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Final Report

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REPORT SUMMARY

One current application of probabilistic safety assessment (PSA) in the nuclear industry is risk informed in-service inspection (RI-ISI) of piping systems. EPRI has developed an approach to RI-ISI that has been implemented in full plant pilot studies and in applications of American Society of Mechanical Engineers (ASME) Code Case N-560. This approach uses insights from service experience to evaluate pipe failure potential based on a systematic search for damage mechanisms. This is a non proprietary version of TR-111880.

Background

In a previous project, an independent review was performed on the EPRI RI-ISI method. That review called for strengthening the technical basis for assessing pipe failure potential in RI-ISI applications. It also called for confirmation that implementing the EPRI method would result in an acceptable change in risk. In response to this review, EPRI incorporated a project into its research program to enhance use of service experience for RI-ISI. This report presents current results of that research.

Objectives

- To support RI-ISI applications by enhancing capabilities of piping reliability assessments.
- To develop piping reliability models and supporting data that will be accountable to service experience.
- To create models and data that help confirm that changes to piping inspection programs can be made that enhance inspection effectiveness.
- To develop models and data that confirm changes to inspection programs have an acceptable impact on risk as measured by core damage frequency and large early release frequency.
- To provide models and databases that are easily useable by piping system and in-service inspection program engineers at nuclear power plants.
- To take into account uncertainties inherent in predictions of passive component reliability.
- To demonstrate the capabilities of the methods and databases that support updates to PSAs for sequences involving piping system failures.

Approach

The approach followed in developing piping system reliability models employs Markov models that permit time-dependent issues associated with aging processes to be addressed. In this application, the project team used Markov models to establish several discreet states of a piping system involving various levels of piping system degradation. Transitions associated with piping

failure, inspection, and repair processes were modeled explicitly. Relationships were established between time-dependent pipe rupture frequencies and observable parameters that describe the failure, inspection, and repair processes. To support application of the models, the team developed a piping reliability database grounded in the cumulative operating experience of light water reactor (LWR) piping systems. A companion report, EPRI TR-110161, covers data analysis methods, parameter estimation techniques, and an overview of the service experience data used. This current report provides a set of piping system failure rates and rupture frequencies generated from the methods described in EPRI TR-110161. The database permits the models to be applied to all four LWR reactor vendors, all existing piping systems, and all observed pipe failure mechanisms. Based on information collected for the EPRI ANO-2 pilot study, practical application of the resulting models and initial database was demonstrated on the Reactor Coolant System at the ANO-2 plant and is documented in TR-110161.

Results

This report's results demonstrate that careful analysis of service data with due regard to uncertainties can support order of magnitude estimates of pipe rupture frequencies and failure rates due to a wide set of failure mechanisms. In EPRI TR-110161, the first version of these data and models was used to evaluate potential risk impacts due to implementation of a risk informed inspection program.

EPRI Perspective

EPRI has performed ongoing research in piping system reliability. In addition, EPRI has long supported industry efforts to apply PSA technology in a variety of risk informed applications that help improve resource allocation and address safety and regulatory matters. This report provides important technical bases for EPRI's approach to risk informed in-service inspection of piping systems. The report's data also should prove useful in bench-marking piping reliability assessments based on alternate approaches such as probabilistic fracture mechanics.

Keywords

RI-ISI
Markov
In-service inspection
Failure rates

ABSTRACT

One of the current applications of probabilistic safety assessment in the nuclear industry is RI-ISI. EPRI has developed an approach to RI-ISI that has been implemented in full plant pilot studies and in applications of ASME Code Case N-560. EPRI report TR-110161 provides important technical bases for this approach by establishing models and databases for piping system reliability assessment that utilize service experience from the first 2,000 reactor years of light water reactor (LWR) operating experience. The approach that was followed was to employ Markov reliability models that permit the role of inspections and the time dependent issues associated with aging processes to be addressed. Relationships are established between the time dependent pipe rupture frequencies and observable parameters that describe the failure, inspection and repair processes. A piping reliability database based on the cumulative operating experience of LWR piping systems was developed to support application of the models. Failure rates and rupture frequencies derived from this database are presented in this report. This database permits the application to all four LWR reactor vendors, all existing piping systems and all the observed pipe failure mechanisms. Practical application of the initial models and databases was demonstrated in a companion report EPRI TR-110161. This is a non proprietary version of TR-111880.

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1
INTRODUCTION

Background

The purpose of this report is to document the development and application of new reliability models and supporting databases to estimate the frequencies of piping failures in nuclear power plants. These models were developed as part of an effort to implement RI-ISI strategies for piping systems.

For the past several years, EPRI has been developing a risk informed approach to optimize programs in which piping systems are inspected in accordance with Section XI of the ASME code. The goal of the RI-ISI program is to optimize the resources for inspection of piping systems in a manner that will maintain the risks of pipe ruptures at acceptably low levels while minimizing inspection costs and personnel radiation exposures associated with the performance of these inspections. As with other risk informed applications, the intent is to reallocate resources away from areas with marginal safety impact to areas that have a greater potential for effecting risk.

The existing requirements for inspections of safety related piping systems are derived from the ASME code Section XI and augmented inspection programs. These programs have been added to address failure mechanisms that were not envisioned in Section XI but yet have been experienced. Examples include flow accelerated corrosion (FAC) and intergranular chloride stress corrosion cracking (IGSCC). The technical approach to RI-ISI adopted by EPRI is to take advantage of insights from service experience from over 2,000 reactor years of commercial LWR piping systems [1, 2]. This experience includes documented evidence of more than 1,500 piping system failures including about 100 that were severe enough to be classified as ruptures with leak flows in excess of 50 gpm, with the remaining failures involving smaller leak rates. This study found that essentially all the leaks and ruptures that have occurred in this experience are the result of a well defined set of failure mechanisms, whose causative factors are known. These failure mechanisms include various combinations of degradation mechanisms such as FAC and IGSCC, as well as mechanisms induced by normal and transient loading conditions such as vibration fatigue, water hammer and over-pressurization transients.

There are important technical issues in RI-ISI related to the question of how to estimate the frequency and risk impacts of pipe failures, especially pipe ruptures. It is unclear how current inspection programs and leak detection programs have influenced the observed frequency of pipe ruptures, or whether such pipe failures have occurred in spite of such inspection programs. There is a growing concern that current inspection programs are geared to locations and failure mechanisms that generally do not occur or only occur rarely, while other failures occur in

locations by failure mechanisms not accounted for in the current inspection programs. A related concern is derived from the insight that for a non-destructive examination to assure a high degree of reliability, the inspection must be geared to look for specific degradation mechanisms. A goal of this program is to feedback insights from the current inspection program to effect an inspection for cause approach. Such an approach should be more effective in reducing failure and rupture frequencies in those locations that are selected in a risk informed process for inspections. At the same time, costs and personnel exposures can be reduced as locations with minimal safety importance are eliminated from the inspection program.

The application of probabilistic safety assessment (PSA) technology in this approach has two important roles. The first is to assess the risk impacts of piping failure modes on a segment by segment basis to understand the relationship between where and how inspections are currently carried out and the risk of pipe failures in these segments. The second role is to better understand the risk impacts of the proposed changes to the inspection program. It is necessary to assess these changes to assure that any risk increases that may result from reduced inspections in some pipe segments are offset by appropriate risk reductions from increased and enhanced inspections in more risk sensitive segments. It is this second role of PSA that is supported by the work presented in this report, and a companion report EPRI TR-110161 [3].

Objectives

The primary objectives of this report and its companion report EPRI TR-110161 [3] are to:

- Develop methods that provide realistic estimates of pipe failure frequencies for all systems within the scope of RI-ISI programs for piping systems.
- Develop a piping reliability database that can be used with these methods to ensure that piping reliability estimates obtained from models will be well anchored and benchmarked against the results of service experience.
- Account for service experience with pipe failures including flaws, leaks and ruptures from the first 2,000 reactor years of U.S. commercial LWR experience through 1995.
- Explicitly model the impacts of inspections for flaws and leaks in piping systems so that changes in pipe failure frequencies due to changes in inspection and leak detection strategies can be determined.
- Support the capability for plant specific estimation of pipe failure rates when adequate data are available.
- Account for uncertainties in the pipe rupture frequency estimates.
- Ensure that all assumptions are clearly documented to effect independent review.
- Account for the "leak before break" characteristic of applicable failure mechanisms when appropriate.
- Provide a set of tools that can be easily applied by utility engineers in current and future RI-ISI projects and any application to estimate rupture frequencies; and, provide guidance to optimize inspection programs.

While a great deal of work has been done by other authors to contribute to our current capabilities to estimate pipe failure frequencies, none of the existing available methods has really met all the above objectives.

Technical Approach

There are several different approaches that have been applied to estimate pipe failure frequencies. The most straightforward approach is to simply obtain statistical estimates of pipe element failure rates. This is the most common approach to this problem [4, 5, 6]. The primary limitation of a statistical approach is that past historical data reflects some indeterminate impact of previous inspection programs. In order to propose changes to these programs, such changes may render the previous failure rate estimates invalid.

A second approach is to make use of probabilistic fracture mechanics models to predict crack initiation and growth from existing flaws. To date, pipe rupture frequencies estimated with this approach have not been fully reconciled against service experience. Moreover, the technology to perform the necessary computations and the requisite input data are generally not available to utility piping engineers. Conversely, new models based on service experience will be useful as a benchmark to help validate current efforts to refine fracture mechanics based approaches.

During a recent project [7] to review the EPRI RI-ISI methodology [8], an idea emerged to utilize an established reliability modeling technique, known as Markov modeling, for a piping reliability assessment. The objective of this approach is to explicitly model the interactions between failure mechanisms that produce failures, and the inspection, detection and repair strategies that can reduce the probability that failures occur, or the probability that failures will progress to ruptures before being repaired. This Markov modeling technique starts with a representation of a "system" in a set of discrete and mutually exclusive states. At any instant in time, the system is permitted to change state in accordance with whatever competing processes are appropriate for that plant state. In this application of the Markov model, the states refer to various degrees of piping system degradation, i.e., the existence of flaws, leaks or ruptures. The processes that can create a state change are the failure mechanisms operating on the pipe and the processes of inspecting or detecting flaws and leaks, and repair of damage prior to the progression of the failure mechanism to rupture.

This idea was evaluated and determined to be capable of meeting all the objectives outlined in the previous section. Importantly, the task of estimating the model parameters was found to be straightforward and feasible. With this information, it was decided to launch a project to demonstrate the capabilities of this approach and to develop a piping reliability database that can be used to support its application. In the database development effort, the project took advantage of the multidisciplinary team that EPRI formed to work on the RI-ISI project. This team was comprised of experts in piping degradation mechanisms, NDE inspection techniques, probabilistic safety assessment and reliability model development, and reliability database development. This team examined a number of issues confronted in previous piping reliability studies such as the need for an integrated approach for treating degradation mechanisms that act on welds, those that act on piping base metal, and those that act on entire systems and subsystems in imposing severe pipe loading conditions. Additional issues that were addressed

include consideration of leak before break characteristics of certain pipe failure mechanisms, and how to specialize a piping analysis to make it plant and system specific.

A final issue addressed in this project is the use of the reliability assessment tools, the supporting piping reliability database, and the results of an application of RI-ISI to perform a risk evaluation of the impacts of changes to the inspection program. To prove the usefulness of the results of this project, an evaluation was performed of the Reactor Coolant System (RCS) at ANO-2. This evaluation provided useful insights in refining the EPRI approach to RI-ISI.

The results of this project are presented in the rest of this report and in a companion report EPRI TR-110161. The purpose of this document is to provide an assessment of the frequency of the major piping and weld damage mechanisms at U.S. nuclear power plants. The assessment of these mechanisms was utilized in EPRI TR-110161 [3] which was published in December 1998. The final report incorporated comments and ideas that stemmed from industry discussion and use of the draft version of that document. The intent of this report is to 1) provide additional information not published in the companion report, EPRI TR-110161, 2) incorporate industry suggestions to better characterize and analyze the failure mechanisms, and 3) perform enhancements to the database.

This document is organized into two sections. Section 1 reviews the changes that were made to the Swedish Nuclear Power Inspectorate (SKI) database described in SKI Report 96:20 [1] that eventually resulted in what is now called the EPRI '97 database. Section 2 presents in detail the methodology used to calculate the pipe failure and rupture rates for various failure mechanisms, including an example. Appendix A presents the complete set of results for all the mechanisms, systems, and reactor vendors. Finally, Appendix B describes the software that was used to perform the Bayesian updating.

Application of the initial failure rates and rupture frequency estimates that are updated and presented in this report is provided in EPRI TR-110161 [3]. In this application, the risk impacts of implementing the EPRI RI-ISI program in one of the pilot plant projects are examined.

An independent technical review of this work was performed by the University of Maryland. This review covered the methods that were developed for piping system reliability and for estimating parameters of these models that describe pipe failure mechanisms, inspections and leak detection processes, and repair of pipes prior to occurrence of pipe rupture. The capability of the existing data to support estimation of the parameters of these new reliability models was also in the scope of the review. As a result of this review, which is documented in Appendix A of EPRI TR-110161, enhancements to the piping reliability models are proposed, the technical validity of the work was affirmed, and the reasonableness of the assumptions made were corroborated.

Evolution of the EPRI Database for Piping Systems

One of the cornerstones of the EPRI RI-ISI methodology is the use of service experience to estimate the frequency of pipe ruptures. With over 2,000 reactor years of operating experience for the U.S. commercial nuclear plants alone, there is sufficient data available to develop pipe

SKI 96:20 Database Review and Screening

A brief review of the SKI 96:20 database with which we started revealed a number of issues that needed to be addressed before the development of statistical information that could be used to suppo piping reliability evaluations. These issues included:

- Leaks and ruptures events of components other than pipes
- Events that occurred before commercial operation or after the final plant shutdown before decommissioning
- Events involving pipe failures with a working fluid other than water or steam
- Events involving small instrument lines that are not of interest in RI-ISI applications
- Events in which the cause of the failure was not clearly defined or listed as "unknown"
- Possible errors or discrepancies in one or more of the database fields

A brief discussion of what was done to address each of these issues that resulted in changes to the original database is presented below.

Non-Piping Events. A brief comment is entered for each event in the SKI 96:20 database. A review of these comments revealed that many of the events in the database involved the failure of components other than piping. Although these events may provide valuable information related to the frequency of leaks and ruptures in water and steam systems in general, they are not applicable to the assessment of pipe failures. Also, these components are not within the scope of ASME Section XI or EPRI RI-ISI programs. Therefore, these "non-piping" events were screened from this analysis. Some examples of these non-piping components are: tubing, hoses, gaskets, nipples, valves and heat exchangers, among others. Care was taken not to delete events involving the failure of a weld connection between a pipe segment and some other component such as a valve or pump. A total of 245 events were screened out of the failure rate analysis based on the "non-piping" criteria. To preserve the completeness of the database, no records were deleted, only modified so that filters could be applied for the purpose of statistical failure rate development.

Screening Based on Year of Plant Operation. The SKI 96:20 database included events that occurred as much as seven years prior to initial criticality, and events that occurred several years after a plant was permanently shutdown. To ensure consistency in the data, only operating experience within one year prior to initial criticality, and before the permanent shutdown of plant, was used. There were 14 failure events in the SKI 96:20 database that occurred prior to one year before initial criticality, and 7 events that occurred after the plant had been shutdown. These events were screened from the database.

Include Water and Steam Piping Only. The EPRI RI-ISI pipe failure analysis is for water and steam system piping only. Several failures in the SKI 96:20 database involve instrument air piping or fuel lines. There were 13 failure events of this type in the SKI 96:20 database screened from the analysis.

RCS Leaks with Unknown Failure Mechanisms. The SKI 96:20 database included a failure mechanism category for pipe failures caused by unknown ("UNK") causes. In the process of performing the ANO 2 pilot project [9], analysts were able to identify the failure mechanism for seven reactor coolant system leaks at Combustion Engineering plants (previously considered unknown) by communication with plant personnel. Four of these events were classified as primary water stress corrosion cracking (SC) and three as vibrational fatigue (VF). The exercise of researching the subset of failure events classified as failure mechanism "unknown" validates that there are very few if any pipe failures where the failure mechanism is unknown. In most cases, the failure mechanism is not specifically reported, and the documentation of the event is not detailed enough to identify the failure mechanism. The failure mechanism "UNK" in the SKI 96:20 database is replaced by unreported cause ("UNR") in the EPRI '97 database. These failure events are not screened from the analysis.

Review of Rupture Events. After applying the screening criteria described above, there remained 87 rupture events in the database. A preliminary reference search secured copies of the SKI cited reference for 72 of these ruptures. Unfortunately 10 of these references either did not correspond to the listed event or did not contain enough information to properly classify the event.

The review of the references did lead to some enhancements to the database. A number of changes and corrections to the database were made. These changes included: reclassifying several rupture (>50 gpm) events as leak (<50 gpm) events (4), screening some events as non-piping events (13), and correcting the date of occurrence or system classification for a number of events. One rupture event was listed twice in the database; therefore, the duplicate was deleted. An additional comment field was added to document any changes made to the original SKI data.

SKI 96:20 Database Enhancements

A number of additional changes have been made to the events database. One change is the result of a concern that the D&C events, on closer inspection, might actually be other failure mechanisms with a misidentified root cause. Consequently, as many of the primary sources of the events were located and reviewed. Of the original 76 failures attributed to D&C, 56 had a reference listed. Of the 56 with a known reference, 49 were obtained and reviewed. As a result of this effort, only 12 events were reclassified. Two events were re-categorized as corrosion, one was re-categorized as dynamic-design loading (DDL), and one DDL event was re-categorized as unknown. The remaining eight events were non-piping component failures such as gaskets or valves and were screened form the analysis. This review confirms that the categorization of the

A review of the rupture events in the system group AUXC was also performed. All of the events were confirmed as appropriately classified except one event. This event was moved to the leak category from the rupture category. The root cause remains unreported.

Summary of Changes. The following table summarizes in a global sense the impact of the review changes on the original SKI 96:20 database.

Markov model presented in Reference [3], a material flaw must develop in the piping before a leak or rupture can occur. Therefore, detection and repair of flaws will influence the likelihood of leak and rupture events. Non-destructive examinations (NDE) of piping are performed on a per weld or foot of pipe basis; therefore, it is appropriate to assess the influence of the inspection program on the pipe rupture frequency at the same elemental level. Another feature exhibited by these mechanisms is the strong "leak before break" characteristic demonstrated in the service experience data (i.e., the observed frequency of leak type failure modes is much greater than the rupture type failure mode).

The pipe rupture frequencies for the degradation mechanisms discussed above are expressed in terms of the product of a pipe failure frequency and a conditional probability of pipe rupture given failure. The calculation of the rupture rate, therefore, requires estimates of the failure (i.e., leak or rupture) frequency and the conditional probability of rupture given failure. This model, referred to as Model A in Table 2-2, has a number of benefits. The failure frequency that is calculated through the application of this model can be used to estimate the transition state from the flaw state to the leak state in the Markov model presented in Reference [3]. This model also takes advantage of the relative abundance of pipe leak events in the experience database, as compared to the number of rupture events. Rupture rates are calculated for each combination of reactor vendor, system group, and failure mechanism as discussed below. Failure events, which are well represented in the experience database (i.e., over 1,100 events), are collected for specific vendor and system group combinations to estimate the failure frequency for each failure mechanism. The conditional probability of rupture given failure is estimated based on the number of rupture and failure events grouped as discussed below. This assumes that the rupture given failure probability for a given degradation mechanism is not dependent on the vendor but the system.

Table 2-2 provides a summary of the different models used to estimate failure rates in the EPRI RI-ISI program for each failure mechanism. The failure mechanisms are grouped into two classes. The degradation mechanisms share the characteristic of requiring the occurrence of a flaw prior to the occurrence of a leak or a rupture, and they are amenable to inservice inspection programs. The design and construction defects (D&C) failure mechanism was included in the degradation mechanism class due to similarities to other degradation mechanisms. The great majority of the D&C failure events in the database were attributed to weld flaws introduced through fabrication errors. Although these fabrication defects would not be specifically targeted by an inspection program, inspections for other degradation mechanisms provide the opportunity to detect the D&C flaws, thereby making D&C failures amenable to in-service inspection (ISI).

Two additional failure mechanisms are listed in Table 2-2. Vibration fatigue (VF) failures are caused by loading conditions, and are not amenable to ISI in the sense that inspection for flaws will not impact the leak or rupture rate. But VF does exhibit a strong "leak before break characteristic" in the failure events database. Therefore, Model A is used to estimate the rupture rate for VF in units of ruptures per system year. The unreported cause failure mechanism is by definition difficult to classify and has been included under failure mechanism class "Other."

Both Model A and Model B use a Bayesian approach to estimate failure and rupture frequencies based on the prior state of knowledge and the evidence contained in the failure event database. This approach generates an uncertainty distribution for each of the rupture rates quantified. A summary of the data collection done to support the rupture rate calculations is presented in the section entitled Data Requirements and sources to support failure rate estimates.

Time Dependent Pipe Failure Rates

Before failure rates and rupture rates were calculated, the events database was reviewed for the presence of time trends in the failure rates for each of the failure mechanisms. This was done because it was postulated that the behavior of one or more of the mechanisms might be correlated to the age of the plant.

The pipe failure event database was reviewed to identify trends in event frequencies correlated to the length of time the plant had been in service. Figure 2-1 shows the distribution of all pipe failure events by year of operation. Year 1 of operation is assumed to begin with the initial criticality date of the plant. This initial view of the data clearly shows a reduced number of failures as the plant operating time increases, but this perception is skewed by the relative abundance of data for plants in the early years of operation. Figure 2-2 shows the number of plant years of experience in the database for each operating year. This chart shows the lack of experience for plants that have been operating for more than 20 to 25 years. For example, only 1.7% (35 years) of the total experience base comes from plants operating 25 years or longer. Figure 2-3 presents the pipe failure rate per year, by year of plant operation. These failure rates are calculated as the number of failure events divided by the number of years of experience. Based on qualitative arguments, such as the effects of "break-in" periods on equipment reliability, and the apparent decrease in annual failure rates with increasing plants maturity as shown in Figure 2-3, a more detailed review of the failure event database was performed. Clear trends in the failure rates would justify partitioning the data and developing failure rates dependent on two or more ranges of operational years.

- *The fraction of elements (i.e., welds and feet of pipe) susceptible to each degradation mechanism for each system.* Failure rates are calculated on a susceptible element basis. Therefore, in addition to the number of element years represented in the EPRI '97 database, the fraction of those elements susceptible to each degradation mechanism must be estimated. These susceptibility fractions were estimated as part of the EPRI RI-ISI pilot application studies performed at Arkansas Nuclear One (ANO) [9] and James A. Fitzpatrick (JAF) [12] sites. The estimates from these pilot studies, along with expert opinion were the information sources used in this analysis. Susceptibility fractions for PWR plants based on the ANO data, and those for BWR plants based on the JAF data, are presented in Appendix A, Table A-6.

For both Models A and B, a Bayesian update procedure is used to include uncertainties associated with the available data sources in the resultant failure rates. The primary sources of uncertainty in this data are:

- Possible omissions of failure events, or incorrect and incomplete information in source reports such as the Licensee Event Reports or other reports used to develop the pipe failure database.
- Lack of precise estimates of the quantities of piping materials including lengths of pipe, numbers of welds in system groups for various plant types, and the fractions of these pipe elements that are susceptible to different damage mechanisms.
- Lack of precise recorded information on the amount of time different piping systems have spent in maintenance or in different plant operating modes.
- Statistical uncertainty due to the sparse data and the use of prior experience to predict the outcome of future periods of operation. It is assumed that the prior experience is representative of the population of all past and future pipe experience.
- Uncertainties introduced by modeling assumptions such as the constancy of failure rates during various time intervals of data collection.
- Uncertainties represented in the prior state of knowledge before service data from LWR experience were collected.

Enhancements to the Methodology

Several changes from the methodology presented in Reference 1 have been made to improve the results. One of the major changes to the analysis is the exclusion of piping less than two inches in diameter. Piping of this size is excluded from the analysis for several reasons. First, the major audience for this document is expected to be those performing inservice inspections, which are limited to large bore piping. Consequently, the addition of small bore piping data to the analyses provides unnecessary additional data which may, at worst, skew the results for the majority of the users who are concerned with inservice inspections of large bore piping. Secondly, because inservice inspections are not performed on small bore piping, there is no current failure mechanism specific susceptibility data for piping of this size. Without susceptibility data, conditional failure rates cannot be calculated with confidence. Thirdly, there is evidence that small bore piping does indeed behave differently than the larger piping. The database has nearly as many events for piping less than two inches in diameter as it has for all piping sizes greater

sufficient evidence to support separate analyses for E/C and SC. D&C remains separate because there is significantly more evidence than is found for either COR, E-C, or TF. VF1 and VF2 remain separate because they are dynamic loading mechanisms that have no place being grouped with degradation mechanisms.

Grouping of the event data was also done with respect to the calculation of rupture rates for those failure mechanisms that employ Model B, where the rupture rate is calculated directly from the rupture events. These failure mechanisms include WH1, WH2, OVP, FP, DDL, HE, and UNR. For the Reactor Coolant and Safety Injection Recirculation system groups, no event grouping was done. For all other system groups, the events were divided into two groups, PWR events and BWR events.

Another change that was instituted is, again, based on the desire to avoid drawing conclusions based on the use of sparse amounts of data. CF has had no ruptures and only five failures in the entire service experience of U.S. nuclear plants. Because of this observation and the similarity CF has with TF, it was decided to merge the two mechanisms.

It was also decided to divide D&C into two groups: D&C and dynamic design loads (DDL). The events classified as D&C are caused by weld fabrication flaws and behave as other degradation mechanisms. The DDL group represents a class of construction errors that present themselves like the other dynamic loading mechanisms (e.g. FP, OVP, or WH). The database evidence suggests the same: two rupture events out of six total failures. That is, the probability of rupture given failure is significantly higher than for the degradation mechanisms.

A last change that was incorporated is the addition of a new generic prior for the conditional probability of rupture calculation. See Table 2-3. E-C, COR, SC, and TF use a prior beta distribution with a mean of 0.1. This particular beta distribution was chosen because of the sparseness of data. The use of the flat prior beta distribution (mean of 0.5) as used for E/C, D&C, and VF would yield conditional probabilities of ruptures that are too conservative and that do not match our engineering understanding of the mechanisms. For the mechanisms E/C, D&C, and VF, the flat prior allows the evidence to shape the resultant distribution and not vice versa.

Figure 2-6
Bayesian Update: Failures per Susceptible Weld Year Based on Lower Bound Estimate for Population Data

Figure 2-7
Bayesian Update: Failures per Susceptible Weld Year Based on Best Estimate for Population Data

Figure 2-8
Bayesian Update: Failures per Susceptible Weld Year Based on Upper Bound Estimate for Population

Figure 2-9
Sample Failure Frequency Distribution for Thermal Fatigue

This Monte Carlo merge technique is used in the failure frequency development for all failure mechanisms that are calculated on a susceptible weld or foot of pipe basis.

For those rupture frequencies calculated via Model A, the failure frequency must be multiplied by the conditional probability of rupture given failure to obtain the rupture frequency.

Model A: Rupture Given Failure Probability Calculation

The rupture given failure probability for a specific failure mechanism, in the case of this example thermal fatigue for a RCS of a CE plant, is also calculated using Bayesian analysis. The prior distribution used for COR, E-C, SC, and TF failure mechanisms is a non-informative Beta distribution with parameters A equal to one and B equal to nine. This is a distribution ranging from 0.0 to 1.0 with a mean value of 0.1. The evidence from the failure events database needed to perform the update can be found in Table 2-5, the summary of failure events for all vendors (add together the failures for RCS and SIR for mechanisms COR, E-C, and TF). For thermal fatigue, the database contained eight failures of which none are classified as a rupture event.

The Beta distribution is a conjugate function; therefore, the parameters A and B of the updated distribution can be calculated directly based on the evidence via the following equations:

A (posterior) = A (prior) + (# of Rupture Events)

B (posterior) = B (prior) + (# of Failure Events) - (# of Rupture Events)

Therefore, the posterior distribution, representing the conditional probability of rupture given failure for the thermal fatigue failure mechanism, is represented by a Beta distribution with parameters A and B equal to 1 and 17, respectively. The BART™ software tool can be used to show the prior distribution and the resultant rupture given failure distribution. This result is presented in Figure 2-10.

Model A: Rupture Frequency Calculation

The frequency of rupture per year for welds susceptible to thermal fatigue in CE RCS, is calculated as the product of the failure frequency distribution (Figure 2-9) and the probability of rupture given failure distribution (Figure 2-10). This product is calculated using the Crystal Ball software. The failure frequency is input as a lognormal distribution, preserving the mean value and the ratio of the 95 percentile and the 50 percentile. The rupture given failure probability is input as a Beta distribution with parameters A and B calculated as described in the previous section. The resultant distribution for the rupture frequency due to TF in CE RCS is presented in Figure 2-11.

Figure 2-10
Bayesian Update: Thermal Fatigue Rupture Given Failure Probability

Figure 2-11
Sample Rupture Frequency Distribution for Thermal Fatigue

event data identified a trend in the occurrence of water hammer failure events that warranted the separation of the events into two categories. Category WH1 represents water hammer failures that occurred in the time period from one year prior to the plant's initial criticality, to three years following initial criticality. Category WH2 represents the period of operation including the fourth year of operation following initial criticality and beyond. For this example, the rupture rate is calculated for category WH2. Table A-4 in Appendix A shows that there was one water hammer (category WH2) rupture event in an AUXC system of a GE plant. The number of system years in the EPRI '97 database associated with AUXC systems of GE plants can be determined from Tables 2-6 and 2-7. Table 2-6 indicates that the database contains 607 GE plant years of data, for years of operation four and older. To determine the total system years of data, the number of plant years must be multiplied by the number of systems or sub-systems contained in System Group AUXC. Table 2-7 lists the assumed number of systems contained in each system group. System Group AUXC is assumed to represent a single system, therefore, the total system years is equal to the years of plant operating experience (i.e., 607 years). Given the prior distribution for ruptures per system year and the experience of 1 rupture in 607 years, the update can be performed to calculate the water hammer (WH2) rupture frequency per system year for GE AUXC systems. Figure 2-13 presents the results of this Bayesian update.

Figure 2-13
Model B Sample Result: Ruptures per System Year

3
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A
PIPE FAILURE RATE DISTRIBUTIONS

Section 2 of this report documents the calculation of the piping reliability failure rates and the mean value results of the calculations are presented in this appendix. Associated with each mean value is an underlying probability distribution. This appendix also provides the parameters associated with each of the resultant probability distributions.

The evidence of leaks and ruptures for each system by reactor vendor is presented in Tables A-1 through A-4, one for each vendor. This is the information used to support the failure rate and conditional probability of rupture given failure (for Model A) calculations. It is also used for the direct rupture rate calculations used in Model B.

Table A-5 presents the best estimate weld count and piping feet count per system. These values are used in developing "success" data to support the Bayesian updates for failure rates.

The susceptibility fractions for welds and pipe feet for each of the failure mechanisms is found in Tables A-6 and A-7. Table A-6 is the set of values for pressurized water reactors, and Table A-7 is for boiling water reactors. These values are used to develop failure rates that are conditioned on the suspected presence of the failure mechanism. The conditional pipe failure rates, that make use of the susceptibility fractions, are found in Tables A-8 through A-11. These tables are specific to each vendor type: Combustion Engineering, Westinghouse, Babcock and Wilcox, and General Electric.

Similarly, Tables A-12 through A-15 are the unconditional pipe failure rates for each of the four reactor vendors: Combustion Engineering, Westinghouse, Babcock and Wilcox, and General Electric, respectively. The unconditional pipe failures are not pre-conditioned on the suspected presence of the failure mechanism. The susceptibility fractions are not used in the calculations.

The unconditional pipe rupture rates are presented in Tables A-16 through A-19. Again, the four tables correspond to each of the reactor vendors: Combustion Engineering, Westinghouse, Babcock and Wilcox, and General Electric. The unconditional pipe rupture rates are not conditioned upon the number of welds or systems modeled in the system group.

Tables A-20 through A-51 present the pipe failure rate distributions quantified to support the EPRI RI-ISI program. Each table presents the failure rate distributions associated with a specific vendor and system group combination. For example, Table A-20 presents the results for Reactor Coolant System Group (RCS) of Combustion Engineering plants. Each of these 32 tables are very similar in format, and this general format is discussed in the following paragraphs.

Tables A-20 through A-27 present the piping reliability distributions for Combustion Engineering plants, one table for each system group. Tables A-28 through A-35 are applicable to Westinghouse plants, Tables A-36 through A-43 are for Babcock and Wilcox plants, and Tables A-44 through A-51 provide the failure frequency distributions for General Electric plants.

Pipe rupture distributions are presented for each of the 13 failure mechanisms identified in Section 2. Section 2 presents two models, Model A and Model B, for calculating the pipe rupture rates dependent on the failure mechanism. Model A calculates the rupture rate as a function of the failure frequency and the rupture given failure probability. Uncertainty parameters are presented for the failure frequency and the probability of rupture given failure, as well as for the resultant rupture frequency for failure mechanisms that employ Model A.

The units of the failure rates are also dependent on the failure mechanism. Frequency units of per susceptible weld year or per susceptible pipe foot year are, as indicated, conditional on the susceptibility of the piping element to the failure mechanism. Those failure rates presented in units of per system year are not conditioned on the presence of the associated damage mechanism.

The uncertainty results (i.e., mean value, 5th, 50th and 95th percentiles) presented for the failure and rupture frequencies were calculated as described in Section 2. Log-normal distribution parameters (mean and range factor) are provided for each of the failure and rupture distributions. The log-normal function, with the parameters supplied, provide a reasonable approximation to the results of the uncertainty analysis. The mean value and the ratio of the 95th and 50th percentile are preserved by the log-normal distribution defined for each failure frequency.

The rupture given failure probabilities are calculated as the result of a Bayesian update, where the prior distribution is assumed to be a Beta distribution. The Beta distribution is a conjugate function, where the Bayesian update of a Beta prior distribution produces a Beta posterior distribution. Therefore, the Beta parameters provided in Tables A-20 through A-51 can be used to directly reproduce the uncertainty results for the corresponding rupture given failure probabilities.

In summary, the following definitions should be considered when reviewing or making use of the tables presented in this appendix:

- Leaks are defined as the passage of fluid through a flaw in a pipe or weld with a flow rate less than 50 gpm.
- A rupture is a leak with a flow rate greater than 50 gpm.
- A failure is a leak or a rupture.
- System group definitions can be found in Table 2-1.
- Failure rates and rupture rates for mechanisms VF1 and WH1 are based on data within three years of initial criticality. Hence, these values should be used by plants with less than three years of operating experience.

where

$P(x|E,E_0) \equiv$ Probability of x being the true value of an unknown quantity in light of the new evidence E and the prior body of knowledge E_0

$L(E|x,E_0) \equiv$ likelihood of observing the new evidence E, assuming that the true value of the unknown quantity is x

$P(x|E_0) \equiv$ probability of x being the true value of the unkown quantity based on the state of knowledge, E_0, available prior to receiving the evidence E

Finally, the coefficient of equation B.1 is the normalizing factor required to ensure the probability sums to 1.0:

$$K = \int_x L(E|x,E_0) * P(x|E_0)dx \qquad \textbf{Eq. B-2}$$

Bayes' Theorem is directly applicable to the process of developing application specific data. In an engineering sense, the initial state of knowledge captured by the generic prior, is the expected performance of the component as viewed from the perspective of the industry. The generic prior is an opinion (i.e., engineering judgment) of the performance of the component in question without the prejudice of the application specific information. It is important to reinforce this point: the development of the generic prior is not to be polluted by the evidence gathered to perform the update. If the evidence taints the development of the prior, then the evidence will reinforce the evidence, overemphasizing and unnaturally reinforcing the data. Bayesian updating is then a quantitative customization of the generic value to make it useful to a specific application.

Application of Bayes' Theorem in BART™

BART™ allows the user to update generic prior probability distributions with new evidence, generating an updated, application specific, posterior distribution. BART™ can generate a report that includes a graphical display of both the prior and posterior probability distributions, as well as a summary of the input data, and key parameters of the posterior distribution. The prior and posterior distributions are also available in tabular format. This section discusses how BART™ performs these functions by making use of Bayes' Theorem.

For the time based and demand based Bayes analyses, the user provides a prior distribution in the form of a lognormal distribution, as well as two values representing the evidence. Other forms of prior distributions are not supported except for beta distributions. The Beta Distribution module must be used to process prior distributions that are in the form of a beta distribution. The following steps are then performed by BART™ after the required input has been supplied.

1. Assume input (prior) distribution is lognormal for time based and demand based calculations or assume a beta distribution for the beta distribution update.

2. Determine the endpoints of the prior: values corresponding to the cumulative probability of 1.0E-6 (1-9.99999E-1) to 9.99999E-1 for a lognormal or beta distribution.

3. Discretize the distribution over this range by evaluating 100 point per decade. This step ensures that distributions with long tails get adequate representation.

4. Determine the cumulative probability between each point calculated in 3).

5. Determine the likelihood of the each point 4) in light of the evidence provided as input. The binomial likelihood function is used for demand based failure rate calculations and the Poisson likelihood function is used for random (time) based failures. The beta distribution update does not make use of a likelihood function.

6. Take the product of 4) and 5) point by point (cell by cell).

7. Normalize 6) from the sum of the results from 4).

8. Correct each value in 6) by the normalization factor from 7). This step is essential because we are not integrating over the entire curve, so the total probability from 4) does not sum to 1.0.

9. Calculate the statistics.

Software Use

BART™ consists of five Microsoft EXCEL spreadsheet files. To use BART™, the user must open the file BART.XLS using EXCEL. The screen presented in Figure B-1 appears when BART™ is first started. The opening screen has a choice of seven buttons representing the four Bayesian analysis options, as well as three batch processing options available to the user. The various options are described in the following paragraphs.

Figure B-3
Time Based Frequency Calculation Report Preview

There are several other buttons on this input sheet. The second button after the Calculate button is the Print Bayes Update Report. This button sends the calculation report to the printer. The Calculate button must be depressed before the report can be sent to the printer. Occasionally, the user may like to customize the axes of the BART™ output graph. By clicking on Format X-Axis or Format Y-Axis, either or both of the scales and titles can be changed. If selected, the user is prompted to enter a new minimum value, maximum value, and title. None of the values are required, and the user can quit at any time.

The last button is Save Results to Database. Once selected, the key attributes of the resultant distribution are sent to a file called Database.xls. The purpose of the database is to provide an easy interface with CAFTA, and as a means to store results as the users data is systematically processed by BART™. The BART™ database is depicted in Table B-1. The mean and error factor for lognormal results are entered in columns called Rate and Parm2 (EF), respectively. If the Bayes result is a Beta distribution, then the Parm (EF) column is filled in with an N/A. Variable name, units, description, and data source are extracted from the BART™ input screen. The type of distribution is captured in the Dist column, where an "M" denotes a lognormal distribution. Columns Common and Parm1 are currently not used. Lastly, column QA date and time stamps the addition of the new variable to the database.

Figure B-5
Demand Based Frequency Calculation Report Preview

Note: because BART™ uses the EXCEL binomial function, the range of input parameters that BART™ can process is limited by EXCEL.

The Demand Based Failure Rates and Frequencies input screen contains the same buttons as those described in Time Based Failure Rates and Frequencies.

Common Cause Beta Factor. This option is specialized to the update of common cause failure beta parameters used in the Multiple Greek Letter (MGL) methodology. Although the MGL methodology can employ parameters beyond the beta factor (i.e., gamma, delta, epsilon, etc.) this feature in BART™ targets the beta factor. The beta factor is also considered the most important of the MGL factors. The savvy BART™ user can tackle other MGL factors, but it is not considered a fundamental feature in BART™. Figure B-6 shows the initial screen for Common Cause Beta Factor module.

The input screen is divided into basically three parts. The top part includes options for saving results, canceling, calculating, and opening the screen for inputting data for calculations six through ten. The center part of the screen is where the input data is entered including fields for: variable name, description, units, mean and range factor (or error factor) of the prior, and failure and demand evidence. The only required data is the variable name (if the results are to be saved), mean, range factor, failures and demands. This same nomenclature (i.e., failures and demands) is used for both time based and demand based calculations. At this time, BART™ can only accommodate lognormal prior distributions described by a mean and a range factor. The lower part of the screen is where BART™ fills in the results of the calculations. BART™ provides the 5th, 50th, and 95th percentiles of the resultant distribution.

If the "Save Results" button is chosen, BART™ will save the results of the calculation to a separate EXCEL file with name provided as the variable name. A variable name is required, and a calculation must have been performed in order to save the correct results.

Method 2

The second method of performing batch Bayesian analyses is to choose the "Process File" option. In this option, the user provides all of the necessary information in a file with a specific structure. With this option, the number of calculations the user is limited to is simply the number of rows in an EXCEL spreadsheet minus two rows for column headings: 16382. Table B-1 shows the structure necessary in a user supplied input file. The column headings are not used by BART™, but BART™ expects the associated information in each of the columns. In addition, BART™ automatically ignores the first two rows and begins processing on the third row. Calculations are performed from the third row and down.

When the "Process File" option is selected, BART™ asks whether the input file is already open. If the user selects the radio button corresponding to Yes, BART™ then prompts the user to indicate how many calculations are to be performed. This window is shown in Figure B-10.

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Nuclear Power

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