhenius relationship is the Larson-Miller parameter such that:

$$T(C + \log_{10} t) = f(\sigma) \qquad [30]$$

Where T is the temperature in degrees Rankine, *t* is the time in hours, and C, an empirical constant, equals 20 for low alloy steel.[62] The Larson-Miller parameter is used for creep testing since it can be used for short times and higher temperatures in test conditions to represent longer times and lower temperatures in service. However, since the predominant failure mode of DMWs is creep, the Larson-Miller parameter was incorporated as a basis of damage prediction when considering the primary stress, i.e., the axial stress in the tube.[57] The Larson-Miller parameter is usually plotted against the applied stress as shown in Figure 29, a Larson-Miller plot for Cr-Mo steel base metal.



(Originally From G. V. Smith Supplemental Report on Elevated-Temperature Properties of Cr-Mo Steels ASTM DS 6S2 and unpublished data from KKK & Sumitomo, Japan)

Figure 29. A Larson-Miller Plot for Cr-Mo Steels [57]

In addition to the primary stress, other stresses on the DMW are secondary stresses, such as bending loads, and stresses associated with the differences in the thermal expansion between the low alloy and stainless steel base metal. The mean coefficient of thermal expansion for T-11 and T-22 steels is 8.1 x 10^{-6}/°F between 70° F and 1000° F. The mean coefficient of thermal expansion of austenitic stainless steel is approximately 27% greater at 10.3 x 10^{-6}/°F in that same temperature range.[(37)]

The EPRI creep damage prediction program for DMWs, called PODIS for Prediction of Damage In Service, considered all three categories of damage: (1) intrinsic damage, (2) primary system load damage, and (3) secondary system load damage. The damage associated with each category was calculated with the aid of some empirically derived constants, and the total damage was the sum of the damage associated with each of the three categories. The intrinsic load damage was the damage from strains caused by the local differences in the thermal expansion between different materials at the weldment, and considered the number of cycles and the temperature variation of the associated cycle. The primary system load damage considered the primary stresses and the Larson-Miller parameter. The secondary system load damage considered the strain range based on the stresses at the DMW and the number of cycles through that strain range. For the primary loads, the calculated stress was used, and for secondary loads the calculated strain range was used. For the intrinsic damage (thermal expansion), actual stress or strain values were not used, only the number of cycles and the magnitude of the temperature difference through the cycle was considered.[(57)]

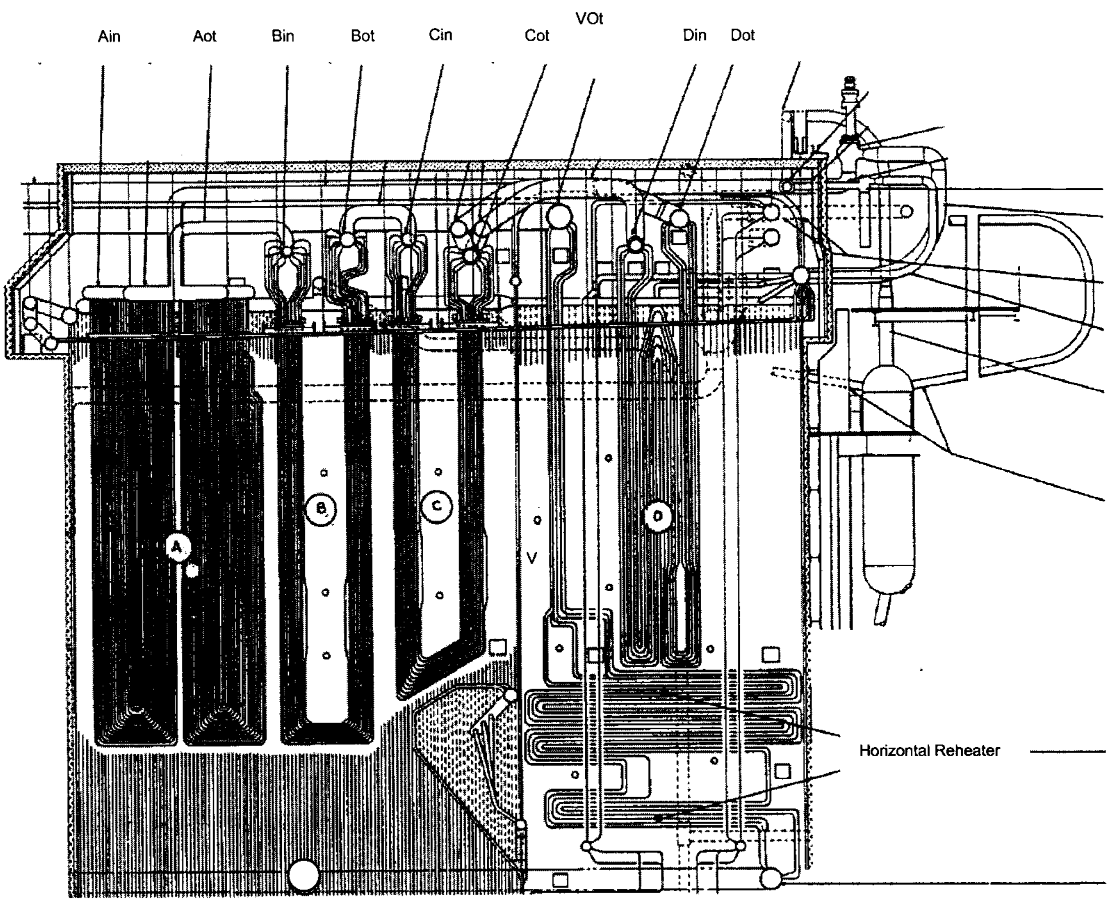
4.3.4.4 Quantitative Risk Assessment for DMWs in a Fossil-Fuel-Fired Boiler

In this case study, an assessment was performed to address the risks associated with potential failures of DMWs using austenitic stainless steel filler metal in the superheater portion of a fossil-fuel-fired power plant. The plant was rated at approximately 680 MWe and had been in service for approximately 30 years although some of the major components had been replaced after approximately 10 years of operation as part of design modifications. A diagram of the plant is shown in Figure 30.

The plant had experienced a number of DMW failures in the past, and the cause of all of those failures had been identified as creep. Therefore, the evaluation focused on the creep damage mechanism alone as the basis for determining the relative probability of failure. The previous failures had all been DMWs with austenitic filler metal. The few DMWs with nickel filler metal were not considered since they were replacement welds and, therefore, had experienced fewer operating hours than welds from the original fabrication. Further, those nickel filler metal welds have a much longer expected life.

The objectives of this evaluation were to: 1) estimate the probability of failure for the various dissimilar metal welds, excluding the more resistant nickel-based welds, 2) determine the economic consequence of failure at various DMW locations, 3) calculate the risk associated with the DMW failures for various components of the boiler, and 4) use the calculated risks to rank the components for maintenance planning.

The evaluation was performed using only the readily available data at the plant site, and, as a result, was based mostly on the primary loads. Since neither detailed tube support data for determining axial stresses from dead weight nor secondary stress data



AIn	Division Panel Inlet	AOt	Division Panel Outlet
BIn	Platen Inlet	BOt	Platen Outlet (Penthouse)
CIn	Pendant Platen Inlet	COt	Pendant Platen Outlet
DIn	Final Superheater Inlet	DOt	Final Superheater Outlet
A	Division Panels	B	Platen Superheater
C	Pendant Platen Superheater	D	Final Superheater
H	Horizontal Reheater	V	Vertical Reheater
		VOt	Vertical Outlet - Reheater

Figure 30. Superheater Portion of a Fossil-Fuel-Fired Boiler

for various DMW locations were available, the stress estimate was based solely on the axial pressure stress (plug stress) in most cases. Although the secondary stresses were unknown, in order to acknowledge those stresses due to potential bending loads, an additional 20% stress factor was used for welds on horizontal runs. The assignment of a secondary load factor was somewhat arbitrary and probably low, but it provided some contribution of secondary loading to the ranking, and incorporated secondary stresses in the Excel model which could be modified in the event that actual secondary stress data are obtained. To acknowledge the significance of the structural discontinuities associated with the DMWs, where the difference between the tube wall thickness between one side of the DMW joint and the other exceeded 20%, an additional 20% stress factor was applied. The plant had experienced numerous cycles over its service life, but since the actual number of cycles was not known, it was not considered.

The hours of operation and the operating temperature information were used to calculate the Larson-Miller parameters for DMWs in the various components of the boiler.

The failure probability considered the axial pressure stress at each dissimilar metal weld joint and the Larson-Miller parameter (P), calculated based on pressure and temperature as described in Section 4.3.4.2. The axial stress due to pressure (plug stress) was used for the evaluation and was calculated for a thick-walled tube as follows: [57, 63]

$$S = P_i\,(R_i)^2/(R_o)^2\text{-}(R_i)^2 \qquad [31]$$

Where P_i = pressure
R_i = inside radius
R_o = outside radius

The Larson-Miller parameter and stress data for operating and failed DMWs in the subject plant, as well as several similar failed DMWs from the EPRI study, is shown in Figure 31. The trend toward a strong qualitative correlation between the Larson-Miller parameter and DMW failure is also illustrated in Figure 31. The influence of stress is not as apparent, probably because the lack of actual secondary stress data resulted in lower stresses overall. For example, failure data point at 34.6 results does not accurately reflect that the failed Platen Superheater Outlet DMW configuration was highly restrained due to its proximity to the penthouse floor penetration, i.e., the DMW was a foot or less from the penthouse floor penetration. A DMW failure in a similar location at the Conemaugh Unit 2 plant was attributed to restraints at the pentetration.(57) The supporting failure data for Figure 31 is shown in Appendix B. For the plant evaluated in this case study, data from DMWs removed because of significant damage revealed by ultrasonic or radiographic inspection, were considered as failures.

The probability of a DMW failure was based on the Larson-Miller parameter data from failed DMWs reported by EPRI and from the plant's failure data, as tabulated in Appendix B and shown in Figure 31.

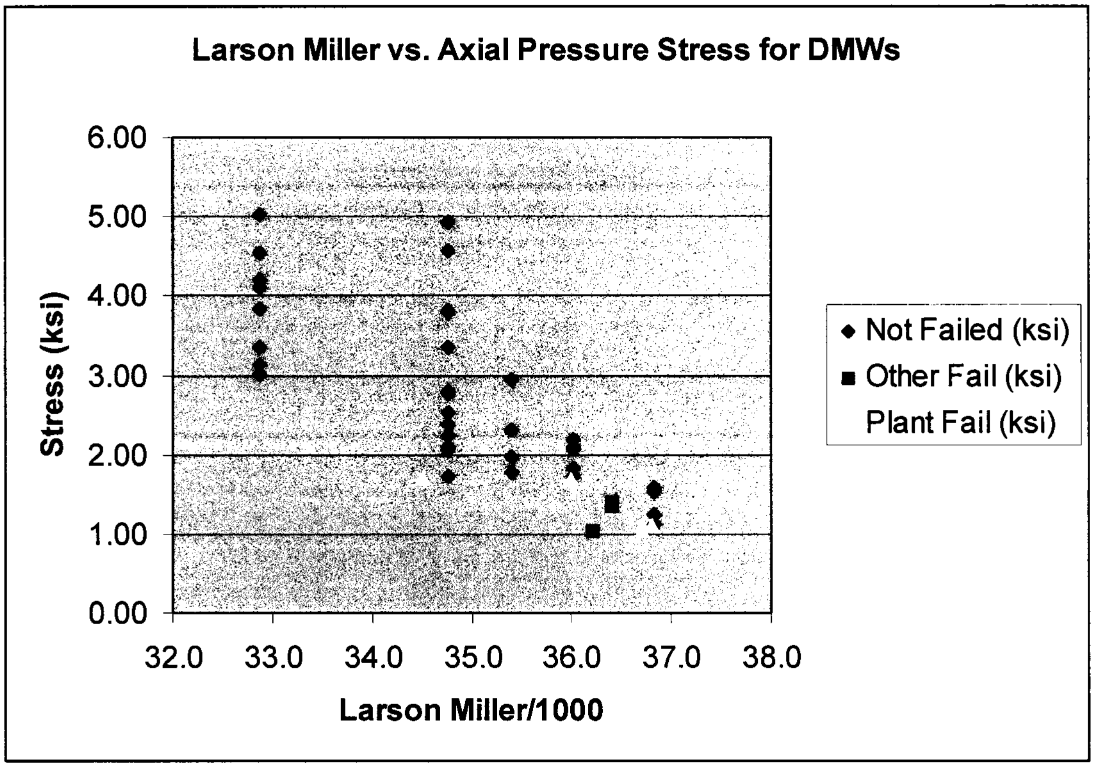


Figure 31. Larson-Miller Plot for Failed and In-Service DMWs

The distribution of the Larson Miller parameters from the DMW failures is shown in Figure 32. The Platen Superheater Outlet DMW failure mentioned above was not included as a data point for Figure 32, since that configuration is highly restrained but had not been modeled, and, therefore, was not considered representative of the general condition. Because there were so few DMW failure data points, the mean and confidence interval shown in Figure 33, based on the small sample t-test, was considered a best effort for this evaluation. The lower bound 95% confidence of the mean, 36.5 (Larson-Miller/1000), was selected as the point at which failure should be strongly considered. To assign a probability of failure for DMWs with Larson-Miller parameter

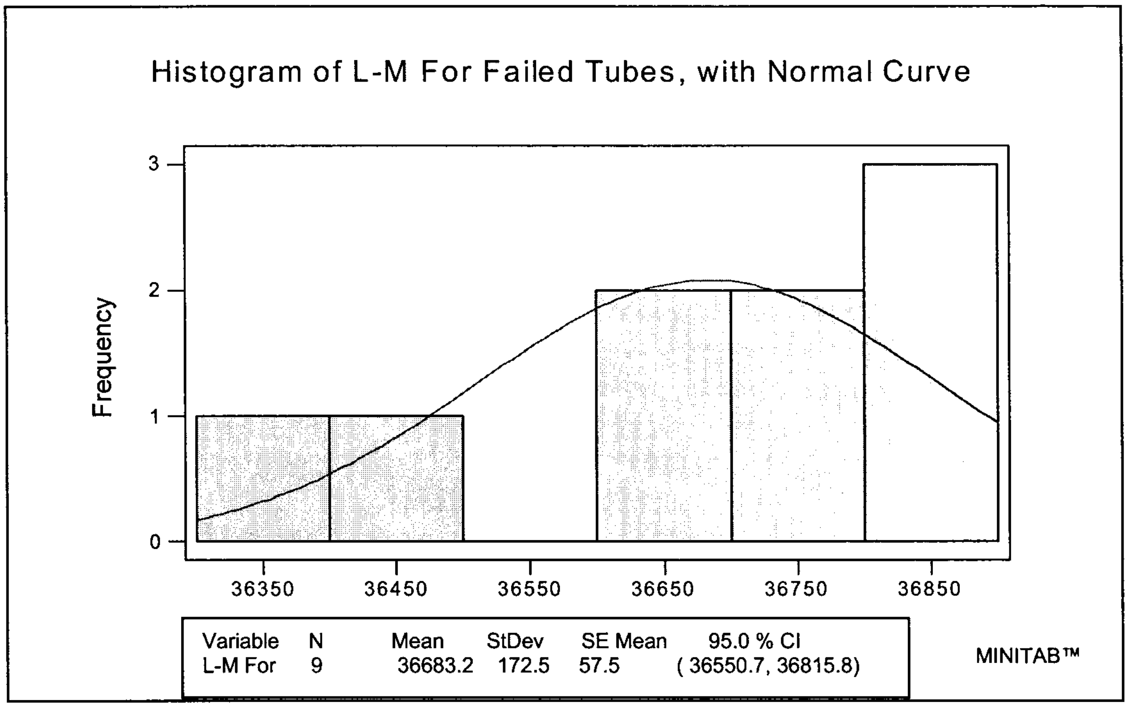


Figure 32. Distribution of Larson-Miller Parameters for Failed DMWs

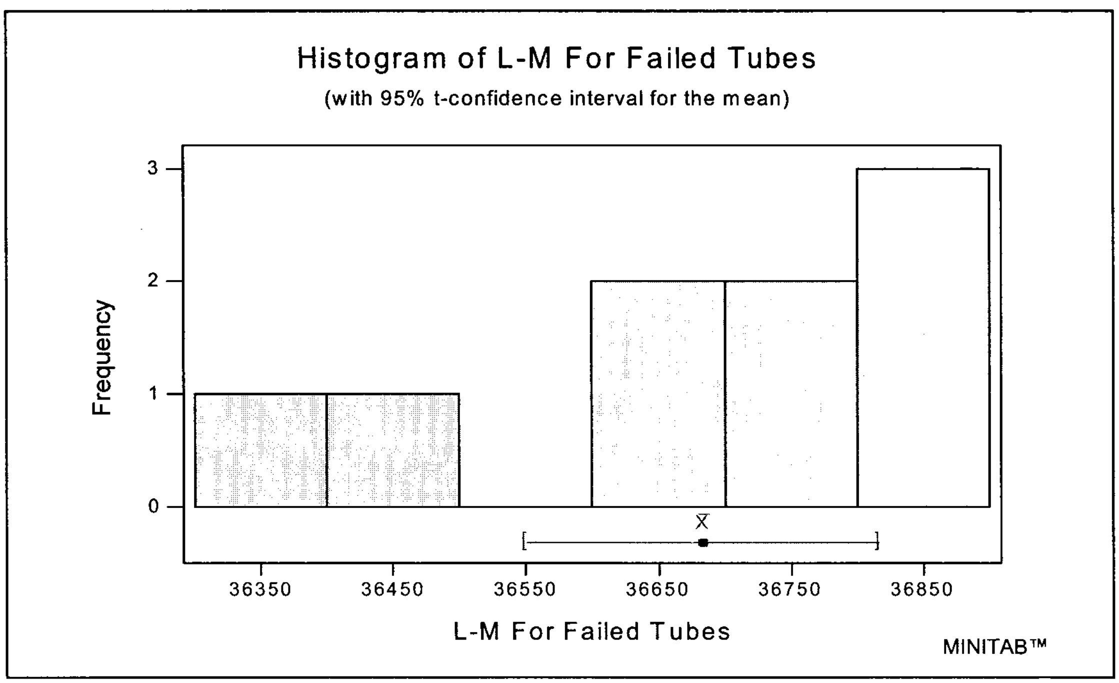


Figure 33. Small Sample t-Test for Larson-Miller Parameters of Failed DMWs

values greater than 36.5 x 10^3 , the data from Appendix B data for the Final Superheater (FSH) Outlet Header was used. The FSH Outlet Header has 882 DMWs; since two failed in 2001, one in 2002, and 2 in 2003, a probability of 0.002, or 2 in 1000, was assigned to indicate probability of a DMW failure in one year. Although the Final Superheater platen experienced two failures in 1996 with a smaller total number of DMWs, which would have generated a higher probability, most of those DMWs, all located on the outside lanes, were preventively replaced using nickel-based filler metal. The remaining welds in the Final Superheater were not in the same outside lane area, which was possibly exposed to higher and more direct heat; therefore, the lower probability based on the FSH Outlet Header was still considered acceptable for the remaining Final Superheater DMWs. For DMWs with Larson-Miller parameter values less than 35.6 x 10^3, a probability of 0.0001, or 1 in 10,000, was assigned. The usefulness of the axial pressure stress values appeared limited but to acknowledge that a stress contribution exists, an additional probability factor of 0.0005, or 5 in 1000, was added to the probability if the stress exceeded 4 ksi. If more detailed stress information is obtained in the future, more meaningful treatment of stresses can be developed for this model.

A series reliability model was used based on the operating assumption that a single DMW failure would cause failure of the component. Also, the risk was calculated based on the probability of a DMW failure for each component. Therefore, the risk increased if the Larson-Miller threshold of 35.6 x 10^3 for a DMW was exceeded and increased proportionally by the number of DMWs in the component as well.

The economic consequences were calculated based on a combination of the estimated costs for the actual repair work plus the penalty for plant downtime in hours, an amount set by the state regulator. Assumptions regarding costs and time associated with repairs are shown in Table 9.

Table 9. Repair Costs & Time

Area	Cost of Repair Work	Time to Repair (MTTR)	Delay Time (DT) (ADT + LDT)	Remarks
Penthouse	$40K	132 hrs (5.5 days)	78 hrs (~3 days)	single tube - no collateral damage
Firebox & Reheater	$80K	132 hrs (5.5 days)	102 hrs (~4 days)	scaffolding ($40K)
Firebox (D) inaccessible tubes	$40K	132 hrs (5.5 days)	142 hrs (~6 days)	single tube - 40 hrs to identify tube; scaffolding ($80K)
Firebox (D or C)	$140K	336 hrs (14 days)	126 hrs (~5 days)	collateral damage ($100K) scaffolding ($40K)
Firebox (A or B)	$180K	336 hrs (14 days)	150 hrs (~6 days)	collateral damage ($100K) extensive scaffolding ($80K)

The cost of time required for forced outages was estimated using a penalty of $10,000/hr of downtime. The economic consequence considered the probability of collateral damage events as 1/100 based on the experience and judgment of plant experts.

It was assumed that the failure of a single DMW constituted the failure of the component. For that purpose the following items were identified as components:

AIn	Division Panel Inlet	AOt	Division Panel Outlet
BIn	Platen Inlet	BOt	Platen Outlet (Penthouse)
CIn	Pendant Platen Inlet	COt	Pendant Platen Outlet
DIn	Final Superheater Inlet	DOt	Final Superheater Outlet
A	Division Panels	B	Platen Superheater
C	Pendant Platen Superheater	D	Final Superheater
H	Horizontal Reheater	V	Vertical Reheater

The configuration of those components in the boiler is shown in Figure 30.

An example of consequence determination for components is shown in Table 10. Considering the Division Panels and Platen Superheater in the Furnace area, for a single tube failure without collateral damage, the repair work alone is $80,000 and is identical to the consequence when no penalty charge applies (Consequence 2 in Table 10). If the plant has already been unavailable for a time greater than that negotiated with the regulator for a given year, a penalty charge is assessed for the time lost due to the DMW failure. That penalty charge is represented by the mean time to repair (MTTR) and the delay time (DT) multiplied by the penalty charge. The cost of repair plus the penalty charge is shown in Table 10 as Consequence 1. Since a tube failure might cause collateral damage, the collateral damage consequence are shown in the next row. For determination of the assumed consequences in Table 10, the collateral damage was considered by adding 1/100 of its consequence value (based on its probability) to the consequence of a single DMW failure. For the Final Superheater in the furnace and the Penthouse header, collateral damage was not a possibility due to configuration and was not included.

An example of component risk determination is shown for the Platen Outlet in Table 11. In the manner illustrated by Table 11, the risks were calculated for each component based on the failure probabilities and component-specific failure consequences. The risk calculation tables for all of the components are provided in Appendix C.

Table 10. Consequence Determinations

Consequence: Division Panels & Platen Superheater - Furnace						
Type	Rep Work	MTTR	DT	Penalty	Consequence 1	Consequence 2
Single	80000	132	102	10000	2420000	80000
single failure with collateral damage	180000	336	150	10000	5040000	180000
Assumed					2470400	81800
Consequence: Final Superheater or Pendant Platen Superheater - Furnace						
Type	Rep Work	MTTR	DT	Penalty	Consequence 1	Consequence 2
Single	80000	132	102	10000	2420000	80000
single failure with collateral damage	140000	336	126	10000	4760000	140000
Assumed					2467600	81400
For Reheater - Assume same as above w/o Collateral Damage factor						
Consequence: Final Superheater - Furnace (for inaccessible tubes only)						
Type	Rep Work	MTTR	DT	Penalty	Consequence 1	Consequence 2
Single	40000	132	142	10000	2780000	40000
Consequence: - Headers - Penthouse						
Type	Rep Work	MTTR	DT	Penalty	Consequence 1	Consequence 2
Single	40000	132	78	10000	2140000	40000

Consequences in dollars - Consequence 1 is with Penalty Charge, Consequence 2 is without

Table 11. Component Risk Determination

BOt Platen Outlet (Penthouse)										
Larson-Miller/1000	Stress (ksi)	No. of DMWs	Probability from L-M	Stress Factor	Prob Singl DMW	Reliability	Cons Equation 1*	Risk 1*	Cons Equation 2*	Risk 2*
34.8	3.80	73	0.0001	0	0.0001	0.993				
34.8	1.72	124	0.0001	0	0.0001	0.988				
34.8	2.06	292	0.0001	0	0.0001	0.971				
Total		489				0.952	2140000	102133.59	40000	1909.04

consequence and risk values in dollars

The relative risk ranking for the various components from the data in Appendix C is shown in Table 12. A combination of the probability of a tube failure and the number of tubes in the component with a high probability of failure drives the risk ranking.

Table 12. Risk of DMW Failure by Component

Segment	Location	Risk w/Penalty	Risk w/o Penalty
Vertical Outlet -Reheater	Furnace	1864399	61633
Final Superheater Pendant Outlet	Penthouse	1773941	33158
Final Superheater Pendant	Furnace	708741	10198
Platen Superheater	Furnace	397084	13148
Horizontal Reheater	Furnace	299905	9914
Division Panels	Furnace	174415	5775
Pendant Platen Inlet	Penthouse	154261	2883
Platen Outlet	Penthouse	102134	1909
Pendant Platen Outlet	Penthouse	76499	1430
Pendant Platen Superheater	Furnace	4438	146

Risk in dollars

The vertical outlet reheater ranks highest due to its large number of tubes having a high failure probability, i.e., those with a Larson-Miller parameter exceeding 36.5×10^3. That vertical outlet reheater had been overlooked because there had not yet been any tube failures; the evaluation showed that preventive maintenance should be strongly considered. Although the Platen Outlet DMWs are highly restrained and that restraint contributed to past failures, insufficient data was obtained to quantify the magnitude of the effect at the time of this evaluation. That limitation was identified so that it could be considered separately in a qualitative manner. Finally, an additional consequence related to the personnel hazard associated with a penthouse DMW failure, i.e., a penthouse outer wall rupture, was also noted for qualitative consideration.

The risk evaluation model developed for this plant was designed such that modifications could be made easily as the hours of operation increase and as more data becomes available for this boiler. Despite its limitations, the risk ranking can be used to evaluate component replacement alternatives and to prioritize mitigation efforts, such as inspection and strategic DMW replacement.

4.4 Discussion

A basic assumption in this study was that risk evaluation methodology could be reasonably adapted to the process of maintenance. Further, it was assumed that the methods identified in the study were sufficiently representative of the available methodologies used in industry, and that they could be recognized and adapted for project-specific use by most readers. These are reasonable assumptions because risk is simply a function of the probabilities and consequences associated with all maintenance activities. The methodology for risk evaluation is well established and has been proven in other applications, which are discussed in this study; therefore, it was reasonable to assume that the methodology could also be applied to maintenance projects.

The application of risk evaluation has been demonstrated for a variety of maintenance scenarios in examples in Section 4.2 and the case histories in Section 4.3. Each example and case involved unique components and circumstances; therefore, the detailed discussion of each risk evaluation was included within each example and case history. The examples and case histories enabled various evaluation tools to be utilized, including MIL-STD-882, presented as a modern graphic risk analysis tool, and probabilistic risk-ranking using fault tree analysis. Fitness for service evaluations were integrated into the overall risk evaluation process through several examples, and the case histories involved scheduling and decision making in a risk-based context. Nevertheless there were some limitations.

The scope of the study was limited by economic and time constraints that prevented more on-site facility visits and investigations of failures that might have been of interest for the research. Access to literature was limited because many of the publications on this topic are unavailable in libraries to which the researcher has reasonable access; a number had to be purchased by the researcher. The scope was also limited by the background and experience of the researcher, a metallurgical engineer, such that the study focused on mechanical systems and equipment and emphasized material considerations.

Chapter 5

Summary, Conclusions, and Recommendations

5.1 Summary

Based on a study of the existing body of work performed specifically to research risk and related work in maintenance systems engineering, fitness for service evaluation, scheduling methods, and decision analysis, a general approach to risk-based maintenance engineering and management has been developed in this study. The approach includes the system or component definition, quantitative and/or qualitative risk assessment, a maintenance needs analysis, and a maintenance scheduling analysis. Various tools, such as the Failure Modes and Effects Analysis (FMEA), fault trees, fitness for service evaluation, and PERT and CPM scheduling methods, were identified and applied in support of developing a robust general approach. This study has described the application of risk assessment methods in the context of various maintenance engineering and management issues.

The application of the approach consisted of using some of the various tools in a series of examples, and applying the approach in four maintenance case histories. As maintenance case histories showed, each maintenance project has its own unique issues, and the generalized approach is easily tailored to address those issues. The studies included two systems studies and two component studies. The system studies, power plant piping for erosion-corrosion and boiler tubes for dissimilar metal weld degradation, emphasized risk assessment and ranking in support of managing the ongoing issue. The component studies, the remote handling equipment for cable degradation, and jet pump beams for stress corrosion cracking, addressed the assessment of risk, but emphasized maintenance needs and scheduling. The case studies also illustrated that aspects of risk assessment may vary over the maintenance life cycle. For example, the jet pump beam was a mature product with a prior history of failures, a background of qualitative risk studies, and maintenance needs analysis in terms of replacement recommendations. The project activity for that case study in the field focused mainly on the immediate maintenance needs and scheduling analysis. The application of the risk-based approach addressed the unique aspects involved in each of the case studies.

5.2 Conclusions

The application of risk-based methods to aspects of maintenance engineering and management was successfully developed. The variety of maintenance scenarios presented in this study is an illustration of the versatility and benefit of risk evaluation as a tool for maintenance engineering and management. Those examples and case histories showed that maintenance needs and scheduling can be optimized through risk assessment. The generalized approach was demonstrated to be a useful and flexible context in which to use tools appropriate to each maintenance project. The overall approach has been demonstrated to be a practical means of approaching maintenance issues. The flexible application of the approach allows the risk assessment, maintenance needs analysis, and scheduling analysis to fit almost any project.

5.3 Implications

The risk-based approach to maintenance engineering and management developed in this study can be successfully applied to reduce the costs of plant maintenance. The concept of risk-based engineering and management has been made accessible by the development of this approach and may be easily understood and applied by maintenance engineers in the field.

5.4 Recommendations for Further Research

A number of possibilities for further research in the field of risk-based maintenance engineering and management follow from this study. Additional case studies involving industries other than those discussed in this study, such as buildings and grounds facilities management, and highway and transportation applications can yield more issue-specific approaches and may involve the identification and development of additional tools. As this study emphasized iron and nickel alloys, studies involving other materials, plastics and composites, for example, that are becoming more widely used and in some cases replacing their metallic counterparts, would be a useful contribution to the body of research. Study of the risks involved in various types of maintenance contracts analyzed from the perspective of both the risk from the customer and vendor would complement the study of risk presented in this study, which emphasized maintenance of the hardware from the owner's perspective.

Because risk analysis is a useful context in which to understand issues, and can result in informed decision making and cost savings, this area of research is expected to expand and many additional opportunities for research will be identified.

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Appendix A

INDEXED COST CALCULATION

Annual Interest (i) and inflation (j)				
i =	0.07	interest (MARR)	1+i =	1.07
j =	0.04	inflation	1+j =	1.04

d = [(1+i)/(1+j)]-1 =	0.028846	years (n) =	10	Annual Est. Cost =	10650
α = 1/(1+d) =	0.971963	α^n =	0.752481	Estimated cost ignoring inflation	

The indexed costs discounted using an interest rate of 7% and an inflation rate of 4% for n=10 years is

PW = cost (1 - α^n/d) = 91383.97

Method of Reference 48, see pages 113-117

Appendix B

DMW Failure Data from Evaluated Power Plant and Other Plants

DMW Failure Data

L-M & Stress Calcs for Failed Tubes from EPRI CS-4252

Sample	Location	Material	Temp F	Temp R	Time (hr)	Log Time	P	Press (psi)	Dia (in)	Wall (in)	Inside Rad	Ro2	Ri2	Stress (ksi)
Conemaugh 2 - EPRI CS-4252, Vol 4, p B4-4**	Penthouse below Superheater head	T-22/347H w/309	1005	1464.7	50000	4.70	36176.3	3800	1.5	0.400	0.350	0.563	0.1225	1.058
Bridgeport 3 - EPRI CS-4252, Vol 4, p A1-6*	Platen Pendant	T-22/347H w/309	1000	1459.7	88000	4.94	36411.2	2060	2	0.360	0.640	1.000	0.4096	1.429
River Rouge 3 - EPRI CS-4252, Vol 4, p A3-9***	Superheater Platen	T-22/321H w/310-15	1000	1459.7	84000	4.92	36381.7	2725	2.5	0.531	0.719	1.563	0.517	1.347

Notes:

*Failures were recorded, but specific date data was not given in all cases - hours were based on the service hours experienced by the samples

**Conemaugh PH DMWs were highly restrained - within one foot of the roof - failures occurred at "less than 50,000 hrs" - 10 failed

*** Unit sarted in 1958 - had 119000 hours by 1982 - failure occurred in 1975; assume 4958 hrs/ year - 17 years (84286 hrs) until failure

P = T(R) * [(20 + LogTime (hr))]

L-M & Stress Calcs for Failed Tubes and Tubes Not Fit for Service from power plant evaluated

Sample	Location	Material	Temp F	Temp R	Time (hr)	Log Time	P	Press (psi)	Dia (in)	Wall (in)	Inside Rad	Ro2	Ri2	Stress (ksi)
Failure in 1996	Final Superheater Platen	T-22/347H w/309	1000	1459.7	1E+05	5.13	36674.7	4000	1.875	0.46	0.4775	0.879	0.228	1.401
Failure in 1996	Final Superheater Platen	T-22/347H w/309	1000	1459.7	1E+05	5.13	36674.7	4000	1.875	0.42	0.5175	0.879	0.2678	1.753
Failure in 1996*	Platen Superheater Outlet	T-11/347H w/309	925	1384.7	91138	4.96	34561.4	4000	1.880	0.425	0.515	0.884	0.2652	1.716
R 3 Tube F 0.350" depth - Apr 2001	Final Superheater Oultet	T-22/347H w/309	1000	1459.7	2E+05	5.19	36772.9	4000	1.875	0.5	0.4375	0.879	0.1914	1.114
R 134 Tube A 0.380" depth - Apr 2001	Final Superheater Oultet	T-22/347H w/309	1000	1459.7	2E+05	5.19	36772.9	4000	1.875	0.5	0.4375	0.879	0.1914	1.114
R 134 Tube A 0.380" depth - Apr 2002	Final Superheater Oultet	T-22/347H w/309	1000	1459.7	2E+05	5.21	36803.4	4000	1.875	0.5	0.4375	0.879	0.1914	1.114
R 2 Tube B 0.200" depth - Jan 2003	Final Superheater Oultet	T-22/347H w/309	1000	1459.7	2E+05	5.23	36828.7	4000	1.875	0.5	0.4375	0.879	0.1914	1.114
R 5 Tube E 0.400" depth - Jan 2003	Final Superheater Oultet	T-22/347H w/309	1000	1459.7	2E+05	5.23	36828.7	4000	1.875	0.5	0.4375	0.879	0.1914	1.114

*Note: BOt weld was highly restrained - close to PH floor penetration - within one foot

P = T(R) * [(20 + LogTime (hr))]

Appendix C

Individual Component Data for DMW Risk Determination

Component Data

AIn Division Panel Inlet			Figure 30				In service since Jul. 1981 (Replaced)					No DMWs	
AOt Division Panel Outlet			Figure 30				In service since Jul. 1981 (Replaced)					No DMWs	
A Division Panels		Dwg. 04680-1C1293 Rev. 00				In service since Jul. 1981 (Replaced)							
B Panel													
Weld Group ID Type*	Dwg Loc	LA Mat'l	LA size	LA min wall	SS Mat'l	SS size	SS min wall	Δ-wall	%Δ wall	num tubes	Quantity	Loc	Remark
DW-1	E-6	T-11	2.50	0.338	347H	2.5	0.360	0.022	6.5	1	6	Furn	outside loop
DW-2	E-6	T-11	2.25	0.338	347H	2.25	0.310	0.0	8.3	2	12	Furn	
DW-2	F-5/6	T-11	2.25	0.338	347H	2.25	0.310	0.0	8.3	2	12	Furn	
DW-4	F-5	T-11	2.25	0.338	347H	2.25	0.310	0.0	8.3	4	24	Furn	
DW-1	D-5	T-11	2.50	0.360	347H	2.5	0.360	0.0	0.0	1	6	Furn	outside loop
A-9	n/a	T-11	2.25	0.320				0.0	0.0	9	54		not on dwg
A-4	n/a	T-11	2.25	0.380				0.0	0.0	4	24		not on dwg
A-3	n/a	T-11	2.25	0.388				0.0	0.0	3	18		not on dwg
A-3	n/a	T-11	2.25	0.320				0.0	0.0	3	18		not on dwg
Total										29	174	Furn	
A Panel													
Weld Group ID Type*	Dwg Loc	LA Mat'l	LA size	LA min wall	SS Mat'l	SS size	SS min wall	Δ-wall	%Δ wall	num tubes	Quantity	Loc	Remark
DW-1	D-5	T-11	2.50	0.360	347H	2.50	0.360	0.0	0.0	1	6	Furn	outside loop
DW-2	F-5	T-11	2.25	0.338	347H	2.25	0.310	0.0	8.3	2	12	Furn	
DW-2	F-4/5	T-11	2.25	0.338	347H	2.25	0.310	0.0	8.3	2	12	Furn	
DW-2	F-4	T-11	2.25	0.338	347H	2.25	0.310	0.0	8.3	2	12	Furn	
DW-2	E-4	T-11	2.25	0.338	347H	2.25	0.310	0.0	8.3	2	12	Furn	
DW-2	F-3/4	T-11	2.25	0.338	347H	2.25	0.310	0.0	8.3	2	12	Furn	
DW-1	E-3	T-11	2.50	0.338	347H	2.25	0.310	0.0	8.3	1	6	Furn	outside loop
DW-1	F-3	T-11	2.12	0.380	347H	2.12	0.360	0.0	5.3	1	6	Furn	horizontal
A-13	n/a	T-11	2.25	0.320				0.0000	0.0	13	78	Furn	not on dwg
Total										26	156	Furn	

Component Data

A Division Panels		Figure 30				In service since Jul. 1981 (Replaced)			hours =	126524				
B Panel														
Dwg Loc	Temp F	Temp R	Time (hr)	Log Time	P	Press (psi)	Dia (in)	Wall (in)	Inside Rad	Ro2	Ri2	Axial Press. Stress (ksi)	stress multiplier for bending	stress mult. for Δ-Wall
E-6	850	1309.69	126524	5.10	32876	4000	2.50	0.338	0.912	1.563	0.832	4.553		0.000
E-6	850	1309.69	126524	5.10	32876	4000	2.25	0.338	0.787	1.266	0.619	3.834		0.000
F-5/6	850	1309.69	126524	5.10	32876	4000	2.25	0.338	0.787	1.266	0.619	3.834		0.000
F-5	850	1309.69	126524	5.10	32876	4000	2.25	0.338	0.787	1.266	0.619	3.834		0.000
D-5	850	1309.69	126524	5.10	32876	4000	2.50	0.360	0.89	1.563	0.792	4.113		0.000
D-5	850	1309.69	126524	5.10	32876	4000	2.25	0.320	0.805	1.266	0.648	4.197		0.000
D-5	850	1309.69	126524	5.10	32876	4000	2.25	0.380	0.745	1.266	0.555	3.124		0.000
D-5	850	1309.69	126524	5.10	32876	4000	2.25	0.388	0.737	1.266	0.543	3.007		0.000
D-5	850	1309.69	126524	5.10	32876	4000	2.25	0.320	0.805	1.266	0.648	4.197		0.000
APanel														
Dwg Loc	Temp F	Temp R	Time (hr)	Log Time	P	Press (psi)	Dia (in)	Wall (in)	Inside Rad	Ro2	Ri2	Axial Press. Stress (ksi)	stress multiplier for bending	stress mult. for Δ-Wall
D-5	850	1309.69	126524	5.10	32876	4000	2.50	0.360	0.89	1.563	0.792	4.113		0.000
F-5	850	1309.69	126524	5.10	32876	4000	2.25	0.338	0.787	1.266	0.619	3.834		0.000
F-4/5	850	1309.69	126524	5.10	32876	4000	2.25	0.338	0.787	1.266	0.619	3.834		0.000
F-4	850	1309.69	126524	5.10	32876	4000	2.25	0.338	0.787	1.266	0.619	3.834		0.000
E-4	850	1309.69	126524	5.10	32876	4000	2.25	0.338	0.787	1.266	0.619	3.834		0.000
F-3/4	850	1309.69	126524	5.10	32876	4000	2.25	0.338	0.787	1.266	0.619	3.834		0.000
E-3	850	1309.69	126524	5.10	32876	4000	2.50	0.338	0.912	1.563	0.832	4.553		0.000
F-3	850	1309.69	126524	5.10	32876	4000	2.12	0.380	0.68	1.124	0.462	2.797	0.200	0.000
n/a	850	1309.69	126524	5.10	32876	4000	2.25	0.320	0.805	1.266	0.648	4.197	0.200	0.000

Component Data

A Division Panels									
B Panel									
Larson-Miller/1000	Stress (ksi)	Number of Tubes	Probability from L-M	Stress cont. to Prob.	Prob Singl Tube	Reliability	Conseqence 1	Risk 1	Conseqence 2	Risk 2
32.9	4.55	6	0.0001	0.0005	0.0006	0.996405				
32.9	3.83	12	0.0001	0	0.0001	0.998801				
32.9	3.83	12	0.0001	0	0.0001	0.998801				
32.9	3.83	24	0.0001	0	0.0001	0.997603				
32.9	4.11	6	0.0001	0.0005	0.0006	0.996405				
32.9	4.20	54	0.0001	0.0005	0.0006	0.96811				
32.9	3.12	24	0.0001	0	0.0001	0.997603				
32.9	3.01	18	0.0001	0	0.0001	0.998202				
32.9	4.20	18	0.0001	0.0005	0.0006	0.989255				
subtotal		174				0.942315	2470400	142504.82	81800	4718.626
A Division Panels										
A Panel										
Larson-Miller/1000	Stress (ksi)	Number of Tubes	Probability from L-M	Stress cont. to Prob.	Prob Singl Tube	Reliability	Conseqence 1	Risk 1	Conseqence 2	Risk 2
32.9	4.11	6	0.0001	0.0005	0.0006	0.996405				
32.9	3.83	12	0.0001	0	0.0001	0.998801				
32.9	3.83	12	0.0001	0	0.0001	0.998801				
32.9	3.83	12	0.0001	0	0.0001	0.998801				
32.9	3.83	12	0.0001	0	0.0001	0.998801				
32.9	3.83	12	0.0001	0	0.0001	0.998801				
32.9	4.55	6	0.0001	0.0005	0.0006	0.996405				
32.9	3.36	6	0.0001	0	0.0001	0.9994				
32.9	5.04	78	0.0001	0.0005	0.0006	0.954265				
Total		330				0.929398	2470400	174414.84	81800	5775.23

Component Data

BIn Platen Inlet			Figure 30				In service since Jul. 1981 (Replaced)					No DMWs		
BOt Platen Outlet			Figure 30				In service since Jul. 1981 (Replaced)							
Weld Group ID Type*	Dwg Loc	LA Mat'l	LA size	LA min wall	SS Mat'l	SS size	SS min wall	Δ-wall	%Δ wall	per SH	Quantity	Loc	Remark	Number Replaced
DW-1	B-5	T-11	2.25	0.340	347H	2.12	0.380	0.040	11.8	1	73	Pent		
DW-2	B-5	T-11	1.88	0.425	347H	1.88	0.425	0.000	0.0	2	124	Pent	22Repl.w/Inc.96/7	22
DW-4	B-5	T-11	1.88	0.425	347H	2.00	0.270	0.155	36.5	4	292	Pent		
Total										7	489	Pent		
B Platen Superheater			Figure 30				In service since Jul. 1981 (Replaced)							
Weld Group ID Type*	Dwg Loc	LA Mat'l	LA size	LA min wall	SS Mat'l	SS size	SS min wall	Δ-wall	%Δ wall	per SH	Quantity	Loc	Remark	
DW-2	C-5	T-11	2.00	0.450	347H	2.00	0.270	0.180	40.0	2	146	Furn -B		
DW-1	F-5	T-11	2.12	0.380	347H	2.00	0.311	0.069	18.2	1	73	Furn -B	horizontal	
DW-2	D-4/5	T-11	2.12	0.380	347H	2.00	0.311	0.069	18.2	2	146	Furn -B		
DW-2	F/G-4/5	T-11	2.12	0.380	347H	2.12	0.420	0.040	10.5	2	146	Furn -B	not on as-found	
DW-2	F/G-4	T-11	2.12	0.320	347H	2.12	0.420	0.100	31.3	2	146	Furn -A		
DW-1	F-3/4	T-11	2.00	0.425	347H	2.00	0.300	0.125	29.4	1	73	Furn -A	bifurcation	
DW-1	F-3/4	T-11	2.00	0.425	347H	2.00	0.300	0.125	29.4	1	73	Furn -A	near bifurcation	
DW-2	E/F-3/4	T-22	2.00	0.395	347H	2.00	0.270	0.125	31.6	2	146	Furn -A		
DW-1	G-3	T-22	2.12	0.425	347H	2.13	0.360	0.065	15.3	1	73	Furn -A	outside loop	
Total										14	1022	Furn		

Component Data

BOt Platen Outlet			Figure 30			In service since Jul. 1981 (Replaced)			hours =	126524				
Dwg Loc	Temp F	Temp R	Time (hr)	Log Time	P	Press (psi)	Dia (in)	Wall (in)	Inside Rad	Ro2	Ri2	Axial Press. Stress (ksi)	stress multiplier for bending	stress mult. for Δ-Wall
B-5	925	1384.69	126524	5.10	34759	4000	2.25	0.340	0.785	1.266	0.616	3.796		0.000
B-5	925	1384.69	126524	5.10	34759	4000	1.88	0.425	0.515	0.884	0.265	1.716		0.000
B-5	925	1384.69	126524	5.10	34759	4000	1.88	0.425	0.515	0.884	0.265	1.716		0.200

B Platen Superheater			Figure 30			In service since Jul. 1981 (Replaced)			hours =	126524				
Dwg Loc	Temp F	Temp R	Time (hr)	Log Time	P	Press (psi)	Dia (in)	Wall (in)	Inside Rad	Ro2	Ri2	Axial Press. Stress (ksi)	stress multiplier for bending	stress mult. for Δ-Wall
C-5	925	1384.69	126524	5.10	34759	4000	2.00	0.450	0.55	1.000	0.303	1.735		0.200
F-5	925	1384.69	126524	5.10	34759	4000	2.12	0.380	0.68	1.124	0.462	2.797	0.200	0.000
D-4/5	925	1384.69	126524	5.10	34759	4000	2.12	0.380	0.68	1.124	0.462	2.797		0.000
F/G-4/5	925	1384.69	126524	5.10	34759	4000	2.12	0.380	0.68	1.124	0.462	2.797		0.000
F/G-4	925	1384.69	126524	5.10	34759	4000	2.12	0.320	0.74	1.124	0.548	3.803		0.200
F/G-3/4	925	1384.69	126524	5.10	34759	4000	2.00	0.425	0.575	1.000	0.331	1.976		0.200
F-3/4	925	1384.69	126524	5.10	34759	4000	2.00	0.425	0.575	1.000	0.331	1.976		0.200
E/F-3/4	925	1384.69	126524	5.10	34759	4000	2.00	0.395	0.605	1.000	0.366	2.309		0.200
G-3	925	1384.69	126524	5.10	34759	4000	2.12	0.425	0.635	1.124	0.403	2.239		0.000

Component Data

BOt Platen Outlet										
Larson-Miller/1000	Stress (ksi)	Number of Tubes	Probability from L-M	Stress cont. to Prob.	Prob Singl Tube	Reliability	Conseqence 1	Risk 1	Conseqence 2	Risk 2
34.8	3.80	73	0.0001	0	0.0001	0.992726				
34.8	1.72	124	0.0001	0	0.0001	0.987676				
34.8	2.06	292	0.0001	0	0.0001	0.971221				
Total		489				0.952274	2140000	102133.59	40000	1909.04

B Platen Superheater										
Larson-Miller/1000	Stress (ksi)	Number of Tubes	Probability from L-M	Stress cont. to Prob.	Prob Singl Tube	Reliability	Conseqence 1	Risk 1	Conseqence 2	Risk 2
34.8	2.08	146	0.0001	0	0.0001	0.985505				
34.8	3.36	73	0.0001	0	0.0001	0.992726				
34.8	2.80	146	0.0001	0	0.0001	0.985505				
34.8	2.80	146	0.0001	0	0.0001	0.985505				
34.8	4.56	146	0.0001	0.0005	0.0006	0.916103				
34.8	2.37	73	0.0001	0	0.0001	0.992726				
34.8	2.37	73	0.0001	0	0.0001	0.992726				
34.8	2.77	146	0.0001	0	0.0001	0.985505				
34.8	2.24	73	0.0001	0	0.0001	0.992726				
Total		1022				0.839263	2470400	397083.68	81800	13148.25

Component Data

CIn Pendant Platen Inlet				Figure 30				In service since Jul. 1981 (Replaced)					
Weld Group ID Type*	Dwg Loc	LA Mat'l	LA size	LA min wall	SS Mat'l	SS size	SS min wall	Δ-wall	%Δ wall	per SH	Quantity	Loc	Remark
DW-1	B-5	T-11	1.88	0.388	347H	2.00	0.270	0.1	30.4	1	18	Pent	18 remaining
DW-1	B-5	T-11	1.88	0.388	347H	2.00	0.270	0.1	30.4	1	73	Pent	
DW-2	B-5	T-11	2.12	0.320	347H	2.00	0.270	0.1	15.6	2	146	Pent	
DW-1	B-5	T-11	1.88	0.388	347H	2.00	0.270	0.1	30.4	1	73	Pent	
DW-1	B-5	T-22	2.50	0.360	347H	2.25	0.440	0.080	22.2	1	73	Pent	outside loop
Total										6	383	Pent	
C Pendant Platen Superheater				Figure 30				In service since Jul. 1981 (Replaced)					
Weld Group ID Type*	Dwg Loc	LA Mat'l	LA size	LA min wall	SS Mat'l	SS size	SS min wall	Δ-wall	%Δ wall	per SH	Quantity	Loc	Remark
DW-1	D-6	T-22	2.00	0.475	347H	2.00	0.310	0.165	34.7	1	18	Furn -A	18 remaining-T9
													not on as-found
Total										1	18	Furn	
COt Pendant Platen Outlet				Figure 30				In service since Jul. 1981 (Replaced)					
Weld Group ID Type*	Dwg Loc	LA Mat'l	LA size	LA min wall	SS Mat'l	SS size	SS min wall	Δ-wall	%Δ wall	per SH	Quantity	Loc	Remark
DW-1	B-7	T-22	2.50	0.507	347H	2.25	0.440	0.067	13.2	1	73	Pent	outside loop
DW-1	B-7	T-22	1.88	0.425	347H	2.00	0.300	0.125	29.4	1	73	Pent	
DW-1	B-7	T-22	2.25	0.507	347H	2.00	0.300	0.207	40.8	1	73	Pent	
DW-1	B-7	T-22	1.88	0.425	347H	2.00	0.282	0.143	33.6	1	73	Pent	
DW-1	B-7	T-22	1.88	0.425	347H	2.00	0.282	0.143	33.6	1	18	Pent	18 remaining
DW-2	B-6	T-22	2.00	0.450	347H	2.00	0.282	0.168	37.3	2	36	Pent	18 remaining
DW-1	B-6	T-22	2.25	0.507	347H	2.25	0.507	0.000	0.0	1	18	Pent	18 remaining
Total										8	364	Pent	

Component Data

CIn Pendant Platen Inlet			Figure 30			In service since Jul. 1981 (Replaced)			hours =	126524				
Dwg Loc	Temp F	Temp R	Time (hr)	Log Time	P	Press (psi)	Dia (in)	Wall (in)	Inside Rad	Ro2	Ri2	Axial Press. Stress (ksi)	stress multiplier for bending	stress mult. for Δ-Wall
B-5	925	1384.69	126524	5.10	34759	4000	1.88	0.388	0.552	0.884	0.305	2.105		0.200
B-5	925	1384.69	126524	5.10	34759	4000	1.88	0.388	0.552	0.884	0.305	2.105		0.200
B-5	925	1384.69	126524	5.10	34759	4000	2.12	0.320	0.74	1.124	0.548	3.803		0.000
B-5	925	1384.69	126524	5.10	34759	4000	1.88	0.388	0.552	0.884	0.305	2.105		0.200
B-5	925	1384.69	126524	5.10	34759	4000	2.50	0.360	0.89	1.563	0.792	4.113		0.200
C Pendant Platen Superheater			Figure 30			In service since Jul. 1981 (Replaced)			hours =	126524				
Dwg Loc	Temp F	Temp R	Time (hr)	Log Time	P	Press (psi)	Dia (in)	Wall (in)	Inside Rad	Ro2	Ri2	Axial Press. Stress (ksi)	stress multiplier for bending	stress mult. for Δ-Wall
D-6	975	1434.69	126524	5.10	36013.8	4000	2.00	0.475	0.525	1.000	0.276	1.522		0.200
COt Pendant Platen Outlet			Figure 30			In service since Jul. 1981 (Replaced)			hours =	126524				
Dwg Loc	Temp F	Temp R	Time (hr)	Log Time	P	Press (psi)	Dia (in)	Wall (in)	Inside Rad	Ro2	Ri2	Axial Press. Stress (ksi)	stress multiplier for bending	stress mult. for Δ-Wall
B-7	975	1434.69	126524	5.10	36013.8	4000	2.50	0.507	0.743	1.563	0.552	2.185		0.000
B-7	975	1434.69	126524	5.10	36013.8	4000	1.88	0.425	0.515	0.884	0.265	1.716		0.200
B-7	975	1434.69	126524	5.10	36013.8	4000	2.25	0.507	0.618	1.266	0.382	1.729		0.200
B-7	975	1434.69	126524	5.10	36013.8	4000	1.88	0.425	0.515	0.884	0.265	1.716		0.200
B-7	975	1434.69	126524	5.10	36013.8	4000	1.88	0.425	0.515	0.884	0.265	1.716		0.200
B-6	975	1434.69	126524	5.10	36013.8	4000	2.00	0.450	0.55	1.000	0.303	1.735		0.200
B-6	975	1434.69	126524	5.10	36013.8	4000	2.25	0.507	0.618	1.266	0.382	1.729		0.000

Component Data

CIn Pendant Platen Inlet										
Larson-Miller/1000	Stress (ksi)	Number of Tubes	Probability from L-M	Stress cont. to Prob.	Prob Singl Tube	Reliability	Conseqence 1	Risk 1	Conseqence 2	Risk 2
34.8	2.53	18	0.0001	0	0.0001	0.998202				
34.8	2.53	73	0.0001	0	0.0001	0.992726				
34.8	3.80	146	0.0001	0	0.0001	0.985505				
34.8	2.53	73	0.0001	0	0.0001	0.992726				
34.8	4.94	73	0.0001	0.0005	0.0006	0.957133				
Total		383				0.927915	2140000	154261.01	40000	2883.38

C Pendant Platen Superheater										
Larson-Miller/1000	Stress (ksi)	Number of Tubes	Probability from L-M	Stress cont. to Prob.	Prob Singl Tube	Reliability	Conseqence 1	Risk 1	Conseqence 2	Risk 2
36.0	1.83	18	0.0001	0	0.0001	0.998202				
Total		18				0.998202	2467600	4437.91	81400	146.40

COt Pendant Platen Outlet										
Larson-Miller/1000	Stress (ksi)	Number of Tubes	Probability from L-M	Stress cont. to Prob.	Prob Singl Tube	Reliability	Conseqence 1	Risk 1	Conseqence 2	Risk 2
36.0	2.19	73	0.0001	0	0.0001	0.992726				
36.0	2.06	73	0.0001	0	0.0001	0.992726				
36.0	2.07	73	0.0001	0	0.0001	0.992726				
36.0	2.06	73	0.0001	0	0.0001	0.992726				
36.0	2.06	18	0.0001	0	0.0001	0.998202				
36.0	2.08	36	0.0001	0	0.0001	0.996406				
36.0	1.73	18	0.0001	0	0.0001	0.998202				
Total		364				0.964253	2140000	76499.09	40000	1429.89

Component Data

D Final Superheater Pendant				Figure 30				In service since Dec. 1972						
Weld Group ID Type*	Dwg Loc	LA Mat'l	LA size	LA min wall	SS Mat'l	SS size	SS min wall	Δ-wall	%Δ wall	per SH	Quantity	Loc	Remark	Number Replaced
DW-2	E-7	T-22	1.875	0.460	347H	1.875	0.460	0.000	0.0	2	294	Furn	Repl.w/Inc.96/7	294
DW-1	D-7	T-22	1.875	0.438	347H	1.875	0.438	0.000	0.0	1	147	Furn	inaccessible	
DW-3	D/E-7	T-22	1.875	0.420	347H	1.875	0.480	0.060	14.3	3	441	Furn	Repl.w/Inc.96/7	441
Total										6	882		Total Replaced:	735
DOt Final Superheater Pendant Outlet				Figure 30				In service since Dec. 1972						
Weld Group ID Type*	Dwg Loc	LA Mat'l	LA size	LA min wall	SS Mat'l	SS size	SS min wall	Δ-wall	%Δ wall	per SH	Quantity	Loc	Remark	
DW-6	B/C-7	T-22	1.875	0.500	347H	1.875	0.500	0.000	0.0	6	882	Pent		
Total										6	882			

Component Data

D Final Superheater Pendant			Figure 30			In service since Dec. 1972			hours =	170021				
Dwg Loc	Temp F	Temp R	Time (hr)	Log Time	P	Press (psi)	Dia (in)	Wall (in)	Inside Rad	Ro2	Ri2	Axial Press. Stress (ksi)	stress multiplier for bending	stress mult. for Δ-Wall
E-7	1000	1459.69	170021	5.23	36828.7	4000	1.88	0.460	0.4775	0.879	0.228	1.401		
D-7	1000	1459.69	170021	5.23	36828.7	4000	1.88	0.438	0.4995	0.879	0.250	1.586		0.000
D/E-7	1000	1459.69	170021	5.23	36828.7	4000	1.88	0.420	0.5175	0.879	0.268	1.753		
DOt Final Superheater Pendant Outlet				Figure 30			In service since Dec.1972			hours =	170021			
Dwg Loc	Temp F	Temp R	Time (hr)	Log Time	P	Press (psi)	Dia (in)	Wall (in)	Inside Rad	Ro2	Ri2	Axial Press. Stress (ksi)	stress multiplier for bending	stress mult. for Δ-Wall
B-7	1000	1459.69	170021	5.23	36828.7	4000	1.88	0.500	0.4375	0.879	0.191	1.114		0.000

Component Data

D Final Superheater Pendant										
Larson-Miller/1000	Stress (ksi)	Number of Tubes	Probability from L-M	Stress cont. to Prob.	Prob Singl Tube	Reliability	Conseqence 1	Risk 1	Conseqence 2	Risk 2
36.8	1.59	147	0.002	0	0.002	0.745057				
Total		147				0.745057	2780000	708741.2	40000	10197.72
DOt Final Superheater Pendant Outlet										
Larson-Miller/1000	Stress (ksi)	Number of Tubes	Probability from L-M	Stress cont. to Prob.	Prob Singl Tube	Reliability	Conseqence 1	Risk 1	Conseqence 2	Risk 2
36.8	1.11	882	0.002	0	0.002	0.171056				
Total		882				0.171056	2140000	1773940.9	40000	33157.77

Component Data

H Horizontal Reheater				Figure 30				In service since Jul. 1980 (Replaced)					
Weld Group ID Type*	Dwg Loc	LA Mat'l	LA size	LA min wall	SS Mat'l	SS size	SS min wall	Δ-wall	%Δ wall	per RH	Quantity	Loc	Remark
DW-2	E-6/7	T-22	2.5	0.220	304H	2.5	0.220	0.000	0.0	2	294	Furn	
DW-2	E-6	T-22	2.5	0.203	304H	2.5	0.203	0.000	0.0	2	294	Furn	
DW-1	E-5	T-22	2.5	0.220	304H	2.5	0.220	0.000	0.0	1	147	Furn	
DW-1	E/F-4	T-22	2.5	0.220	304H	2.5	0.220	0.000	0.0	1	147	Furn	
DW-1	E/F-4	T-22	2.5	0.203	304H	2.5	0.220	0.017	8.4	1	147	Furn	
DW-1	F-4	T-11	2.5	0.148	304H	2.5	0.148	0.000	0.0	1	147	Furn	
DW-1	F-4	T-11	2.5	0.203	304H	2.5	0.148	0.055	27.1	1	147	Furn	
Total										9	1323		
V Vertical Reheater				Figure 30				In service since Dec. 1972				No DMWs	
VOt Vertical Outlet -Reheater				Figure 30				In service since Dec. 1972					
Weld Group ID Type*	Dwg Loc	LA Mat'l	LA size	LA min wall	SS Mat'l	SS size	SS min wall	Δ-wall	%Δ wall	per RH	Quantity	Loc	Remark
DW-3	B-4	T-22	2.375	0.238	304H	2	0.2	0.038	16.0	3	441	Pent	
DW-2	B-4	T-22	2.375	0.203	304H	2	0.2	0.003	1.5	2	294	Pent	
Total										5	735		

Component Data

H Horizontal Reheater			Figure 30			In service since Jul. 1980 (Replaced)			hours =	126524				
Dwg Loc	Temp F	Temp R	Time (hr)	Log Time	P	Press (psi)	Dia (in)	Wall (in)	Inside Rad	Ro2	Ri2	Axial Press. Stress (ksi)		
E-6/7	950	1409.69	126524	5.10	35386.3	700	2.50	0.220	1.03	1.563	1.061	1.481	0.200	0.000
E-6	950	1409.69	126524	5.10	35386.3	700	2.50	0.203	1.047	1.563	1.096	1.646	0.200	0.000
E-5	950	1409.69	126524	5.10	35386.3	700	2.50	0.220	1.03	1.563	1.061	1.481	0.200	0.000
E/F-4	950	1409.69	126524	5.10	35386.3	700	2.50	0.220	1.03	1.563	1.061	1.481	0.200	0.000
E/F-4	950	1409.69	126524	5.10	35386.3	700	2.50	0.203	1.047	1.563	1.096	1.646	0.200	0.000
F-4	950	1409.69	126524	5.10	35386.3	700	2.50	0.148	1.102	1.563	1.214	2.442	0.200	0.000
F-4	950	1409.69	126524	5.10	35386.3	700	2.50	0.203	1.047	1.563	1.096	1.646	0.200	0.200
VOt Vertical Outlet - Reheater			Figure 30			In service since Dec. 1972			hours =	170021				
Dwg Loc	Temp F	Temp R	Time (hr)	Log Time	P	Press (psi)	Dia (in)	Wall (in)	Inside Rad	Ro2	Ri2	Axial Press. Stress (ksi)	stress multiplier for bending	stress mult. for Δ-Wall
B-4	1000	1459.69	170021	5.23	36828.7	700	2.38	0.238	0.9495	1.410	0.902	1.241		0.000
B-4	1000	1459.69	170021	5.23	36828.7	700	2.38	0.203	0.9845	1.410	0.969	1.539		0.000

Component Data

H Horizontal Reheater										
Larson-Miller/1000	Stress (ksi)	Number of Tubes	Probability from L-M	Stress cont. to Prob.	Prob Singl Tube	Reliability	Conseqence 1	Risk 1	Conseqence 2	Risk 2
35.4	1.78	294	0.0001	0	0.0001	0.971027				
35.4	1.97	294	0.0001	0	0.0001	0.971027				
35.4	1.78	147	0.0001	0	0.0001	0.985407				
35.4	1.78	147	0.0001	0	0.0001	0.985407				
35.4	1.97	147	0.0001	0	0.0001	0.985407				
35.4	2.93	147	0.0001	0	0.0001	0.985407				
35.4	2.30	147	0.0001	0	0.0001	0.985407				
Total		1323				0.876072	2420000	299904.9	80000	9914.21
VOt Vertical Outlet - Reheater										
Larson-Miller/1000	Stress (ksi)	Number of Tubes	Probability from L-M	Stress cont. to Prob.	Prob Singl Tube	Reliability	Conseqence 1	Risk 1	Conseqence 2	Risk 2
36.8	1.24	441	0.002	0	0.002	0.413589				
36.8	1.54	294	0.002	0	0.002	0.55511				
Total		735				0.229587	2420000	1864399	80000	61633.02

Appendix D

Excerpts from API 579

This appendix contains background information and excerpts related to API 597 used to support examples in the study.

The example in 4.2.1.1 used Curve B from API 597 Figure 3.3, as represented in this study by Figure D1.

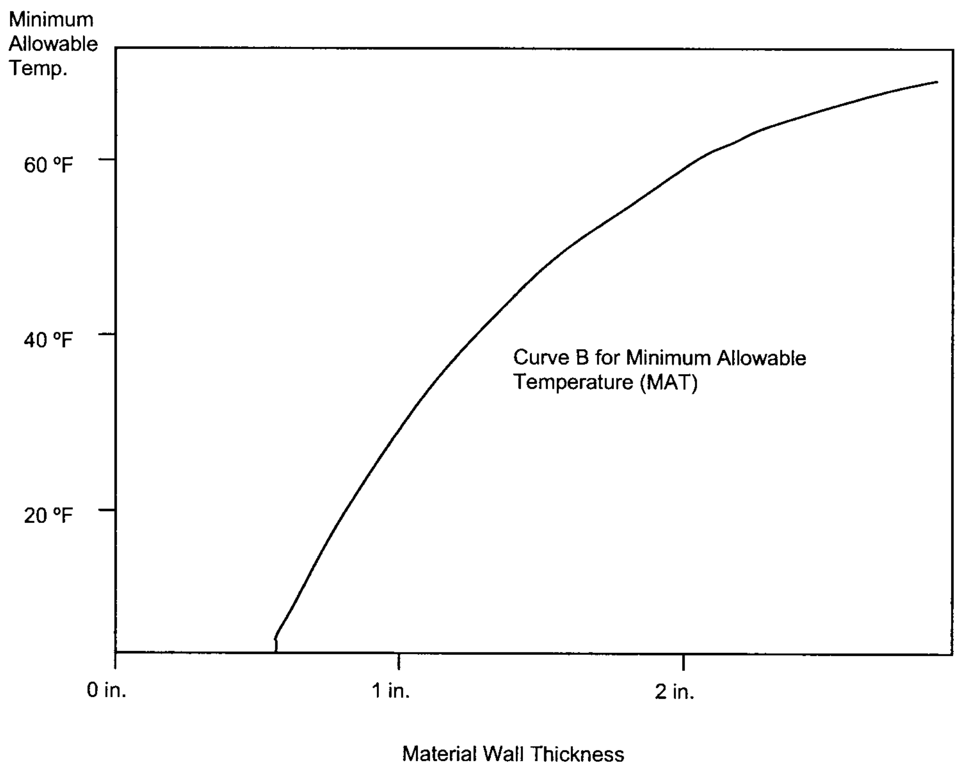


Figure D-1. Minimum Allowable Temperature vs. Wall Thickness from API 597, Figure 3.3

Table 3.3, also referenced in the example is not shown because it is simply a cross reference that identifies curve B from Figure 3.3 for many carbon steels, including SA-333 Grade 6.

The following equations and figures from API 579 were used in Section 4.3.1.3.

$$t_{min} = \max\left(t_{min}^{C}, t_{min}^{L}\right)$$

Equation A.190 (terms defined in Section 4.3.1.3)

$$t_{min}^{C} = \frac{PD_o}{2\left(\frac{SE}{L_f} + PY\right)} + MA$$

Equation A.193

where S = the allowable stress, E = joint efficiency, P = the pressure, and Y = coefficient for temperatures, Do = outside diameter. The term MA is a mechanical allowance for threaded components and is zero for the welded piping example in Section 4.3.1.3.

$$t_{min}^{L} = \frac{PD_o}{4(SE + PY)} + t_{sl} + MA$$

Equation A.187

where the terms are the same as in Equation A.193 except t_{sl} is the supplemental thickness for supplemental loads.

$$L_f = \left(\frac{\frac{R_b}{R_m} + 0.5}{\frac{R_b}{R_m} + 1.0}\right)$$

Equation A.198 (terms defined in Section 4.3.1.3)

$$R_t = \frac{t_{mm} - FCA}{t_{min}}$$

Equation 5.3 (terms defined in Section 4.3.1.3)

$$\lambda = \frac{1.285s}{\sqrt{Dt_{min}}}$$

Equation 5.4 (terms defined in Section 4.3.1.3)

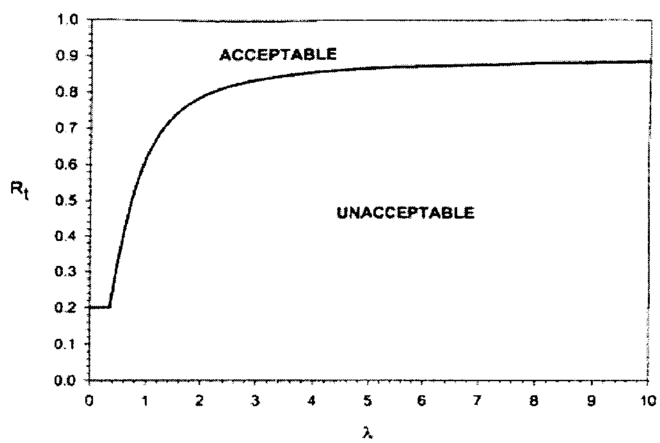
$R_t \geq 0.20$ Equation 5.5

$t_{mm} - FCA \geq 2.5\ mm\ (0.10\ inches)$ Equation 5.6

$L_{msd} \geq 1.8\sqrt{Dt_{min}}$ Equation 5.7

Equations 5.5, 5.6, and 5.7 are flaw size criteria; terms are defined in Section 4.3.1.3.

API 579 Figure 5.6, shown below is the longitudinal screening criteria for localized metal loss in a shell for a level 1 evaluation. The criteria of Figure 5.6 is based on a remaining strength factor of 0.9.

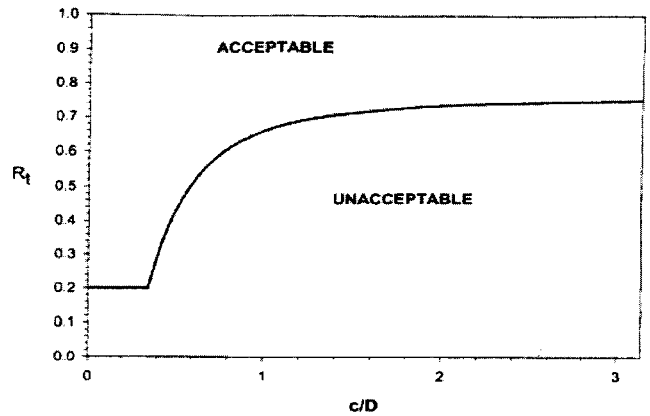


Notes:

1. Nomenclature.

s	=	Meridional (axial) dimension of the region of local metal loss (mm:in)
D	=	Inside diameter (see paragraph 5.4.2.1.a), (mm:in),
FCA	=	Future corrosion allowance (mm:in),
R_t	=	$\frac{t_{mm} - FCA}{t_{min}}$
t_{min}	=	Required thickness per applicable code (mm:in),
t_{mm}	=	Minimum measured thickness (mm:in), and
λ	=	$\frac{1.285s}{\sqrt{Dt_{min}}}$

API 579 Figure 5.7, shown below is the circumferential screening criteria for localized metal loss in a shell for a level 1 evaluation.



The screening criteria of Figure 5.7 is described in Equations 5.64 and 5.65.

$$R_t = 0.2 \qquad for\ c/D \le 0.348$$

Equation 5.64 (terms defined in Section 4.3.1.3)

$$R_t = \frac{-0.73589 + 10.511(c/D)^2}{1.0 + 13.838(c/D)^2} \qquad for\ c/D > 0.348$$

Equation 5.65 (terms defined in Section 4.3.1.3)

Appendix E

Definitions, Abbreviations, and Symbols

The following provides definitions for some abbreviations and specific terms as they are used for the purposes of this study.

ADT: Administrative Delay Time

API: American Petroleum Institute

ANSI: American National Standards Institute

ASME: American Society of Mechanical Engineers

BWR: Boiling Water Reactor

Consequence: The impact of a failure of a system or component, e.g., death, injury, environmental damage, or economic loss.

CPM: Critical Path Method of scheduling

Cutset: A specific combination of failures associated with specific components and specific causes, leading to an ultimate or resulting system failure.

Deterministic: A term that applies to analysis based on calculated values and other factors which are used as bounding criteria from which decisions are made. The term also applies to requirements resulting from deterministic analysis.

DMW: Dissimilar Metal Weld

EPRI: Electric Power Research Institute

FEMA: Federal Emergency Management Agency

Failure Modes, Effects, and Criticality Analysis (FMECA): An analysis method utilizing diagramming techniques to identify all of the possible failure modes. Synonymous with Failure Modes and Effects Analysis (FMEA) for purposes of this study

Fault Tree: A diagram, incorporating logical gates, which is used to identify combinations of events that lead to a top, undesired resulting event.

FV: The Fussell Vesely risk importance measure

Hazard and Opportunity (HAZOP): A type of risk evaluation study

IGSCC: Intergranular Stress Corrosion Cracking

INPO: Institute of Nuclear Power Operations

LDT: Logistics Delay Time

Minimum Cutset: A cutset having the minimum number of failures required to produce the undesired consequence.

MDT: Maintenance Down Time

Mpt: Mean Preventive maintenanceTime

MTBF: Mean Time Between Failures

MTBM: Mean Time Between Maintenance

MTTF: Mean Time To Failure

MTTR: Mean Time To Repair

Probabilistic Risk Assessment (PRA): An analysis that identifies combinations of events that can lead to an ultimate or resulting failure, estimates the frequency with which such events occur, and the consequences of such events in terms of the ultimate or resulting failure.

PERT: Program Evaluation and Review Technique for scheduling

PRAISE: Piping Reliability Analysis Including Seismic Events, a computer code for structural reliability and risk assessment of piping systems.

PWR: Pressurized Water Reactor

Some standard abbreviations, such as those used for units of measurement, stress, etc., have not been included. In the main body of the study, abbreviations are defined as a parenthetical reference the first time they are used.

SLC: The Standby Liquid Control system for boron injection in a BWR.

The following provides definitions for some symbols as they are used for the purposes of this study.

α: Alpha, represents the area under the tail of the curve for the normal and t distributions. For the Wiebull distribution, α is a distribution parameter.

λ: Lambda, a variable used in the exponential function and used to represent the failure rate when the probability of failure follows an exponential distribution.

μ: Mu, used to represent the mean or arithmetic average of a population.

σ: Sigma, used to represent the standard deviation (or variance when squared); also used to represent stress.

Q: The probability of failure of a component or system; also used to represent the flaw shape parameter in fracture mechanics calculations.

R: The reliability of a component or system.

t: The standardized random variable for the t distribution.

z: The standardized random variable for the normal distribution.