ELECTRIC POWER RESEARCH INSTITUTE

Improved Basis and Requirements for Break Location Postulation

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Improved Basis and Requirements for Break Location Postulation

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Abstract

Appendix A of 10 CFR 50, General Design Criteria 4, requires that structures, systems, and components in nuclear plants be designed to accommodate dynamic effects associated with pipe rupture. Guidelines are provided in NUREG-0800 Standard Review Plans 3.6.1 and 3.6.2 and associated Branch Technical Positions (BTP) 3-3, 3-4, and MEB 3-1 (MEB 3-1, Revision 1 is provided as an attachment to NRC Generic Letter 87-11). These guidelines are generally unchanged from those developed in the early 1970s and focus on thermal fatigue as the primary damage mechanism of concern that could lead to pipe rupture. Since the 70s, a significant amount of industry experience has accumulated that identifies various mechanisms in addition to thermal fatigue as more likely to result in high energy pipe failure.

The objective of the work performed for this study was to establish a recommended approach that can be used to supersede or provide an alternative approach to existing requirements. To understand the urgency and timeframe for implementing a revised approach, a survey was conducted of existing plants, plants seeking license renewal, and applications for new plant designs to determine the impact of the continued use of the current requirements.

The report discusses insights obtained from operating plant experience. It also describes a study of the relationship among cumulative fatigue usage factors, leak probability, and risk for a limited set of components. Finally, a suggested approach to address postulated pipe rupture is outlined for consideration in the development of future regulations applicable for design of nuclear power plants.

Keywords
High energy line break
Environmental fatigue
License renewal
New plant design
General design criteria

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Section 1: Introduction

A loss of coolant accident (LOCA) is an event postulated to occur following a pipe break/rupture of the reactor coolant pressure boundary (RCPB). LOCA events are integral to the design philosophy of nuclear power plants, which is why criteria were established to define break locations.

1.1 Current Requirements

In accordance with the NRC Standard Review Plan (SRP), NUREG-0800, Sections 3.6.1 and 3.6.2, circumferential and longitudinal breaks are to be postulated in high energy lines. High energy lines are defined as lines in which the maximum normal operating temperature exceeds 200°F and the maximum normal operating pressure exceeds 275 psig.

Breaks are postulated in high energy ASME Code, Section III Class 1 lines at the following locations:

- Terminal ends
- Intermediate locations where ASME Code, Section III, Subarticle NB-3653 stress equations (10) and either (12) or (13) exceed $2.4S_m$ for Level A and Level B operating conditions
- Locations where the cumulative usage factor (CUF) exceeds 0.1 for Level A and Level B operating conditions

In accordance with 10CFR50 Appendix A General Design Criterion (GDC) 4, structures systems and components (SSCs) important to safety have to be designed to accommodate the environmental and, unless they are qualified for leak-before-break (LBB), the dynamic effects of the postulated breaks. The dynamic effects of pipe breaks are:

- Jet impingement
- Pipe whip
- Subcompartment pressurization
- Blowdown transient in the broken pipe

1.2 Challenges

The introduction of environmental fatigue effects, and the extension of the operating life of nuclear power plants beyond their initial licensed period, will result in an exceedance of the 0.1 CUF criterion at more locations than originally postulated in the design of these plants. This will, in turn, result in a larger number of postulated breaks in high energy Class 1 lines, and the addition of pipe break protection hardware (pipe whip restraints, jet shields, barriers, pipe supports and anchors).

It is generally agreed that the addition of pipe break protection hardware, when based on the current arbitrary break criterion of a CUF greater than 0.1, negatively impacts the reliability and safety of the plant by increasing congestion, and increasing the risk of interference between the pipe and the break protection structures. This concern was at the basis of application of the LBB approach.

1.3 Objective

The objective of this study is to outline a technically-based alternative to replace the current CUF criterion, which appears to be arbitrary, possibly avoiding the need to add pipe break protection hardware.

In the context of the above, the following objectives were initially established for this project:

Develop an improved basis and requirements for break location postulation, for those high energy piping locations currently defined by SRP 3.6.2 BTP MEB 3-1 (MEB 3-1 Revision 1 is an attachment to NRC Generic Letter 87-11).

1. Apply a risk-informed approach, consistent with the NRC's policy statement on probabilistic risk assessment (PRA) and Regulatory Guide 1.174.
2. Develop a methodology that can be universally applied to currently licensed US nuclear power plants and new reactor plants which have submitted a combined license application to the NRC.

These objectives were pursued by performing the following tasks described below:

1. Identification of plants adversely impacted by existing requirements and plants which have successfully obtained NRC approval for exceptions to existing requirements. This involved a review of License Amendment Requests, Design Certification Documents and internal SI documents.
2. Examine the relationship between fatigue usage, including environmental effects, and the potential for pipe failure. Based on the potential for pipe failure, examine the relative risk impact associated with a larger fatigue usage value than the current 0.1 CUF criterion.
3. Defining an approach that could be used to supersede or provide an alternative approach to existing requirements, which is expected to be acceptable to the NRC.

Additional improvements to the high energy line break criteria may be achieved in future work in the areas of (a) the definition of high energy and (b) the stress-based limit of 80% of the allowable.

1.4 Acronyms

ABWR – Advanced Boiling Water Reactor

ADAMS - Agencywide Documents Access and Management System (NRC)

APWR – Advanced Pressurized Water Reactor

ASME – American Society of Mechanical Engineers

BINP – Battelle Integrity of Nuclear Piping

BTP – Branch Technical Position

CDF – Core Damage Frequency

CFR – Code of Federal Regulations

CUF – Cumulative Usage Factor (fatigue)

DBA – Design Basis Accident

DCD – Design Certification Document(s)

ECCS – Emergency Core Cooling System(s)

EPR – Evolutionary Power Reactor

ESBWR – Economic Simplified Boiling Water Reactor

ESF – Engineered Safety Feature

FAC – Flow-Accelerated Corrosion

GDC – General Design Criteria

GE – General Electric Company (aka GE Energy, GE Hitachi)

GSI – Generic Safety Issue

HCF – High Cycle Fatigue

IGSCC – Intergranular Stress Corrosion Cracking

IPE – Individual Plant Examination

LAR – License Amendment Request

LBB – Leak-Before-Break

LERF – Large Early Release Frequency

LOCA – Loss of Coolant Accident

LRA – License Renewal Application

MEB – Mechanical Engineering Branch (NRC)

NP – Nuclear Power

NRC – Nuclear Regulatory Commission

NUREG – Reports prepared for the Nuclear Regulatory Commission

OEM – Original Equipment Manufacturer

PNNL – Pacific Northwest National Laboratory

PWSCC – Primary Water Stress Corrosion Cracking

RCPB – Reactor Coolant Pressure Boundary

RI-ISI – Risk Informed In-Service Inspection

SAR – Safety Analysis Report

SEM – Scanning Electron Microscope

SEN – Significant Event Notice

SI – Structural Integrity Associates

SRP – Standard Review Plan

SSC – Structures, Systems and Components

TGSCC – Transgranular Stress Corrosion Cracking

TLAA – Time Limited Aging Analysis

TR – Technical Report

Section 2: Background

Laboratory tests have indicated that the effects of reactor coolant environment were not adequately included in the ASME Code fatigue design curves used in the original design of reactor coolant pressure boundary components. The design analyses for license renewal as well as for new nuclear power plants require the application of environmental fatigue life correction factors (F_{en}) to the cumulative usage factors (CUF). An environmentally adjusted cumulative usage factor is then determined as CUF_{en} = (CUF) x (F_{en}). This issue has led to more regulatory requirements for license renewal, where applicants are required to assess a number of locations for the effects of environmental conditions as part of their fatigue monitoring programs.

A CUF of 0.1 is the current fatigue criterion for class 1 piping break location postulation. One area of impact, due to the requirement for fatigue analysis that includes environmental effects, is that CUF_{en} will likely exceed 0.1. Another factor to consider is the increase in CUF associated with an increase in the number of transient cycles associated with a longer period of operation.

A criterion of CUF $<$ 0.4 has been applied in some cases where the effect of environment is considered. This is not necessarily an equivalent standard, however, and with the current requirements for application of environmental fatigue, this standard is expected to be more restrictive than use of CUF $<$0.1 had been when used in combination with air curves for fatigue. This can result in an undesirable increase in the number of postulated break locations and attendant needs for designing, constructing, installing and working around whip restraints for the remaining life of the plant. At a minimum, significantly increased analytical costs, such as finite element analysis of multiple piping locations, will likely be required to meet the high energy line break postulation criterion.

The NRC has acknowledged that the CUF $<$ 0.1 break location criterion does not have a well defined or documented basis (see Generic Safety Issue A-18). Therefore, it is desirable to develop pipe break postulation requirements with a sound technical basis that would result in appropriate impact on fatigue design, pipe whip restraint requirements, and ultimately plant safety.

Both deterministic and probabilistic approaches are used when considering postulated break events. The deterministic approach establishes design limits to prevent or mitigate the consequences of these events. The special category of events called “design basis events” are those which are not expected to occur during the lifetime of the reactor, but which are postulated as the basis for the design of systems that perform safety-related functions (or engineered safety features).

In deterministic safety analysis, the selection of LOCA events is based on the use of bounding values of essential plant variables, to show by analysis that the criteria are met for the defined set of initiating events. For example, the SRP requires that a double-ended rupture of the largest pipe in the RCPB is considered in the evaluation of postulated accidents.

The approach taken by probabilistic analysis, such as that described in NRC Generic Letter 88-20, is fundamentally different. In a probabilistic analysis, all events are considered as possible events with varying probability of occurrence based on best estimate values. A probabilistic analysis will consider effects of a whole spectrum of loss of coolant accidents, and a comparable evaluation will be done for the more or less frequent events, and their consequences estimated. Thus, the probabilistic approach addresses events ranging from events with higher frequencies and lower consequences like very small LOCAs to the extremely improbable events with significantly greater consequences such as the double-ended rupture of the largest pipe in the RCPB.

Early attempts at using a probabilistic approach were limited by the availability of a robust set of failure data, computer codes and computer processing power. Consequently, engineering judgment was used and, at the time the industry guidance for pipe rupture was being developed, the reactor coolant piping design transients were considered to be a primary contributor to the potential for piping failure.

Branch Technical Position (BTP) 3-4 of the NRC Standard Review Plan (SRP), NUREG-0800, states that pipe ruptures are to be postulated *"at locations having relatively higher potential for failure, such that an adequate and practical level of protection may be achieved."* In the case of high energy line breaks, SRP 3.6.3 specifies that a fatigue usage factor greater than 0.1 be the basis for postulating "intermediate" pipe breaks for locations other than terminal ends of the piping. This value represents a significant margin to the ASME Code limit of 1.0 and provides a significantly higher margin than the 0.8 multiplier on piping stresses that is specified by BTP MEB 3-1 for postulating intermediate breaks on a stress basis. Due to the aforementioned limitations, there was no defined technical basis for the fatigue usage criterion of 0.1. It was an arbitrary judgment made at the time that the guidance was developed.

Section 3: Insights Obtained From a Review of Industry Operating Experience

There has been about 40 years of nuclear plant operating experience since the original pipe rupture guidance was developed. Available public and proprietary database and information sources on piping system failures were searched for relevant information. This included the NRC Agencywide Documents Access and Management System (ADAMS) database. The ADAMS search was performed based on an attempt to identify and quantify experience related to actual failure modes (crack, leak, failure, rupture). Once a list of records was identified, piping systems were excluded if they did not contain high energy fluids (e.g., service water system, closed loop cooling, instrument air, diesel generator, fire protection). Secondary systems (i.e. non-RCPB piping) both inside and outside containment were included to ensure all reported failures in high energy piping were considered. In addition, reactor coolant pressure boundary components other than piping (e.g., steam generators, pumps) were excluded from review, since these are not within the scope of NUREG-0800 Section 3.6.3.

The sets of keywords (and their various forms) used in the search were:

- pipe crack
- pipe fatigue
- pipe failure
- pipe rupture

In addition, a number of other documents were reviewed, which included information from ADAMS and other sources not limited to the ADAMS database. These documents are listed in the references and included:

- EPRI publications:
  - EPRI Technical Report, "Corrosion Fatigue of Water-Touched Pressure Retaining Components in Power Plants," TR-106696.
  - EPRI 1001006, "Operating Experience Regarding Thermal Fatigue of Unisolable Piping Connected to PWR Reactor Coolant Systems *(MRP-25)*.

- EPRI 1013141, "Pipe Rupture Frequencies for Internal Flooding PRAs.
- EPRI TR-1015010 (MRP-235), "Fatigue Management Handbook."

▪ U.S. National Laboratory publications:

  - NUREG/CR-6674 (PNNL-13227), "Fatigue Analysis of Components for 60-Year Plant Life."
  - NUREG/CR-6679 (BNL-NUREG-52587), "Assessment of Age-Related Degradation of Structures and Passive Components for U.S. Nuclear Power Plants."
  - NUREG/CR-6837, "The Battelle Integrity of Nuclear Piping (BINP) Program Final Report."
  - NUREG/CR-6936 (PNNL-16186), "Probabilities of Failure and Uncertainty Estimate Information for Passive Components – A Literature Review."
  - PNNL-16625, Probabilistic Fracture Mechanics Evaluation of Selected Passive Components – Technical Letter Report.

▪ Swedish Nuclear Power Inspectorate (SKI) reports:

  - SKI Report 95:59, "Reliability of Piping System Components, Volume 2: PSA LOCA Database Review of Methods for LOCA Evaluation since the WASH-1400."
  - SKI Report 95:61, "Reliability of Piping System Components, Volume 4: The Pipe Failure Event Database."

Although there is a considerable amount of information on operating experience with piping systems, the level of information provided varies widely. Records contain information based on some combination of detailed root cause failure analysis and less comprehensive "apparent cause" analysis. Another consideration is that some reports contain a subset of information from other reports.

From this information, a matrix was constructed that provides a listing of fatigue cracks, large pipe leaks, and ruptures in nuclear system piping (both RCPB and non-RCPB). The objective was to see what correlation could be established between design and actual CUF (where available), as well as any identified causal mechanism that includes information such as whether there was a non-design basis fatigue load present (and nature of the load), non-fatigue contributing factors, and known (or discovered) fabrication defects. Where known, currently postulated break locations were indicated and what criterion was used to postulate the break location (0.1 CUF, terminal end, etc).

Unfortunately, the information contained in these reports was sparse concerning calculated CUF and postulated break locations. From the data reviewed, there was no direct quantitative comparison between the estimated versus actual fatigue usage where thermal fatigue was the failure mechanism. Several reports noted that failures associated with thermal fatigue have generally been due to stresses not anticipated during plant design (thermal stratification, turbulence penetration into branch piping, thermal striping, etc.).

Based on the data limitations, this review focused on the mechanisms involved and how these related to design thermal fatigue and inferred postulated break locations where CUF was not the deciding factor (e.g., terminal ends; including branch connections).

One of the more comprehensive databases is the PIPExp database. This database is the source of failure data for calculating pipe failure rates and rupture frequencies in EPRI 1013141. Based on EPRI 1013141, the current PIPExp database evolved from previous SKI-funded research efforts and has since been supported by a continuous, active database maintenance and update effort. It concludes that the PIPExp database is more complete and has benefited from a more rigorous program of validation. Event by event comparisons of this database as well as SKI-96:20 and EPRI TR-111880 databases were performed to reconcile discrepancies as part of EPRI 1013141.

A comparison of the information in each of these databases is shown below in Table 3-1.

Table 3-1
Piping Failure Data Comparison (excerpted from EPRI 1013141)

Database Parameter	Database		
	SKI 96:20 [6]	**EPRI TR-111880 [5]**	**PIPExp [7]**
Commercial Nuclear Power Plant (NPP) Population	U.S. light water reactors (LWRs)	U.S. LWRs	NPPs worldwide incl. LWRs, heavy water reactors and Soviet designed reactors
Data Collection Period	1961 to 1995	1961 to 1995	1970 to date
Reactor Critical Years Experience covered	2,100	2,100	Ca. 9,179 (12/04)
Number of Pipe Failure Events	1,511 (part-through wall, small leaks, large leaks and rupture events) • ASME Class 1: 137 • ASME Class 2: 497 • ASME Class 3: 548 (about 10% H/X tubes/coils) • Non-Code: 329	1,145 (part-through wall, small leaks, large leaks and rupture events)	As of December 2004: 4,900 records involving part-through wall cracks, small leaks, large leaks and rupture events, plus 465 water hammer events; • ASME Class 1: 1381 • ASME Class 2: 1512 • ASME Class 3: 952 • Non-Code: 1055
Number of Pipe Rupture Events (complete, sudden loss of structural integrity)	119 • No breakdown by ASME class provided	69 • No breakdown by ASME class provided	250 records as of 12/04 • ASME Class 1: 15 (≤ DN50) • ASME Class 2: 36 • ASME Class 3: 22 • Non-Code: 177
Verification and Validation of Data Records?	Unknown	Some non-piping events deleted from SKI 96:20. Most rupture events verified but most leaks and cracks not verified	Extensive verification and validation of all database records
Component Population Data Included (ASME Class)?	No	Yes, generic estimates and actual data from 2 plants (ASME Class 1,2)	Yes, actual data from 25 plants (ASME Class 1,2,3, non-Code piping)

Another database used for this evaluation is the OPFD-2008 database, developed for the EPRI Fatigue Management Handbook (TR-1015010 / MRP-235). In addition, information from NUREG/CR-6936 was used as a complement to the other two data sources.

NUREG/CR-6936 summarizes the number of reported failures (defined as part through-wall and through-wall flaws) from 1970-2005. Figure 3-1 shows the contribution to the total number of reported failures from each of the identified mechanisms.

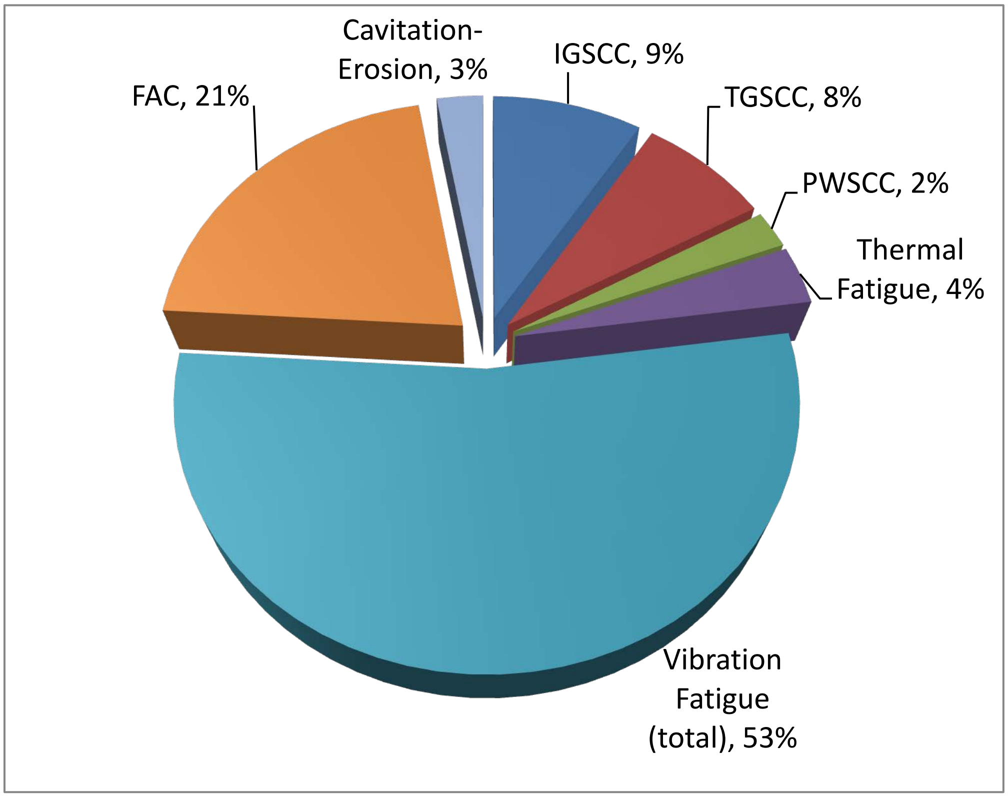

Figure 3-1
Contribution to Reported Piping Failures (obtained from NUREG/CR-6936)

The data from NUREG/CR-6936 compares relatively well to information from SKI Report 95:59 shown in Table 3-2. Although the mechanisms are characterized somewhat differently, thermal fatigue contributes approximately 4% to the total, flow accelerated corrosion approximately 20% and IGSCC roughly 13%. The difference in the fraction assigned to vibration fatigue is unknown, but is likely due to differences in the information sources.

Table 3-2
Piping Failures by Mechanism (excerpted from SKI Report 95:59)

Failure mechanisms relevant for piping in SLAP data base	Occurrence in %
Cavitation/Erosion	0.7 %
Corrosion	8.6 %
Corrosion/Fatigue	1.0 %
Erosion	3.5 %
Erosion/corrosion	19.9 %
Fatigue	2.2 %
Fretting	0.0 %
IGSCC	12.8 %
SCC	7.8 %
Vibration fatigue	35.0 %
Water hammer	4.6 %
Thermal fatigue	3.9 %

From this operating experience review, it's clear that there is a defendable basis for concluding that the potential for high energy line failures is dominated by mechanisms other than thermal fatigue due to design plant thermal transients, which typically constitutes nearly all of the calculated fatigue usage in piping systems. In addition, piping ruptures were associated with non-RCPB systems (extraction steam, feedwater heater drains, etc.). Therefore, the use of a fatigue usage value significantly below the ASME Code limit of 1.0 is judged to be overly restrictive from a design standpoint.

Section 4: Review of License Amendment Requests and Design Certification Documents

To assess the impact of the CUF criterion on current holders of a nuclear operating license, a review of License Amendment Requests associated with License Renewal and other License Amendment Requests associated with break locations was performed. In addition, a review was performed for Design Certification Documents submitted for new operating licenses in the U.S. Finally, a review of internal Structural Integrity Associates (SI) documents was performed to identify whether any work had been performed to assist in seeking relief from the current HELB fatigue criterion, since SI performs a significant amount of work associated with fatigue usage calculations and HELB evaluations.

4.1 License Amendment Request Review for Existing Plants

The License Renewal Applications (LRAs) for plants that have applied for extended operation were reviewed. A total of 77 units were examined, where 20 units had applications pending and 57 units had been granted 20 additional years of operation. The details of the information collected are presented in Appendix A. Based on this review, none of the plants had sought relief from the 0.1 CUF break exclusion criteria. It should be noted that it is likely that rather than seek such relief, the utility performed more detailed analysis in order to meet the CUF criteria, incurring additional expense.

As part of this review, discussions were held with personnel involved with license renewal activities at several plants. Specific questions were asked regarding the issue of break locations. In one case, discussion occurred regarding whether or not the HELB evaluation was a time-limited aging analysis (TLAA). Following further discussions on this question between plant personnel and the NRC staff, agreement was reached that the HELB evaluation is a TLAA, which will be managed by monitoring fatigue usage at controlling locations to see if there are any additional areas that may exceed the 0.1 CUF criterion. If monitoring identifies a projected CUF of greater than 0.1, additional evaluation will be performed prior to reaching that time to determine the most appropriate course of action.

Environmental fatigue life correction factors have not been applied to these values. The issue of environmental fatigue has been addressed by implementing the evaluations at areas identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components" and by review of other fatigue sensitive locations to ensure limiting locations for the plant have been identified.

This is consistent with work performed in support of plant license renewal. That is, HELB fatigue values are considered TLAA and fatigue monitoring is performed to identify instances where the CUF is projected to exceed 0.1. In no cases the authors are familiar with do plants also apply environmental multipliers to HELB locations, unless they also happen to be the locations identified in NUREG/CR-6260 or otherwise identified as limiting fatigue locations for the plant, when considering environmental effects on fatigue.

A search of the NRC Agencywide Documents Access and Management System (ADAMS) information system was also conducted. A search for the document type License Amendment and the phrase "break location" was completed. This search yielded 165 hits, of which there was considerable overlap with the LRAs. Documents other than LRAs were more closely examined, and none revealed any instances where plants sought relief from the current criteria.

4.2 Design Certification Document (DCD) Review for New Plants

The DCDs for the AREVA U.S. Evolutionary Power Reactor (U.S. EPR), Westinghouse AP1000, U.S.-APWR, General Electric-Hitachi ESBWR, and the General Electric ABWR were also reviewed. Only the ESBWR DCD was identified as having a CUF criterion of > 0.1. The other DCDs each used a CUF criterion of 0.1.

The ESBWR DCD states *"Criteria defining postulated pipe rupture locations and configurations inside containment are in accordance with BTP 3-4. For the piping system with reactor water, if the environmental fatigue is included in accordance with Regulatory Guide (RG) 1.207, the fatigue usage limit should be ≤ 0.40 as the criterion instead of ≤ 0.10 for determining pipe break locations."* The technical basis for the larger value was not provided in the DCD.

4.3 Review of SI Internal Documents

A review of internal calculation documents was performed based on a search of SI's document records database, which includes nearly all domestic nuclear plants as well as nuclear plants in Europe and Asia. Search terms included "HELB", "LBB", "rupture" "break", "break location" and "break exclusion". While hundreds of documents were identified in the search, no calculations were identified that involved any changes to the 0.1 CUF break exclusion criteria.

Section 5: Relationship Between Fatigue Usage, Leak Probability and Risk

Although the operating experience review information provided in Section 3 shows that the potential for high energy line failures is dominated by mechanisms other than thermal fatigue due to design plant thermal transients, design fatigue usage is evaluated here to be consistent with the objective of this study to outline a technically-based alternative to replace the current CUF criterion.

5.1 Methodology

A limited study of the relationship between cumulative usage factors and leak probabilities is performed using the pcPRAISE software. pcPRAISE is a probabilistic fracture mechanics software for evaluating pipe leak and rupture probability due to a variety of degradation mechanisms (refer to NUREG/CR-2189, vol. 5, NUREG/CR-5864 and NUREG/CR-6674). In the current study, fatigue crack initiation and growth is specifically evaluated, using random crack initiation and growth properties. Both leak (existence of a through-wall crack) and rupture (sudden and complete pipe severance) probabilities can be obtained, with leak probabilities being more straightforward to evaluate. Leak probabilities are used as a conservative surrogate for rupture probability.

For consistency with prior work used to evaluate fatigue crack initiation, the methodology developed in NUREG/CR-6674 is used in the study. This methodology is appropriate because it considers the effects of reactor water environment on fatigue usage, relates leak probability to core damage frequency and addresses a range of fatigue sensitive locations for various plant designs representing 47 components. The only changes in this study from the NUREG/CR-6674 methodology are in the environmental strain life relationships and the fatigue curves used. The purpose of this was to apply the latest available information used by the NRC for evaluating environmental effects on fatigue. A comparison of the approach used in NUREG/CR-6674 and this study is presented below in Table 5-1.

Table 5-1
Comparison of NUREG/CR-6674 Approach with Current Study

	NUREG/CR-6674	Current Study
Stresses, fatigue cycles	As reported in NUREG/CR-6674	Same as NUREG/CR-6674
Reactor water environment (strain rate, dissolved oxygen content and temperature)	As reported in NUREG/CR-6674	Same as NUREG/CR-6674
Strain-life relation	From NUREG/CR-6335	From NUREG/CR-6909
Cumulative usage factor	As reported in NUREG/CR-6674, for air and environment	Evaluated using new ASME design curve AND using 0.1% strain-life fractile curve
Leak probability	pcPRAISE as in NUREG/CR-6674, values reported therein	pcPRAISE with NUREG/CR-6909 strain-life relation
Core damage frequency (CDF) given leak	As reported in NUREG/CR-6674	Same as NUREG/CR-6674

This study used NUREG/CR-6674 stresses and fatigue cycles, along with the environment (strain rate, oxygen content and temperature) for the components evaluated. CUF values are first calculated using the specific stresses and cycles for each component location listed in Table 5-2 using the new ASME design fatigue curve. Fatigue damage was assumed to accumulate linearly with time. CUFs were then adjusted for reactor coolant environmental conditions to obtain environmentally assisted fatigue (EAF) values.

Leak probabilities in NUREG/CR-6674 were obtained using pcPRAISE with probabilistic strain life relationships from NUREG/CR-6335. In order to reflect the most current information regarding reactor water environment on fatigue life, the current study uses updated strain life relationships from NUREG/CR-6909. Using the NUREG/CR-6674 methodology, cyclic stresses, fatigue cycles and environmental conditions, pcPRAISE was again used to calculate leak probabilities associated with operating time.

Core damage frequency (CDF) was estimated using the methodology from NUREG/CR-6674 where information on the CDF given the occurrence of a pipe rupture is provided. The method used to establish this relationship is provided in Appendix B. As mentioned above, the leak probability is used as a surrogate for the rupture probability, because its evaluation is more straightforward (fewer assumptions and fewer Monte Carlo trials). The core damage frequency is then compared to values provided in the EPRI PSA Applications Guide (TR-105396) and Regulatory Guide 1.174.

Five components were selected from NUREG/CR-6674 for evaluation. All of these locations are locations evaluated in NUREG/CR-6260. These components are listed in Table 5-2, along with selected results from NUREG/CR-6674.

The following factors were considered in the selection of these components:

- Material (low alloy, LS, and austenitic stainless steel, SS)
- High and low cumulative usage factors
- High and low environmental effects (the environmental effects are based on a comparison of air and reactor water results from NUREG/CR-6674).

Table 5-2
Components Selected for Evaluation, Including Related Results from NUREG/CR-6674

#	name	NUREG/ CR-6260 Section	matl	EAF(60)	Env air	Plk(60)	comment
4	CE-new surge line elbow	5.1.3	SS	3.90	2.65	0.998	high failure prob.
24	W-new charging nozzle	5.4.4	SS	5.06	4.08	0.963	
14	CE-old charging nozzle	5.2.4	SS	0.843	2.11	6.0x10-4	low CUF, low env
39	GE-new RHR straightpipe	5.6.6	LAS	16.9	27.66	0.621	high CUF, big env
28	W-old RPV inlet	5.5.2	LAS	0.453	2.23	0.0504	low CUf, low env

EAF(60) is the CUF considering environmental effects at 60 years of plant operation. *env/air* is the ratio of fatigue usage factor with environment to value in air and $P_{lk}(60)$ is the probability of leakage at 60 years.

5.2 Results

As is discussed further below, the results of this study show that there is no direct correlation between CUF and leak probability, which was used as a conservative surrogate for pipe rupture probability. Similarly, there is no direct correlation between CUF and CDF. This is noteworthy because a key objective of this work was to establish a technically-based alternative to replace the current CUF criterion of 0.1, which lacks a technical basis.

The pcPRAISE results for leakage probability based on using NUREG/CR-6909 are provided in Figure 5-1. This plot shows the cumulative leak probabilities [$P_{lk}(t)$] as a function of time [t] for each component evaluated. The calculations include times up to 60 years.

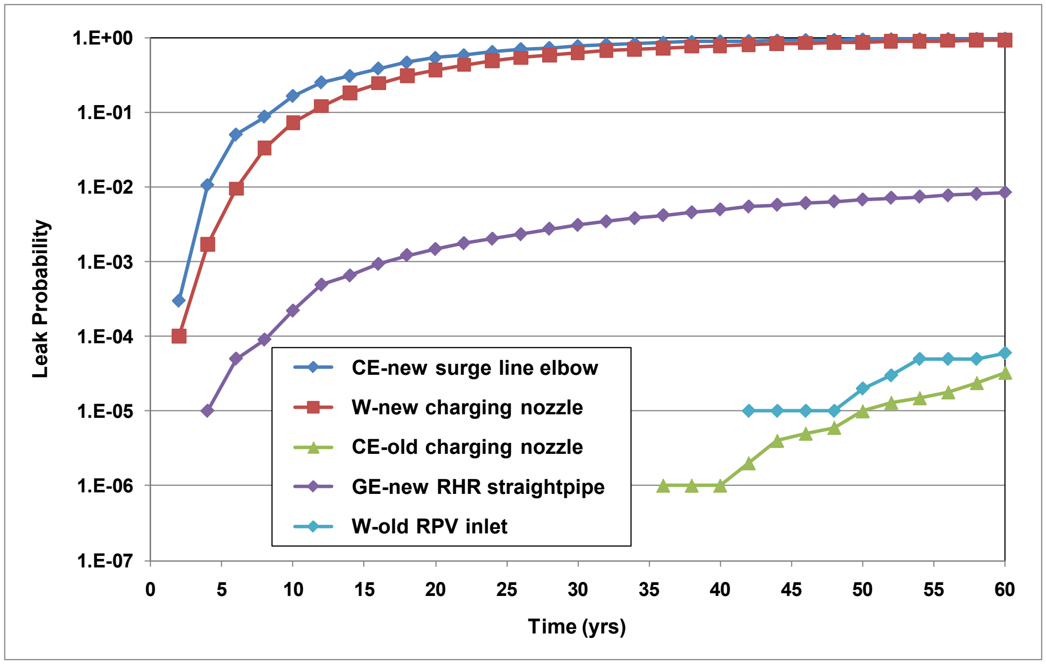

Figure 5-1
Plot of Cumulative Leak Probability as Obtained Using NUREG/CR-6909 Fatigue Correlations

NUREG/CR-6674 provides information on the probability of core damage given a pipe failure, with Table 5-3 summarizing the relevant results for the components of interest.

Table 5-3
Core Damage Probability Given Occurrence of Leakage, from NUREG/CR/6674

#	name	matl	*P(CD|leak)*
4	CE-new surge line elbow	SS	2.85×10^{-5}
24	W-new charging nozzle	SS	8.00×10^{-6}
14	CE-old charging nozzle	SS	8.00×10^{-5}
39	GE-new RHR straightpipe	LAS	9.02×10^{-9}
28	W-old RPV inlet	LAS	2.70×10^{-6}

The cumulative leak probability in Figure 5-1 is converted to leak frequency by taking the slope of the curve, $dP_{lk}(t)/dt$. Once a leak frequency is obtained as a function of time, the estimated core damage frequency (CDF) is related to the leak frequency by multiplying by the factor in the right-hand column of Table 5-3. The resulting core damage frequency can be plotted as a function of the environmentally enhanced fatigue usage factor (EAF) by assuming that cycles accumulate linearly with time. Figure 5-2 provides the results.

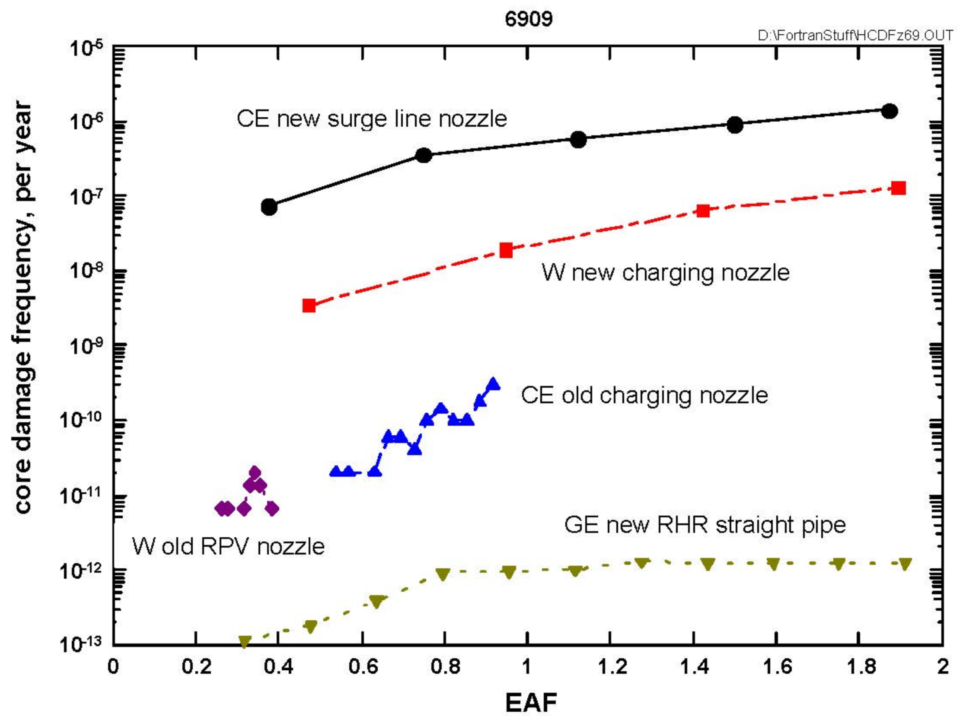

Figure 5-2
Core Damage Frequency vs. EAF as Derived From Figure 5-1 and Table 5-2

One would expect that as fatigue usage increases there would be a corresponding increase in the Core Damage Frequency. Figure 5-2 shows a lack of correlation for the various components and inconsistency between components. This compromises the ability to use specific EAF values as a criterion for risk ranking of components or for using specific values as a threshold for postulating HELB locations.

To understand why there is a lack of correlation between leakage and EAF, it is important to recognize that there are many factors that are involved in establishing the relationship between leak frequency and calculated ASME CUF/EAF values. These include:

- Stress profile (membrane, bending, radial gradient thermal)
- Geometry (use of stress indices)
- CUF methodology (strain-life correlations)
- Stress evaluations (NB-3600 vs. NB-3200)
- Crack growth considerations
- Material, temperature
- Crack growth relationships

The reason for the lack of correlation between CDF and EAF is mainly due to two factors:

- ASME design curves do not follow a line of constant crack initiation probability due to the method by which the ASME Code curve is developed and the multipliers on stress and cycles employed. The ASME design curve is not consistent with initiation probability fractiles based on statistical analysis of fatigue data as reported in NUREG reports and used in pc-PRAISE initiation and leak probability calculations. This is shown in Figure 5-3.
- The probability of CDF for a given leakage probability varies for the components themselves. Even if there were a good correlation between leak probability and EAF, agreement would not be consistent due to differences in the *P(CD|leak)* values shown in Table 5-3.

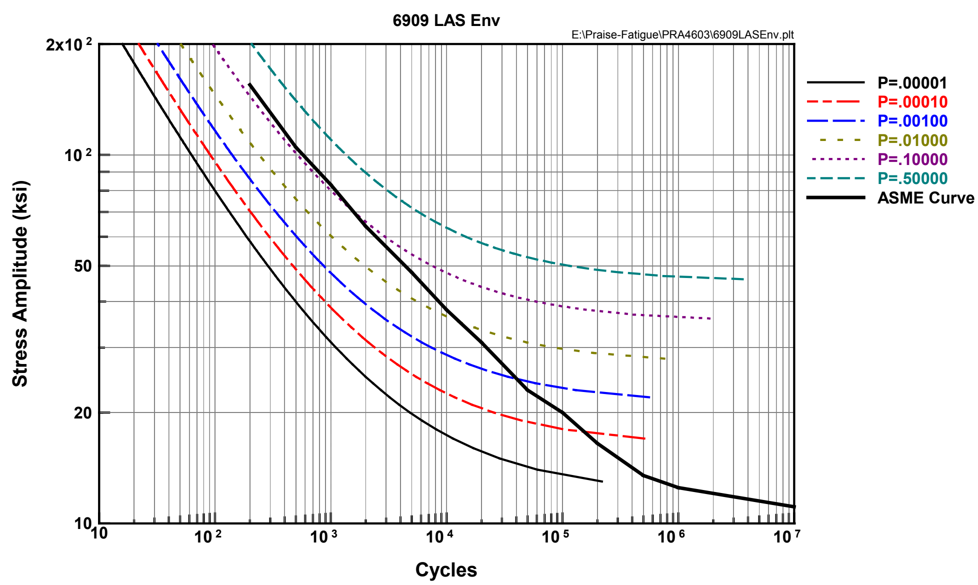

Figure 5-3
Comparison of ASME Fatigue Curve to Fatigue Probability Curve

Further information is provided in Figure 5-4 and Figure 5-5. Figure 5-4 shows initiation probabilities plotted against EAF values using the ASME design curve. There is no direct correlation due to inconsistencies between the fatigue initiation fractiles and the ASME design curve, as previously shown in Figure 5-3. Figure 5-5 is a plot of initiation probability versus EAF computed using the 0.1% fractile fatigue curve. As expected, Figure 5-5 shows much better correlation between initiation probability and EAF values. There is still some separation observed between LAS and SS materials.

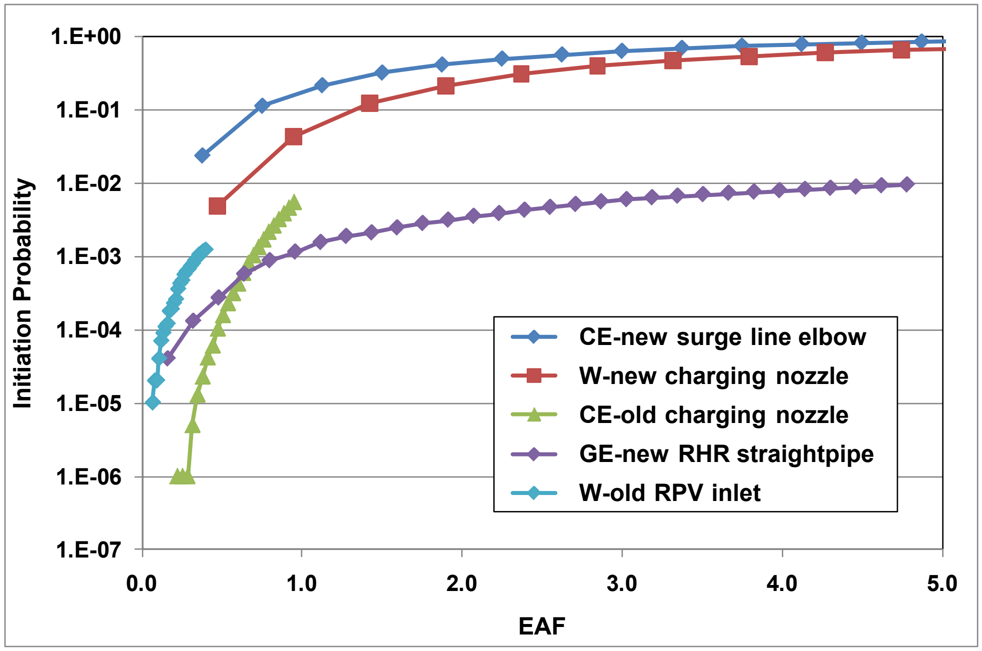

Figure 5-4
Comparison of Initiation Probability to EAF using ASME Design Curve

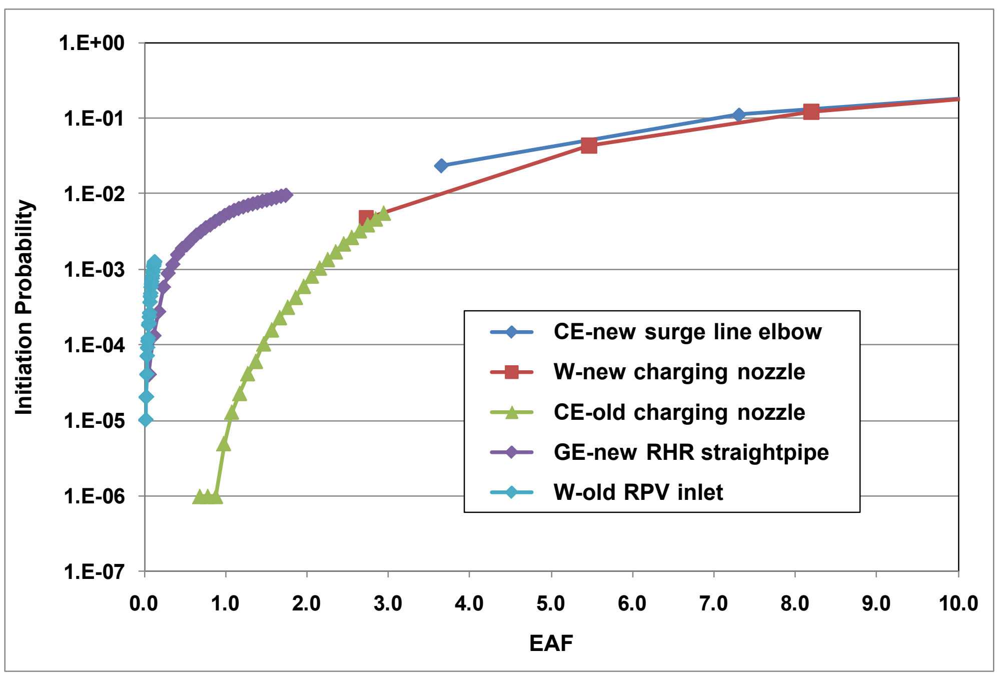

Figure 5-5
Comparison of Initiation Probability to EAF using 0.1% Fractile Fatigue Curve

Figure 5-6 and Figure 5-7 provide plots of the leak frequency corresponding to Figure 5-4 and Figure 5-5.

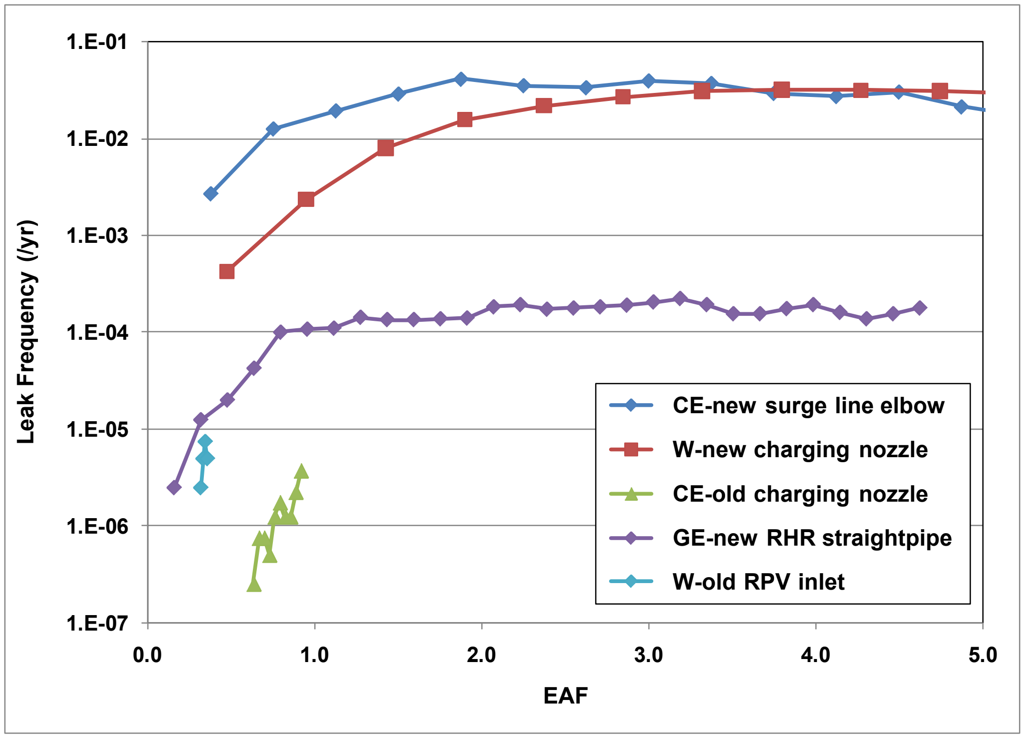

Figure 5-6
Comparison of Leak Probability to EAF using ASME Design Curve

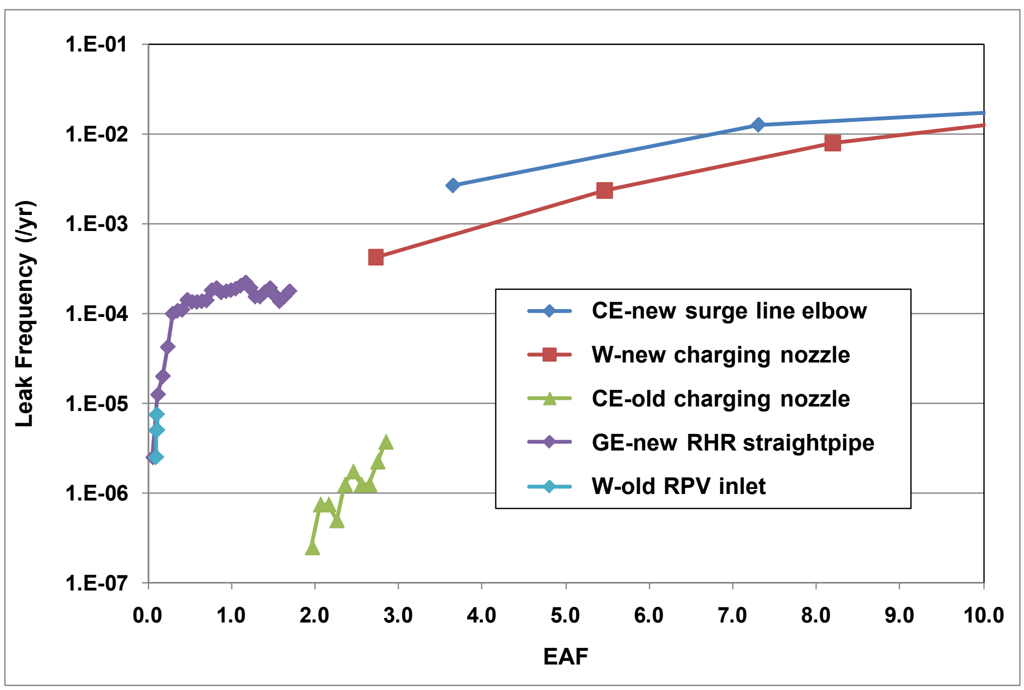

Figure 5-7
Comparison of Leak Probability to EAF using 0.1% Fractile Fatigue Curve

The already poor relation between initiation probability and EAF is further degraded when considering leaks, because other factors than those entering into the EAF become important. Such factors include fatigue crack growth and spatial stress gradients.

Since it is clear that evaluating the change in EAF criterion from that of 0.1 to some other value is significantly hampered due to inconsistencies in the impact of EAF on initiation, leak probability, and core damage frequency, another approach was considered. The impact of an EAF value of 1.0 was evaluated, consistent with the ASME Code and what is considered to be acceptable for other plant locations, in accordance with the NUREG-1801.

For new plants, SRP Chapter 19 provides guidance for performing probabilistic risk assessment and severe accident evaluation. As part of the applicants PRA and severe accident evaluation, the risk associated with the design of the plant is compared to the NRC's CDF goal of less than $1x10^{-4}$/year and LERF goal of $1x10^{-6}$/year. These goals were promulgated in SECY-90-016 and are not a regulatory requirement.

For determining acceptable risk at existing plants, the guidance in the EPRI PSA Applications Guide and NRC Regulatory Guide 1.174 were applied. The risk-acceptance guidelines presented in this regulatory guide are based on the principles and expectations for risk-informed regulation discussed further in Section 6, and they are structured as follows.

Regions are established in the two planes generated by a measure of the baseline risk metric (CDF or LERF) along the x-axis, and the change in those metrics (e.g., ΔCDF) along the y-axis (Figure 5-8) and acceptance guidelines are established for each region as discussed below.

These guidelines are intended for comparison with a full-scope (including internal events, external events, full power, low power, and shutdown) assessment of the change in risk metric, and when necessary, as discussed below, the baseline value of the risk metric (CDF or LERF). However, it is recognized that many PRAs are not full scope and PRA information of less than full scope may be acceptable.

There are two sets of acceptance guidelines, one for CDF and one for LERF. Both sets should be used, but as NUREG/CR-6674 provided only a method for relating CDF as a function of leakage, only CDF is used for this example.

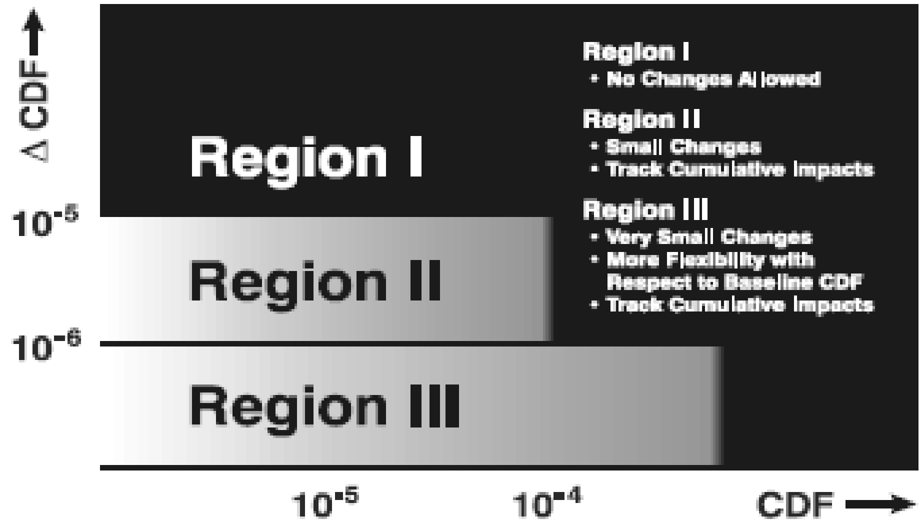

Figure 5-8
Acceptance Guidelines for Core Damage Frequency (CDF)

When the calculated increase in CDF is very small, which is taken as being less than 10^{-6} per reactor year, the change will be considered regardless of whether there is a calculation of the total CDF (Region III). While there is no requirement to calculate the total CDF, if there is an indication that the CDF may be considerably higher than 10^{-4} per reactor year, the focus should be on finding ways to decrease rather than increase it. Such an indication would result, for example, if (1) the contribution to CDF calculated from a limited scope analysis, such as the individual plant examination (IPE) or the individual plant examination of external events (IPEEE), significantly exceeds 10^{-4}, (2) a potential vulnerability has been identified from a margins-type analysis, or (3) historical experience at the plant in question has indicated a potential safety concern.

When the calculated increase in CDF is in the range of 10^{-6} per reactor year to 10^{-5} per reactor year, applications will be considered only if it can be reasonably shown that the total CDF is less than 10^{-4}per reactor year (Region II).

Applications that result in increases to CDF above 10^{-5} per reactor year (Region I) would not normally be considered.

For the purposes of our example, the total CDF is conservatively taken as the change in CDF. Referring to Figure 5-2 and Figure 5-8, it can be seen that use of an EAF of 1.0 achieves an acceptable risk for all components evaluated. Based on this, the approach in Section 6 is suggested.

Section 6: Suggested Approach For Development of Future Regulations

Although piping constitutes a significant portion of the reactor coolant system boundary, because of its robust design and the protection afforded by other engineered safety systems, piping failures generally make relatively small contributions to core damage frequency (CDF) or large, early release frequency (LERF).

Since there is no documented technical basis for the current requirements for postulated break locations where CUF exceeds 0.1, the authors propose that an alternate methodology be used. In addition to the CUF criterion, the authors feel that it would be worthwhile to reconsider the criterion for postulating breaks at terminal ends, the stress based limit of 80% of allowable at intermediate locations and the current definition for high energy lines in MEB 3-1. This alternate risk-informed methodology would consider revised fatigue limits as well as inclusion of the effects of other potential damage mechanisms.

6.1 Proposed Methodology

The proposed methodology is intended to demonstrate that safety objectives (such as core damage frequency) are met using risk-informed principles. The intent is to use NRC-approved RI-ISI methods and insights as much as possible and supplement this, as needed, based on industry operating experience. It is important to recognize that past operation must be considered in the application of this methodology in order to ensure that any pre-existing degradation (e.g., CUF) is accounted for.

In developing and applying any risk-informed approach, the overall objective is to achieve a level of risk below some specified criteria (i.e., those described in Regulatory Guide 1.174) and apply risk insights in the decisionmaking process. Risk insights are obtained based on an improved understanding of what contributes to risk, which is the product of probability and consequence. This approach is used to complement deterministic methods.

For the purpose of establishing postulated break locations, the proposed methodology takes a four phase approach. The first three phases are more deterministic in nature and the fourth phase uses a probabilistic evaluation such as that described in Section 5 of this report.

Phase 1 eliminates locations where the consequence of failure is low, regardless of the potential for failure, since the overall risk impact would be minimal and there would be no net benefit to any changes in plant design to further mitigate the effects of failure. Phase 2 identifies relevant damage mechanisms and eliminates any locations which can reasonably be demonstrated to have a slow rate of propagation, based on operating experience and literature review. A Leak-Before-Break evaluation is one example of this. Any mechanism which cannot result in a gross failure will also be eliminated from further review. Use of existing criteria for establishing break exclusion regions is also considered acceptable for relevant damage mechanisms. Phase 3 will eliminate locations which have damage mechanisms that can propagate rapidly, if it can be demonstrated that these mechanisms can be effectively mitigated. Finally, phase 4 will perform a probabilistic evaluation against the criterion in Regulatory Guide 1.174 (described in more detail in Section 6.2), eliminating any locations which demonstrate an acceptable level of risk.

Based on the preceding discussion, break (aka rupture) locations will be postulated based on their potential safety and risk impact as follows:

1. The first phase consists of an initial screening process to reduce the number of locations requiring a more detailed evaluation based on potential consequence of rupture (e.g., high mass release lines, or lines which can directly cause problems due to jet impingement). The approach used for RI-ISI to identify locations with low consequence would screen out.
2. For the locations that screen in, the second phase conducts a systematic review of the degradation mechanisms that can lead to breaks based on currently published literature. Based on a literature search [30 through 38], the following pipe failure mechanisms/causes require further evaluation for their pipe rupture and rupture mitigation potential:
    1. Fabrication defects (due to improper material selection, defective materials and poor workmanship).
    2. Overload (pressure)
    3. Brittle fracture due to low service temperature associated with cold water injection and/or change in material properties due to neutron fluence or other mechanisms.
    4. Water hammer.
    5. Fatigue (including thermal and vibration-induced fatigue).
    6. Stress-corrosion cracking (e.g., IGSCC, IASCC, PWSCC, TGSCC, ECSCC)
    7. Flow-Assisted Corrosion (including cavitation erosion, liquid impingement erosion and abrasive erosion).
    8. Seismic loads.

9. Microbiologically Influenced Corrosion (MIC).
10. Thermal aging (e.g., Cast Austenitic Stainless Steel).
11. Creep/stress rupture.
12. Others which may be identified as part of the literature review.

The evaluation will focus on those mechanisms which may lead to a loss of reactor coolant (i.e., pressure boundary function of reactor coolant piping systems) due to gross failures such as guillotine breaks, large axial splits. Therefore, leaks which do not result in a significant threat to the ability to perform required safety functions will be excluded from further evaluation.

Note: The only degradation mechanisms that would lead to a pipe rupture ("gross failure"), as opposed to a crack or leak, are FAC and Water Hammer per RI-ISI methodologies approved by the NRC in ASME Code Cases N-577-1 and N-578-1. An examination should be performed to determine whether there are other degradation mechanisms which could also cause a gross failure such as those mentioned above (e.g., overpressure) or other significant consequence which require evaluation. Consideration should also be given to the benefit of actions such as those currently applied by SRP Section 3.6.2 and MEB 3-1 for establishing break exclusion regions.

For each mechanism, consideration is given as to whether the mechanism potentially results in rapid propagation or if propagation is slow (i.e., can be reasonably managed via the RI-ISI program, including applicable augmented inspection programs such as FAC inspections required by Generic Letter 89-08). Per EPRI TR-112657, the mechanisms that can be effectively managed via RI-ISI include Thermal Fatigue, Stress Corrosion Cracking, MIC, Pitting, Crevice Corrosion, Erosion-Cavitation and FAC.

3. This phase of the evaluation would next focus on those mechanisms leading to pipe rupture where rapid propagation can occur. The intent of this step is to mitigate those mechanisms, where feasible. Where reliable methods exist to detect the onset of conditions leading to the mechanism, existing or new programs may be used to manage the mechanism. For mechanisms where the ability to detect the onset of conditions that lead to the mechanism is not highly reliable, a risk-informed management strategy may be implemented, and programs and processes for reducing the probability of failure identified. Some examples follow for the pipe failure mechanisms/causes identified in item number 2:

    1. Fabrication defects (due to improper material selection, defective materials and poor workmanship).
       Methods intended to reduce the probability of failure would include quality controls and testing performed prior to placing piping systems in service. Quality controls include requirements that materials are procured from qualified sources, material testing is performed which confirms the material properties meet design specifications, pre-service and in-service examination of welds are performed and hydrostatic leak testing is conducted. Controls on water quality used for leak testing are intended

to ensure that chlorides or other halogens as well as other deleterious contaminants, which can lead to intergranular attack or pitting of piping materials, are not introduced.

2. Overload (pressure)
   The probability of failure due to overload is minimized by existing ASME Code primary stress limits and rules for overpressure protection.

3. Brittle fracture due to low service temperature and/or change in material properties due to neutron fluence
   Methods intended to reduce the probability of failure include controls on operation and regulatory guidance for evaluating the effects of neutron fluence on reactor coolant materials such as those in ASME Code Section XI Appendix G, 10CFR50 Appendix G and 10CFR50.61.

4. Water hammer
   Methods intended to reduce the probability of failure include designs which minimize the potential for air to be trapped, use of pressure surge devices, controlling the speed of valves, etc.

5. Fatigue (including thermal and vibration-induced fatigue)
   The probability of failure due to fatigue is reduced by considering thermal and pressure cycles as part of the design of the piping system as part of the ASME Code fatigue analysis with a CUF allowable of 1.0, and by incorporating information from the EPRI Fatigue Management Handbook.

6. Stress-corrosion cracking (e.g., IGSCC, IASCC, PWSCC, TGSCC, etc.).
   Methods intended to reduce the probability of failure include designs which include resistant materials or provide compressive stress fields at susceptible locations. Controls on reactor coolant water chemistry that essentially eliminate the presence of halogens and minimize its electrochemical potential also reduce the probability of failure.

7. Flow-Assisted Corrosion (including cavitation erosion, liquid impingement erosion and abrasive erosion)
   Methods intended to reduce the probability of failure include designs which minimize the number of areas with high levels of turbulent flow, incorporation of alloying constituents more resistant to erosion/corrosion and improved water chemistry controls. This is supplemented by inspection of susceptible areas to identify and monitor piping wall thickness loss.

8. Seismic loads
   The potential for failures due to seismic loads are minimized by existing ASME Code primary and primary plus secondary stress limits.

9. Microbiologically Influenced Corrosion (MIC)
   The probability of failure due to MIC is minimized by use of closed water systems which are treated and/or do not contain raw water and implementation of actions required to address Generic Letter 89-13.

10. Thermal aging (e.g., Cast Austenitic Stainless Steel)

11. Creep/stress rupture (This will not be evaluated further since the coincident conditions of time/temperature/stress for light water reactor coolant piping does not result in creep damage.)

As part of this phase, the ability to detect the onset of the mechanism or parameters which lead to break mechanisms where break propagation is rapid will be assessed. An example where rapid propagation can occur that is difficult to detect is high-cycle fatigue (HCF). HCF has a relatively long incubation period where crack initiation is difficult to detect and is followed by a significantly shorter period of crack propagation where often the first indication of a problem is leakage. For this mechanism, sources of HCF would need to be identified by the plant or plant designer based on plant design specifications and a literature review (including industry operating experience). These sources would then need to be accounted for in fatigue calculations wherein the calculated alternating stress must be below the material endurance limit or the consequences of HCF failures are evaluated as acceptable based on a plant-specific risk assessment. Furthermore, relevant plant programs (e.g., corrective action, operating experience, equipment reliability, fatigue management) must contain guidance for identifying new sources of HCF and ensuring they are appropriately addressed.

4. This phase applies a probabilistic approach to assess risk (product of probability and consequence). Specific details regarding the various methods for calculating failure probabilities and consequences of failure are beyond the scope of this report. One approach which could be applied is that outlined in Section 5 using the methodology in NUREG/CR-6674. NRC-approved methods applied in RI-ISI applications that calculate failure probabilities and consequence are also acceptable. For each plant, the current licensing basis documents are applied, as applicable (e.g., NUREG documents used for evaluating the effects of reactor water environment on fatigue).

   An acceptable level of risk is achieved by reducing the consequence of failure, the probability of failure due to relevant damage mechanisms or some combination of these as described below:

   1. For lines having higher failure consequences, the failure probability must be maintained at a sufficiently low level. For these cases, the parameters that have the most significant impact on failure probability will be identified by reviewing the literature (such as NUREG/CR-6837), using analysis insights from the use of structural reliability software such as pcPRAISE and applying industry experience from RI-ISI programs.

   2. For lines with lower failure consequences, a somewhat higher failure probability can be tolerated so long as the overall impact on core damage frequency (CDF) and large early release frequency (LERF) is low when compared to guidance in EPRI TR-105396 and Regulatory Guide 1.174.

6.2 Basis for Regulatory Acceptance of Proposed Methodology

As noted in the NRC policy statement for the use of Probabilistic Risk Assessment (PRA) methods in nuclear regulatory activities and Regulatory Guide 1.174: *"PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices…It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised."*

Five principles are associated with NRC acceptance of a risk-informed approach, as outlined in Regulatory Guide 1.174 and illustrated in Figure 6-1 (excerpted from Regulatory Guide 1.174):

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change, i.e., a "specific exemption" under 10 CFR 50.12 or a "petition for rulemaking" under 10 CFR 2.802 (presumption of adequate protection),
2. It is consistent with the defense-in-depth philosophy,
3. It maintains sufficient safety margins,
4. When proposed changes result in an increase in CDF or risk, it is small and consistent with the intent of the Commission's Safety Goal Policy Statement, and
5. Will be monitored using performance measurement strategies.

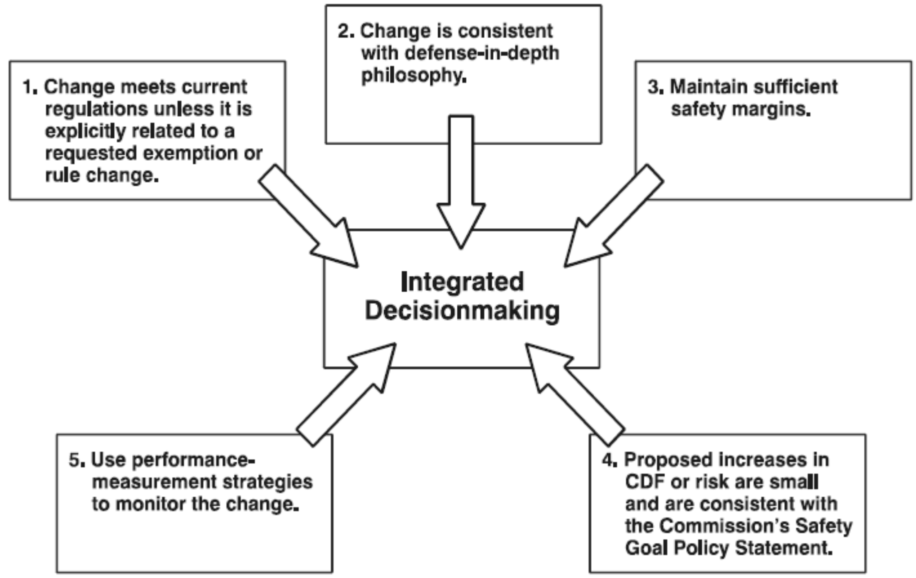

Figure 6-1
Principles of Risk-Informed Integrated Decision Making

The suggested approach provided above will require plant-specific risk analysis insights to be applied. Since existing plants are required to have an Individual Plant Examination (IPE) performed to meet NRC Generic Letter 88-20 and, for new plants, SRP Chapter 19, information on assumed break probabilities and sensitivity of line breaks on CDF/LERF should be available.

Furthermore, in implementing these principles, the NRC has established expectations in Regulatory Guide 1.174, which include the following:

- Safety impacts of the proposed change are evaluated in an integrated manner as part of an overall risk management approach in which the licensee is using risk analysis to improve operational and engineering decisions broadly by identifying and taking advantage of opportunities to reduce risk, and not just to eliminate requirements the licensee sees as undesirable. For those cases when risk increases are proposed, the benefits should be described and should be commensurate with the proposed risk increases.
- The scope, level of detail, and technical acceptability of the engineering analyses (including traditional and probabilistic analyses) conducted to justify the proposed change should be appropriate for the nature and scope of the change, should be based on the as-built and as-operated and maintained plant, and should reflect operating experience at the plant.
- The plant-specific PRA supporting the licensee's proposals has been subjected to quality assurance methods and quality control methods.
- Appropriate consideration of uncertainty is given in analyses and interpretation of findings, including using a program of monitoring, feedback, and corrective action to address any significant uncertainties.

- The use of core damage frequency (CDF) and large early-release frequency (LERF) as bases for PRA acceptance guidelines is an acceptable approach to addressing Principle 4.
- Increases in estimated CDF and LERF resulting from proposed LB changes will be limited to small increments. The cumulative effect of such changes should be tracked and considered in the decision process.
- The acceptability of proposed changes should be evaluated by the licensee in an integrated fashion that ensures that all principles are met.
- Data, methods, and assessment criteria used to support regulatory decisionmaking must be well documented and available for public review.

Section 7: Conclusions

A background on the current regulations is provided, noting the lack of a defined technical basis for the cumulative fatigue usage criterion of 0.1 used to postulate pipe breaks in high energy reactor coolant piping. The objective of this report was to establish the technical basis for a fatigue usage criterion.

Since various methods and level of detail are used when calculating stresses and fatigue usage, operating experience information was reviewed in an attempt to establish some correlation to fatigue usage and identify what other failure mechanisms have resulted in pipe breaks in high energy piping. This review indicated that fatigue usage based on design basis calculations is, at most, a minimal contributor to the potential for pipe failures.

A review of LRAs, LARs and DCDs from NRC publically available information was performed. There were no instances identified where existing plants had sought relief from the current requirements and one instance where a DCD has been submitted seeking approval for the use of a fatigue usage criterion of 0.4 when environmental factors were applied (ESBWR).

The strategy for current operating plants seeking license renewal is to perform fatigue monitoring for limiting locations to ensure that any locations that are expected to exceed the current 0.1 CUF criterion are identified in advance. To date, environmental factors have not been applied to these locations and are applied to locations identified in NUREG/CR-6260 and assessed for other plant locations as discussed in NUREG-1801.

Although no instances were identified where exceptions to the current rules were requested for current operating plants, this does not mean that the current regulations have not caused utilities to incur additional expenses to meet the CUF allowable. The likelihood that plants will incur additional expense to meet the current CUF allowable increases as plants seek to extended their operating licenses beyond the original license term.

A review of industry operating experience was also performed. The operating experience review clearly indicates that the potential for high energy line failures is dominated by mechanisms other than thermal fatigue due to design plant thermal transients, which typically constitutes nearly all of the calculated fatigue usage in piping systems. In addition, piping ruptures were associated with non-RCPB systems (extraction steam, feedwater heater drains, etc.).

An evaluation of the relationship between fatigue usage (considering effects from the reactor water environment), leak probability, and risk was performed for selected component locations using the methodology previously developed in NUREG/CR-6674 and current NRC guidance for evaluating the effects of reactor water environment on fatigue. The component locations account for variation in materials, stress history, geometry and design fatigue usage.

The results of this evaluation showed that consideration of fatigue usage by itself is not a reliable approach to predict crack initiation or leakage. Therefore, it is also an unreliable parameter for estimating rupture. However, in all cases evaluated, the use of a CUF criterion of 1.0 resulted in a minimal impact on core damage frequency (CDF) within limits commonly found to be acceptable to the NRC.

Based on these results, a proposed methodology to be used in NRC regulatory guidance for postulating line breaks is offered for consideration based on the NRC's policy statement on probabilistic risk assessment and Regulatory Guide 1.174. This methodology applies both deterministic and probabilistic methods which could also be used to replace the criterion for postulating breaks at terminal ends and the stress based limit of 80% of allowable at intermediate locations in MEB 3-1 (MEB 3-1 Revision 1 is attachment to NRC Generic Letter 87-11).

Section 8: References

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25. License Renewal Applications (LRAs) and associated Safety Evaluation Reports (SERs), available from NRC website at http://www.nrc.gov/reactors/operating/licensing/renewal.html

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28. EPRI TR-105396, “PSA Applications Guide,” August 1995.

29. NRC Regulatory Guide 1.174, Revision 1, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” July, 2002.

30. EPRI Report NP-3944, "Erosion/Corrosion thermal aging in Nuclear Plant Steam Piping: Causes and Inspection Program Guidelines,"

31. ASM Handbook, Volume 11, Failure Analysis and Prevention.

32. EPRI Report 112657, Revised Risk-Informed Inservice Inspection Evaluation Procedure.
33. ASME Code Section XI Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components, Appendix G.
34. Appendix G to Part 50 – Fracture Toughness Requirements.
35. 10CFR50.61, Fracture toughness requirements for protection against pressurized thermal shock events and 10CFR50.61a, Alternate fracture toughness requirements for protection against pressurized thermal shock events.
36. EPRI Report 1015010, Material Reliability Program: Fatigue Management Handbook (MRP-235).
37. EPRI Report NP-3944, "Erosion/Corrosion thermal aging in Nuclear Plant Steam Piping: Causes and Inspection Program Guidelines."
38. NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," July, 1989.
39. NRC Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities 10CFR50.54(f)," November 1988.
40. NUREG/CR-5864, "Theoretical and User's Manual for pc-PRAISE, A Probabilistic Fracture Mechanics Code for Piping Reliability Analysis", July 1992.
41. NUREG-0933, "Resolution of Generic Safety Issues," Supplement 33, Section 2. Task Action Plan Items, TMI Action Plan Item A-18, August 2010.
42. Mechanical Engineering Branch (MEB) Technical Position MEB 3-1, Revision 1, transmitted by NRC Generic Letter 87-11, on June 11, 1987.
43. NRC, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," Federal Register, Vol. 60, p. 42622 (60 FR 42622), August 16, 1995.
44. NRC, SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements." January 12, 1990.
45. NRC Regulatory Guide 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," January, 2007.
46. NUREG/CR-2189, Vol. 5, "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant, Vol. 5: Probabilistic Fracture Mechanics Analysis", 1981.
47. NUREG/CR-6335, "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments", 1995.

48. NUREG/CR-6909, “Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials,” February 2007.

49. ASME Code Case N-577-1, “Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method A,” Approval date March 28, 2000.

50. ASME Code Case N-578-1, “Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B,” Approval date March 28, 2000.

51. NRC Generic Letter 84-04, “Safety Evaluation of Westinghouse Topical Reports dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops,” February, 1984.

52. NRC Generic Letter 89-08, “Erosion/Corrosion Induced Pipe Wall Thinning,” May, 1989.

8.2 Glossary

The definitions provided in this appendix were obtained from the references listed in the report.

C

Cavitation Erosion. Cavitation erosion is an attack of metal surfaces caused by the collapse of cavitation bubbles on the surface of the liquid and characterized by pitting.

Core Damage. Uncovery and heatup of the reactor core to the point where prolonged oxidation and severe fuel damage is anticipated.

Core Damage Frequency (CDF). An expression of the likelihood that, given the way a reactor is designed and operated, an accident could cause the fuel in the reactor to be damaged.

Cumulative Usage Factor (CUF). The cumulative usage factor (CUF) is the sum of the individual usage factors, and the ASME Code Section III requires that the CUF at each location must not exceed 1.

D

Design Basis Accident (DBA). A DBA is a postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety.

Design Certification Documents (DCDs). The review process for new reactor designs involves certifying standard reactor designs, independent of a specific site. DCDs are the documents submitted by applicants. Design certification applicants must provide the technical information necessary to demonstrate compliance with the safety standards set forth in applicable NRC regulations. The DCDs must include a detailed analysis of the design's vulnerability to certain accidents or events, and inspections, tests, analyses, and acceptance criteria to verify the key design features.

Deterministic (probabilistic). Consistent with the principles of "determinism," which hold that specific causes completely and certainly determine effects of all sorts. As applied in nuclear technology, it generally deals with evaluating the safety of a nuclear power plant in terms of the consequences of a predetermined bounding subset of accident sequences. The term "probabilistic" is associated with an evaluation that explicitly accounts for the likelihood and consequences of possible accident sequences in an integrated fashion. See also Probabilistic Risk Assessment (PRA).

E

Emergency Core Cooling Systems (ECCS). The ECCS includes reactor system components (pumps, valves, heat exchangers, tanks, and piping) that are specifically designed to remove residual heat from the reactor fuel rods in the event of a failure of the normal core cooling system (reactor coolant system).

Environmentally Adjusted Cumulative Usage Factor (CUF_{en}). The environmentally adjusted cumulative usage factor (CUF_{en}) is the sum of the individual usage factors multiplied by appropriate fatigue life correction factor (F_{en}). The CUF_{en} at each location must not exceed 1.

Erosion Corrosion. Erosion corrosion is an acceleration in the rate of corrosion attack in metal due to the relative motion of a corrosive fluid and a metal surface. The increased turbulence caused by pitting on the internal surfaces of a tube can result in rapidly increasing erosion rates and significant thinning.

F

Fatigue Life Correction Factor (F_{en}). The effects of reactor coolant environments on fatigue life have been expressed in terms of a fatigue life correction factor, F_{en}, which is the ratio of the life in air at room temperature to that in water at the service temperature. The value of F_{en} is a function of a set of variables such as the sulfur content, dissolved oxygen content, strain rate, and temperature.

Flow Accelerated Corrosion (FAC). Flow accelerated corrosion, also known as flow-assisted corrosion, is a corrosion mechanism in which a normally protective oxide layer on a metal surface dissolves in fast flowing water. The underlying metal corrodes to re-create the oxide, and thus the metal loss continues. The rate of FAC depends on the flow velocity. FAC most often affects carbon steel piping carrying pure, deoxygenated water or wet steam.

I

Intergranular Stress Corrosion Cracking (IGSCC). IGSCC occurs as a result of the effects of stress and environment on a susceptible material. Type 304 and type 316 austenitic stainless steel piping has been identified as being susceptible materials.

L

Leak-Before-Break (LBB). Subject to certain limitations, Appendix A to 10CFR50 Criterion 4 allows dynamic effects associated with postulated pipe ruptures to be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate the probability of fluid system piping rupture is extremely low under design basis conditions. These analyses are commonly referred to as "leak-before-break" analyses.

Loss of Coolant Accident (LOCA). Those postulated accidents that result in a loss of reactor coolant at a rate in excess of the capability of the reactor makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.

P

robabilistic Risk Assessment (PRA). PRA is a systematic method for assessing three questions that the NRC uses to define risk. These questions consider (1) what can go wrong, (2) how likely it is, and (3) what its consequences might be. These questions allow the NRC to understand likely outcomes, sensitivities, areas of importance, system interactions, and areas of uncertainty, which the staff can use to identify risk-significant scenarios. The NRC uses PRA to determine a numeric estimate of risk to provide insights into the strengths and weaknesses of the design and operation of a nuclear power plant.

R

Reactor Coolant Pressure Boundary (RCPB). The reactor coolant pressure boundary provides a barrier against the release of radioactivity generated within the reactor. It consists of the reactor pressure vessel and the pressure-retaining portion of systems directly connected to the reactor pressure vessel and exposed to the reactor coolant.

Risk. Risk encompasses what can happen (scenario), its likelihood (probability), and its level of damage (consequences).

Risk-Informed Decisionmaking. An approach to regulatory decisionmaking, in which insights from probabilistic risk assessment are considered with other engineering insights

S

Standard Review Plan (SRP). The SRP is a document that provides guidance to the staff for reviewing an application to obtain an NRC license to construct or operate a nuclear facility or to possess or use nuclear materials. NUREG-0800 is the SRP for plants licensed to the requirements of 10CFR50 and NUREG-1800 is the SRP for plants being licensed to the requirements of 10CFR54.

T

Thermal Stratification. Thermal stratification results from a temperature differential across a pipe or other component cross-section where the upper fluid stream is at a significantly higher temperature than the lower fluid stream. This condition occurs under relatively low flow conditions, by hot fluid injection into a stagnant cold region or vice versa. This induces gross thermal bending moments across the pipe cross-section and results in a bowing deflection.

Time Limited Aging Analysis (TLAA). Time limited aging analyses consist of calculations and analyses relevant to the plant's licensing bases that include time-limited assumptions defined by the current operating term involving aging effects for systems, structures and components within the scope of license renewal. 10CFR54.3 contains the specific definition of TLAA related to license renewal. Fatigue calculations are an example of a TLAA, as defined in 10CFR54.3.

Thermal Striping. Thermal striping is a cyclic temperature variation which results from the existence of a hot/cold interface layer during stratified flow condition, in conjunction with thermal stratification bending. The level of the interface layer raises and lowers as the thermal stratification condition initiates and abates; creating heat-transfer induced thermal stresses in the pipe wall. The interface layer may fluctuate rapidly and locally.

V

Vibration Fatigue. High cycle fatigue due to vibrational loading. The vibrational loading may be either steady state or transient.

Appendix A: HELB Information Associated with License Renewal Applications

This appendix provides a summary of information related to License Renewal Applications and, where issued, Safety Evaluation Reports issued by the NRC as of December, 2010. It should be noted that plants which have applied LBB are not affected by the consideration of environmental multipliers to CUF values, since LBB evaluates critical flaw size under design load conditions and not the development of initial cracking.

Table A-1
Review of HELB Criteria for Plants which have submitted License Renewal Applications

Plant	**LRA Section/ Table References**	**SER Issued?**	**Piping Design Code**	**HELB Criteria**	**Additional Locations where 60-year Projected CUF>0.1?**
ANO-1 Unit 1	Section 4.3.4, Thermal Fatigue Section 4.8.3, Leak Before Break	Y	ASME Section III / B31.7	LBB approach used per NUREG-1061 (LBB)	No, since projected 60 year cycles are less than original 40 year cycles.
ANO-1 Unit 2	Section 4.3.1, Class 1 Fatigue Table 4.3.1, RCS Design Transients Section 4.7.1, RCS Piping Leak Before Break	Y	ASME Section III	LBB approach used per CEN-367-A (LBB)	No, since projected 60 year cycles are less than original 40 year cycles.
Beaver Valley Unit 1 & 2	Section 4.3.1, Class 1 Fatigue Section 4.7.3, Leak Before Break Section B.2.27, Metal Fatigue of Reactor coolant Pressure Boundary	Y	B31.1 except pressurizer surge line (Unit 1) ASME Section III (Unit 2)	LBB approach used per Generic Letter 84-04 WCAP-11317 WCAP-11923	No, since projected 60 year cycles are less than original 40 year cycles (except Unit 2 RHR and charging line piping).
Browns Ferry Units 1, 2 & 3	Section 4.3.3, Piping and Component Fatigue Analysis Section B.3.2, Fatigue Monitoring Program	Y	B31.1	Not discussed in LRA.	No. Plant piping was designed per B31.1, which allows up to 7,000 cycles without taking a penalty. No reduction in allowable stress limits was identified.
Brunswick Unit 1 & 2	Section 4.3.4, Reactor Coolant Pressure Boundary Piping and Component Fatigue Analyses Table 4.3-1, Reactor Design Transients and 60-Year Cycle Projections	Y	B31.1	Not discussed in LRA.	No. Plant piping was designed per B31.1, which allows up to 7,000 cycles without taking a penalty. The only reduction was for the feedwater system, which was accounted for in the original design basis for the plant.
Calvert Cliffs Unit 1 & 2	Appendix A, Table 5, Potential TLAAs Associated with Codes, Standards and Regulatory	Y	ASME Section III / B31.7	Not discussed in LRA.	LRA discusses pipe rupture as being related to erosion corrosion concerns. CUF discussion relates to CUF values

Plant	LRA Section/ Table References	SER Issued?	Piping Design Code	HELB Criteria	Additional Locations where 60-year Projected CUF>0.1?
	Documents Appendix A, 4.1, Reactor Coolant System, Group 4 (fatigue)				being less than 1.0. LBB discussion is limited to CASS components.
Catawba Unit 1 & 2	Section 4.3, Metal Fatigue Section 4.7.2, Leak Before Break Analyses	N	ASME Section III	LBB approach used per WCAP-10546	No. In addition, committed to monitor cycles to ensure design cycles are not exceeded.
Columbia	Section 4.3.3, Reactor Coolant Pressure Boundary Piping and Piping Component Fatigue Analysis Table 4.3-5, CUFs for Reactor Pressure Boundary Piping and Piping Components Section B.2.24, Fatigue Monitoring Program	N	ASME Section IIII	CUF >0.1 per FSAR Section 3.6.2.	Although none were specifically identified, the LRA states that this will be monitored using the Fatigue Monitoring Program and actions taken to address the new break locations, if required.
Cooper	Section 2.1.4, Generic Safety Issues Section 4.3.1.4, Class 1 Piping Table 4.3-2, CUFs of Record for CNS Class 1 Components Section B.1.15, Fatigue Monitoring	T	ASME Section III (replaced recirculation piping) / B31.1 (remainder)	GSI-156.6.1	LRA references GSI for postulated pipe breaks, but in references AMP for flow accelerated corrosion and for fatigue monitoring, uses CUF of 1.0.
Crystal River Unit 3	Table 4.1-2, Review of Generic TLAAs Listed on Table 4.1-2 and 4.1-3 of NUREG 1800.	N	ASME Section III / B31.7	GSI-156.6.1	LRA references GSI for postulated pipe breaks, but in references AMP for flow accelerated corrosion and for fatigue monitoring, uses CUF of 1.0. Table 4.1-2 states that HELB postulation did not meet TLAA criteria.

Plant	LRA Section/ Table References	SER Issued?	Piping Design Code	HELB Criteria	Additional Locations where 60-year Projected CUF>0.1?
Duane Arnold	Section 4.3.2, Fatigue of Class 1 Piping Table 4.3-2, Usage Factors Section A.18.3.2.2, Fatigue of Class 1 Piping	Y	B31.7 / B31.1	GSI-156.6.1	LRA references GSI for postulated pipe breaks, but in references AMP for flow accelerated corrosion and for fatigue monitoring, uses CUF of 1.0.
D. C. Cook Unit 1 & 2	Section 4.3, Metal Fatigue Table 4.3-1, RCS Design Transients – Projection to 60 Years Section 4.7.1, Reactor Coolant System Piping Leak-Before Break	Y	B31.1 / ASME Section III (pressurizer surge line, charging line, aux spray line per NRC Bulletin 88-08 and 88-11)	LBB approach used per WCAP-15131 WCAP-15434	No; projected 60 year cycles are less than original 40 year cycles.
Diablo Canyon Unit 1 & 2	Section 4.3, Metal Fatigue Analysis Section 4.3.2.7, Reactor Coolant Pressure Boundary Piping Table 4.3-2, DCPP Units 1 and 2 Transient Cycle Count and 60-Year Projections Section 4.3.2.12, TLAAs in Fatigue Crack Growth Assessments and Fracture Mechanics Stability Analyses for Leak-Before-Break Elimination of Dynamic Effects of Primary Loop Piping Failures	N	B31.1	LBB approach used per WCAP-13039 WCAP-15434	No; projected 60 year cycles are less than original 40 year cycles.
Dresden Unit 2 & 3	Section 4.3.3, Reactor Coolant Pressure Boundary Piping and Component Fatigue Analysis	Y	B31.1 / ASME Section III (replaced recirculation piping at Unit 3)	Not discussed in LRA.	No. Plant piping was designed per B31.1, which allows up to 7,000 cycles without taking a penalty. No reduction in allowable stress limits was identified.

Plant	LRA Section/ Table References	SER Issued?	Piping Design Code	HELB Criteria	Additional Locations where 60-year Projected CUF>0.1?
Farley	Section 2.1.5.16, Time-Limited Aging Analyses Supporting Information for License Renewal Applications (ISG-16) Section 4.3, Metal Fatigue Section 4.5.2, Leak Before Break Section A.4.2, Metal Fatigue Analysis	Y	B31.1 / ASME Section III (pressurizer surge line, charging nozzles, RHR/SI nozzles per NRC Bulletin 88-08 and 88-11)	LBB approach used per WCAP-12825 WCAP-12835	No; projected 60 year cycles are less than original 40 year cycles.
Fitzpatrick	Section 4.3.1, Class 1 Fatigue Section 4.3.1.3, Class 1 Piping and Components	Y	B31.1 / ASME Section III (portion of replaced Core Spray piping)	Not discussed in LRA.	No. Plant piping was designed per B31.1, which allows up to 7,000 cycles without taking a penalty. No reduction in allowable stress limits was identified.
Ft. Calhoun	Section 4.3.1, Reactor Coolant and Associated Systems Thermal Fatigue Section 4.7.2, Leak Before Break (LBB) Analysis for Resolution of USI A-2 Section 4.7.3, High energy Line Break (HELB)	Y	B31.1 / B31.7	Locations with CUFs greater than 0.1 for 40 years were selected as break locations based on stresses exceeding the other selection criteria of 2Sm.	No. The Class I portions are wrapped in steel "barrel slat" enclosures to prevent lateral pipe movement and the formation of longitudinal and axial jets, which could impact nearby structures and equipment. Pipe whip restraints are installed to limit pipe movement due to circumferential breaks. A potential exception considered were the piping connections to the isolation valves, however, the CUFs at these nodes are less than 0.001 and will not exceed a CUF of 0.1.
Ginna	Section 4.3, Metal Fatigue	Y	B31.1 / ASME Section III (pressurizer surge line per NRC Bulletin 88-11)	LBB approach used per WCAP-12928 WCAP-15837	No. LBB approach used which was projected to 60 years with acceptable results.

Plant	LRA Section/ Table References	SER Issued?	Piping Design Code	HELB Criteria	Additional Locations where 60-year Projected CUF>0.1?
Harris	Section 4.3, Metal Fatigue Section 4.3.1.7, Reactor Coolant Pressure Boundary Piping Section 4.3.4, RCS Loop Piping Leak-Before-Break Analysis Section A.1.2.2.11, RCS Loop Piping Leak-Before-Break Analysis	Y	B31.1 / ASME Section III (pressurizer surge line per NRC Bulletin 88-11)	LBB approach used per WCAP-14549-P	No, since projected 60 year cycles are less than original 40 year cycles except for pressurizer piping which was projected to 60 years with acceptable results.
Hatch Unit 1 & 2	Section 4.2, Pipe Stress Time-Limited Aging Analyses Table 4.2.2-1, ASME Codes Applicable for Class 1 Piping	Y	B31.7 (Unit 1) / ASME Section III (Unit 2)	CUF >0.1 per MEB 3-1	<u>Yes.</u> Per the NRC SER (NUREG-1803) locations were identified that may exceed 0.1. Three bounding locations were identified for continued monitoring.
Hope Creek	Section 4.3.3, Reactor Coolant Pressure Boundary Piping and Component Fatigue Analysis Table 4.3.3-1 Fatigue Monitoring Locations for HCGS RCPB Piping Components and Estimated CUFs	Y	ASME Section III	CUF >0.1 per Section 3.6 of the UFSAR	<u>Yes.</u> Four locations were identified and other locations were identified for continued monitoring.
Indian Point Unit 2 & 3	Section 4.3.1, Class 1 Fatigue Section 4.3.1.8, Class 1 Piping and Components Section 4.7.2, Leak Before Break	Y	B31.1 / ASME Section III (pressurizer surge line per NRC Bulletin 88-11)	LBB approach used per WCAP-10977 (U2) WCAP-10931 (U2) WCAP-8228 (U3)	No. In addition, the number of significant transients projected for 60 years of operation was determined to be acceptable.
Kewaunee	Section 4.3.1, Fatigue of Class 1 Components Section 4.3.1.3, Reactor Coolant Loop Piping Section 4.7.3, Leak Before Break	Y	B31.1 / ASME Section III (pressurizer surge line per NRC Bulletin 88-11)	LBB approach used per WCAP-15311 WCAP-16040-P WCAP-16738	No; projected 60 year cycles are less than original 40 year cycles.

Plant	LRA Section/ Table References	SER Issued?	Piping Design Code	HELB Criteria	Additional Locations where 60-year Projected CUF>0.1?
McGuire Unit 1 & 2	Section 4.3, Metal Fatigue Section 4.7.2, Leak Before Break Analyses	Y	ASME Section III	LBB approach used per WCAP-10585	No. In addition, committed to monitor cycles to ensure design cycles are not exceeded.
Millstone Unit 2	Section 4.3, Metal Fatigue Section 4.7.4, Leak Before Break Table 4.3-2, Millstone Unit 2 – 60-Year Projected Transient Cycles	Y	B31.7	LBB approach used per CEN-367-A	No; projected 60 year cycles are less than original 40 year cycles.
Millstone Unit 3	Section 4.3, Metal Fatigue Section 4.7.4, Leak Before Break Table 4.3-2, Millstone Unit 3 – 60-Year Projected Transient Cycles	Y	ASME Section III	LBB approach used per WCAP-9558 WCAP-9787	No; projected 60 year cycles are less than original 40 year cycles.
Monticello	Section 4.3.3, ASME Section III Class 1 Reactor Coolant Pressure Boundary (RCPB) Piping and Fatigue Analysis Table 4.1-1, List of MGNP Time-Limited Aging Analyses (TLAAs)	Y	B31.1 / ASME Section III (for several RCPB piping systems)	Not specifically listed in LRA.	No. Break locations postulated on pipe size and time of operation, not fatigue criteria.
Nine Mile Point Unit 1	Section 4.3, Metal Fatigue Analysis	Y	B31.1	Not discussed in LRA.	No. Plant piping was designed per B31.1, which allows up to 7,000 cycles without taking a penalty. No reduction in allowable stress limits was identified. In addition, NMP1 was licensed prior to the issuance of 10 CFR 50 Appendix A.

Plant	LRA Section/ Table References	SER Issued?	Piping Design Code	HELB Criteria	Additional Locations where 60-year Projected CUF>0.1?
Nine Mile Point Unit 2	Section 4.3.2, ASME Section III Class 1 Piping and Components Fatigue Analysis Table 4.3-5, NMP2 ASME Section III Class 1 Piping – CUF Bounding Location	Y	ASME Section III	CUF >0.1	Yes. Several locations have a CUF for the 40 year design that are near or above 0.08. Piping locations with a CUF of >0.04 will be monitored by the fatigue monitoring program.
North Anna Unit 1 & 2	Section 4.3, Metal Fatigue Section 4.7.3, Leak Before Break	Y	ASME Section III	LBB approach used per Westinghouse evaluation (WCAP not listed in references for Section 4).	No. In addition, the number of transients projected for 60 years of operation was determined to be acceptable.
Oconee Unit 1, 2 & 3	Section 5.4, Time-Limited Aging Analyses for the Reactor coolant System and Class 1 Components	Y	ASME Section III / B31.7	CUF >0.1 BAW-2243A BAW-1847-R1	Potentially. Although none were specifically identified, the NRC SER states that this will be monitored using the Fatigue Monitoring Program and actions taken to address the new break locations, if required.
Oyster Creek	Section 4.3.3, Reactor Coolant Pressure boundary Piping and Component Fatigue Analysis	Y	B31.1	Not discussed in LRA.	No. Plant piping was designed per B31.1, which allows up to 7,000 cycles without taking a penalty. No reduction in allowable stress limits was identified.

Plant	LRA Section/ Table References	SER Issued?	Piping Design Code	HELB Criteria	Additional Locations where 60-year Projected CUF>0.1?
Palisades	Section 4.3.1, Evaluation of Fatigue in Vessels, Piping and Components Table 4.3-1, Primary Coolant System Design Transients Section 4.3.8, ASME III Class A Primary Coolant Piping Fatigue Analyses Section 4.3.12, Absence of a TLAA for ASME III Class 1 HELB Locations and Leak-Before-Break Analyses Based on Fatigue Usage Factor	Y	ASME Section III / B31.1	>2.4Sm (per NRC Generic Letter 87-11)	No. Intermediate break locations were postulated based on deterministic criteria (longitudinal break with greatest impingement loading, circumferential break with greatest pipe whip) and exceeding stress criterion in NRC Generic Letter 87-11. This was determined not to be a TLAA (in both the LRA and SER).
Palo Verde Unit 1, 2 & 3	Section 4.3.2, ASME III Class 1 Fatigue Analysis of Vessels, Piping and Components Section A2.1, Metal Fatigue of Reactor Coolant Pressure Boundary Section A3.2.1.12, High Energy Line Break Postulation Based on Fatigue Cumulative Usage Factor	N	ASME Section III	LBB approach used and CUF >0.1	Potentially. A LBB approach was used for the RCPB piping and a CUF>0.1 for other Class 1 piping. Although none were specifically identified, the LRA states that this will be monitored using the Fatigue Monitoring Program and actions taken to address the new break locations, if required.
Peach Bottom Unit 2 & 3	Section 4.3.3, Piping and Component Fatigue and Thermal Cycles SER section 4.1, Identification of Time-Limited Aging Analyses	Y	B31.1 / ASME Section III (replaced recirculation piping and portions of the RHR system to address IGSCC concerns)	CUF >0.1	No. Piping analyzed per ASME Section III had been replaced and based on number of years projected through 60 years of plant operation no locations are expected to exceed the 0.1 CUF criterion.

Plant	LRA Section/ Table References	SER Issued?	Piping Design Code	HELB Criteria	Additional Locations where 60-year Projected CUF>0.1?
Pilgrim	Section 4.3.1, Class 1 Fatigue Section 4.3.1.3, Class 1 Piping and Components	Y	B31.1 / ASME Section III (replaced recirculation piping and portions of the RHR system to address IGSCC concerns)	Not discussed in LRA.	No. Plant piping was designed per B31.1, which allows up to 7,000 cycles without taking a penalty. No reduction in allowable stress limits was identified. The LRA only discusses HELB in terms of scoping.
Point Beach Unit 1 & 2	Section 4.3, Metal Fatigue Section 4.4.4, Reactor Coolant System Main Loop Piping Leak-Before-Break Analysis Section 4.4.5, Pressurizer Surge Line Leak-Before-Break Analysis Section 4.4.6, Accumulator Injection Line Piping Leak-Before-Break Analysis Section 4.4.7, Class 1 RHR Line Leak-Before-Break Analysis	Y	B31.1 / ASME Section III (pressurizer surge line per NRC Bulletin 88-11)	LBB approach used per WCAP-13509 WCAP-14439 WCAP-15065-P-A WCAP-15107-P-A WCAP-15105-P-A	No; the number of transients projected for 60 years of operation was determined to be acceptable.
Prairie Island Unit 1 & 2	Section 4.3.1, Class 1 Fatigue Table 4.3-1, PINGP Units 1 and 2 Design and Projected Number of Design Cycles Section 4.7.1, RCS Piping Leak-Before-Break Analyses	Y	B31.1 / ASME Section III (pressurizer surge line per NRC Bulletin 88-11)	LBB approach used per WCAP-10639-P WCAP-12876-NP WCAP-12877-P WCAP-10928-NP	No; projected 60 year cycles are less than original 40 year cycles.
Quad Cities Unit 1 & 2	Section 4.3.3, Reactor Coolant Pressure Boundary Piping and Component Fatigue Analysis	Y	B31.1	Not discussed in LRA.	No. Plant piping was designed per B31.1, which allows up to 7,000 cycles without taking a penalty. No reduction in allowable stress limits was identified.

Plant	LRA Section/ Table References	SER Issued?	Piping Design Code	HELB Criteria	Additional Locations where 60-year Projected CUF>0.1?
Robinson	Section 4.3, Metal Fatigue Section 4.6, Other Plant-Specific Time-Limited Aging Analyses	Y	B31.1 / ASME Section III (pressurizer surge line per NRC Bulletin 88-11 and Aux Feedwater Line)	LBB approach used per WCAP-15628	No; projected 60 year cycles are less than original 40 year cycles.
Salem Unit 1 & 2	Section 4.3, Metal Fatigue of Piping and Components Section 4.3.4, Supplementary ASME Section III, Class 1 Piping and Component Fatigue Analyses Section 4.4.3, Leak-Before-Break Analyses	Y	B31.1 / ASME Section III (pressurizer surge line per NRC Bulletin 88-11 and 88-08 for the auxiliary spray, charging and SI lines)	LBB approach used. Evaluation document not specifically listed in LRA.	No; projected 60 year cycles are less than original 40 year cycles.
Seabrook	Section 4.3, Metal Fatigue Analysis of Piping and Components Section 4.7.3, Leak-Before-Break Analyses Section 4.7.4, High Energy Pipe Break Postulation Based on	N	ASME Section III	LBB approach used per and CUF >0.1 per UFSAR section 3.6	No; projected 60 year cycles are less than original 40 year cycles.
St. Lucie Unit 1 & 2	Section 4.3, Metal Fatigue Section 4.6.1, Leak-Before-Break for Reactor Coolant System Piping	Y	B31.7 (Unit 1) / ASME Section III (Unit 2) In addition, pressurizer surge lines were evaluated per ASME Section III per NRC Bulletin 88-11.	LBB approach used per CEN-367-A	No; projected 60 year cycles are less than original 40 year cycles.
Surry Unit 1 & 2	Section 4.3, Metal Fatigue Section 4.7.3, Leak-Before-Break	Y	B31.1 / ASME Section III (pressurizer surge line per NRC Bulletin 88-11)	LBB approach used. Evaluation document not specifically listed in LRA.	No; projected 60 year cycles are less than original 40 year cycles.

Plant	LRA Section/ Table References	SER Issued?	Piping Design Code	HELB Criteria	Additional Locations where 60-year Projected CUF>0.1?
Susquehanna Unit 1 & 2	Section 4.3.4, Reactor Coolant Pressure Boundary Piping Fatigue Analyses Table 4.3-2, Fatigue Usage for Limiting RCPB Locations Section 4.7.2, High Energy Line Break Cumulative Fatigue Usage Factors	Y	ASME Section III	CUF >0.1	Yes. Additional piping locations were identified that are projected to exceed a 0.1 CUF. These locations will be monitored by the fatigue monitoring program and actions taken as determined to be necessary.
Three Mile Island Unit 1	Section 4.3, Metal Fatigue of Piping Components Section 4.4, Leak-Before-Break Analysis of Primary System Piping	Y	B31.7 / ASME Section III (pressurizer spray line per NRC Bulletin 88-11)	LBB approach used per BAW-1999 BAW-1847-R1	Potentially. Although the number of cycles for 60 years was not specifically compared with the number of design cycles, the LRA states that this will be monitored using the Fatigue Monitoring Program.
Turkey Point Unit 3 & 4	Section 4.3, Metal Fatigue Section 4.7.3, Leak-Before-Break for Reactor Coolant System Piping	Y	B31.1 / ASME Section III (pressurizer surge line per NRC Bulletin 88-11)	LBB approach used per WCAP-14237	No; the number of transients projected for 60 years of operation was determined to be acceptable.
Vermont Yankee	Section 4.3, Metal Fatigue Table 4.3-1, Cumulative Usage Factors	Y	B31.1	Not discussed in LRA.	No. Plant piping was designed per B31.1, which allows up to 7,000 cycles without taking a penalty. No reduction in allowable stress limits was identified. The LRA only discusses HELB in terms of scoping.

Plant	**LRA Section/ Table References**	**SER Issued?**	**Piping Design Code**	**HELB Criteria**	**Additional Locations where 60-year Projected CUF>0.1?**
Vogtle Unit 1 & 2	Section 4.3, Metal Fatigue Section 4.3.1.7, High-Energy Line-Break Postulated Locations Based on Fatigue Cumulative Usage Factor (and related SER section) Section 4.7.1, Leak-before-Break Analysis	Y	ASME Section III	LBB approach used per WCAP-10551 and CUF >0.1 for CVCS lines	No; the projected 60 year cycles are less than original 40 year cycles. The SER suggests that use of the environmental multiplier with a resulting CUF of <1.0 is acceptable for locations with a CUF of 0.1.
Wolf Creek	Section 4.3, Metal Fatigue Analysis 4.3.2.10, High-Energy Line-Break Postulation Based on Fatigue Cumulative Usage Factor Section 4.3.2.11, Fatigue Crack Growth Assessment for the Leak-Before-Break (LBB) Elimination of Dynamic Effects of Primary Loop Piping Failures	Y	ASME Section III	LBB approach used per WCAP-10691 and stress >2.4Sm or CUF >0.1.	No; there are currently no locations where the CUF exceeds 0.1 and the number of transients projected for 60 years of operation was determined to be acceptable. This will be monitored by the fatigue monitoring program to ensure the assessment remains valid.

Appendix B: Methodology for Calculating Core Damage Frequencies (Excerpt from NUREG/CR-6674)

As noted in Section 5.1, this appendix provides the methodology used for calculating core damage frequencies, which was obtained from Section 8 of NUREG/CR-6674. Appendix B of NUREG/CR-6674 provides tables that detail the inputs and results for CDF calculations for the components covered in the NUREG, a subset of which was evaluated in this report.

Through-wall cracks can cause core damage if the leakage rate through the crack exceeds the leakage rates corresponding to plant criteria for small or large LOCAs. The previous sections have described the methodologies and results for calculating frequencies of through-wall cracks, the approach used to assign through-wall cracks to the categories of small and large LOCAs, and the estimates for conditional core damage probabilities.

The following equations were used to develop the tables of Appendix B:

$$CDF = CDF_{LARGE\ LOCA} + CDF_{SMALL\ LOCA}$$

where CDF = core damage frequency contribution from through-wall cracks

$CDF_{LARGE\ LOCA}$ = core damage frequency due to large LOCAs

$CDF_{SMALL\ LOCA}$ = core damage frequency due to small LOCAs

The core damage frequency for a large LOCA is calculated as follows:

$$CDF_{LARGE\ LOCA} = F_{TWC} \times FRAC_{LARGE\ LOCA} \times COND\_CDF_{LARGE\ LOCA}$$

where F_{TWC} = frequency of through-wall cracks (per year)

$FRAC_{LARGE\ LOCA}$ = fraction of through-wall cracks that result in large LOCA

$COND\_CDF_{LARGE\ LOCA}$ = conditional core damage probability given a large LOCA.

Similarly, the core-damage frequency for a small LOCA is calculated as

$$CDF_{SMALL\ LOCA} = F_{TWC} \times FRAC_{SMALL\ LOCA} \times COND\_CDF_{SMALL\ LOCA}$$

where F_{TWC} = frequency of through-wall cracks (per year)

$FRAC_{SMALL\ LOCA}$ = fraction of through-wall cracks that result in small LOCA

$COND\_CDF_{SMALL\ LOCA}$ = conditional core damage probability given a small LOCA.

Both small and large LOCAs can contribute to CDFs. The consequences of large LOCAs are much greater than the consequences for small LOCAs. However, small LOCAs have much higher frequencies of occurrence, making it possible for small LOCAs to make significant contributions to core damage.

Appendix C: Translated Table of Contents

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改良的破裂位置假定的基本要求與規定

摘要

10 CFR 50 在附錄 A 中的 General Design Criteria (一般設計準則) 4 要求核電廠在結構、系統以及零組件的設計上，必須能夠承受管路破裂有關的動態效應。在 NUREG-0800 標準審查規範 (Standard Review Plan) 3.6.1 和 3.6.2 以及相關的部門技術立場 (Branch Technical Position, BTP) 3-3、3-4 和 MEB 3-1 (MEB 3-1，修訂版 1 係以附件形式提供在 NRC Generic Letter 87-11 內) 中有提供相關指導原則。一般而言，這些指導原則在 1970 年代初期制定之後就不曾更動，並且將導致管路破裂的主要損壞機制集中歸因於熱疲勞。自從 70 年代以來，業界不斷累積許多重要經驗，找出除了熱疲勞外，較有可能導致高能量管路失效的各種不同機制。

針對此項研究的工作執行目標是確立出一套可行的建議作法，能夠用來取代或提供現有要求的替代方法。為了解實施修訂方法可能遇到的緊急事件或是需要的時程，將針對現有電廠、尋求執照更新的電廠以及申請新電廠設計者進行調查研究，以確定繼續使用目前規定所造成的影響。

本報告將討論根據電廠營運經驗所獲得的深入分析。報告中同時還會分析針對累積疲勞耗用係數、洩漏概率以及零組件組數目受限所產生的風險之間有何關聯的研究。最後會列舉出假設管路破裂時解決問題的建議作法，作為日後設計核電廠時制定適用法規的考慮因素。

關鍵詞
高能管路破裂
環境性疲勞
執照更新
新電廠設計
一般設計準則

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Amélioration des bases et des impératifs du postulat sur l'emplacement des ruptures

Résumé

L'Annexe A de 10 CFR 50, Critères de conception générale 4, exige que les structures, les systèmes et les composants des centrales nucléaires soient conçus afin de prendre en compte les effets dynamiques associés à la rupture de canalisations. Des directives sont fournies dans Plans d'examen standard 3.6.1 et 3.6.2 de NUREG-0800 et dans Positions techniques de la direction générale (BTP) associées 3-3, 3-4 et MEB 3-1 (MEB 3-1, révision 1, est jointe à la Lettre générique 87-11 du NRC). Ces directives n'ont généralement pas évolué depuis celles mises au point au début des années 1970 et se concentrent sur la fatigue thermique comme le mécanisme principal d'endommagement pouvant entraîner une rupture de canalisations. Depuis les années 70, l'industrie a accumulé une expérience considérable qui identifie maintenant différents mécanismes en complément à la fatigue thermique pouvant plus vraisemblablement entraîner la rupture de canalisations à haute énergie.

L'objectif du travail effectué pour cette étude consistait à mettre en œuvre une approche recommandée pouvant être utilisée en remplacement ou en alternative aux impératifs existants. Afin de comprendre l'urgence et le calendrier de mise en œuvre d'une approche révisée, une enquête a été menée sur des centrales existantes, sur des centrales recherchant un renouvellement de licence et sur des demandes relatives à de nouveaux concepts de centrales afin de déterminer l'impact de la poursuite de l'utilisation des obligations actuelles.

Le rapport traite d'informations tirées de l'expérience des centrales en exploitation. Il décrit également une étude sur la relation entre les facteurs cumulatifs liés à la fatigue pendant l'exploitation, la probabilité de fuites et les risque pour un ensemble limité de composants. Pour terminer, une suggestion d'approche afin de répondre à une rupture concevable de canalisations est formulée pour une prise en compte lors du développement de réglementations ultérieures applicables à la conception des centrales nucléaires.

Mots clés

Rupture de canalisations à haute énergie
Fatigue environnementale
Renouvellement de licence
Nouveaux concepts de centrales
Critères généraux de conception

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破断箇所の条件に対する改善された根拠と要件

要約

10 CFR
50の付録A一般設計条件4は、パイプの破断に関連した動的作用を確保するように、原子力プラントの構造体、システム、および構成部品を設計する必要があります。ガイドラインは、NUREG-
0800企画レビュー計画3.6.1および3.6.2、ならびに関連するで示されているガイドラインおよび部門内規(BTP) 3-3、3-4、およびMEB 3-1(MEB 3-1、改訂1はNRC包括書簡87-
11に添付されている)に示されています。これらのガイドラインは、1970年代初めに開発されたものから一般的に変更されておらず、パイプの破断を導く可能性のある主要な破損メカニズムとして、熱疲労に焦点を当てています。70年代以降、業界の経験を積み、高エネルギーパイプの障害を導く可能性が高い熱疲労に加えて、種々のメカニズムを識別できるようになりました。

この研究に対する作業の目的は、既存の要件に対する代替のアプローチを超える、またはそれを提供するために使用できる推奨アプローチを設定することでした。変更されたアプローチを実施するための緊急性とタイムフレームを理解するために、既存のプラント、ライセンス更新を予定しているプラント、および新しいプラント設計のアプリケーションに対する調査が行われ、現在の要件を引き続き使用する影響を判断しました。

このレポートでは、プラントの運転経験から取得した洞察について説明します。また、蓄積疲労使用要素、漏洩の可能性、構成部品の制限されたセットのリスクの関係の研究も説明します。最終的に、仮想のパイプ破断に対応する推奨アプローチは、原子力プラントの設計に適用可能な将来の規則の開発における考慮事項を概説します。

キーワード
高エネルギーラインの破損
環境的な疲労
ライセンスの更新
新しいプラント設計
一般設計条件

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파열 위치 가정에 대한 개선된 근거 및 요건

초록

10 CFR 50의 첨부 A '일반 설계 기준 4'는 핵발전소들의 구조, 시스템, 구성품들이 관균열과 연관된 역학적 영향을 수용할 수 있도록 설계되어야 한다고 규정하고 있습니다. 가이드라인은 NUREG-0800 Standard Review Plans 3.6.1과 3.6.2 그리고 관련된 Branch Technical Positions (BTP) 3-3, 3-4, 및 MEB 3-1에서 제공됩니다 (MEB 3-1, 제1 개정판은 NRC Generic Letter 87-11의 별첨으로 제공됩니다). 이러한 가이드라인은 1970년대 초에 수립된 상태에서 대개 변경되지 않은 상태이며 관균열의 원인이 될 수 있다는 우려의 1차 손상 메커니즘으로서 열 피로에 초점을 맞추고 있습니다. 70년대 이래 상당한 규모의 업계 경험이 축적되어, 열 피로 말고도 다양한 메커니즘이 고 에너지 관 고장의 원인으로 더 지목될 수 있음을 인식하고 있습니다.

본 연구를 위해 수행된 작업의 목적은 기존 요건에 대한 대안 접근법을 대체하거나 제공하는 데 사용될 수 있는 권장 접근법을 수립하는 것이었습니다. 개정된 접근법의 구현에 필요한 시급성과 일정을 이해하기 위한 설문조사가 기존 발전소, 라이센스 갱신을 모색하는 발전소, 그리고 신규 발전소 설계를 위한 애플리케이션에 대해 현재 요건의 계속적 사용에 대한 영향을 알아내기 위해 수행되었습니다.

보고서는 발전소 가동 경험에서 얻어진 통찰을 논합니다. 또한 제한된 일단의 구성품들의 누적 피로 계수 인자, 누출 확률, 그리고 위험성 사이의 관계에 대한 연구를 서술합니다. 마지막으로, 가정된 관균열을 파악하기 위한 제안된 접근방식이 향후 핵발전소 설계에 적용할 수 있는 규정의 수립을 고려하여 요약되어 있습니다.

주요 어휘
고 에너지 계통 파열
환경 피로
라이센스 갱신
신규 발전소 설계
일반 설계 기준

목차

그림 목록

표 목록

Requisitos y fundamentos mejorados para deducir la ubicación de la rotura

Resumen

El apéndice A de 10 CFR 50 (Criterios de diseño general 4) determina que las estructuras, sistemas y componentes de las centrales nucleares deben diseñarse para tener en cuenta los efectos dinámicos asociados a la rotura de tuberías. Las directrices se incluyen en los Planes de revisión estándar 3.6.1 y 3.6.2 de NUREG-0800 y en las posiciones técnicas sectoriales (Branch Technical Positions, BTP) 3-3, 3-4 y MEB 3-1 (la Revisión 1 de la MEB 3-1 se proporciona como un archivo adjunto del documento genérico de la NRC 87-11). Estas directrices no han sufrido ninguna modificación a nivel general desde que se redactaron a principios de los años 70 y se centran en la fatiga térmica como el principal mecanismo de daños que se debe tener en cuenta en lo referente a la rotura de tuberías. Desde la década de los 70, se ha adquirido un importante grado de experiencia en la industria que ha contribuido a identificar otros mecanismos, además de la fatiga térmica, que con mayor probabilidad provocan fallos en las tuberías de alta energía.

El objetivo del trabajo llevado a cabo para este estudio fue establecer un enfoque recomendado que se pudiera utilizar como sustituto de los requisitos existentes o proporcionar un enfoque alternativo a los mismos. Para comprender la urgencia y el plazo necesarios para implementar un enfoque revisado, se llevó a cabo una encuesta sobre centrales existentes, centrales que tienen prevista una renovación de licencia y solicitudes para el diseño de nuevas centrales con el objetivo de determinar las repercusiones de un uso continuado de los requisitos actuales.

El informe analiza los conocimientos obtenidos de la experiencia de las centrales operativas. Asimismo, describe un estudio sobre la relación entre los factores de uso de la fatiga acumulativa y el riesgo y la probabilidad de fugas relativos a un conjunto limitado de componentes. Finalmente, se ha perfilado un enfoque propuesto para abordar una supuesta rotura de tuberías con el objetivo de que se tenga en cuenta en la elaboración de normativas futuras aplicables al diseño de centrales nucleares.

Palabras clave
Rotura de tuberías de alta energía
Fatiga ambiental
Renovación de licencia
Diseño de nueva central
Criterios de diseño general

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