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January 17, 2003

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Division of Regulatory Improvement Programs
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Interim Staff Guidance (ISG): Recommendations for Fatigue Environmental Effects in a License Renewal Application

PROJECT NUMBER: 690

Dear Dr. Kuo:

The draft Interim Staff Guidance (ISG) 13 for aging management of environmental fatigue for carbon/low-alloy steel is attached for your review, per the commitment in our November 12, 2002 correspondence.

Provided is Draft ISG-13 "Environmental Assisted Fatigue for Carbon/Low-Alloy Steel," a long with recommendations for appropriate revisions to NUREG-1800 and NUREG-1801. The use of the ISG process was identified during the September 18, 2002, management meeting between the Nuclear Energy Institute (NEI), members of the EPRI Materials Reliability Program (MRP) Fatigue Issue Task Group (ITG), and the Nuclear Regulatory Commission (NRC).

The EPRI MRP Fatigue ITG continues to work on similar interim staff guidance and supporting technical documentation for austenitic stainless steel and Ni-Fe-Cr high-nickel alloy components.

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The industry looks forward to working with the NRC staff in efforts to implement ISG 13. The MRP Fatigue ITG is available to meet with the NRC in efforts to clarify any areas of the ISG and to discuss in more detail the technical basis. If you have any questions, or please call me at (202) 739-8110 or Email me at apn@nei.org.

Sincerely,

Alan Nelson

Enclosures

INTERIM STAFF GUIDANCE-13

ENVIRONMENTAL ASSISTED FATIGUE FOR CARBON/LOW-ALLOY STEEL

EXECUTIVE SUMMARY

Recommendations are provided below in the form of Interim Staff Guidance (ISG) that modifies the current staff guidance in NUREG-1800 (Standard Review Plan for License Renewal) and NUREG-1801 (Generic Aging Lessons Learned Report). These recommendations are intended to support the continued use of existing programs to manage fatigue, including the effects of reactor water environments, of carbon and low-alloy components with one surface in contact with primary coolant. Attachment 1 to these recommendations for Interim Staff Guidance contains the technical basis to support the industry findings with respect to environmental effects on carbon and low-alloy steel components.

The technical basis for resolution of the environmental fatigue issue for carbon and low-alloy steel locations, and for the interim staff guidance, are based on four sets of mutually supportive data:

- Results from the re-calculation of fatigue crack initiation and through-wall cracking probabilities, and core damage frequency for carbon and low-alloy steel component locations that were originally evaluated in NUREG/CR-6260 and NUREG/CR-6674; these results, documented in EPRI report 1003667 (MRP-74), are summarized here with the complete MRP-74 report provided in Attachment 2;
- Review and assessment of laboratory fatigue data obtained under simulated reactor water environmental conditions, in terms of thresholds on temperature, strain amplitude, specimen surface strain distribution, strain rate, simulated coolant dissolved oxygen content and oxidation potential, and – in particular – simulated coolant flow rate; these results are contained in a separate EPRI report (MRP-49), but are summarized in Attachment 1 as a complete and consistent set;
- Examination of structural/component scale fatigue tests with at least one surface in contact with the simulated coolant environment, including evaluation of size and surface finish effects; again, these results are contained in MRP-49, but are provided in Attachment 1 as a complete and consistent set; and
- Review and comparison with plant operating experience and failure data on light-water reactor components in the United States.

Based on these findings, the current programs used to manage fatigue can be continued from the current term through the license renewal term, with no need for explicit incorporation of reactor water environmental effects by license renewal applicants, as a part of the 10 CFR 54.21 fatigue aging management program evaluation, for carbon and low-alloy steel component locations for either PWR or BWR plants.

NUREG-1800 Recommended Changes

In order to implement the industry findings with respect to carbon and low-alloy steels, the following changes are recommended for Section 4.3 (Metal Fatigue Analysis) of NUREG-1800 (the Standard Review Plan for License Renewal). The changes to the existing text are indicated by inserted bold face italics or deletion marks.

In Section 4.3.1.2 (Generic Safety Issue), the fourth, fifth, and sixth paragraphs should be changed to read:

> "The scope of GSI-190 included design basis fatigue transients. It studied the probability of fatigue failure and its effect on core damage frequency (CDF) of selected metal components for 60-year plant life. The ***original analysis*** results showed that some components have cumulative probabilities of crack initiation and through-wall growth that approach one within the 40- and 60-year period. The maximum failure rate (through-wall cracks per year) was in the range of 10^{-2} per year, and those failures were generally associated with high cumulative usage factor locations and components with thinner walls, i.e., pipes more vulnerable to through-wall cracks. In most cases, the leakage from these through-wall cracks is small and not likely to lead to core damage. ***These failure rates have been recalculated for carbon and low-alloy steel components [16], using more refined and accurate assumptions, confirming the very low contribution to core damage frequency and revising downward by three to six orders of magnitude the probabilities of through-wall cracking and leakage.*** It was concluded that no generic regulatory action is required and that GSI-190 is resolved based on results of probabilistic analyses and sensitivity studies, interactions with the industry (NEI and EPRI), and different approaches available to licensees to manage the effects of aging (Refs. 11 and 12).
>
> However, the calculations supporting resolution of this issue, which included consideration of environmental effects, indicate the potential for an increase in the frequency of pipe leaks ***for austenitic stainless steel and Ni-Fe-Cr high nickel alloy component locations*** as plants continue to operate. Thus, the staff concluded that licensees are to address the effects of coolant environment on austenitic ***stainless steel and Ni-Fe-Cr high nickel alloy*** component ***location*** fatigue life as aging management programs are formulated in support of license renewal. ***Because of the low probabilities of through-wall cracking and leakage shown in Ref. 16, no explicit consideration of the effects of coolant environment on carbon and low-alloy steel component fatigue life are necessary for aging management programs formulated in support of license renewal.***
>
> The applicant's consideration of the effects of coolant environment on austenitic ***stainless steel and Ni-Fe-Cr high nickel alloy*** component ***location*** fatigue life for license renewal is an area of review."

Section 4.3.2.2 (Generic Safety Issue) should be changed to read:

> "The ***original*** staff recommendation for the closure of GSI-190 is contained in a December 26, 1999 memorandum from Ashok Thadani to William Travers (Ref. 11). The staff recommended ***at that time*** that licensees address the effects of the

coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. One method acceptable to the staff for satisfying this recommendation is to assess the impact of the reactor coolant environment on a sample of critical components. These critical components should include, as a minimum, those selected in NUREG/CR-6260 (Ref. 10). The sample of critical components can be evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses. Formulas for calculating the environmental life correction factors for carbon and low-alloy steels are contained in NUREG/CR-6583 (Ref. 14) and those for austenitic are contained in NUREG/CR-5704 (Ref. 15). ***However, based on the more recent calculation of through-wall cracking and leakage probabilities for carbon and low-alloy component locations in Ref. 16, only the critical austenitic stainless steel and Ni-Fe-Cr high nickel alloy component locations selected in NUREG/CR-6260 (Ref. 10) need to be included. Formulas for calculating the environmental life correction factors for austenitic stainless steels and Ni-Fe-Cr high nickel alloys are contained in NUREG/CR-5704 (Ref. 15)."***

Section 4.3.3.2 (Generic Safety Issue) should be changed to read:

"The reviewer verifies that the applicant has addressed the ***original*** staff recommendation for the closure of GSI-190 contained in a December 26, 1999 memorandum from Ashok Thadani to William Travers (Ref. 11) ***as supplemented by more recent information (Ref. 16).*** The reviewer verifies that the applicant has addressed the effects of the coolant environment on ***austenitic stainless steel and Ni-Fe-Cr high nickel alloy*** component ***location*** fatigue life as aging management programs are formulated in support of license renewal. If an applicant has chosen to assess the impact of the reactor coolant environment on a sample of critical ***austenitic stainless steel and Ni-Fe-Cr high nickel alloy*** components ***locations***, the reviewer verifies the following:

1. The critical components ***locations*** include, as a minimum, those ***austenitic stainless steel and Ni-Fe-Cr high nickel alloy component locations*** selected in NUREG/CR-6260 (Ref. 10).

2. The sample of critical components has~~ve~~ been evaluated by applying ***appropriate*** environmental correction factors to the existing ASME Code fatigue analyses.

3. Formulas for calculating the environmental life correction factors are those contained ~~in NUREG/CR-6583 (Ref. 14) for carbon and low-alloy steels, and~~ in NUREG/CR-5704 (Ref. 15) for austenitic ***stainless steels and Ni-Fe-Cr high nickel alloys*** ~~SSs~~."

In Section 4.3.6 (References), a new Reference 16 should be added, as shown:

"16. Materials Reliability Program: Re-Evaluation of Results in NUREG/CR-6674 for Carbon and Low-Alloy Steel Components (MRP-74), EPRI, Palo Alto, CA and U.S. Department of Energy, Washington, D.C. 1003667."

In Table 4.3-2 (TLAA Evaluation), the text should be changed, as shown:

Table 4.3-2. Example of FSAR Supplement for Metal Fatigue TLAA Evaluation

10 CFR 54.21(c)(1)(iii) Example

TLAA	Description of Evaluation	Implementation Schedule*
Metal fatigue	The aging management program monitors and tracks the number of critical thermal and pressure test transients, and monitors the cycles for the selected reactor coolant system components ***For austenitic stainless steel and Ni-Fe-Cr high nickel alloy components,*** ~~T~~*t*he aging management program will address the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components ***locations*** that include, as a minimum, those ***austenitic stainless steel and Ni-Fe-Cr high nickel alloy*** components ***locations*** selected in NUREG/CR-6260. The sample of critical components can be evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses. Formulas for calculating the environmental life correction factors are contained in ~~NUREG/CR-6583 for carbon and low-alloy steels and~~ in NUREG/CR-5704 for austenitic stainless steels and Ni-Fe-Cr high nickel alloys.	Evaluation should be completed before the period of extended operation

* An applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

NUREG-1801 Recommended Changes

In addition to the changes recommended for NUREG-1800, recommendations for changes to Chapter X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary of NUREG-1801 (Generic Aging Lessons Learned Report) are provided below. The changes to the 'Program Description' portion of X.M1 are indicated by inserted bold face italics or deletion marks.

"In order not to exceed the design limit on fatigue usage, the aging management program (AMP) monitors and tracks the number of critical thermal and pressure transients for the selected reactor coolant system components.

The AMP addresses the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical ***austenitic stainless steel and Ni-Fe-Cr high nickel alloy*** components ***locations*** that includes, as a minimum, those ***austenitic stainless steel and Ni-Fe-Cr high nickel alloy*** components ***locations*** selected in NUREG/CR-6260. The sample of critical components can be evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses. Formulas for calculating the environmental life correction factors are contained ~~in NUREG/CR-6583 for carbon and low-alloy steels and~~ in NUREG/CR-5704 for austenitic stainless steels ***and Ni-Fe-Cr high nickel alloys.***

As evaluated below, this is an acceptable option for managing metal fatigue for ***austenitic stainless steel and Ni-Fe-Cr high nickel alloy component locations in*** the reactor coolant pressure boundary, considering environmental effects. Thus, no further evaluation is recommended for license renewal if the applicant selects this option under 10 CFR 54.21(c)(1)(iii) to evaluate metal fatigue for ***austenitic stainless steel and Ni-Fe-Cr high nickel alloy component locations in*** the reactor coolant pressure boundary."

In addition to the above text changes, a fourth reference should be added to the reference list, a reference to MRP-74. The complete reference is:

Materials Reliability Program: Re-Evaluation of Results in NUREG/CR-6674 for Carbon and Low-Alloy Steel Components (MRP-74), EPRI, Palo Alto, CA and U.S. Department of Energy, Washington, D.C. 1003667.

ATTACHMENT 1

INTERIM STAFF GUIDANCE-13
ENVIRONMENTAL ASSISTED FATIGUE FOR CARBON/LOW-ALLOY STEEL

TECHNICAL BASIS DOCUMENT

1.0 Introduction

This document establishes the technical basis to remove the requirements on license renewal applicants to incorporate reactor water environmental effects into fatigue evaluations of carbon and low-alloy steel components performed for the purpose of demonstrating the adequacy of aging management programs for license renewal. The information provided in this document and in the referenced material applies only to carbon and low-alloy components with one surface in contact with primary coolant. Information regarding component locations fabricated from stainless steel and high-nickel alloy materials will be supplied at a later date.

The technical basis provided in this document is organized in a logical sequence, beginning with background information (Section 2.0) on the various issues related to fatigue of metal components at U. S. nuclear power plants, leading to eventual closure of Generic Safety Issue 190 in December 1999. The remaining document sections are described as follows:

- Section 3.0 Closure of Generic Safety Issue 190. This section discusses GSI-190 closure resolution, which concluded that no safety issue was remaining but placed explicit environmental fatigue requirements on license renewal applicants because of probabilistic estimates of through-wall cracking and associated leakage from NUREG/CR-6674.
- Section 4.0 Industry/EPRI Materials Reliability Program Efforts. This section describes the overall activity underway in the EPRI Materials Reliability Program Fatigue Issue Task Group to address environmental fatigue issues.
- Section 5.0 Re-Evaluation of NUREG/CR-6674 Results. This section summarizes an MRP effort to re-calculate probabilistic estimates for through-wall cracking and associated leakage, and to re-calculate core damage frequencies, for carbon and low-alloy steel component locations from NUREG/CR-6260 and NUREG/CR-6674. Based on realistic assumptions, results show that through-wall cracking and associated leakage probabilities are, in fact, insignificant for both 40 and 60 years of operation, in agreement with industry operating experience. The re-evaluation also shows significant reductions in core damage frequency.
- Section 6.0 Laboratory Data Evaluation. This section discusses the confirmation of carbon and low-alloy steel component resistance to fatigue crack initiation and growth, including reactor water environment effects, through the critical review of laboratory fatigue data under simulated reactor water environmental conditions. Included is the comparison of the simulation conditions for

temperature, strain amplitude, surface strain amplitude distribution, strain rate, dissolved oxygen and associated oxidation potential, and – most importantly – coolant flow rate with actual plant operating conditions.

- Section 7.0 Structure/Component Fatigue Tests. This section provides demonstrable confirmation of carbon and low-alloy steel component resistance to fatigue crack initiation and growth through structural/component fatigue test results. These fatigue tests incorporate size and surface finish considerations, with one surface of the structure or component in contact with oxygenated water in either stagnant or flowing conditions that realistically simulate actual plant operating conditions.
- Section 8.0 Summary
- Section 9.0 References

The results of this comprehensive program demonstrate that no requirements for explicit consideration of reactor water environmental effects should be placed on license renewal applicants relative to evaluation of carbon and low-alloy steel component fatigue crack initiation and growth as an aging effect to be managed during the renewal term.

2.0 Background

One of the most significant technical issues that potentially affect the ability to renew the operating licenses of commercial nuclear power plants in the United States is fatigue of metal components. Two aspects of this issue have received considerable attention in recent years – the observed effects of transient thermal loading not anticipated during the component design process and the potential influence of the reactor water environment on fatigue crack initiation and growth. The first of these became a concern about twenty years ago, as the result of stratified flow conditions in feedwater piping that caused premature crack initiation and growth, and was documented by the U. S. Nuclear Regulatory Commission (NRC) in Inspection and Enforcement (IE) Bulletin 79-13 [1]. Unanticipated thermal transients were identified later as a concern also for reactor coolant system and primary coolant pressure boundary piping and components, as documented in NRC Bulletins 88-08 [2] and 88-11 [3], and NRC Information Notices 91-38 [4] and 93-20 [5]. This concern was addressed through Generic Issue No. 78 [6].

The concerns about the influence of the reactor water environment are more recent, but first indications extend back more than twenty years. Laboratory and component-scale fatigue crack initiation data under simulated water reactor environmental conditions have been obtained over the past two decades that indicate a significant reduction in cyclic life when compared to fatigue crack initiation data obtained in air environments. An early report on these effects was published by the General Electric Company in 1982 [7], based on carbon steel piping component tests in high-temperature (550°F [288°C]), high dissolved oxygen (8 ppm) and nominal BWR (0.2 ppm dissolved oxygen) environments. The greatest effects were observed at high-amplitude, low-cyclic frequency (i.e., low strain rate) loading at temperature, in particular at loads causing stresses in the plastic range. It was found that an environmental correction factor, K_{en}, to be applied to the stress range, was needed to restore ASME Code fatigue design margins under the worst-case conditions. This factor was not needed when the water temperature was less than 400°F (204°C), nor was the factor needed when the cyclic frequency was relatively rapid, greater than or equal to 0.1 Hz. K_{en} was found to depend on strain amplitude and dissolved oxygen, with a value of 1.0 for small plastic strains. For large plastic strains, K_{en} was found to have a maximum of about 3.4 for 8 ppm dissolved oxygen and about 2.4 for 0.2 ppm dissolved oxygen.

Approximately a decade later, Japanese investigators published a set of fatigue crack initiation data for carbon, low-alloy, and austenitic stainless steels [8]. These data were then presented to ASME Code bodies and to staff of the NRC, leading to concerns about the structural integrity of both existing light-water reactor (LWR) components and potential new construction. The data set included the carbon steel data obtained previously by the General Electric Company, but also included data for low-alloy and austenitic stainless steels showing somewhat lesser but still significant reductions in fatigue life. During the subsequent discussions between the industry and the NRC, in particular within the context of nuclear plant license renewal, the industry concluded that:

- The carbon steel data were well known.
- A procedure for addressing severe BWR reactor water environmental effects was available in the form of the K_{en} stress concentration factor; K_{en} is a maximum of 2.4 for nominal BWR conditions and even less for nominal PWR conditions, well within available ASME Code margins.
- Therefore, K_{en} needed to be applied only under high-strain-amplitude conditions at temperature, with saturated dissolved oxygen, under slow, cyclic loading conditions, a combination of conditions that is rarely encountered in actual operation.
- The reduction in fatigue life for low-alloy and austenitic stainless steels could be accommodated by the recognition that a fraction of the factor of 20 at the low-cycle end of the ASME Code Section III fatigue design curve accounts for some of the environmental effects.

This latter conclusion was based on the statement in the ASME Code Background Document [9] regarding the factor of 20 at the high-strain-amplitude, low-cycle end of the ASME Code fatigue design curve that:

> ***"These factors were intended to cover such effects as environment, size effect, and scatter of data, and thus it is not to be expected that a vessel will actually operate safely for twenty times its specified life."***

Furthermore, the industry believed that the new laboratory data were not supported by actual nuclear power plant component operating experience.

Nevertheless, the NRC staff prepared and implemented a Fatigue Action Plan in 1993 to address technical and regulatory compliance concerns for both the current operating term and for potential extension of the current operating license. Subsequent confirmatory research carried out by the NRC staff and contractors led to the closure of Generic Issue No. 78, with a finding in SECY-95-245 [10] that "the [NRC] staff believe that no immediate staff or licensee action is necessary to deal with the fatigue issues addressed by the [Fatigue Action Plan]." Further, SECY-95-245 found that "fatigue failure of piping is not a significant contributor to core-melt frequency" and "the [NRC] staff does not believe it can justify requiring a backfit of the environmental fatigue data to operating plants." However, with respect to license renewal, SECY-95-245 found that "the [NRC] staff believe that the [Fatigue Action Plan] issues should be evaluated for any proposed extended period of operation for license renewal."

As a result of the completion of the Fatigue Action Plan, the NRC staff technical and regulatory compliance concerns with respect to fatigue for license renewal were subsumed into Generic Safety Issue No. 166 (GSI-166), "Adequacy of Fatigue Life of Metal Components" [11]. Later, this issue was renumbered as GSI-190 [12]. SECY-95-245 provided some guidance with respect to the need to demonstrate that the effects of fatigue will be managed during the license renewal term by stating that "The staff will consider, as part of the resolution of GSI-166,......., the need to evaluate a sample of components with high fatigue usage, using the latest available environmental fatigue data."

As a way of addressing the need for sample locations, Argonne National Laboratory (ANL) prepared a set of modified ASME Code Section III fatigue design curves that were based upon the continuous influence of reactor water environmental effects over the entire life of the component. These curves were published in NUREG/CR-5999 [13]. Idaho National Engineering Laboratory (INEL), now Idaho National Engineering and Environmental Laboratory (INEEL) applied these curves to the evaluation of fatigue-sensitive component locations in all light-water-cooled reactor classes. The work was published in NUREG/CR-6260 [14]. It should be emphasized that the reduced fatigue design curves from Reference 13 were applied in Reference 14, without consideration of thresholds on temperature, strain rate, strain amplitude, etc. Later, Pacific Northwest National Laboratories (PNNL) determined the effects of reactor water environment-shortened fatigue lives on core damage frequency and evaluated the shortened fatigue life on the probabilities of crack initiation and through wall cracking for the extended operating life. This work was published in NUREG/CR-6674 [15].

3.0 Closure of Generic Safety Issue 190

A December 26, 1999, memorandum [16] from Ashok C. Thadani, Director of the NRC Office of Nuclear Regulatory Research (RES), to William D. Travers, NRC Executive Director of Operations, provided the instrument for formal closure of Generic Safety Issue 190. That memorandum stated, in part:

> *The conclusion to close out this issue is based upon the low core damage frequencies from fatigue failures estimated by technical studies making use of recent fatigue data developed on test specimens. The results of these probabilistic analyses and associated sensitivity studies led the staff to conclude that no generic regulatory action is required.*

The probabilistic analyses and associated studies referred to in this statement were published in NUREG/CR-6674 [15].

However, the memorandum went on to state:

> *However, calculations including environmental effects, that were performed to support resolution of this issue, and the nature of age-related degradation indicate the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, the staff concludes that, consistent with existing requirements in 10 CFR 54.21, licensees should address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.*

The requirement for license renewal applicants to address the effects of the coolant environment on fatigue life of metal components, as an element of fatigue aging management programs, is apparently not related to operation for 60 years, as opposed to operation for 40 years during the initial license term. This is evident from the next paragraph in the Thadani memorandum, which states:

> *The advanced light water reactors (ALWRs) that have been certified under 10 CFR Part 52 were designed for a 60-year life expectancy. The associated fatigue analyses accounted for the design cycles based on a 60-year plant life but did not account for the environmental effects as addressed in GSI-190. However, the staff has concluded that there is sufficient conservatism in the fatigue analyses performed for the generic 60-year ALWR plant life to account for environmental effects.*

Therefore, even though no safety issue was identified by the staff related to reactor water effects on metal component fatigue life, and although the existing ASME Code explicit fatigue design rules were deemed to be adequate for 60 years of design life, new requirements were imposed on license renewal applicants. The perceived reason for these new requirements was based on two considerations: (1) potential increases in through-wall leakage caused by fatigue crack initiation and growth, as accelerated by reactor water environmental effects; and (2) the implied requirement in 10 CFR 54.21 to manage potential aging effects, such as fatigue-related through-wall cracking and associated leakage, during the license renewal term.

4.0 Industry/EPRI Materials Reliability Program Efforts

The U. S. nuclear power industry responded to the imposed requirements on license renewal applicants through the EPRI Materials Reliability Program (MRP). Environmental fatigue was incorporated into the scope of the MRP Fatigue Issue Task Group (ITG) during mid-2000, with the first meeting of the ITG that included the expanded scope held in early September of 2000.

The MRP Fatigue ITG activities were divided into two principal areas. The first objective was to provide near-term guidance to future license renewal applicants on how to address environmental fatigue effects in a license renewal application. Prior to the Fall of 2000, the license renewal applications already approved by the NRC each addressed environmental fatigue in a slightly different manner. The near-term objective was pursued to provide guidance for consideration of reactor water environmental effects and minimize the amount of plant-specific work necessary to comply with NRC requirements for addressing this issue in a license renewal application. This was performed with no judgment as to the necessity of considering reactor water effects.

The second objective of the Fatigue ITG was to perform longer-term efforts to directly address the technical issues associated with environmental fatigue and to determine the necessity of considering reactor water effects. It was anticipated that the results of this objective would likely dictate a revision to the near-term guidance developed.

The first immediate concern of the EPRI MRP ITG on Fatigue was the guidance needed for near-term license renewal applicants. For this reason, an activity was initiated on a guidance document [17] completed in draft form in December 2000, and eventually submitted to the NRC staff for formal review in June 2001. During this activity the Fatigue ITG also developed the longer-term set of program activities to directly address the overall technical issue of environmental fatigue. These activities were initiated and are summarized below.

One such activity was the evaluation of laboratory fatigue test data in simulated reactor water environments, and the comparison of those data with structural/component fatigue test results and with actual plant operating experience. This activity was completed in parallel with and, to some extent, in conjunction with a related effort underway by the Pressure Vessel Research Council (PVRC) under the aegis of the ASME Board on Nuclear Codes and Standards (BNCS). The final report on this EPRI project was published in December 2001 [18]. The related PVRC report was to be published in late 2002. Reference 18 has been provided to the NRC staff, and the information has been presented at PVRC and ASME Code meetings in recent months.

Also of high priority was the evaluation of results contained in NUREG/CR-6674. The industry, through the Nuclear Energy Institute (NEI) License Renewal Working Group (LRWG), had alerted the NRC staff in early 1999 that, while the bounding approach used in NUREG/CR-6674 was sufficiently robust to justify estimates of core damage frequency, such an approach was inherently too conservative to provide reasonable and useful estimates of through-wall cracking and potential leakage. Realizing the significance of the very conservative estimates provided in NUREG/CR-6674 to the NRC staff in their GSI-190 deliberations, the industry committed to the recalculation of these estimates, using more realistic assumptions. Much of this work has now been completed under the auspices of the EPRI MRP Fatigue ITG, and a report on the results for carbon and low-alloy steel locations from NUREG/CR-6260 and NUREG/CR-6674 has been published (hereafter referred to as MRP-74) [19]. The major highlights of these recalculated estimates are covered in the next section.

5.0 Re-Evaluation of NUREG/CR-6674 Results

Fifty-eight (58) component locations for seven different types of light-water-cooled reactor designs were selected and analyzed in NUREG/CR-6260 [14], including design fatigue curves reduced by environmental effects. The 58 component locations were chosen as being representative of high design-basis fatigue usage locations with one component surface in contact with the reactor water environment. Twenty-seven (27) of the locations are carbon or low-alloy steel, and thirty-one (31) are austenitic stainless steel or Ni-Fe-Cr high-nickel alloy. Of the 58 component locations, eighteen (18) were found to have a cumulative fatigue usage factor including explicit reactor water environmental effects, greater than 1.0 for either 40 or 60 years (or both) of operation. The 47 component locations analyzed in NUREG/CR-6674 [15] were identical to 47 of the 58 component locations evaluated in NUREG/CR-6260. The stresses and loading conditions were taken directly and extrapolated from the information contained in NUREG/CR-6260. The eleven locations that were analyzed in NUREG/CR-6260, but not analyzed in NUREG/CR-6674, are perceived to have an insignificant contribution to risk.

Several of the 47 component locations evaluated in NUREG/CR-6674 were found to have a relatively high fatigue crack initiation probability and a through-wall cracking (leakage) probability exceeding 0.1 at 40 years. For example, the results for one stainless steel component showed that there was a 50 percent probability for fatigue crack initiation after only approximately ten years of operation, with a significant probability of through-wall cracking (leakage) after about 15 years of operation. These predictions are contrary to industry experience, and are an indication that the analyses used very conservative assumptions.

The most critical of the assumptions in NUREG/CR-6674 is related to the probabilistic representation of the uncertainty in the endurance limit end of the fatigue design curves. In addition, a bounding high temperature of 590^{o}F was used in NUREG/CR-6674. Assumed through-wall stress distributions were also used in NUREG/CR-6674. The evaluation documented in MRP-74 used more realistic alternatives for these assumptions, and also updated the probabilistic calculations to incorporate the most recent laboratory fatigue data [20,21,22].

The objective of the EPRI MRP re-evaluation was to determine if the probability of fatigue crack initiation and growth of cracks to produce leaks would be substantially reduced by the use of less conservative, yet realistic assumptions. In order to assess achievement of the objective, the cumulative probabilities of through-wall cracking (and associated leakage) in 60 years were compared between the NUREG/CR-6674 calculations and the EPRI MRP re-calculations. In particular, the comparative measure of component failure was chosen to be 0.001; i.e., one chance in 1000 that a fatigue crack would initiate and propagate completely across the component wall thickness in 60 years of operation. Stated another way, if a plant has five component locations with a cumulative through-wall cracking probability of 0.001, and if 100 plants are operating with these same components under these same conditions for the same operating period, any one plant out of the total 100-plant population will have less than a 50 % probability of a through-wall crack in any one of the five component locations.

If this measure is used to examine the NUREG/CR-6674 results, 24 of the 47 component locations are found to have a cumulative probability of through-wall cracking (and leakage) greater than 0.001 for 60 years of operation. The other 23 component locations have a cumulative probability of through-wall cracking and leakage less than 0.001 for 60 years of operation, implying an insignificant potential for leakage. Ten of the 24 locations are either carbon or low-alloy steel and 14 are austenitic stainless steel. Based upon the conservative analyses performed in NUREG/CR-6674, it was plausible to recommend environmental fatigue evaluation as a condition for extension of plant operation from 40 to 60 years.

In the initial portion of the EPRI MRP study, only the carbon and low alloy steel component locations in the NUREG/CR-6674 report have been re-analyzed. During this re-analysis, it was found that the probabilistic representation of the fatigue curve endurance limit used in NUREG/CR-6674 was unduly conservative, so an alternate, more realistic value was derived. Figures 1 and 2 illustrate the changes made to provide a realistic probabilistic basis for the endurance limit. Figure 1 shows the probabilistic fatigue curve for low-alloy steel used in NUREG/CR-6674, while Figure 2 shows the alternative (and more physically meaningful) fatigue design curve used in MRP-74. Figures 3 and 4 illustrate the same effect for carbon steel. This single assumption change caused the estimate of crack initiation and through-wall cracking (and leakage) to be reduced by an order of magnitude.

An additional modification to the analysis was to incorporate updated strain-life (fatigue) curve recommendations for carbon and low-alloy steel that became available after publication of NUREG/CR-6674, that is, NUREG/CR-6717 [22]. As a result of these two modifications alone, one location – the RCIC tee in the feedwater line of an older vintage BWR – showed a very slight increase in cumulative probability of leakage at 60 years, while all other locations showed a reduction up to several orders of magnitude.

In addition to the above two modifications (revised endurance limit uncertainty and updated fatigue curves) more realistic loading conditions were derived for two components where details from original stress reports were available and contained sufficient information. For BWR feedwater line components more accurate transient temperature information from NUREG/CR-6260 was also considered. For one BWR component, a feedwater tee that had been evaluated in NUREG/CR-6260, more realistic strain rates for the significant transients were derived.

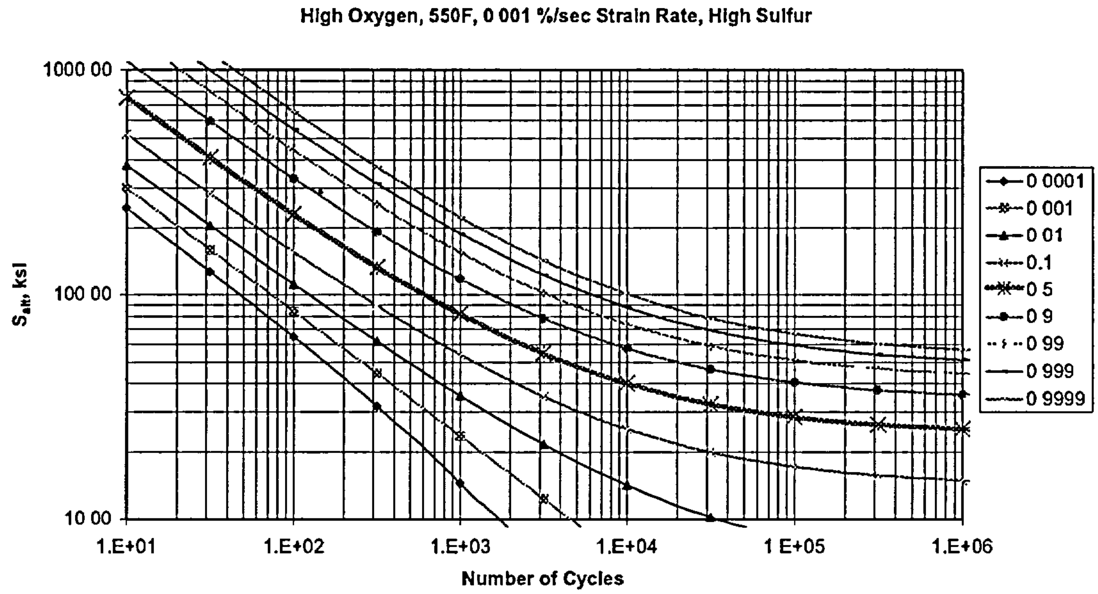

Figure 1. Probabilistic fatigue curve for low-alloy steel used in NUREG/CR-6674 for various quantiles.

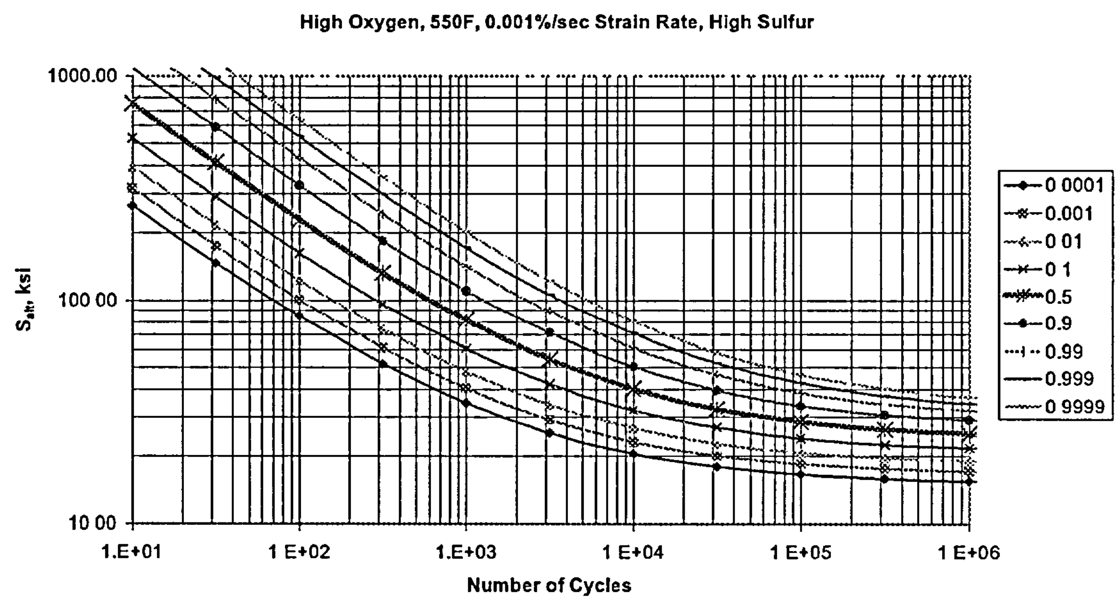

Figure 2. Alternate probabilistic fatigue curve for low-alloy steel used in MRP re-analysis of NUREG/CR-6674 for various quantiles.

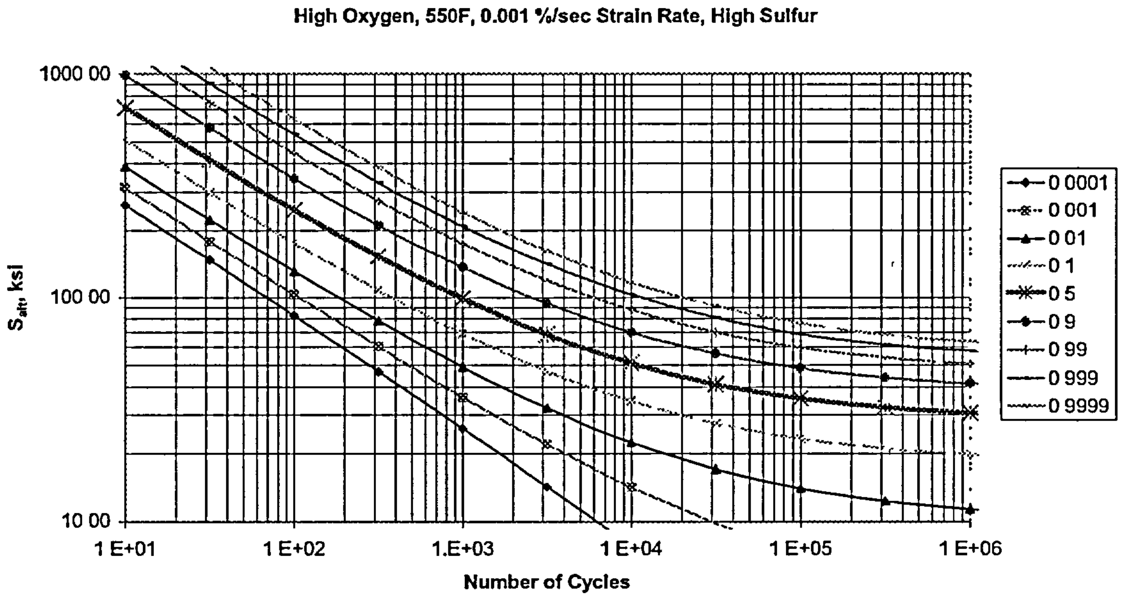

Figure 3. Probabilistic fatigue curve for carbon steel used in NUREG/CR-6674 for various quantiles.

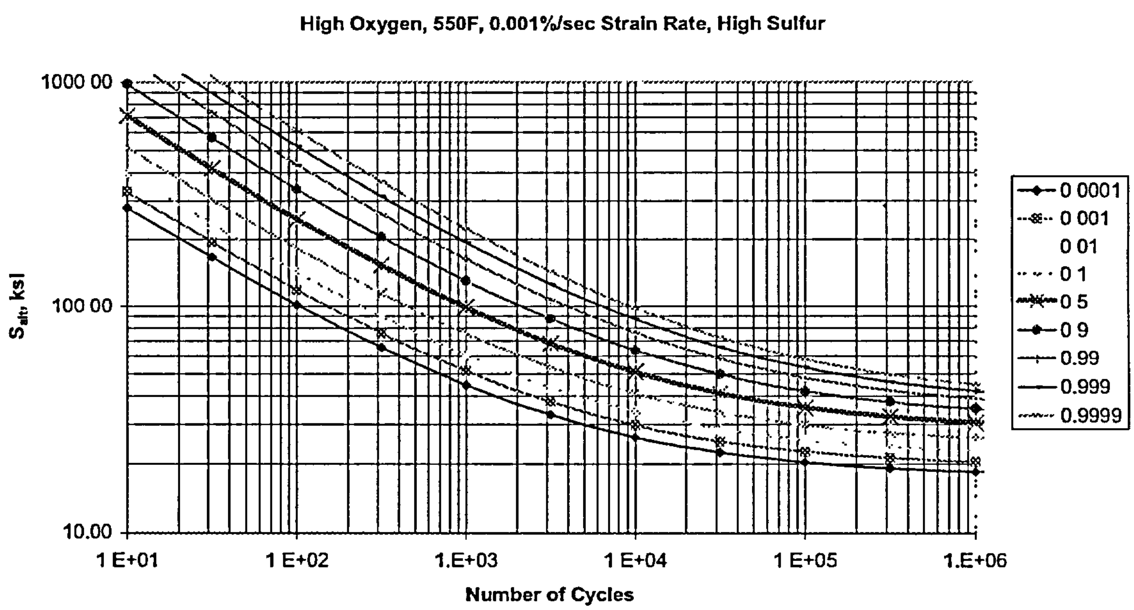

Figure 4. Alternate probabilistic fatigue curve for carbon steel used in MRP re-analysis of NUREG/CR-6674 for various quantiles.

Figures 5 and 6 illustrate the principal results from this study. Figure 5 compares the cumulative probability of initiation at 60 years from NUREG/CR-6674 and MRP-74. A decrease in cumulative probability of initiation of up to approximately 6 orders of magnitude is evident from the EPRI MRP re-analysis. Figure 6 compares the cumulative probability of leakage at 60 years from NUREG/CR-6674 and MRP-74. The reduction in predicted leakage is even more pronounced when realistic assumptions are considered. These results indicate that the probability of leakage at 60 years, when environmental effects are considered, is insignificant and no explicit consideration of reactor water environment is necessary.

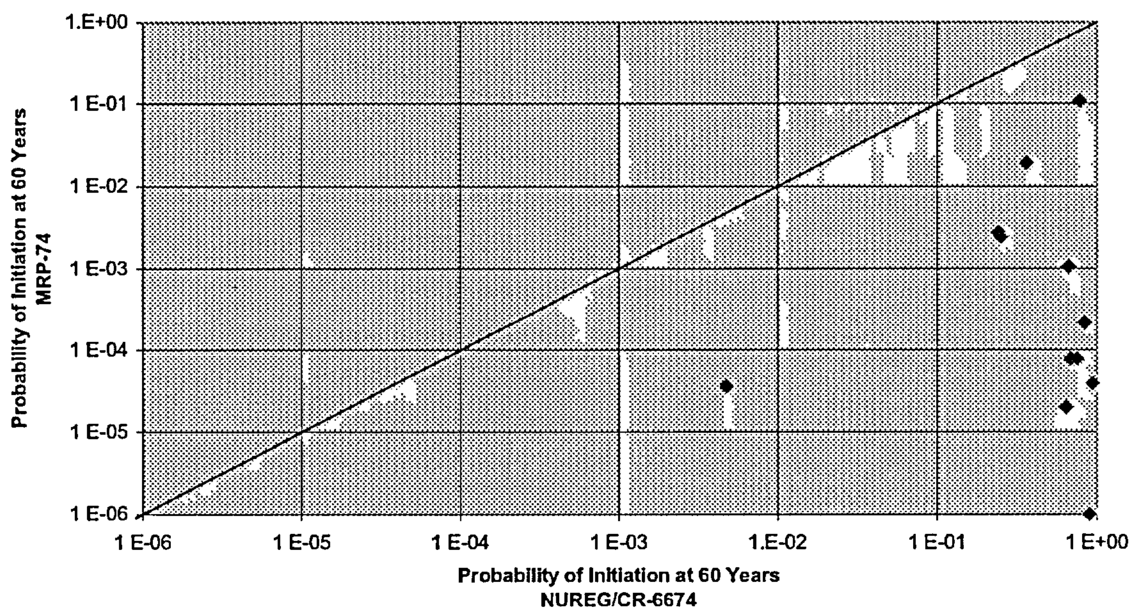

Figure 5. Comparison of cumulative probability of initiation at 60 years from NUREG/CR-6674 and MRP-74

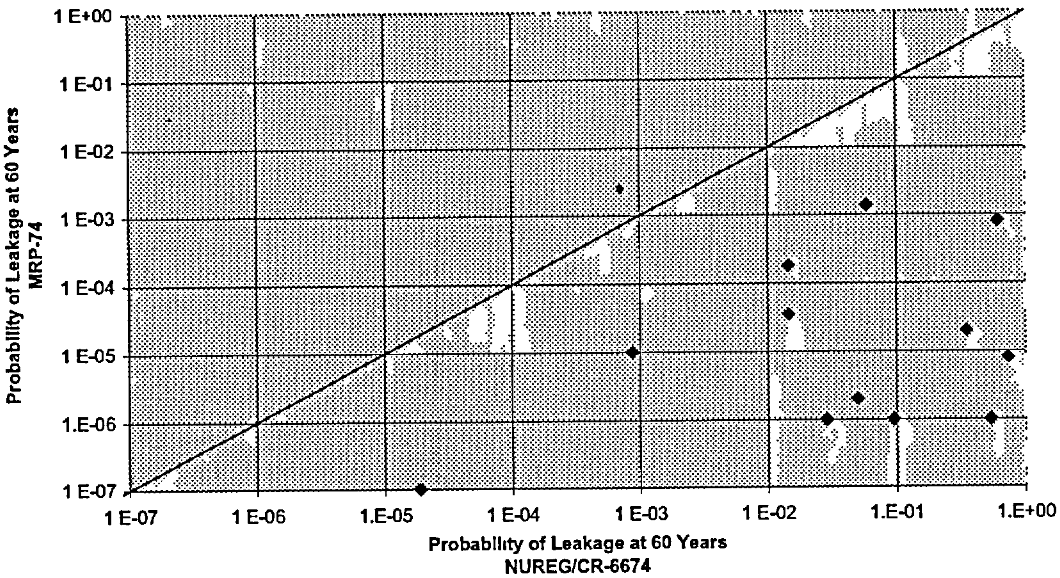

Figure 6. Comparison of cumulative probability of leakage at 60 years from NUREG/CR-6674 and MRP-74

This conclusion is further illustrated in Figures 7 and 8. Figure 7 provides a comparison between the cumulative probabilities of initiation at both 40 and 60 years reported in NUREG/CR-6674 and MRP-74. It is evident that the consideration of more realistic assumptions in the analysis reduces the cumulative probability of initiation, in most cases significantly. Figure 8 provides the same comparison for cumulative probabilities of leakage. The reduction in cumulative probability of leakage is more pronounced when realistic assumptions are considered.

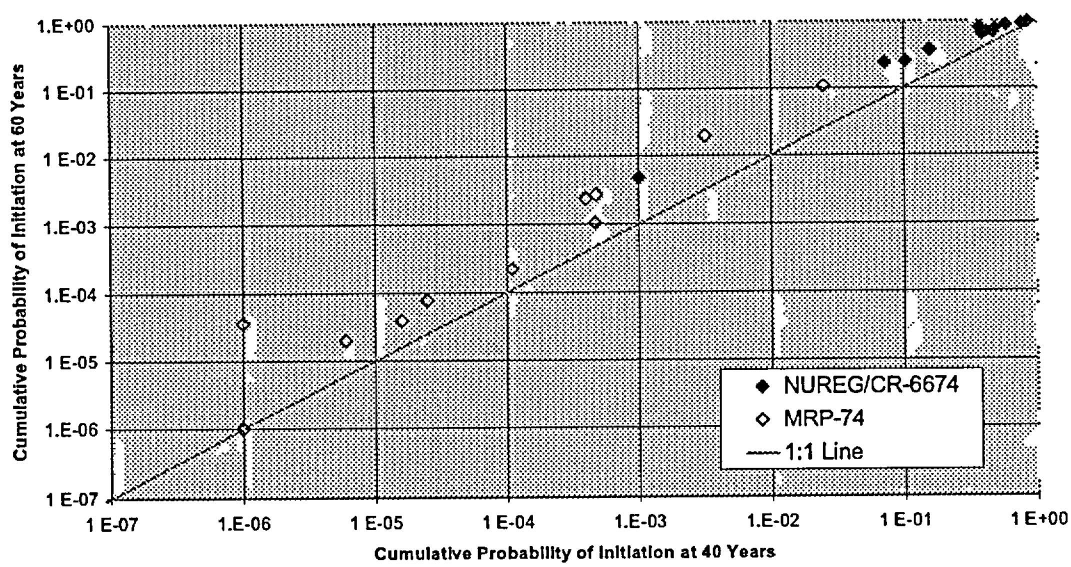

Figure 7. Cumulative probability of initiation at 60 years versus 40 years

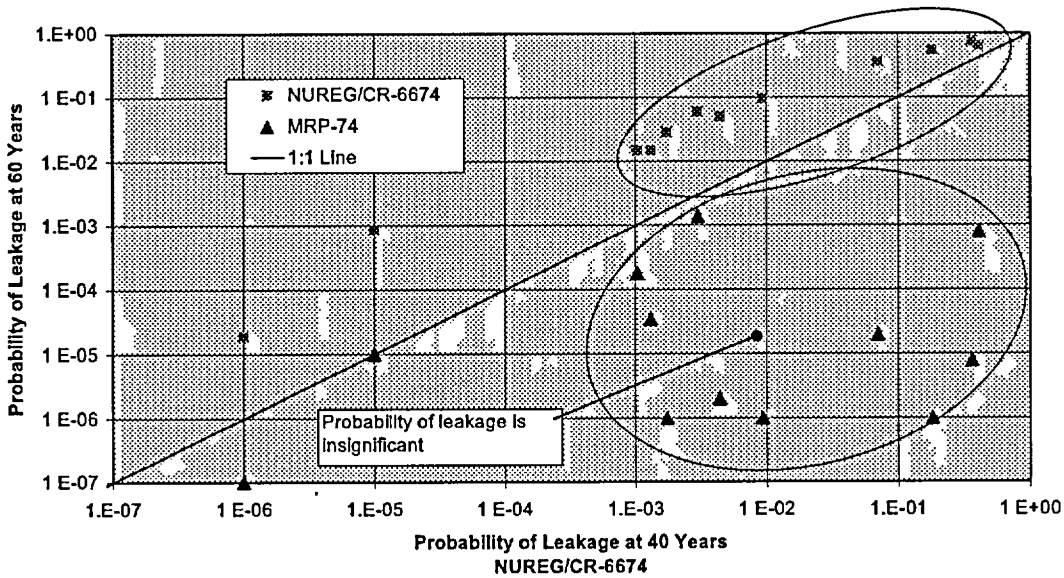

Figure 8. Cumulative probability of leakage at 60 years versus 40 years

Figure 8 compares the probability of leakage for 60 years, as determined in MRP-74 and the original NUREG/CR-6674 analysis, to the probability of leakage for 40 years that was determined in the original NUREG/CR-6674 study. An increase in the predicted leakage from 40 to 60 years is evident from the original PNNL analysis (note the square symbols lie above the 1:1 line). However, this increase is not due solely to the conservative consideration of environmental fatigue. Even without this consideration, an increase would be expected since fatigue is a time-related aging mechanism. (In the original NUREG/CR-6674 analysis, the fatigue usage factor calculated for 40 years was multiplied by 1.5 to derive the predicted 60-year usage factor. This value was then used in the cumulative probability calculations). The use of more realistic assumptions in MRP-74 clearly demonstrates that the anticipated leakage at 60 years is less, in many cases by several orders of magnitude, than the leakage predicted to occur after 40 years in the NUREG/CR-6674 analysis. This is shown by the triangle symbols in Figure 8 all lying below the 1:1 line, and in many cases significantly below the 1:1 line. These results suggest that the present 40-year design basis is maintained and no additional treatment of environmental fatigue should be required.

Figures 7 and 8 also show an increase in probabilities of initiation and leakage during the license renewal period. The increased probabilities are expected since fatigue is an age-related degradation mechanism. The increase is due to the combination of added cyclic life during the license renewal period and the conservative nature in which reactor water effects were considered. However, a critical point to be considered when determining if additional aging management actions are necessary is whether the predicted increase is at a level that would be considered significant. Figure 9 compares the 60-year initiation probabilities for carbon/low-alloy steel components in environment from MRP-74 with results from the NUREG/CR-6674 study in air. In all cases, the re-analysis indicates that consideration of reactor water environment results in initiation probabilities at 60 years that are essentially equal to or lower than predicted in the NUREG/CR-6674 study in air.

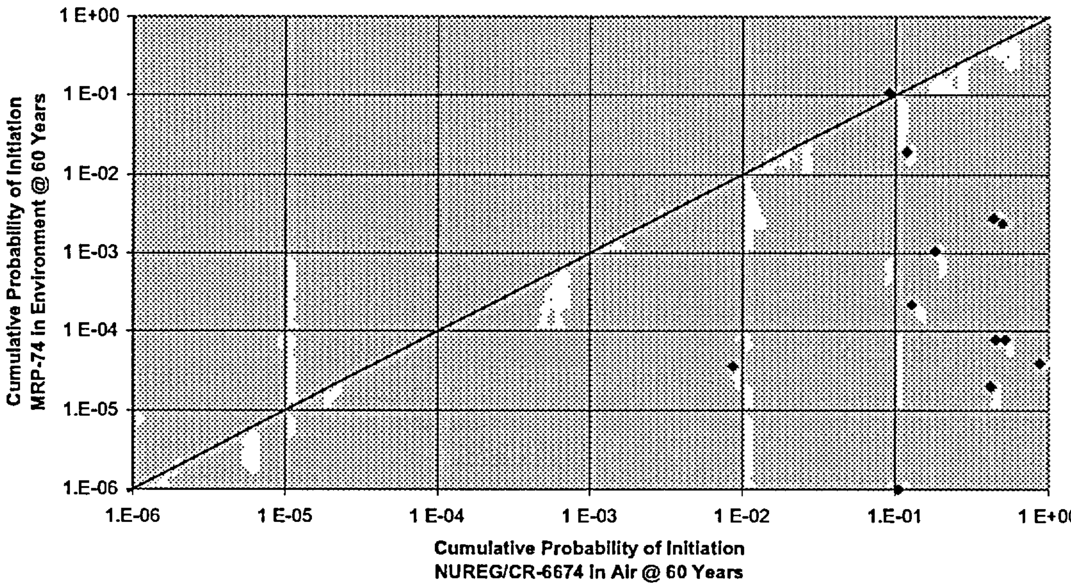

Figure 9. Comparison of 60-year cumulative probabilities of initiation for MRP-74 in environment and NUREG/CR-6674 in air.

Figure 10 provides a similar comparison for leakage probabilities. The re-analysis indicates that consideration of reactor water environment results in leakage probabilities at 60 years that are either lower than predicted in the NUREG/CR-6674 study in air or are at a sufficiently low probability level to be deemed insignificant.

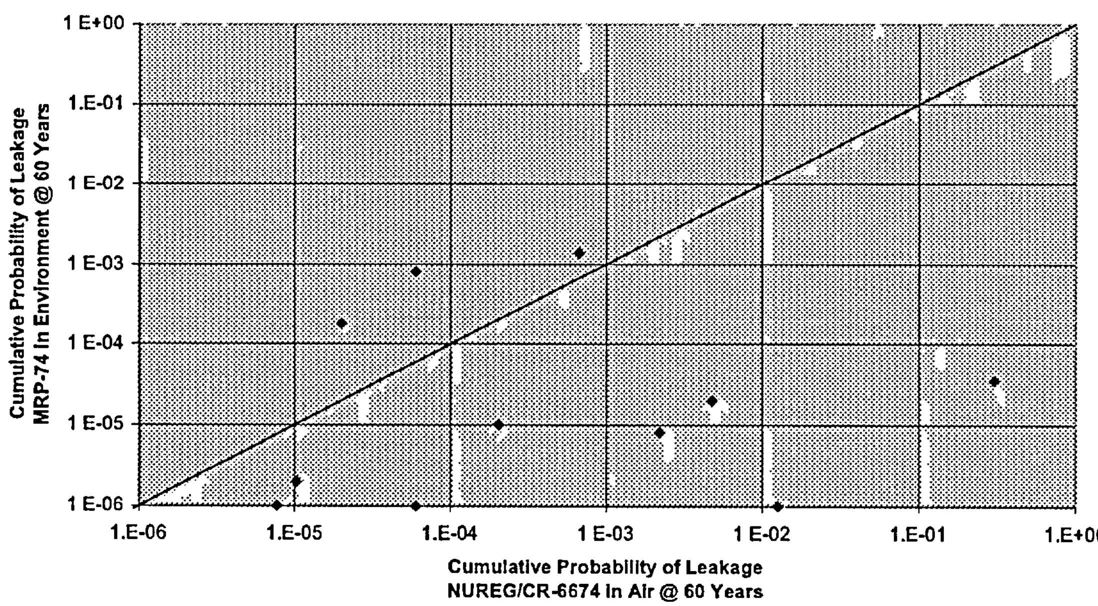

Figure 10. Comparison of 60-year cumulative probabilities of leakage for MRP -74 in environment and NUREG/CR-6674 study in air.

Table 1 provides the calculated results for cumulative probability of leakage from NUREG/CR-6674 and MRP-74. Only one location, the RCIC tee in the feedwater line of an older vintage BWR, has a 60-year cumulative through-wall cracking probability slightly above the threshold value of 0.001 (0.00139, see Table 1). However, since the environmental penalties were applied for all transients at all times, and were based on saturated strain rates, the RHR piping location will have a 60-year through-wall cracking (and leakage) probability well below the threshold for more realistic assumptions.

Table 1. Cumulative probability of leakage predictions for all locations.

Component	40 Year Life		60 Year Life	
	NUREG/CR-6674	MRP-74	NUREG/CR-6674	MRP-74
B&W RPV OUTLET NOZZLE	1.83E-01	< 1 00E-06	5 44E-01	< 1.00E-06
CE-NEW RPV OUTLET NOZZLE	1.74E-03	< 1 00E-06	2.90E-03	< 1 00E-06
CE-NEW SAFETY INJECTION NOZZLE	1.00E-06	< 1.00E-06	1.90E-05	1.00E-07
CE-OLD RPV OUTLET NOZZLE	7.05E-02	< 1.00E-06	3 53E-01	2.00E-05[1]
GE-NEW FEEDWATER NOZZLE SAFE END	1.31E-03	1.00E-06	1.47E-02	3 50E-05
GE-NEW RHR LINE STRAIGHT PIPE	4.10E-01	3.00E-04	6 21E-01	8.00E-04
GE-NEW FEEDWATER LINE ELBOW	1 03E-03	2 00E-06	1.46E-02	1.80E-04
GE-OLD RPV FEEDWATER NOZZLE BORE	1.00E-05	< 1 00E-06	8 80E-04	1.00E-05
GE-OLD FEEDWATER LINE – RCIC TEE	2.99E-03	6.00E-05	5.92E-02	1.39E-03
W-NEW RPV OUTLET NOZZLE	3 65E-01	1 00E-06	7.42E-01	8 00E-06
W-OLD RPV INLET NOZZLE	4 38E-03	< 1 00E-06	5 04E-02	2.00E-06
W-OLD RPV OUTLET NOZZLE	9 33E-03	< 1.00E-06	9 60E-02	1 00E-06

[1]Consideration of actual transient cycles further reduced the probability to < 1 00E-06

The RCIC tee in the feedwater line of an older vintage BWR is the only remaining carbon or low-alloy steel component location with a 60-year cumulative through-wall cracking probability arguably greater than 0.001. Since the stress analysis upon which the probabilistic calculations are based is a piping stress analysis (NB-3600), NUREG/CR-6260 observed that an NB-3200 stress analysis would have reduced the fatigue usage factor considerably. This reduction in stress also reduces the probability of through-wall cracking in 60 years. A very simplified approach for approximating the effects of the NB-3200 stress analysis is to estimate the thermal stress differences more accurately. Such estimations for one RCIC injection transient pair reduced the 60-year through-wall cracking (and leakage) probability from 0.219 to 0.00139. Clearly, the probability of through-wall cracking at this location would be reduced to insignificant levels through a complete NB-3200 stress analysis.

A re-evaluation of core damage frequencies (CDF) that provide a measure of risk contributed by failure of the component was also performed and reported in MRP-74. The methodology used was that reported in NUREG/CR-6674 and considered failure probability (derived from cumulative leakage probability results) to estimate the CDF. Using the revised leakage probabilities calculated in the MRP re-evaluation, the CDF values reported in NUREG/CR-6674 were significantly reduced. In NUREG/CR-6674 the maximum 60-year CDF reported was 1.22×10^{-7}. In MRP-74 the maximum 60-year CDF reported was 7.5×10^{-11}, representing over four orders of magnitude reduction in the maximum estimated CDF.

While the results from NUREG/CR-6674 could have been interpreted to require explicit consideration of reactor water environmental effects in fatigue aging management programs, the re-calculated results do not support such an interpretation. The 60-year cumulative probabilities of through-wall cracking (and leakage) are too low to justify such considerations for carbon and low-alloy steel component locations. These re-calculated results also are supported by plant operating experience.

The re-evaluation performed and documented in MRP-74 demonstrates that the use of more realistic assumptions results in 60-year cumulative probabilities of through-wall cracking (and leakage) that are significantly below the level previously found acceptable for a 40-year period of operation in NUREG/CR-6674. Additionally, a significant reduction in CDF was calculated beyond the already low values reported in NUREG/CR-6674.

Fatigue is a time related degradation mechanism that will require aging management during license renewal. Results of this study indicate that explicit consideration of reactor water effects is not necessary for carbon and low-alloy steel location aging management programs that are formulated for license renewal. Present aging management programs, including transient tracking and cycle counting, are sufficient to manage fatigue for these components.

6.0 Laboratory Data Evaluation

In addition to the potential increased leakage inferred from NUREG/CR-6674 results, the license renewal requirements imposed by the closure of GSI-190 are based, in part, on laboratory fatigue data under simulated BWR and PWR reactor water coolant conditions. Those laboratory data have been used to support arguments for revising the design-basis fatigue curves in the ASME Code Section III, Division 1, for new construction of Class 1 components. Therefore, the EPRI MRP performed a thorough review of the applicability of these data to actual component operating conditions. Reference 18 provides the results of that data review.

Reference 18 was prepared, in part, in parallel with a related effort of laboratory fatigue data evaluation being carried out by the Cyclic Life and Environmental Effects (CLEE) Steering Committee of the PVRC, under direction from the ASME Board on Nuclear Codes and Standards.

Four major elements were included in the MRP laboratory data assessment:

- Review of available laboratory data relative to thresholds for environmental parameters, such as temperature, reactor water oxidation potential, strain rate, strain amplitude, reactor water flow rate, and component metal sulfur content;
- Determination of the relevance of the laboratory data to actual plant operating conditions;
- Review of structure/component scale fatigue tests where one surface of the structure/component is in contact with oxygenated water; and
- Assessment of current ASME Code Section III Class 1 margins to account for the effects of data scatter, surface finish, size, and reactor water environments.

In the following paragraphs, the portions of the laboratory data review that apply to carbon and low-alloy steel are summarized.

6.1 Laboratory S-N Data Review

The results of the laboratory S-N data review show that laboratory fatigue crack initiation data for carbon steels in an air environment at various temperatures are in good agreement with the ASME Code carbon steel mean air curve. In addition, laboratory fatigue crack initiation data for carbon steels in simulated PWR reactor water environments satisfy the PVRC environmental parameter thresholds [18] for moderate environmental effects, with the data points falling within the appropriate region between the ASME Code mean air curve and the ASME Code fatigue design curve assigned to data scatter and moderate environmental effects. Figures 11 and 12 illustrate this finding.

In contrast, most laboratory fatigue crack initiation data for carbon steels in simulated BWR reactor water environments satisfy the PVRC environmental parameter thresholds for moderate environmental effects. However, a few data points actually fall below the ASME Code carbon steel fatigue design curve. The location of these data points is attributed to above normal ranges for dissolved oxygen. Figure 13 illustrates this finding.

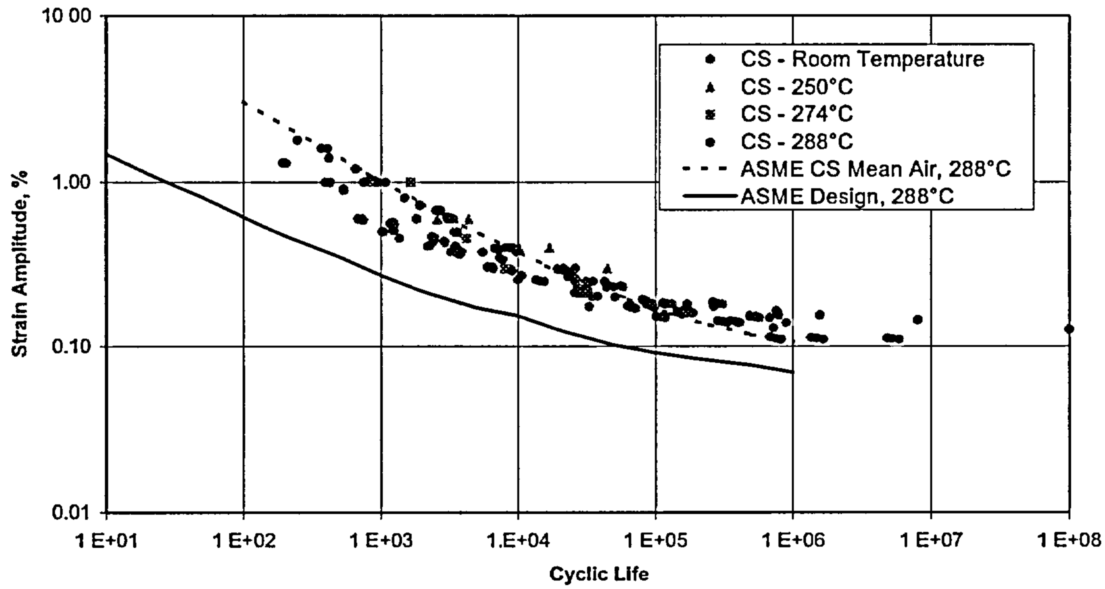

Figure 11. Laboratory data for carbon steel in air.

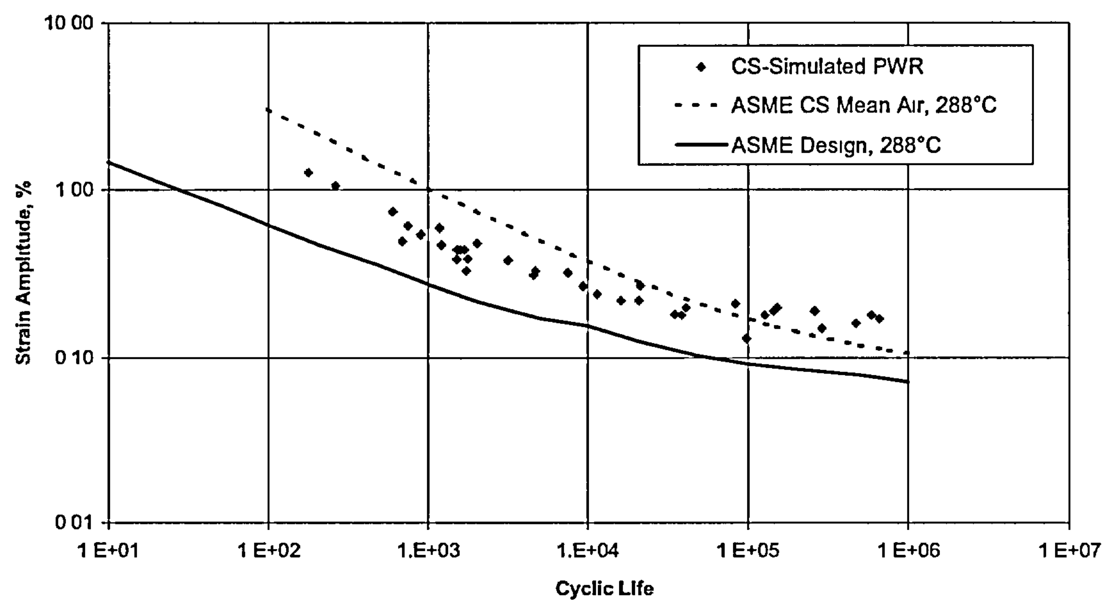

Figure 12. Carbon steel data laboratory obtained under simulated PWR conditions.

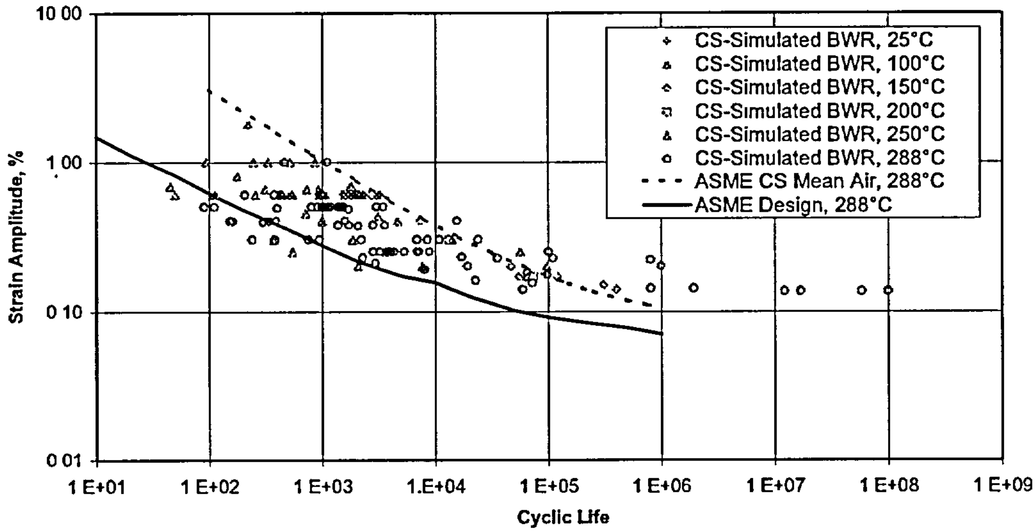

Figure 13. Carbon steel data laboratory obtained under simulated BWR conditions.

Laboratory fatigue crack initiation data for low-alloy steels in an air environment at various temperatures are also in excellent agreement with the ASME Code low-alloy steel mean air curve. Laboratory fatigue crack initiation data for low-alloy steels in simulated PWR reactor water environments satisfy the PVRC environmental parameter thresholds, with the data points falling within the appropriate region between the ASME Code mean air curve and the ASME Code fatigue design curve assigned to data scatter and moderate environmental effects [18]. Figures 14 and 15 illustrate this finding.

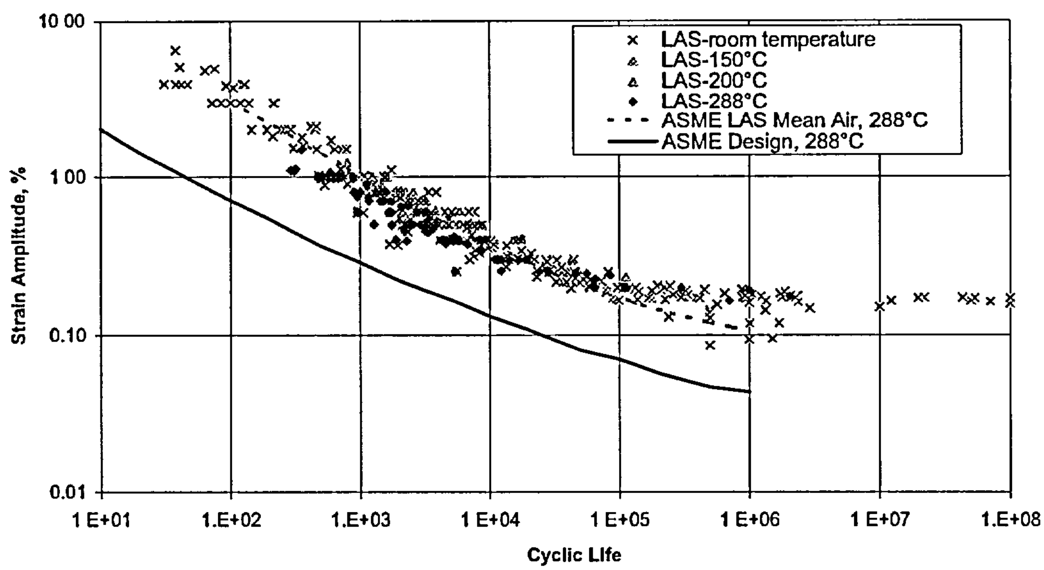

Figure 14. Laboratory data for carbon steel in air.

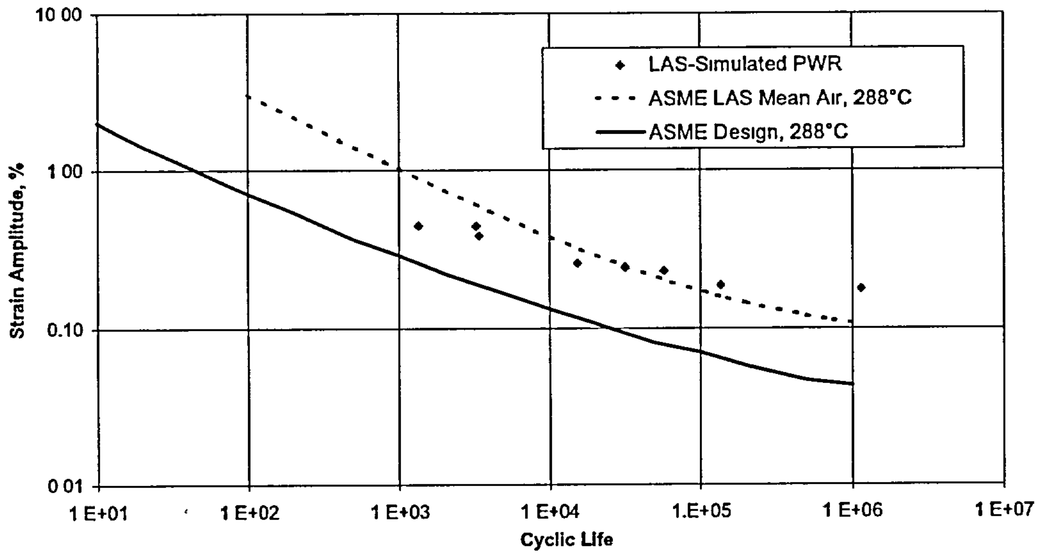

Figure 15. Low-alloy steel laboratory data obtained under simulated PWR conditions.

With the exception of a very few data points obtained at very high strain range (very low cycle fatigue), laboratory fatigue crack initiation data for low-alloy steels in simulated BWR reactor water environments satisfy the PVRC environmental parameter thresholds for moderate environmental effects. Again, a few data points fall below the ASME Code low-alloy steel fatigue design curve and are attributed to high dissolved oxygen. Figure 16 illustrates this finding.

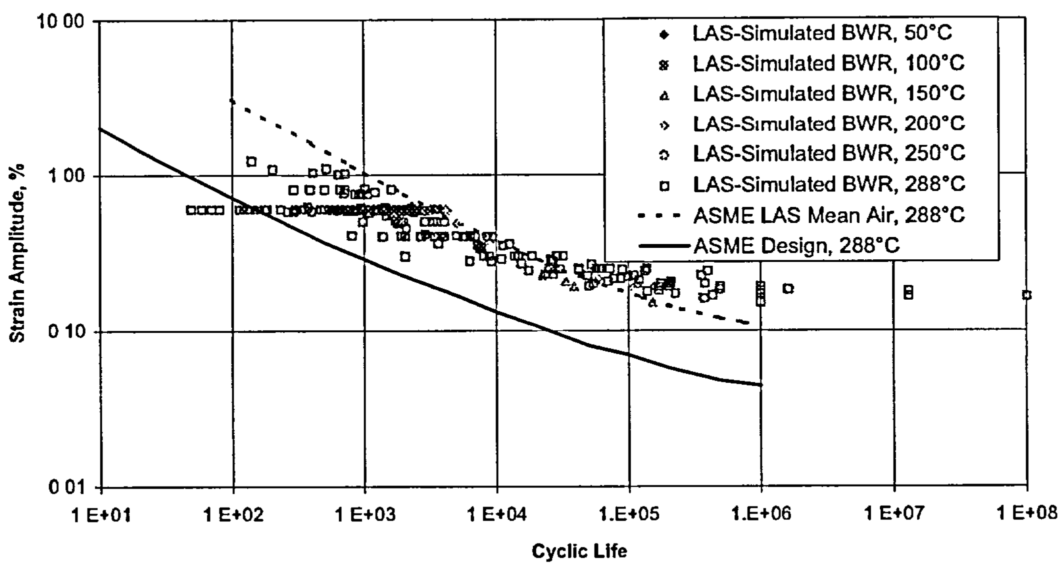

Figure 16. Low-alloy steel laboratory data obtained under simulated BWR conditions.

6.2 Laboratory Data Applicability

The review of laboratory data applicability examined five variables: (1) temperature; (2) applied strain amplitude; (3) applied strain rate; (4) dissolved oxygen content; and (5) coolant flow rate.

6.2.1 Temperature

Testing temperature relative to operating temperature was not deemed to be a concern.

6.2.2 Applied Strain Amplitude

The major concern with respect to applied strain amplitude was deemed to be strain distribution. Surface plastic strains that cause fatigue in actual components tend to be very localized in regions of geometric discontinuity, whereas the plastic strains for low-cycle laboratory fatigue specimens are uniform over the testing gauge length. Therefore, while both cylindrical and hourglass-shaped specimens are valid fatigue test geometries, hourglass-shaped specimens are more representative of actual components in service than cylindrical-shaped specimens.

6.2.3 Applied Strain Rate

Strain rate is a concern because of the ability in the laboratory to apply very high strain amplitudes at very low strain rates. For a component in actual service, high strain amplitudes are generally associated with relatively high strain rates (e.g., thermal shocks), while very low strain rates are associated with low strain amplitudes that cause little fatigue damage. An exception to this rule are thermal stratification interface stresses.

6.2.4 Dissolved Oxygen Content

Reactor coolant chemistry is controlled during normal plant operation to maintain prescribed dissolved oxygen content and other oxidizing species. For PWRs, the nominal dissolved oxygen levels in the reactor coolant system during normal plant operation is of the order of 0.01 to 0.02 ppm, well below the threshold level for carbon and low-alloy steels. These levels also apply for BWRs using hydrogen water chemistry. For other BWRs, the primary coolant pressure boundary is maintained at dissolved oxygen levels of 0.05 to 0.2 ppm during normal plant operation. During plant shutdown, when the reactor coolant system or the primary coolant pressure boundary may be open to atmospheric conditions, it is possible for the dissolved oxygen to reach saturation levels (i.e., 8 ppm). The plant returns quickly to normal operating chemistry during startup and before the plant reaches significant power levels.

6.2.5 Coolant Flow Rate

This is a major concern relative to the interpretation of much of the existing laboratory data under simulated reactor water environmental conditions. This is because the effect of reactor water flow rate has been evaluated for carbon steels in both the laboratory and in large-scale component tests, and found to be a critical environmental parameter.

Typical reactor coolant velocities are of the order of 25 to 200 in/s (0.6 to 5 m/s). On the other hand, flow velocities in the laboratory under simulated reactor water conditions are much lower. As an example, the apparatus used by Argonne National Laboratory (ANL) for their simulated fatigue crack initiation experiments has a volume flow rate of 10 ml/min, with an autoclave volume of 12 ml. Using a length of 2 inches (50 mm) between the inlet and outlet to the autoclave, and ignoring the volume occupied by the specimen, the average flow velocity is about 0.028 in/s (0.0007 m/s), approaching stagnant flow.

Not all simulated reactor water velocities are that low. The recirculating test loop used by General Electric Company for the butt-welded piping tests discussed later had a volume flow rate of 12 gal/min through the NPS 4 piping specimens, implying a velocity of about 25 in/s (0.6 m/s). However, much of the experimental data generated over the past two decades has been obtained at flow rates that are virtually stagnant. These low flow rates, when combined with very low dissolved oxygen, expose the test specimens to extremely low oxidizing potential that could introduce problems at strain amplitudes sufficient to rupture protective oxide layers or passivated surfaces.

6.2.6 Summary

In summary, there are major difficulties with the application of laboratory data under simulated reactor water environmental conditions, with the biggest concerns relative to strain amplitude/strain rate combinations, plastic strain distribution on the surface of the test specimens, and simulated coolant flow rate.

7.0 Structure/Component Fatigue Tests.

Three sets of structural/component fatigue test data were reviewed and compared to laboratory data and plant operating experience. Both carbon and low-alloy steel components had one surface exposed to stagnant, oxygen-saturated water that approximates worst-case BWR reactor water environments. Any potential reduction in fatigue life from specimen size or surface finish effects were explicitly accommodated, since these were structural/component tests and not smooth, laboratory geometry specimens. Laboratory fatigue test results would have predicted crack initiation at or below the ASME Code fatigue design curve. However, results from the structural/component fatigue tests indicate that crack initiation was above and, in some cases, well above the ASME Code fatigue design curve.

7.1 Pressure Vessel Tests

Six carbon steel (A-201) and six low-alloy steel (A-302) pressure vessels were tested by Southwest Research Institute (SWRI) under cyclic hydraulic pressure loading at ambient temperature. The vessels were cylindrical shells 7-ft long with a 36-inch inside diameter and a 2-inch thick wall. The vessels had hemispherical heads, one head containing a 15-inch manway. The vessels contained a number of different nozzle openings and blind holes. Oxygenated water was in contact with the inner surface, since the cyclic loading was obtained by pumping water into the vessels. Extensive stress analyses were performed on the vessels both experimentally with strain gages and by analysis to define the stress and strain ranges at the various locations in the vessels. The vessels were cycled until leakage or failure. In some cases, leaks were repaired and the cycling continued.

The results from these twelve tests are shown in Figures 17 (carbon steel vessels) and 18 (low-alloy steel vessels). While some of the crack initiation results approach the ASME Code fatigue design curve, none of the cycles to crack initiation were less than the design curve. Since the intent of the ASME Code fatigue design curves is to predict the mean line for crack initiation in actual vessels [18], these tests demonstrate that the margins used by the ASME in developing the design curves are conservative, even with exposure of vessel inner surfaces to oxygenated water.

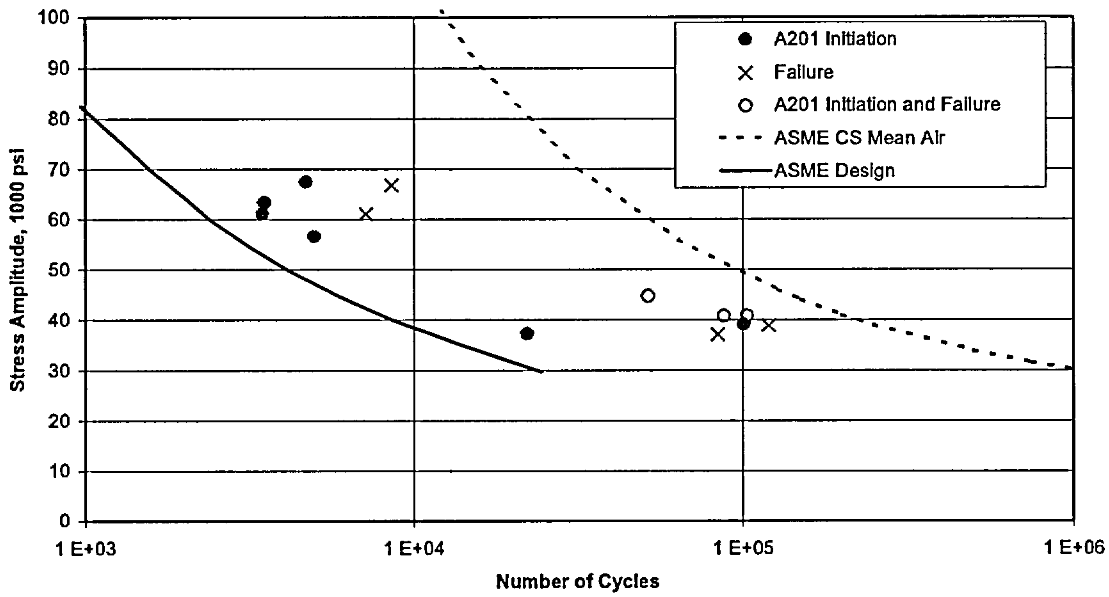

Figure 17. Fatigue testing of full size carbon steel pressure vessels.

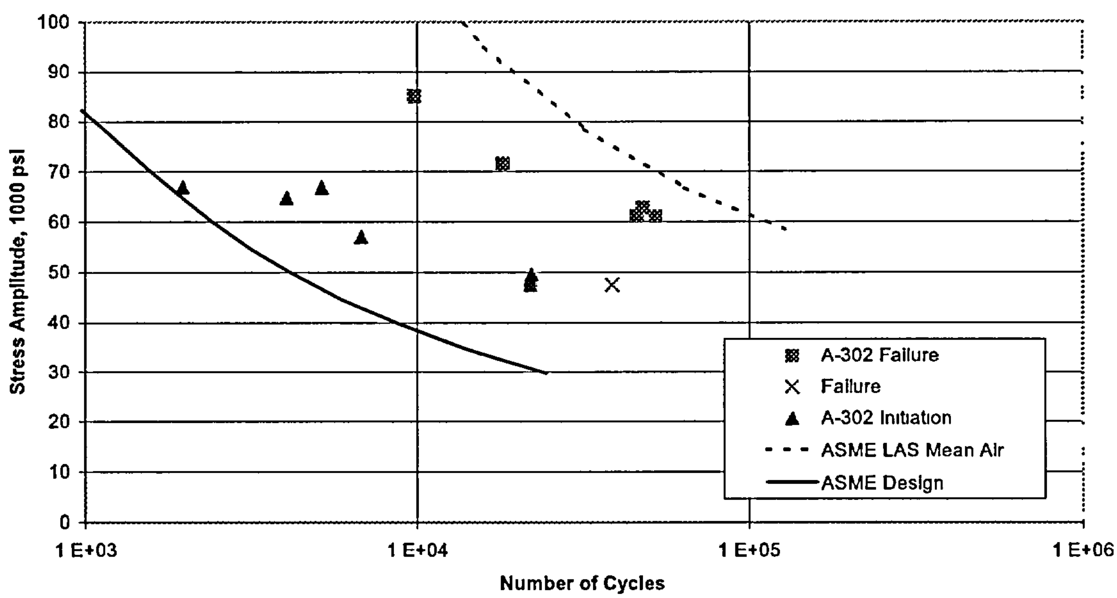

Figure 18. Fatigue testing of full size low-alloy steel pressure vessels.

7.2 Butt-Welded Piping Tests

In the early 1980's, the General Electric Company conducted a combined experimental and analytical program on the fatigue crack initiation behavior of carbon steel components. Included in the program were a series of fatigue experiments on butt-welded pipes in simulated BWR reactor coolant environments. The test specimens were

4-in, Schedule 80 welded pipe. The test section was about 4 ft long and contained 11 butt welds in series spaced about one diameter apart. The welding parameters were typical of those used in the field, and the welds were post-weld ground to reproduce typical field conditions on the inside diameter. The pipes contained 1200 psig water on the inside and were subjected to an externally applied axial stress. In addition to the external load, the pipe welds also experienced welding residual stresses. The pipe tests were conducted in 288°C air and in 0.2 (nominal BWR conditions) and 8 ppm (saturated) oxygenated water. The recirculating test loop had an internal volume flow rate of 12 gal/min (45.4 l/min) through the nominal 4-inch (101.6 mm) diameter piping specimens, implying a velocity of about 25 in/s (0.6 m/s).

The results are shown in Figures 19, which shows the actual fatigue life measured in the experiments. This figure shows that all of the data points, except those that did not fail, were less than the life predicted by the ASME Code fatigue design curve. The major reason for the measured cyclic life to be low is the bounding nature of the test program. When the first of the eleven butt welds failed, the test was terminated, in spite of the fact that ten other butt welds had longer, and perhaps much longer lives. Therefore, these test results must be viewed with the perspective that a complete set of failure points for all of the butt welds would produce a considerably different statistical picture. This is confirmed by the "apparent" scatter in the failure data from these bounding failure points and those points from specimens (with 11 butt welds in series) for which no fatigue failure was observed.

For this reason, the scatter in fatigue life shown in Figure 19 is much greater than the scatter generally seen in fatigue results from laboratory specimens. This is particularly evident at the lowest stress level tested. At this stress level, one pipe section cracked after 127 cycles while two other pipe sections did not crack after 5000 cycles and termination of testing. In addition, the environmental conditions under which most of these tests were conducted were more severe than is typical of a BWR environment. Typical oxygen contents in BWR coolant are 0.2 ppm for normal water chemistry and 0.05 ppm for hydrogen water chemistry. Most of these tests were conduced with 8 ppm oxygen. Only one test was conducted with 0.2 ppm oxygen. That one test exhibited a relatively long fatigue life. The other tests would be expected to display similar behavior in actual BWR oxygenated environments.

7.3 Carbon Steel Tube Bend Tests

A series of tests on 180-degree bends of carbon steel tubing were carried out at KWU laboratories in Erlangen, Germany, over a decade ago, in order to specifically study the effect of simulated coolant flow and oxidation potential on fatigue life. Flow rates through the tubes was varied from essentially stagnant flow up to 0.6 m/s, and dissolved oxygen was varied from PWR conditions (0.01 ppm) up to nominal BWR conditions (0.2 ppm). The data from the test program are shown in Figure 20.

Four different experimental conditions were used – high dissolved oxygen/low flow, very low dissolved oxygen/low flow, 0.2 ppm dissolved oxygen/low flow, and 0.2 ppm dissolved oxygen/moderate flow. With the exception of the very low dissolved oxygen/low flow, all of the low flow data lies close to, or below, the ASME Code fatigue design curve for carbon steel at temperature. For very low dissolved oxygen, approximating stagnant flow in a PWR reactor water environment, the data points are similar to previous component test results from PVRC testing. The effect of even a moderate flow rate is to largely eliminate the environmental effect.

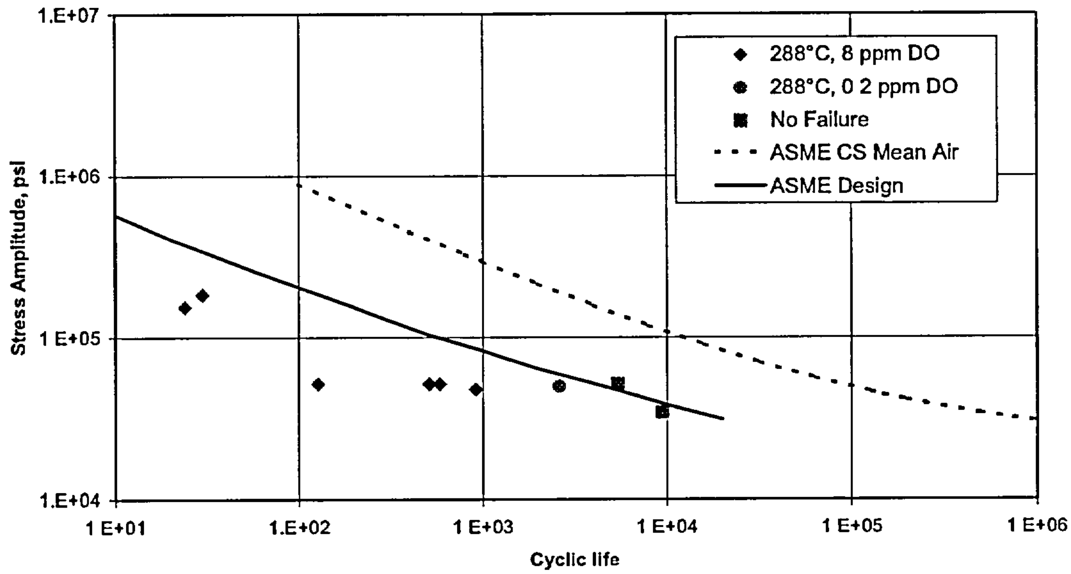

Figure 19. Fatigue testing of butt-welded pipe under simulated BWR conditions.

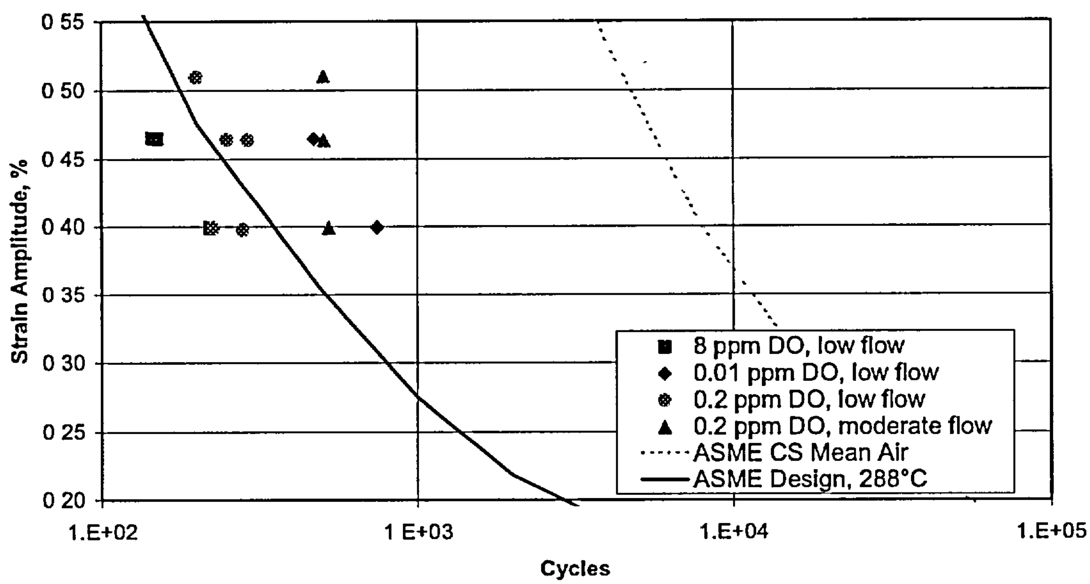

Figure 20. Component-scale fatigue test results for carbon steel illustrating effect of flow rate.

7.4 Structure/Component Test Summary

The structure/component scale data evaluation provides the following conclusions:

- For PWR dissolved oxygen conditions, flow rate effects are not particularly significant for carbon steel components. Even for trickle flow or stagnant flow (SWRI vessel tests), reactor water environmental effects are moderate for PWR dissolved oxygen conditions. There is no basis for requiring environmental correction for carbon steel under these conditions.
- Similar conclusions would apply for low-alloy steel. No environmental correction should be required for low-alloy steel components, even for stagnant or trickle flow conditions, if PWR dissolved oxygen conditions are met.
- The beneficial effect of moderate flow conditions is such that even the moderate reactor water environmental effect is compensated for by the effect of moderate flow.
- For BWR nominal conditions, the effect of moderate flow is sufficient to bring any potential reactor water environmental effect for carbon and low-alloy steel components within the moderate environmental effects envelope. This same conclusion applies to BWR plants operating under hydrogen water chemistry conditions.
- These results are consistent with the butt-welded pipe tests, which were primarily carried out with 8 ppm dissolved oxygen. Therefore, no environmental correction is needed for carbon and low-alloy steel component fatigue evaluations, except possibly for trickle flow, very high dissolved oxygen conditions.

8.0 Summary

Following the closure of GSI-190 and with the recognition of requirements placed on license renewal applicants to explicitly consider reactor water environmental effects in fatigue aging management evaluations, the industry – through the EPRI MRP Fatigue ITG – directed its efforts both toward providing implementation guidelines to meet the requirements and toward systematic analysis of the need for those requirements. Initial emphasis was placed on guidance for license renewal applicants attempting to meet the imposed requirements. However, this document summarizes several MRP activities that provide the technical basis for eliminating those requirements, and that explicit consideration of reactor water environmental effects, for PWR and BWR component locations fabricated from carbon and low-alloy steels should no longer be required. Further work is underway by the industry to complete its assessment for austenitic stainless steel and Ni-Fe-Cr high-nickel alloy component locations.

The technical arguments for resolution of the environmental fatigue issue for carbon and low-alloy steel locations are based on four sets of mutually supportive data:

- Results from the re-calculation of fatigue crack initiation and through-wall cracking probabilities, and core damage frequency for carbon and low-alloy steel component locations that were originally evaluated in NUREG/CR-6260 and NUREG/CR-6674;
- Review and assessment of laboratory fatigue data obtained under simulated reactor water environmental conditions, in terms of thresholds on temperature, strain amplitude, specimen surface strain distribution, strain rate, simulated coolant dissolved oxygen content and oxidation potential, and – in particular – simulated coolant flow rate;
- Examination of structural/component scale fatigue tests with at least one surface in contact with the simulated coolant environment, including evaluation of size and surface finish effects; and
- Review and comparison with plant operating experience and failure data on light-water reactor components in the United States.

There is no need for explicit incorporation of reactor water environmental effects by license renewal applicants, as a part of the 10 CFR 54.21 fatigue aging management program evaluation, for carbon and low-alloy steel component locations for either PWR or BWR plants. Current programs for managing the effects of fatigue, including any reactor water environmental effects, continue to be adequate for managing fatigue effects during the license renewal term.

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EPRI

Materials Reliability Program: Re-Evaluation of Results in NUREG/CR-6674 for Carbon and Low-Alloy Steel Components (MRP-74)

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Technical Report

Materials Reliability Program: Re-Evaluation of Results in NUREG/CR-6674 for Carbon and Low-Alloy Steel Components (MRP-74)

1003667

Final Report, November 2002

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REPORT SUMMARY

This report presents the outcome of a project to review the analysis performed in Nuclear Regulatory Commission (NRC) report NUREG/CR-6674, and presents a re-evaluation of the carbon and low-alloy steel components described in that report. The re-evaluation showed that the use of more realistic, yet conservative, assumptions results in probabilities of crack initiation and leakage that are significantly less than stated in NUREG/CR-6674.

Background

In June 2000, NUREG/CR-6674 was published, reporting the results of a study performed by Pacific Northwest National Laboratory (PNNL) for the NRC. NUREG/CR-6674 assessed failure probabilities and core-damage frequencies due to fatigue cracking with and without light water reactor environmental effects. The study showed that environmental effects produced a significant increase in the probability of leakage, but did not produce a significant increase in predicted core-damage frequency. The stresses and loading conditions were taken from NUREG/CR-6260. Some 47 components were analyzed in NUREG/CR-6674, with several of them showing a high initiation probability and a leakage probability that exceeded 0.5 at 20 years. These predictions are contrary to industry experience and are an indication that the analysis was conducted using very conservative assumptions. A review of the NUREG/CR-6674 analysis was warranted to determine the potential impact of reactor water environment on fatigue crack initiation and leakage under more realistic assumptions.

Objectives

- To re-evaluate the predicted probability of fatigue crack initiation and subsequent leakage for the carbon and low-alloy steel piping locations reported in NUREG/CR-6674
- To revise the pcPRAISE probabilistic fracture mechanics code using more realistic component stress information and operating conditions
- To demonstrate that reactor water environment does not significantly increase the potential for fatigue crack initiation and leakage during the license renewal period

Approach

Through the EPRI Materials Reliability Program (MRP), the probabilistic fracture mechanics code pcPRAISE was modified to account for more realistic assumptions regarding component stress information and operating conditions. Using a revised version of the program, evaluations were conducted to a) duplicate the previous results, b) perform analysis with a modified endurance limit uncertainty using the same environmentally adjusted fatigue curves, c) evaluate the effect of more recent environmentally adjusted fatigue curve recommendations, and d) evaluate modified loading and temperature conditions for cases where the information was available.

Results

Some differences were observed using more recent environmentally adjusted fatigue curve data, although the probabilities of crack initiation and leakage were usually less. Of more significance, the evaluation showed that the fatigue curve endurance limit uncertainty assumed in NUREG/CR-6674 was not realistic. For most locations, a more realistic endurance limit uncertainty representation significantly reduced the probability of crack initiation and leakage. Further evaluation was conducted for a number of components. For the Babcock and Wilcox (B&W) plant reactor pressure vessel outlet nozzle, and for the older Combustion Engineering (CE) plant reactor pressure vessel outlet nozzle, specific stress report information made available to the NRC and reported in NUREG/CR-6260 was obtained. For these locations, removal of conservatisms and implementation of the modified fatigue curves reduced the probability of leakage by about four orders of magnitude, to the level where it is insignificant. For a number of other locations, evaluations using less conservative temperatures than used in NUREG/CR-6674 significantly reduced the probability of leakage by more than an order of magnitude, using the same stresses as in NUREG/CR-6674. Completion of this project has been hampered by a lack of detailed stress information. It has not been possible to obtain the original stress data upon which NUREG/CR-6260 was based. However, from the locations evaluated, it is expected that further detailed analysis to remove conservatisms would establish that the probabilities of crack initiation and leakage from ferritic steel components, when reactor water environment effects are considered, are insignificant.

EPRI Perspective

This report provides an alternative assessment of work sponsored by the NRC and reported in NUREG/CR-6674. The work reported herein uses more recent information and removes excessive conservatisms inherent in the original work to show that the probability of cracking and leakage from Class 1 reactor coolant pressure boundary ferritic steel components, due to fatigue in light water reactor environments, is significantly less than previously suggested. Although this report is based on a subset of the locations originally evaluated, it demonstrates that analysis to remove conservatisms in stress and loading conditions can show that the effects of reactor environment on crack initiation and leakage are not significant. Thus, plant-specific evaluations of environmental effects on carbon and low-alloy steel components in license renewal submittals, and in the extended operating period, are not warranted.

Keywords

Fatigue
Thermal fatigue
Reactor coolant piping
Environmental fatigue
Cracking
Leakage

EXECUTIVE SUMMARY

In May 2000, NUREG/CR-6674 was published by the Nuclear Regulatory Commission (NRC) to describe work performed by Pacific Northwest National Laboratory (PNNL). The results of performing probabilistic fatigue analysis for both air and light water reactor (LWR) environments showed that LWR environment could significantly increase the probability of leakage from reactor coolant pressure boundary components in an extended operating period. However, there was no significant increase in core melt frequency predicted. Based on the results of this study, the NRC has required that licensees submitting applications for license extension evaluate the effects of LWR environment on fatigue in the extended operating period.

The study reported herein examines, in detail, the analysis reported in NUREG/CR-6674 to identify conservatisms that contribute to the high, predicted probabilities of crack initiation and leakage. Of special significance is the conservatism used to define the variance for the high-cycle end of the probabilistic fatigue curves. Because the NUREG/CR-6674 study was undertaken using fatigue data curve fits circa 1995, later curve fits (published by Argonne National Laboratory [ANL]) were also evaluated.

This report addresses only the 12 carbon and low-alloy steel components, addressed in NUREG/CR-6674, that exhibited a relatively high probability of crack initiation and leakage. Analysis using the latest fatigue data curve fits and modified endurance limit variance significantly reduced the probabilities for most components. Further analysis that reduced conservatisms in the operating temperatures and stresses showed further reduction of the predicted probabilities.

Comparison of the revised initiation and leakage probabilities (as affected by LWR environment for a 60-year life) to the air environment results predicted by PNNL for 40-year life showed that LWR environmental effects are not significant. This demonstrates that consideration of LWR effects on carbon and low-alloy steel components in the extended license-operating period is not warranted.

ACKNOWLEDGMENTS

The special contribution of Dr. Fred Simonen (Pacific Northwest National Laboratory) is acknowledged for providing input and assistance in understanding the work originally reported in NUREG/CR-6674.

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1 INTRODUCTION

In June 2000, NUREG/CR-6674 [1] was published, reporting the results of a study performed by Pacific Northwest National Laboratory (PNNL) for the Nuclear Regulatory Commission (NRC) to assess failure probabilities and core-damage frequencies due to Class 1 reactor vessel and piping component fatigue cracking with and without light water reactor (LWR) environmental effects. The study showed that environmental effects produced a significant increase in the probability of leakage but did not produce a significant increase in predicted core-damage frequency. The stresses and loading conditions were taken from NUREG/CR-6260 [2]. Some 47 components were analyzed in NUREG/CR-6674, with several of them showing a high initiation probability and a leakage probability that exceeded 0.1 at 40 years. The results for two components showed a 50% probability for fatigue crack initiation after approximately only 10 years of operation, and a similar probability of leakage after about 15–20 years. These predictions are contrary to industry experience and are an indication that the analysis was conducted using very conservative assumptions.

The objective of the work reported herein is to provide more realistic assumptions and to re-evaluate the analyses in NUREG/CR-6674 to show that the predicted probability of fatigue crack initiation and growth of cracks to produce leaks can be substantially reduced by the use of less-conservative inputs. For a selected set of carbon and low-alloy steel component locations, alternate analyses were conducted. In this evaluation, it was determined that the fatigue curve endurance limit used in NUREG/CR-6674 was unduly conservative, so an alternate, more realistic, value was derived. The revised fatigue curves were used in probabilistic fatigue initiation and crack growth calculations using a modified version of the pcPRAISE (Piping Reliability Analysis Including Seismic Events) computer program [3]. More realistic loading conditions were derived for two components where details from original stress reports were available and contained sufficient information. In addition, minor changes to generating temperature and geometry were made to more accurately reflect actual conditions. Because the NUREG/CR-6674 analysis was based on fatigue data circa 1995, the effects of newer environmental fatigue data (as reported in NUREG/CR-6717 [4]) were also evaluated.

This report describes three major efforts:

- Evaluations were conducted to determine an appropriate variance of the high-cycle (endurance limit) end of the fatigue curve. The original evaluations reported in NUREG/CR-6674 were re-calculated with this modified variance of the fatigue curve endurance limit.

- The evaluations were then repeated using the most recent environmental fatigue data curve fit recommendations from Argonne National Laboratory (ANL), as reported in NUREG/CR-6717 [4].

- Revised input stress and/or temperature conditions were derived for a few locations. The results for these locations were re-calculated to show the effects of using less conservative input conditions.

Section 2 of this report provides additional background and summarizes the work completed in NUREG/CR-6260 and NUREG/CR-6674. Key conservatisms are identified.

Section 3 describes the environmental fatigue curves used in NUREG/CR-6674 and modifications of the curves based on work by ANL since the report was published. Revisions to the variance of the endurance limit are also discussed.

Section 4 presents results of the NUREG/CR-6674 re-evaluation effort. Results of the modified input are compared to the original NUREG/CR-6674 results.

Section 5 discusses the final results and their implications for requiring consideration of LWR environmental effect with carbon and low-alloy steel components.

Section 6 presents overall conclusions from this re-evaluation.

2 BACKGROUND

2.1 Application of Interim Fatigue Curves to Selected Locations

In 1995, NUREG/CR-6260 was published by the NRC [2]. This report, prepared by Idaho National Engineering Laboratory (INEL), now Idaho National Engineering and Environmental Laboratory (INEEL), evaluated the effects of LWR environment on the fatigue life of a selected set of locations in both Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) plants. Under a separate program, the NRC had sponsored research at ANL to develop interim fatigue curves that would quantify the effects of LWR environment, including the effects of oxygen content, strain rate, elevated temperature, and so on. These results were initially published in NUREG/CR-5999 [5] and were the basis for the studies in NUREG/CR-6260.

To support the work presented in NUREG/CR-6260, INEL collected stress report information for six representative components in each of a selection of five PWR plants and two BWR plants. The selection of plants included the following:

- One Babcock and Wilcox (B&W) PWR plant
- One older- and one newer-vintage Combustion Engineering (CE) PWR plant
- One older- and one newer-vintage Westinghouse (W) PWR plant
- One older- and one newer-vintage General Electric (GE) BWR plant

The component locations for PWR plants included:

- Reactor vessel shell and lower head
- Reactor vessel inlet and outlet nozzles
- Pressurizer surge line (including nozzles)
- Charging system nozzle (makeup system in a B&W plant)
- Safety injection nozzle
- Residual heat removal system Class 1 piping

The component locations for the BWR plants included:

- Reactor vessel shell and lower head
- Reactor vessel feedwater nozzle
- Reactor recirculation system piping (including inlet and outlet nozzles)

- Reactor core spray nozzle and associated piping
- Residual heat removal Class 1 piping
- Feedwater line Class 1 piping

INEL recognized that there was conservatism in many of the original fatigue analyses conducted for the initial plant designs. In certain cases, where there was information from component stress analyses to support the re-evaluation, some of these conservatisms were removed. However, there was generally insufficient detail in the component stress reports to support reduction of conservatisms or to derive realistic strain rates or temperatures for transient loading conditions.

In other cases, no explicit fatigue analysis was available (for example, ANSI B31.1 components), so INEL performed detailed stress and fatigue analysis based on loading conditions for similar plants where loading information was available.

In most cases, there was an attempt to provide a revised fatigue assessment that would take into account a more realistic number of plant cycles, as compared to those used in the original plant designs. This step was generally not taken when the fatigue usage with environmental effects was determined to be less than 1.0 for the environmental fatigue calculations.

In summary, their analysis showed that environmental effects could be accommodated in many components by removing existing conservatisms in the stress report fatigue calculations and by taking the actual number of cycles into account. However, their analysis did not conclusively prove that usage factors less than 1.0 could be derived for all locations when considering environmental effects.

It should be noted that later work was performed by ANL [4] to develop modified correlations that would describe fatigue environmental effects for both carbon steel and stainless steel components. There were no further evaluations sponsored by the NRC to assess the effects of these later research findings on the usage factors presented in NUREG/CR-6260.

2.2 Assessment of Fatigue Effects for 60-Year Life

In June 2000, NUREG/CR-6674 was published, documenting the PNNL study to assess the impact of fatigue, including environmental effects, on the probability of fatigue crack initiation and leakage for both a 40-year and 60-year plant life [1]. The major purpose of the PNNL study was to provide an estimate of the effect of LWR environment on core-damage frequency. The effects of LWR environment on crack initiation and leakage probabilities were also reported.

In performing their study, PNNL used the stress amplitudes and cycles for vessel and piping locations reported in NUREG/CR-6260. An enhanced version of pcPRAISE [3] was used to estimate the probability of developing a leak at these locations. Analyses were performed for both 40-year and 60-year reactor lifetimes by multiplying the number of cycles for 40 years by 1.5 to derive 60-year numbers. Probabilistic representation of the strain-life curve that was developed by ANL [6] was incorporated into the modified pcPRAISE software to provide the probability of fatigue crack initiation as a function of time. The previously available crack growth portion of pcPRAISE was used for probabilistic analysis of crack growth when initiation was predicted to occur.

NUREG/CR-6260 provided only the cyclic stress amplitudes at the surface of each location evaluated. This is sufficient information to evaluate the probability of crack initiation at the location, but it is not sufficient to define the subsequent fatigue crack growth. The major piece of information that is lacking is the through-wall distribution of the stress for each condition that makes up the stress range pair. This stress distribution can have a large influence on the time that it takes for an initiated crack to grow through the wall of the component.

Another important piece of information that was lacking was the spatial variation of stresses along the component surface. This can have an important effect on the extent of crack initiation. This can be especially important in piping systems where there might be significant circumferential stress variation due to the contribution of pipe bending stress to the total stress range.

In predicting crack growth, the size of the initiated crack is an important parameter. The ANL correlations were stated to represent the number of cycles to a crack depth of 3 mm (0.12 inch). This defines the initiated crack depth (a), but not the half-length (b). As discussed in the PNNL report, the median value of the initiated length was taken to be 0.30 inches, which results in an initial aspect ratio b/a of 2.5. The difference between the crack half-length and the crack depth (b-a) was assumed to be log-normally distributed with a shape parameter (standard deviation of the natural logarithm) of 0.682. These assumptions were not changed in this study.

The pcPRAISE software, as it existed before the PNNL efforts, did not consider fatigue crack initiation. It considered stress corrosion crack initiation and growth, and the fatigue crack growth of pre-existing crack-like defects. As described in the PNNL report, the software was modified to consider fatigue crack initiation and provisions were made to consider through-wall and circumferential gradients of the peak cyclic stresses that were obtained from the INEL report. The radial gradient considered by the version of pcPRAISE used in the PNNL efforts can be defined in a very general fashion, but the PNNL study used a combination of a uniform through-wall stress and a gradient stress to represent the combined stresses, which was assumed to be uniformly applied around the circumference of the component. A generic gradient to represent thermal stresses was of the form:

$$\sigma = \sigma_0 \left[1 - 3\xi + \frac{3}{2}\xi^2 \right] \qquad \textbf{Eq. 2-1}$$

where,

σ = thermal stress at a specified location through the thickness, psi
σ_o = inside surface thermal stress, psi
ξ = x/t
x = radial distance from inner wall, in.
t = wall thickness, in.

This describes a second-order polynomial stress distribution that is self-equilibrating through the wall with zero slope at the outer surface ($\xi=1$). Such a gradient generally occurs for thermal stresses due to ramp changes to thermal boundary conditions.

The stresses from NUREG/CR-6260 were decomposed into uniform and generic gradient stress distributions. The circumference was broken up into 2-inch (50.8-mm) increments. The relative magnitudes of the uniform and gradient contributors to the stress were identical in each increment around the circumference. The peak stresses from NUREG/CR-6260 were decomposed by the following procedure:

- Cyclic stresses associated with seismic loads were treated as 100 % uniform stress.
- Cyclic stresses greater than 45 ksi (310 MPa) were treated as having a uniform component of 45 ksi (310 MPa) and the remainder was assigned to the gradient category.
- For those transients with more than 1000 cycles over a 40-year life, it was assumed that 50% of the stress was uniform and 50% was due to a through-wall gradient. In addition, for these transients, the uniform stress component was not permitted to exceed 10 ksi (69 MPa).

The PNNL analyses did not consider any circumferential variations of stress, and the minimum stress during a load cycle was taken to be zero (that is, the load ratio, R, was taken to be zero), because NUREG/CR-6260 provided only the alternating stress amplitude for each load set pair.

The PNNL evaluations also took no credit for inservice inspection, which might detect cracks prior to leakage occurring.

The above assumptions allowed a conservative probabilistic fatigue lifetime to be predicted, which included both the initiation and growth portions of the lifetime, using only the minimal amount of information that was contained in NUREG/CR-6260. Because component thickness and diameter were not available in NUREG/CR-6260, assumed values were used by PNNL.

Results of the PNNL evaluations are shown in Table 2-1 for an air environment and Table 2-2 for a water environment. Based on these results and additional results for stainless steel components, PNNL concluded that "environmental effects were predicted to increase through-wall crack probabilities by as much as two orders of magnitude."

Table 2-1
Probabilities of Crack Initiation and Leakage From NUREG/CR-6674 in Air Environment

Component	40-Year Life		60-Year Life	
	Initiation	Leakage	Initiation	Leakage
B&W RPV Outlet Nozzle	7.89E-02	2.92E-03	1.05E-01	1.27E-02
CE – New RPV Outlet Nozzle	2.21E-01	1.00E-07	4.55E-01	7.70E-06
CE – New Safety Injection Nozzle	2.15E-03	1.22E-06	8.69E-03	3.18E-05
CE – Old RPV Outlet Nozzle	7.89E-02	6.72E-04	1.28E-01	4.79E-03
GE – New Feedwater Nozzle Safe End	4.31E-02	2.00E-06	1.19E-01	2.00E-05
GE – New Residual Heat Removal (RHR) Line Straight Pipe	3.71E-01	2.08E-01	4.99E-01	3.02E-01
GE – New Feedwater Line Elbow	6.77E-02	1.00E-05	1.82E-01	6.00E-05
GE – Old RPV Feedwater Nozzle Bore	3 83E-02	4.53E-05	9.13E-02	6.76E-04
GE – Old Feedwater Line – Reactor Core Isolation Cooling (RCIC) Tee	1.37E-01	4.30E-06	4.35E-01	2.06E-04
W – New RPV Outlet Nozzle	7.26E-01	3.00E-04	8.76E-01	2.20E-03
W – Old RPV Inlet Nozzle	2.06E-01	2.00E-07	4.21E-01	1.03E-05
W – Old RPV Outlet Nozzle	2.67E-01	2.00E-05	5 22E-01	6 00E-05

Table 2-2
Probabilities of Crack Initiation and Leakage From NUREG/CR-6674 in Reactor Water Environment

Component	40-Year Life		60-Year Life	
	Initiation	Leakage	Initiation	Leakage
B&W RPV Outlet Nozzle	7.74E-01	1.83E-01	8 99E-01	5.44E-01
CE – New RPV Outlet Nozzle	4.22E-01	1.74E-03	6 89E-01	2.90E-02
CE – New Safety Injection Nozzle	1.01E-03	1.00E-06	4.81E-03	1.90E-05
CE – Old RPV Outlet Nozzle	5.91E-01	7.05E-02	8 46E-01	3 53E-01
GE – New Feedwater Nozzle Safe End	1.04E-01	1.31E-03	2.53E-01	1.47E-02
GE – New RHR Line Straight Pipe	4.73E-01	4.10E-01	6 71E-01	6.21E-01
GE – New Feedwater Line Elbow	1.59E-01	1.03E-03	3.65E-01	1.46E-02
GE – Old RPV Feedwater Nozzle Bore	7.27E-02	1.00E-05	2.42E-01	8.80E-04
GE – Old Feedwater Line – RCIC Tee	3.76E-01	2.99E-03	7.82E-01	5.92E-02
W – New RPV Outlet Nozzle	8.62E-01	3.65E-01	9.49E-01	7.42E-01
W – Old RPV Inlet Nozzle	3 91E-01	4.38E-03	6 44E-01	5.04E-02
W – Old RPV Outlet Nozzle	4.90E-01	9.33E-03	7.53E-01	9.60E-02

3
pcPRAISE MODIFICATIONS

The starting point in the current efforts was pcPRAISE 4.1, which was the version used in the PNNL study (NUREG/CR-6674). This software was modified for the current study in two ways:

- To allow a detailed definition of the radial stress gradient at each of the angular locations to be considered
- To incorporate an updated curve fit to the probabilistic fatigue data. The modified version of pcPRAISE is identified as Version 5.0

The modifications allow much more detailed specification of stress distributions. Version 4.1 only allowed the stress distribution to be scaled up or down at each circumferential location. Thus, the relative contributions of the constant stress and the gradient could not be changed for various angular locations around the component circumference. For instance, in Version 4.1, details of the distribution could not be defined for stress systems that are a combination of bending (such as restraint of thermal expansion) and axisymmetric with a radial gradient (such as radial gradient thermal stresses). This refinement was not believed to be necessary in the PNNL efforts because insufficient information was available to define these details. In fact, the results reported in NUREG/CR-6674 did not consider circumferential variations at all, even though the software was capable of doing so.

The other modifications accomplished in the current effort incorporated the updated strain-life curve fit recommendations provided in NUREG/CR-6717 by ANL [4]. The strain-life curves recommended by ANL are deterministic and represent median behavior. The probabilistic relationship used in the PNNL study was based on the earlier ANL work reported in NUREG/CR-6335 [6]. It was retained and used in this re-evaluation, except for the modifications that are described in the text that follows. The updates became available after the work reported in NUREG/CR-6674 was completed. The later curve fits were incorporated into pcPRAISE and were also used in the analysis. The later curve fits themselves were further modified to use a more realistic representation of the strain-life curve in the high-cycle region. Other modifications were incorporated based on information in NUREG/CR-6717. Overall, the strain-life curves employed in the PNNL study were modified in the following ways:

- The standard deviation of the curve fitting constant used to describe high-cycle behavior was changed to provide a more realistic representation of the endurance limit variance
- Constants in the median curve fits were updated based on values given in the updated ANL report, NUREG/CR-6717

- The following modifications, as discussed in Section 5 of NUREG/CR-6717, were incorporated:
  - The strain environmental threshold described in Section 5.2 of NUREG/CR-6717 was incorporated into the probabilistic strain-life relation
  - Mean stress effects were incorporated according to relations given in the beginning of Section 5.1 of NUREG/CR-6717
  - The strain-life curve in the very high-cycle range was altered to incorporate the procedure suggested in Section 5.2 of NUREG/CR-6717

Other than the first change described, all of these changes are (as suggested in NUREG/CR-6717) considered to be the most recent recommendations for environmental effects. The pcPRAISE modification was accomplished in such a manner that each effect could be included or not included in an evaluation.

3.1 ANL Fatigue Data Curve Fits

The strain-life (ε_a-N) relation in the ANL efforts for each material in LWR environments can be written as follows:

$$\ln(N) = C_0 + C_{env} - \frac{1}{b}\ln(\varepsilon_a - \varepsilon^*) \qquad \textbf{Eq. 3-1}$$

where,

N	= cycles to crack initiation
C_0, b	= constants that describe the curve shape
C_{env}	= term that accounts for environmental effects
ε_a	= strain amplitude (half the peak-to-peak value), %
ε^*	= term representing the endurance limit, %

This equation can be rewritten as

$$\varepsilon_a = \frac{B}{N^b} + \varepsilon^* \qquad \textbf{Eq. 3-2}$$

where,

$$\ln(B) = b(C_0 + C_{env})$$

This form of the equation clearly shows that as N becomes large, the behavior is dominated by ε^*; thus, ε^* is effectively an endurance limit. Randomness is introduced into the relation by considering C_0 and ε^*; thus, to be random variables. However, following the relations used in NUREG/CR-6674, they are not independent random variables. They are taken to be related so that, for example, when the 10th percentile of B is used, the 10th percentile of ε* is also used. This is equivalent to considering them perfectly correlated. Hence, the probabilistic representation of Equation 3-1 can be rewritten as:

$$\ln(N) = [C_0 + C_{env} + s_0 f(p)] - \frac{1}{b}\ln\{\varepsilon_a - [\varepsilon^* + s_e f(p)]\} - \ln(4) \qquad \textbf{Eq. 3-3}$$

where,

s_0 and s_e are the standard deviations of the mean values for C_0 and ε^*, respectively, and $f(p)$ is the inverse cumulative unit normal probability. (The argument of f is a random variable in the range 0 to 1, so that $f(p)$ goes from minus to plus infinity, and $f(0.5)=0$). The above treatment is equivalent to considering the effects of C_0 to be log-normally distributed and the effects of ε^* to be normally distributed.

The ln(4) term is included to incorporate size effects in accordance with the discussion in the PNNL study. NUREG/CR-6674 provides a complete set of constants for Equation 3-3, as also provided in NUREG/CR-6335 [6].

The constants used in the above equations (as used in NUREG/CR-6674) are summarized in Table 3-1 for carbon and low-alloy steel.

Table 3-1
Constants in Strain-Life Relationship From NUREG/CR-6674

Parameter	Low-Alloy Steel	Carbon Steel
C_0	6.091	6.144
1/b	1.813	2.032
ε*	0.080	0.094
s_0	0.52	0.52
s_e	0.026	0.026
C_{env}	$0.1097S^*T^*O^*\dot{\varepsilon}^*$	
S*	S	S<0.015
	0.015	S>0.015
T*	0	T<150
	T-150	T>150
O*	0	DO<0.05
	DO	DO=0.05-0.5
	0.5	DO>0.5
$\dot{\varepsilon}^*$	Ln(0.001)	$\dot{\varepsilon}$<0.001
	Ln($\dot{\varepsilon}^*$)	$\dot{\varepsilon}$=0.001-1
	0	$\dot{\varepsilon}$>1

S = sulfur content, wt %
DO = dissolved oxygen, ppm
T = temperature, °C
$\dot{\varepsilon}$ = strain rate, %/sec

Table 3-2 summarizes the new set of constants from NUREG/CR-6717 that were incorporated into pcPRAISE to represent the latest fit to the environmental fatigue data. However, NUREG/CR-6717 provides only a set of constants for median behavior (that is, s_0 and s_e are not specified). Because these standard deviations are required to perform the probabilistic crack initiation analysis, the value of s_0 of 0.52, which was used in NUREG/CR-6674, was assumed to still be applicable for the new curve fit. The standard deviations above were also reflected in NUREG/CR-6583 [7].

Table 3-2
Constants in Strain-Life Relationship in NUREG/CR-6717

Parameter	Low-Alloy Steel	Carbon Steel
C_0	5.729	6.010
1/b	1.808	1.975
ε^*	0.151	0.113
s_0	not given	not given
s_e	not given	not given
C_{env}	$0.101S^*T^*O^*\dot{\varepsilon}^*$	
S^*	S	DO<1 & S<0.015
	0.015	otherwise
T^*	0	T<150
	T-150	T>150
O^*	0	DO<0.04
	ln(DO/0 04)	DO=0.04-0.05
	ln(12.5)	DO>0.5
$\dot{\varepsilon}^*$	ln(0.001)	$\dot{\varepsilon}$<0.001
	ln(ε^*)	$\dot{\varepsilon}$=0.001-1
	0	$\dot{\varepsilon}$> 1

S =sulfur content, wt %
DO =dissolved oxygen, ppm
T =temperature, °C
$\dot{\varepsilon}$ =strain rate, %/sec

The standard deviations of ε^* and s_e used in NUREG/CR-6674 are believed to be too large. This fact was recognized in NUREG/CR-6583 [7], which stated, "...the standard deviation of 0.026 on strain amplitude obtained from the analysis may be an overly conservative value. A more realistic value of the standard deviation on strain could be obtained by analysis of the fatigue limits of different heats of material."

The value of ε^* is representative of the endurance limit, and the mean and standard deviation in Table 3-1 lead to unreasonable behavior in the high-cycle portion of the fatigue curves. Hence, the values of s_e were reduced for this study. Although data on scatter in the endurance limit are limited, Wirsching [8] states that the coefficient of variation (standard deviation divided by mean) is about 10% for fatigue strength at a given number of cycles. Thus, s_e in Equation 3-3 is taken as $0.1\varepsilon^*$ for this study.

Figure 3-1 shows a typical fatigue curve using the constants in Table 3-1 for low-alloy steel in water at 550°F (288°C) at low strain rate (0.001 %/sec), high dissolved oxygen (0.5 ppm), and high sulfur (0.15 wt %). The mean curve and the 0.001 quantile curves are shown where one sample in 1000 would have a lower number of cycles to initiate. One curve uses $s_e = 0.026$ (original value) and the other set uses $s_e = 0.1\varepsilon^* = 0.008$ (modified value, which is smaller). Figures 3-2 through 3-5 show a range of quantiles for carbon steel and low-alloy steel for both endurance limit assumptions. In these curves, and in other similar ones in this document, the legend markers are provided only to assist in defining a specific curve, not to delineate specific data points.

The modified curves (which use $s_e = 0.1\varepsilon^*$) are more representative of the expected shape of S-N curves. Changing the standard deviation of the endurance strain amplitude does not significantly affect the distribution on the low-cycle end of the curve; the original and modified curves come together at the low-cycle portion of the figure (as seen in Figure 3-1). It is seen that the value of the standard deviation can have a large effect when the stress amplitude is below about 30 ksi (207 MPa) (0.1% strain), which corresponds to a stress range of about 60 ksi (414 MPa). The probability of having a zero or negative endurance limit, which is a physical absurdity, is greatly reduced with the modified endurance limit variance.

Figures 3-6 through 3-9 show a similar set of data plots for the latest environmental curve fits from NUREG/CR-6717 for the same conditions described previously.

Figure 3-10 is a comparison of the original and latest curve fits and 0.001 quantiles for low-alloy steel, as depicted in Figures 3-2 and 3-7. There is an appreciable difference between the two sets of curves. The new environmental fatigue data curve fit will predict many more cycles to initiation for stress amplitudes below about 40–50 ksi (276–345 MPa) (0.15% strain), but fewer cycles to initiation for higher stress amplitudes. With the modified endurance limit uncertainty shown with the new data fit, the probability of initiation below a 30-ksi (207-MPa) stress amplitude will be reduced significantly.

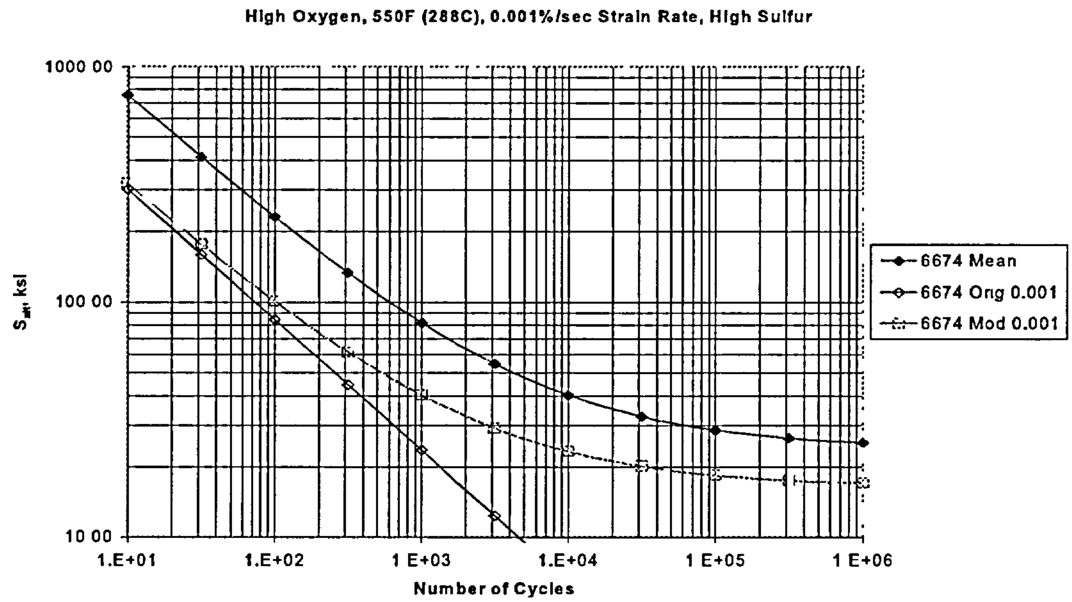

Figure 3-1
Mean and 0.001 Quantiles for Low-Alloy Steel Showing Changes Due to Modified Endurance Limit Variance

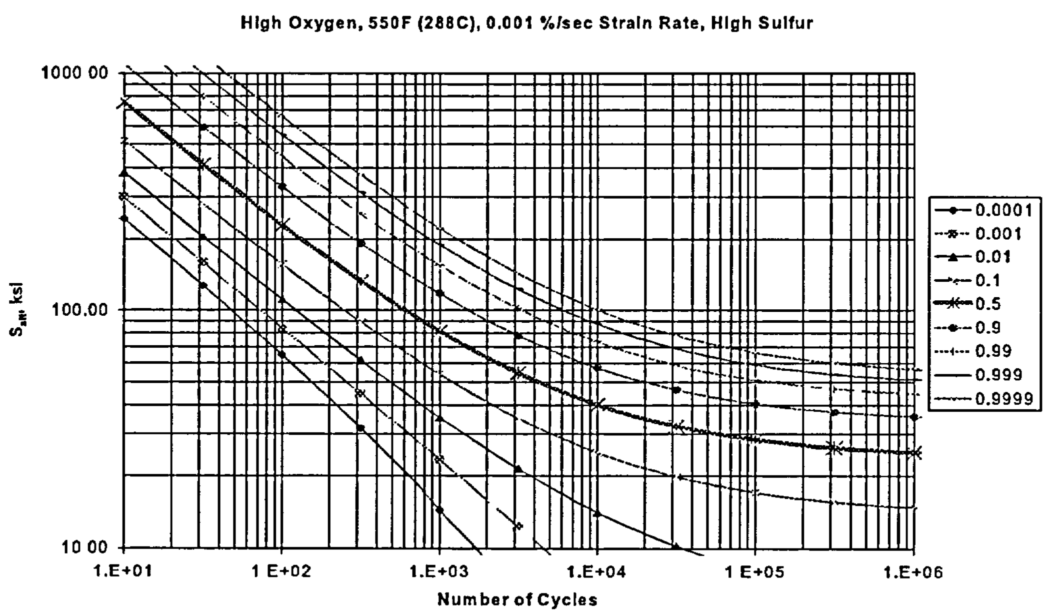

Figure 3-2
NUREG/CR-6674 Low-Alloy Steel Fatigue Curve With Original Endurance Limit Variance for Various Quantiles

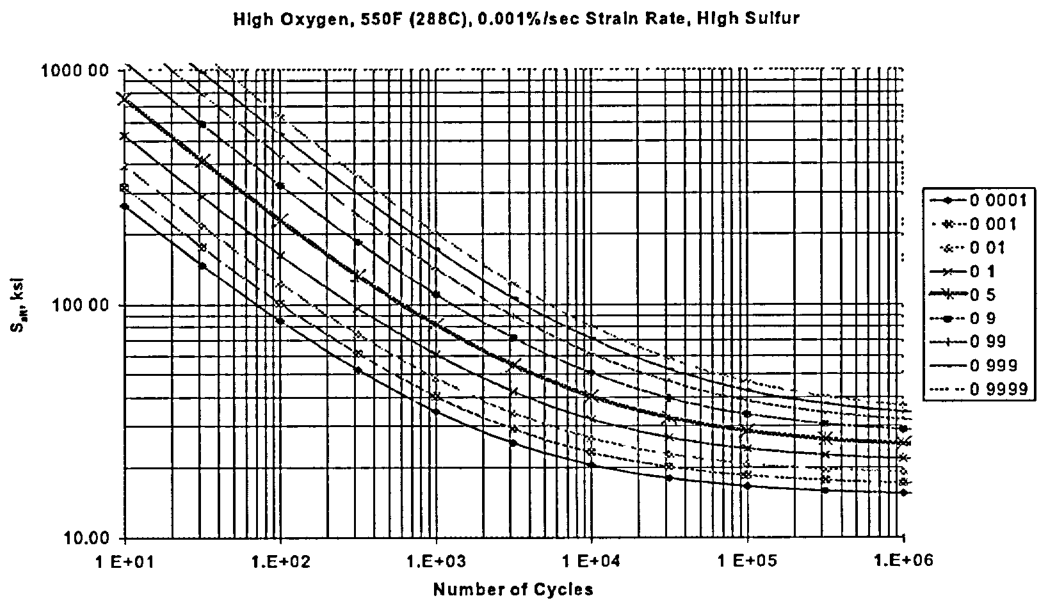

Figure 3-3
NUREG/CR-6674 Low-Alloy Steel Fatigue Curve With Modified Endurance Limit Variance for Various Quantiles

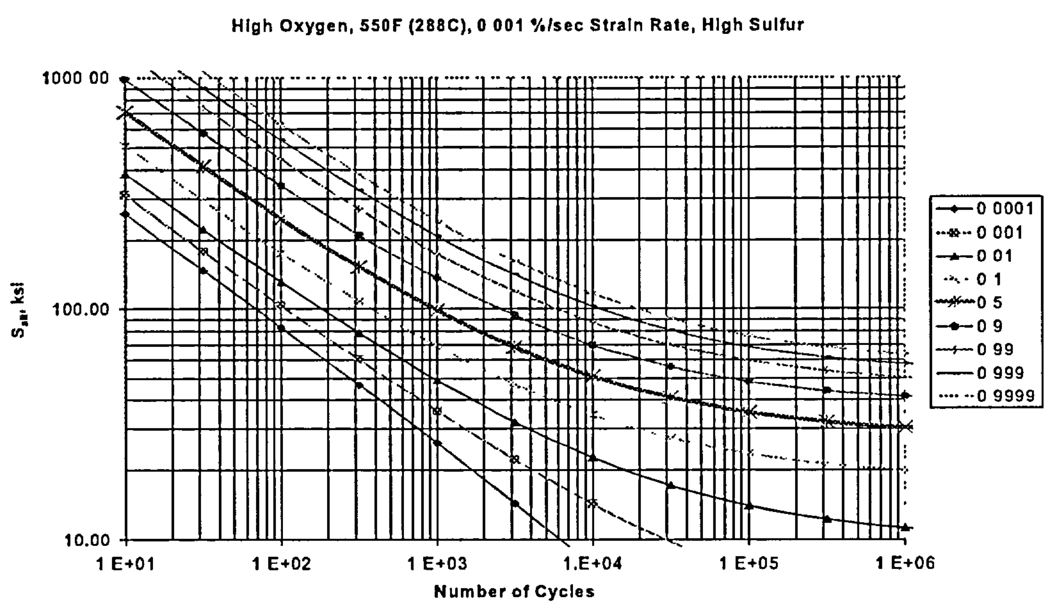

Figure 3-4
NUREG/CR-6674 Carbon Steel Fatigue Curve With Original Endurance Limit Variance for Various Quantiles

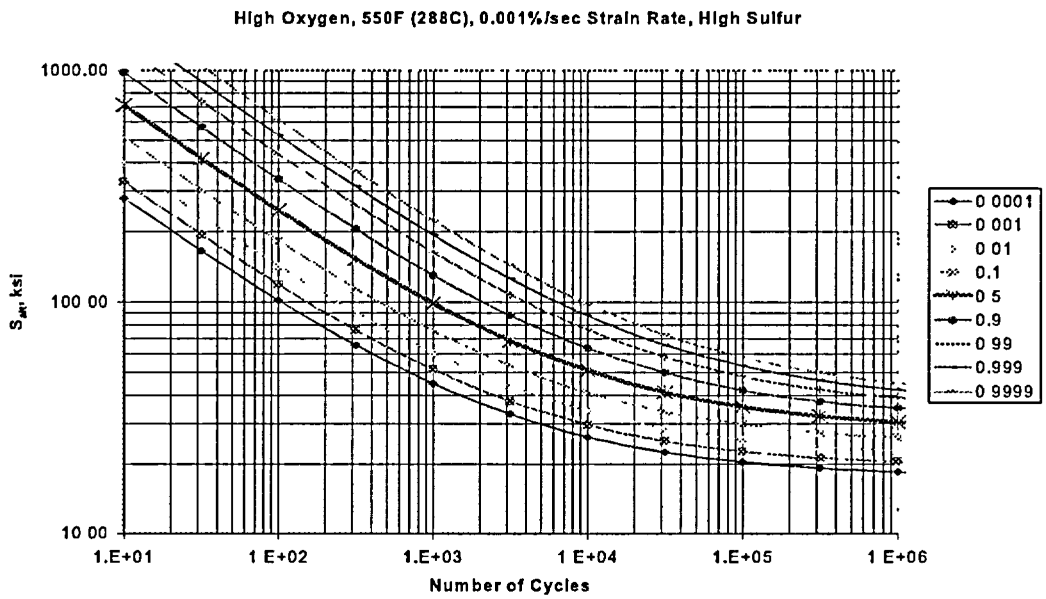

Figure 3-5
NUREG/CR-6674 Carbon Steel Fatigue Curve With Modified Endurance Limit Variance for Various Quantiles

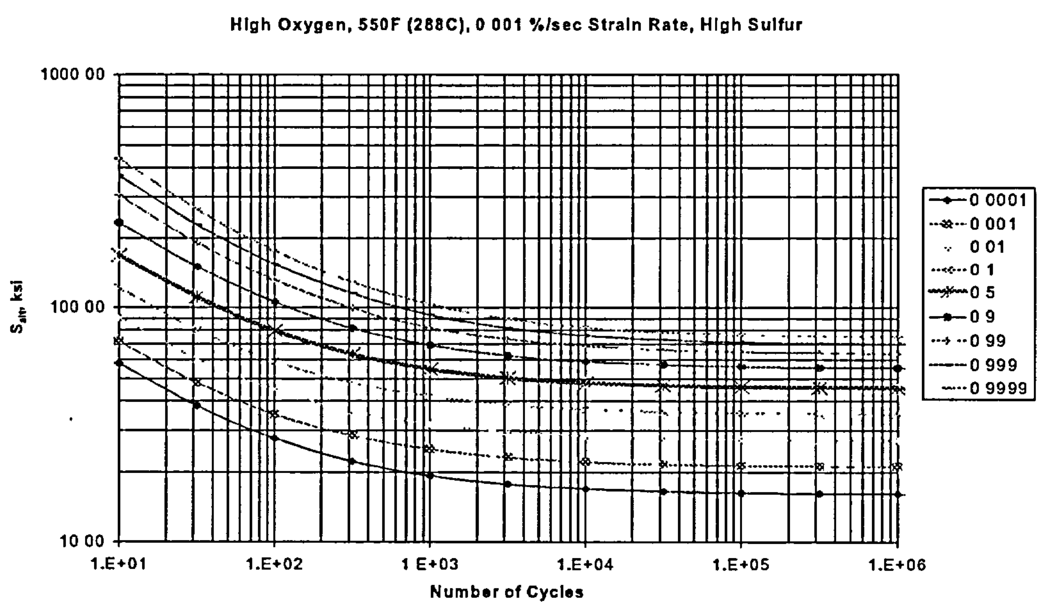

Figure 3-6
NUREG/CR-6717 Low-Alloy Steel Fatigue Curve With Original Endurance Limit Variance for Various Quantiles

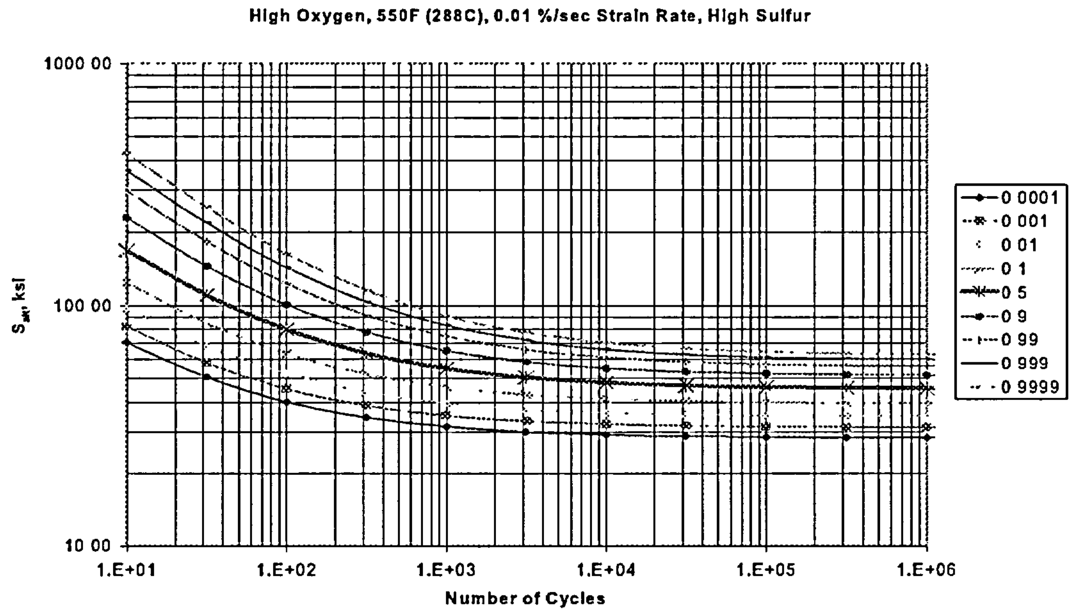

Figure 3-7
NUREG/CR-6717 Low-Alloy Steel Fatigue Curve With Modified Endurance Limit Variance for Various Quantiles

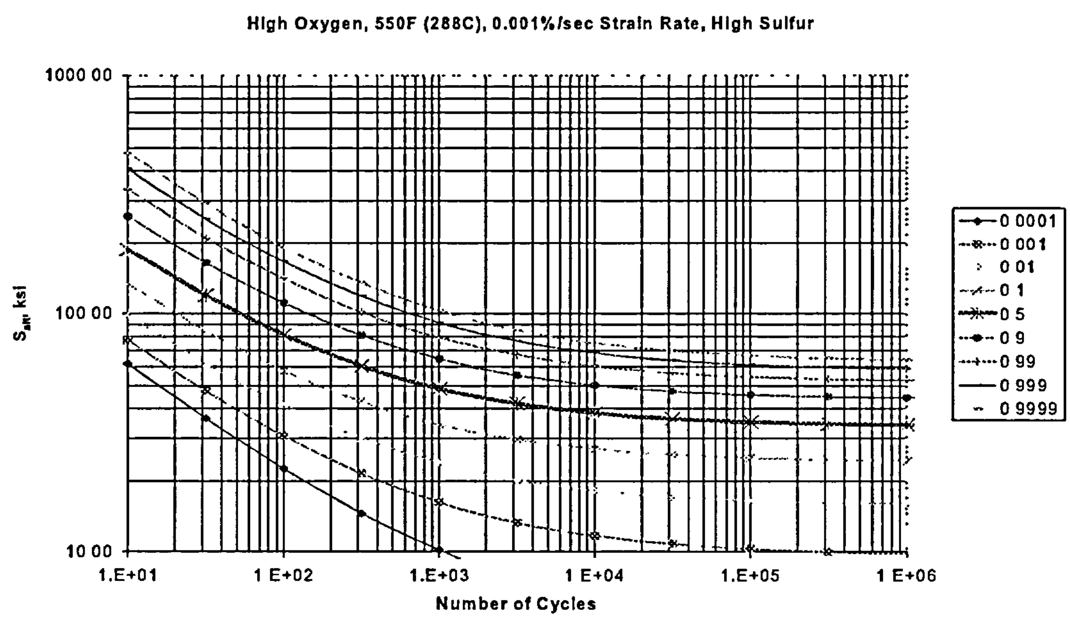

Figure 3-8
NUREG/CR-6717 Carbon Steel Fatigue Curve With Original Endurance Limit Variance for Various Quantiles

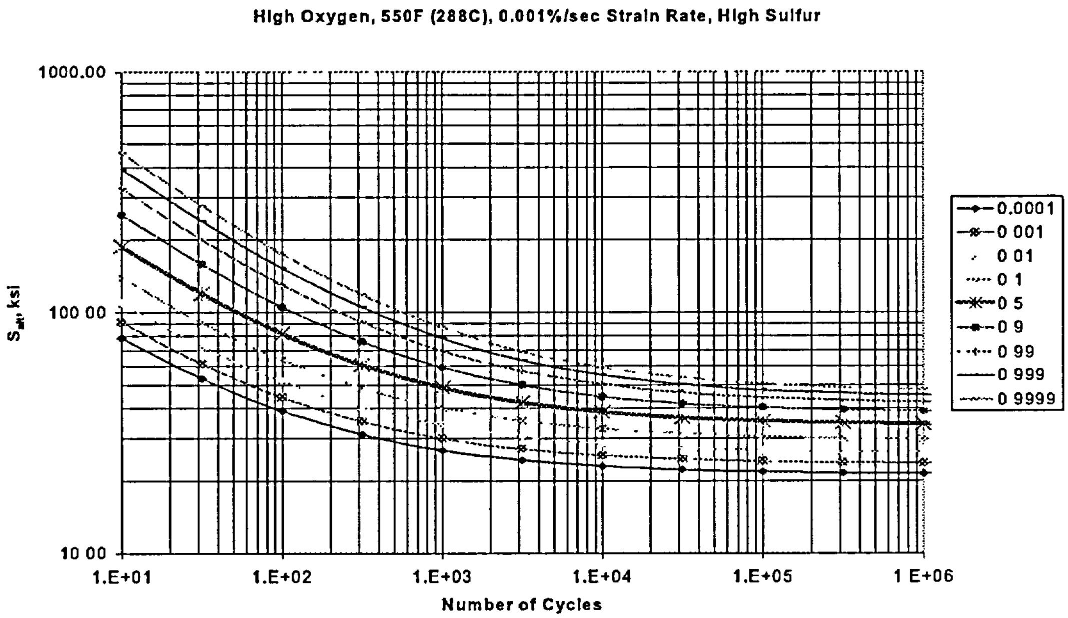

Figure 3-9
NUREG/CR-6717 Carbon Steel Fatigue Curve With Modified Endurance Limit Variance for Various Quantiles

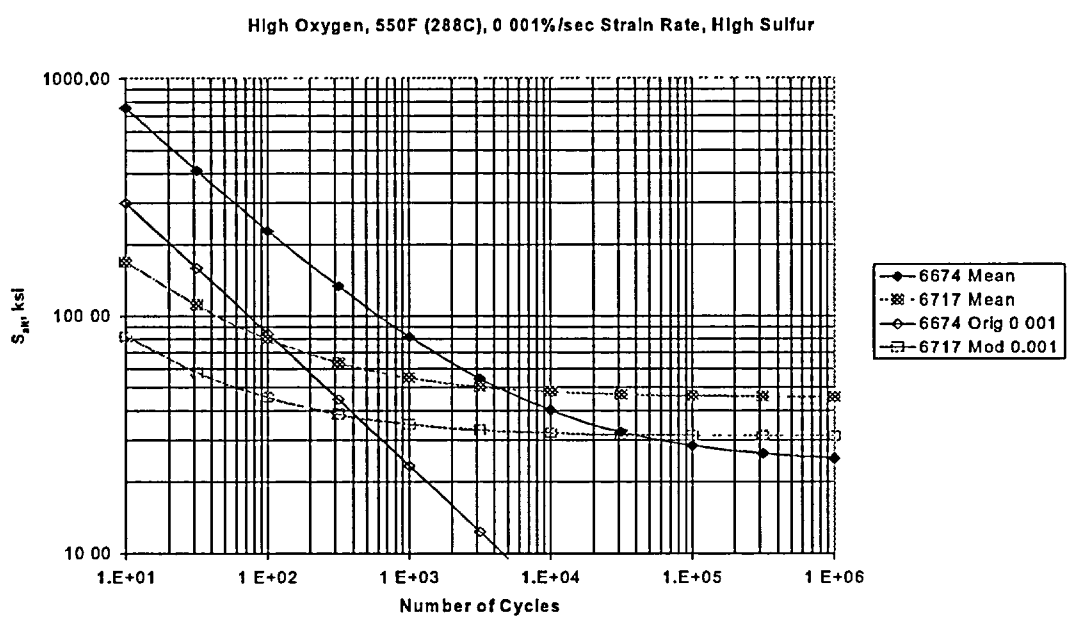

Figure 3-10
Comparison of Original to Latest Fatigue Data for Low-Alloy Steel

3.2 Other Modifications

Section 5.2 of NUREG/CR-6717 [4] describes a strain threshold for environmental effects. This amounts to setting C_{env} in Table 3-1 to zero for strain amplitudes below ε_{lo}, and increasing this from zero to the value of Table 3-1 as strains increase to ε_{hi}. C_{env} has the value in Table 3-1 for $\varepsilon > \varepsilon_{hi}$. The values of ε_{lo} and ε_{hi} are 0.07 and 0.08% for carbon and low-alloy steels. This threshold was not considered in the PNNL study. For example, Figure 3-9 shows that this threshold will not have any effect on median behavior for the new fit for carbon steel, because ε_{hi} of 0.08% (24 ksi, 165 MPa) is below ε^* of 0.113% (33.9 ksi, 228 MPa). However, it could have an effect on behavior at low probabilities, because the low quantile fatigue curves are less than the threshold values.

The strain-life relation in the very high-cycle range was modified according to procedures suggested in Section 5.2 of NUREG/CR-6717 [4]. For fatigue lives up to 10^7 cycles, the above equations are used directly. For longer lives, the following strain life relation is used:

$$\frac{\varepsilon_a}{\varepsilon_{a7}} = \left(\frac{10^7}{N}\right)^{1/100} \qquad \text{Eq. 3-4}$$

where,

ε_{a7} is the strain amplitude for 10^7 cycles

Mean stress effects were also incorporated using procedures from NUREG/CR-6717. This amounts to replacing the applied stress amplitude, σ_a, with an adjusted value when using the strain-life curve. The adjusted value, S_{adj} is:

$$S_{adj} = \begin{Bmatrix} \dfrac{\sigma_{ult}\sigma_a}{\sigma_{ult} - \sigma_{ys} + \sigma_a}, \; for\ \sigma_a < \sigma_{ys} \\ \sigma_a, \qquad for\ \sigma_a \ge \sigma_{ys} \end{Bmatrix} \qquad \text{Eq. 3-5}$$

In this equation, σ_{ys} is the yield strength and σ_{ult} is the ultimate strength. Note that there is an adjustment only if the stress amplitude is less than the yield strength.

A typical adjusted fatigue curve incorporating these effects (without the effects of the thresholds) is shown for carbon steel in Figure 3-11. The mean stress potentially has an effect for the lower quantiles. However, because the strain thresholds are at the 21 to 24 ksi (145 to 165 MPa) level, which is not much less than the yield strength (assumed to be 35 ksi [241 MPa] for this example), the net effect should be small. In fact, only the change in the standard deviation of ε^* and incorporation of the later environmental curves from NUREG/CR-6717 were found to have a significant effect, as will be demonstrated in the next section.

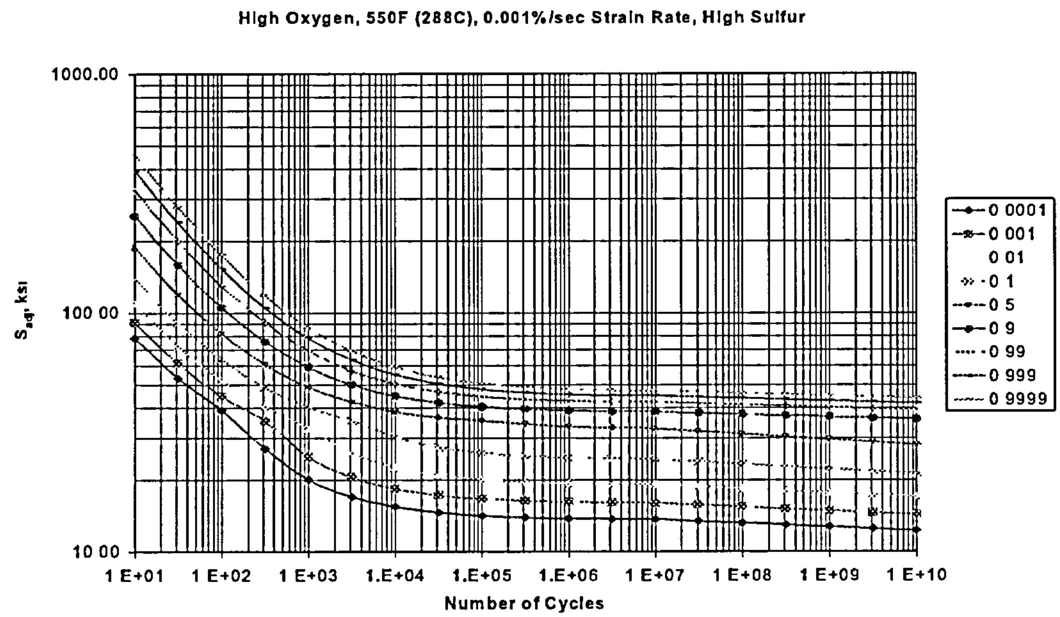

Figure 3-11
Example of Effective Fatigue Curve for Carbon Steel Based on NUREG/CR-6717 and Modified Endurance Limit Variance

4 RE-EVALUATION OF CRACK INITIATION AND LEAKAGE PROBABILITIES

4.1 Re-Run of Components With New Fatigue Correlations

PNNL supplied their input and output files for the components run with pcPRAISE as listed in Table 9.1 of NUREG/CR-6674. Some of the components with very low failure probability in NUREG/CR-6674, Table 9.1, were analyzed by Latin hypercube sampling and did not involve pcPRAISE, so they are not evaluated in this report. The components that PNNL evaluated with pcPRAISE were run again with the same inputs, thus verifying that the PNNL results could be reproduced with the modified version of pcPRAISE developed for this effort.

To evaluate the locations with the changes discussed in Section 3, the same set of components were then re-evaluated with the following modifications:

- The first modification was with the fatigue data curve fit used in NUREG/CR-6674, but with the endurance limit variation coefficient of $\varepsilon^* = 0.1$.
- The second modification was with the revised fatigue data curve fit and other modifications as suggested in NUREG/CR-6717 [4] as discussed in Section 3, plus the endurance limit variation coefficient of $\varepsilon^* = 0.1$.

Table 4-1 and Table 4-2 summarize the probability of initiation and leakage, respectively, for the carbon and low-alloy steel components that PNNL analyzed with pcPRAISE. These tables also show the results from Table 9.1 of NUREG/CR-6674. A value of 0 in the table means that no failures occurred in the 10^6 or 10^7 random sampling trials employed in the pcPRAISE analysis.

Figures 4-1 and 4-2 graphically show the results of modifying the endurance limit coefficient of variation using the fatigue data curve fit in NUREG/CR-6674. For this assessment, the value of the initiation probability can only go down as compared to the earlier PNNL results, because the only difference is that the standard deviation of ε^* has been reduced. The reduction can be small or large, depending on whether the fatigue usage is dominated by cycles near the endurance limit. The reduction in probability of leakage was up to approximately one order of magnitude.

Figures 4-3 and 4-4 graphically compare the results for the complete modification (NUREG/CR-6717 fatigue data curve fit plus endurance limit modification) to the original PNNL results. In this case, the initiation probability can increase or decrease because of shifts in the fatigue curves. For the cases of the *old GE feedwater line – RCIC tee* and the *new GE feedwater line elbow*, the initiation probability slightly increased relative to the PNNL evaluation. On the other hand, there was a significant reduction in initiation and leakage probabilities for the majority of the components. The *old GE feedwater line – RCIC tee* and the *new GE feedwater line elbow*, as well as several other components, are further evaluated in the following sections where additional conservatisms are removed from the analysis.

The above comparisons take no credit for removing any conservatisms other than the endurance limit uncertainty. The effects of the updated fatigue data curve fit are included because there have been several iterations of *new environmental fatigue correlations* since the time that NUREG/CR-6674 was completed. Specifically, NUREG/CR-6583 [7] updated the curve fits as compared to those used in NUREG/CR-6674. The differences between NUREG/CR-6583 and NUREG/CR-6717 are not judged to be significant.

In Section 3.3, further modifications were discussed that considered the use of thresholds, mean stress effects, and extension of the fatigue curves beyond 10^7 cycles. Although these changes were recommended in NUREG/CR-6717, there was no consideration of these effects in NUREG/CR-6583. To demonstrate that these effects were insignificant, a separate set of evaluations was made without these additional considerations. The changes to initiation and leakage probability are shown in Figures 4-5 and 4-6. The comparisons show that removing these effects has no effect for the locations with relatively higher initiation and leakage probabilities. Where the probabilities are low, these additional considerations conservatively predict higher initiation and leakage probabilities, thus demonstrating that their use in this study was conservative.

Table 4-1
Probability of Initiation Predictions for All Locations

Component	40-Year Life			60-Year Life		
	NUREG/ CR-6674	Endurance Limit Modified	NUREG/ CR-6717 + Endurance Limit Modified	NUREG/ CR-6674	Endurance Limit Modified	NUREG/ CR-6717 + Endurance Limit Modified
B&W RPV Outlet Nozzle	7.74E-01	1.78E-01	6.00E-06	8.99E-01	3.35E-01	1.40E-05
CE – New RPV Outlet Nozzle	4.22E-01	6 18E-02	2.50E-05	6.89E-01	1.83E-01	7.80E-05
CE – New Safety Injection Nozzle	1.01E-03	3.36E-05	0.00E+00	4.81E-03	4.38E-04	3.60E-05
CE – Old RPV Outlet Nozzle	5.91E-01	1.40E-01	1.10E-04	8.46E-01	3.62E-01	2.20E-04
GE – New Feedwater Nozzle Safe End	1.04E-01	1.23E-02	1.39E-02	2.53E-01	7.15E-02	5.33E-02
GE – New RHR Line Straight Pipe	4.73E-01	1.12E-01	5.00E-04	6.71E-01	2.63E-01	1.50E-03
GE – New Feedwater Line Elbow	1.59E-01	2.62E-02	1.40E-01	3.65E-01	1.22E-01	4.34E-01
GE – Old RPV Feedwater Nozzle Bore	7.27E-02	1.44E-02	1.13E-02	2.42E-01	7.83E-02	4.00E-02
GE – Old Feedwater Line RCIC Tee	3.76E-01	1.72E-01	7.91E-01	7.82E-01	5.62E-01	9.81E-01
W – New RPV Outlet Nozzle	8.62E-01	2.79E-01	1.60E-05	9.49E-01	4.73E-01	3.90E-05
W – Old RPV Inlet Nozzle	3.91E-01	4.64E-02	6 00E-06	6 44E-01	1.37E-01	2.00E-05
W – Old RPV Outlet Nozzle	4.90E-01	8.51E-02	2.50E-05	7.53E-01	2.38E-01	7.80E-05

Table 4-2
Probability of Leakage Predictions for All Locations

Component	40-Year Life			60-Year Life		
	NUREG/ CR-6674	Endurance Limit Modified	NUREG/ CR-6717 + Endurance Limit Modified	NUREG/ CR-6674	Endurance Limit Modified	NUREG/ CR-6717 + Endurance Limit Modified
B&W RPV Outlet Nozzle	1.83E-01	1.42E-02	2.00E-06	5.44E-01	7.78E-02	3.00E-06
CE – New RPV Outlet Nozzle	1.74E-03	8 00E-05	0 00E+00	2.90E-03	2.02E-03	1.00E-06
CE – New Safety Injection Nozzle	1.00E-06	1.00E-07	0.00E+00	1.90E-05	6.00E-07	1.00E-07
CE – Old RPV Outlet Nozzle	7.05E-02	7.00E-03	0.00E+00	3.53E-01	6.41E-02	2.00E-05
GE – New Feedwater Nozzle Safe End	1.31E-03	1.00E-04	1.00E-04	1.47E-02	1.70E-03	1.70E-03
GE – New RHR Line Straight Pipe	4.10E-01	7.59E-02	4.00E-04	6.21E-01	2.11E-01	1.40E-03
GE – New Feedwater Line Elbow	1.03E-03	0.00E+00	3.00E-04	1.46E-02	1.60E-03	1.07E-02
GE – Old RPV Feedwater Nozzle Bore	1.00E-05	5.00E-06	5.00E-06	8.80E-04	1.22E-04	1.39E-04
GE – Old Feedwater Line – RCIC Tee	2.99E-03	5.70E-04	1.69E-02	5.92E-02	1.98E-02	2.19E-01
W – New RPV Outlet Nozzle	3.65E-01	4.11E-02	1.00E-06	7.42E-01	1.79E-01	8.00E-06
W – Old RPV Inlet Nozzle	4.38E-03	1.80E-04	0.00E+00	5.04E-02	3.64E-03	2.00E-06
W – Old RPV Outlet Nozzle	9.33E-03	4.00E-04	0.00E+00	9 60E-02	9.30E-03	1.00E-06

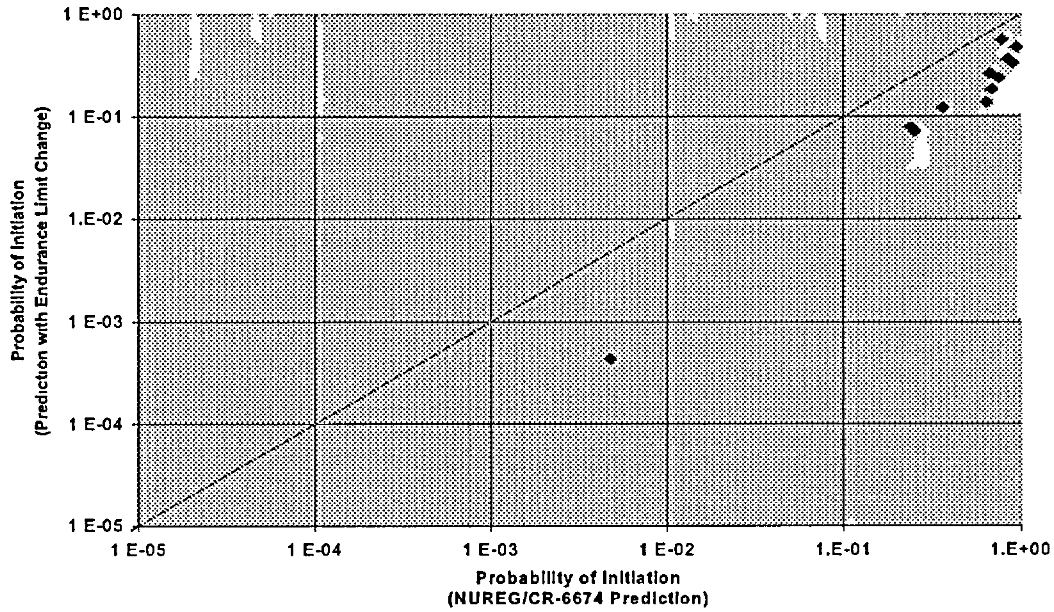

Figure 4-1
Comparison of Probability of Initiation at 60 Years—Endurance Limit Change Versus Original NUREG/CR-6674 Results

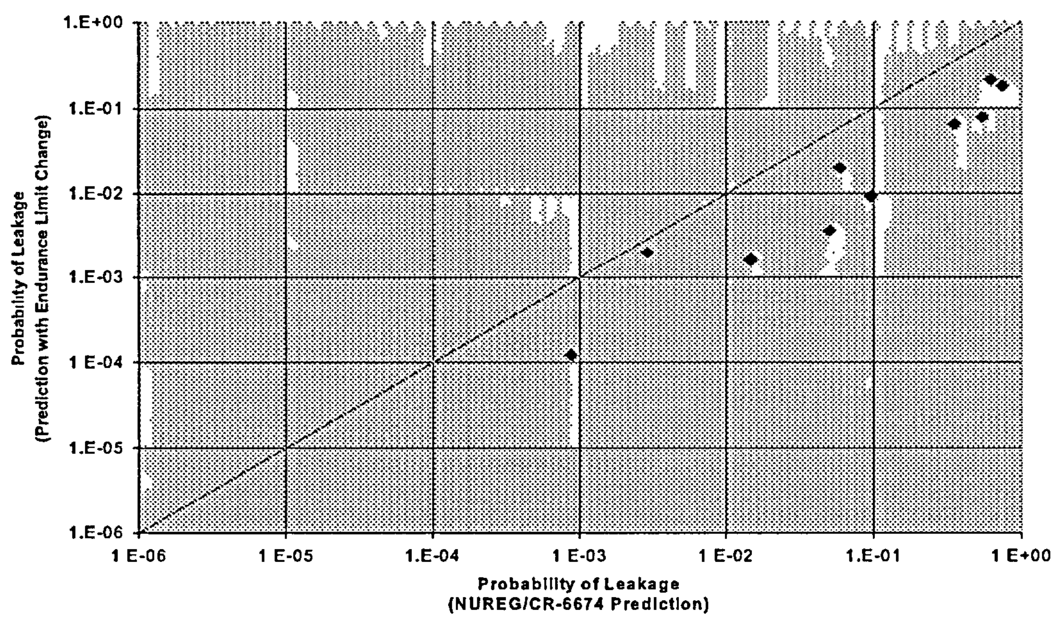

Figure 4-2
Comparison of Probability of Leakage at 60 Years—Endurance Limit Change Versus Original NUREG/CR-6674 Results

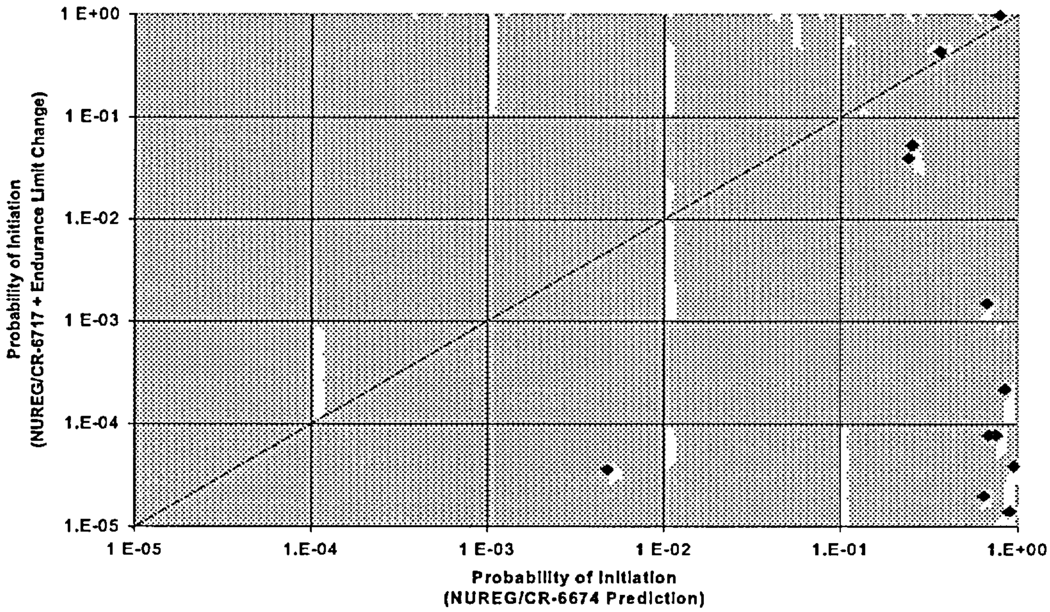

Figure 4-3
Comparison of Probability of Initiation at 60 Years—Updated Curve Fit + Endurance Limit Change Versus Original NUREG/CR-6674 Results

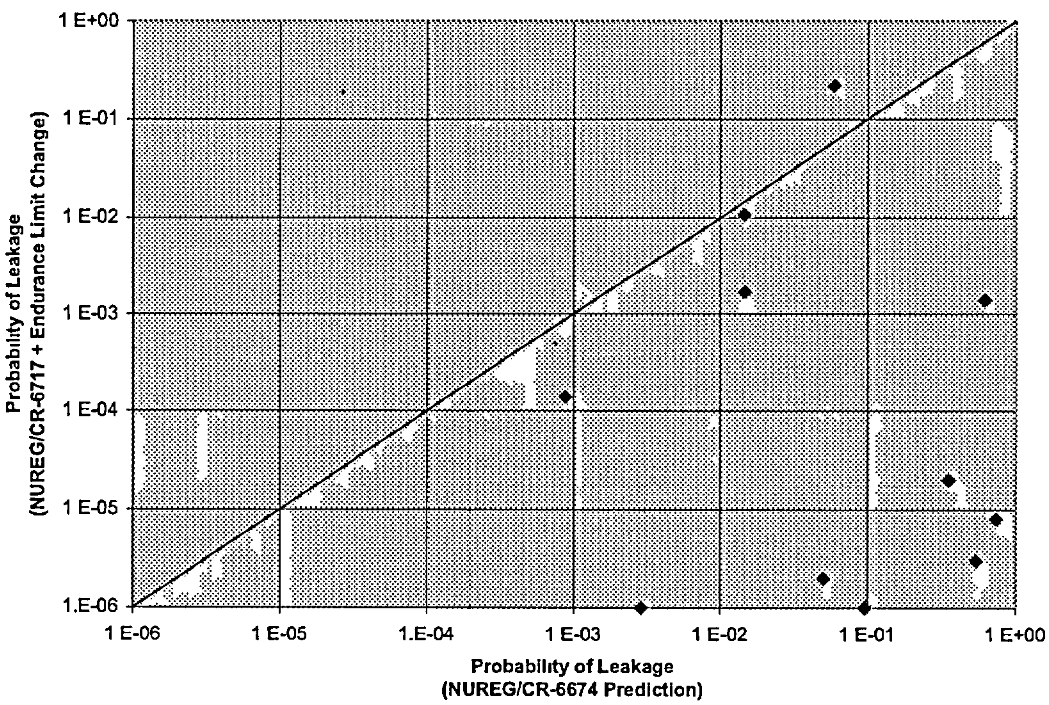

Figure 4-4
Comparison of Probability of Leakage at 60 Years—Updated Curve Fit + Endurance Limit Change Versus Original NUREG/CR-6674 Results

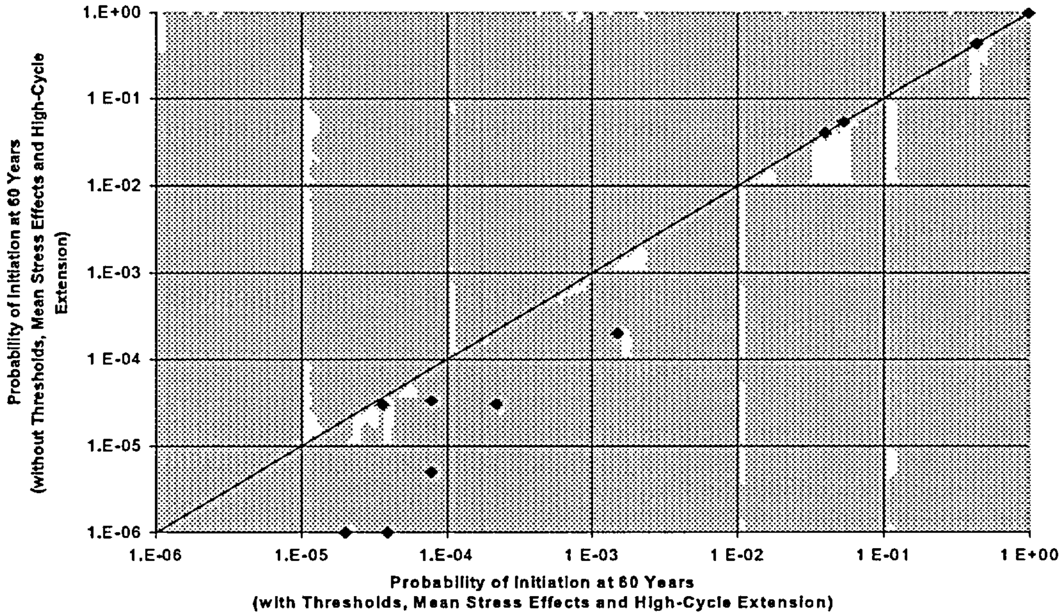

Figure 4-5
Effect of Thresholds, Mean Stress Effects, and High-Cycle Extension on Initiation Probability

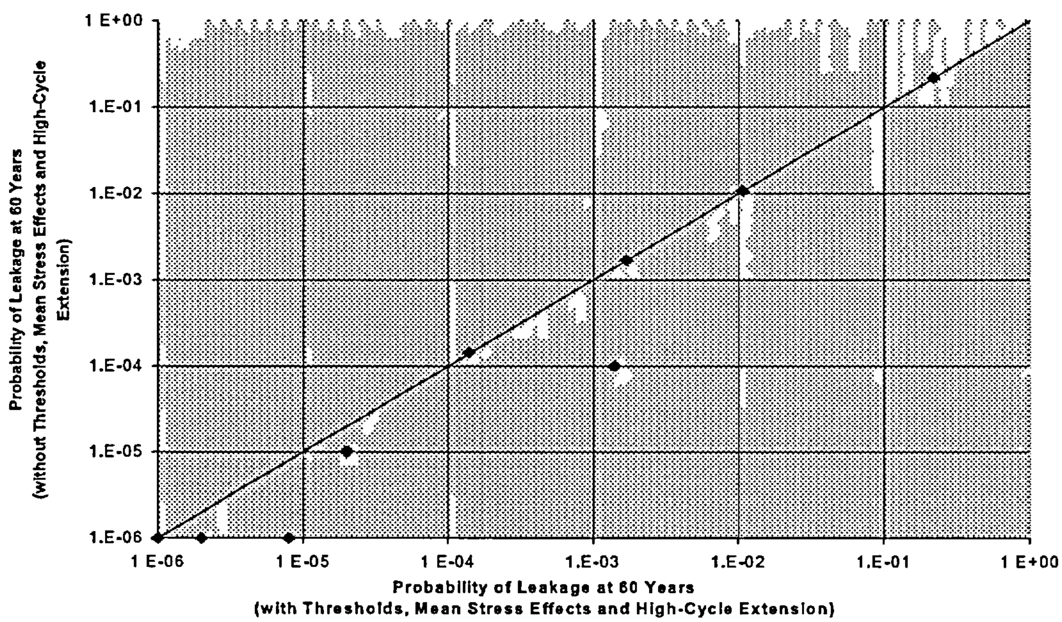

Figure 4-6
Effect of Thresholds, Mean Stress Effects, and High-Cycle Extension on Leakage Probability

4.2 Evaluation of Older-Vintage CE RPV Outlet Nozzle

The stress cycles for the older CE reactor vessel outlet nozzle are summarized on page A.8 of NUREG/CR-6674, and are included here as Table 4-3. It should be noted that the NUREG/CR-6260 load set pair identification is incorrect in that the *reactor trip/plant unloading* pair should be identified as *plant loading/plant unloading* because there are only a total of 400 reactor trips and there are 15,000 plant loading and plant unloading cycles specified for the design. This table shows that the fatigue usage is dominated by plant loading/unloading due to the extremely large number of design cycles.

Table 4-3
Fatigue Analysis of Old CE RPV Outlet Nozzle

Load Pair	Amplitude, ksi (MPa)	Number Per 40 Years	Usage
Loss of Secondary Pressure/Hydro	74.46 (513)	5	0.010
Hydrotest A/Hydrotest B	38.46 (265)	5	0.001
Heatup/Loss of Load	32.41 (223)	40	0.005
Heatup/Loss of Flow	31.73 (219)	40	0.004
Heatup/Cooldown	31.53 (217)	420	0.045
Cooldown/Plant Loading	29.70 (205)	80	0.007
Reactor Trip/Plant Loading	25.83 (178)	400	0.019
Plant Loading/Plant Unloading	23.79 (164)	14,520	0 462
		Total	0.553

Information from the original nozzle stress report from the plant, which was used by INEL, was reviewed for this component. Details of the stresses were available for all stress pairs. The stress distributions for thermal transients were not available, so the generic radial gradient distribution used in NUREG/CR-6674 was assumed. In performing the revised analysis, several changes were made to the pcPRAISE input:

- The actual nozzle thickness and diameter were used. (These were not available to PNNL, as they were not reported in NUREG/CR-6260.) Whereas PNNL used an inner diameter of 24 inches (610 mm) and a thickness of 3 inches (76.2 mm), the actual diameter is 34.6 inches (878.8 mm), with a thickness of 7.65 inches (194.3 mm) at the location where the usage factor is highest. This results in a larger number of potential initiation sites than were noted in the PNNL study. In addition, the thickness is much larger, which results in a much greater distance of potential crack growth required before leakage would be predicted.
- The component stress report provided the individual axial, hoop, and radial stresses at the inside surface of the nozzle due to pressure, thermal, pipe reaction and seismic loads. These stresses were evaluated to determine the membrane and bending stresses through the component wall. The thermal stress distribution was assumed to be identical to that used in the PNNL study. The circumferential variation of combined stresses was not significant due to the large section modulus of the nozzle.

- In the original analysis, a small seismic axial stress of ±1.6 ksi (11 MPa) was included with each heatup/cooldown cycle. For the re-evaluation, this was conservatively included for only five heatup/cooldown events. This was conservative because INEL actually eliminated the seismic cycles for many components in the NUREG/CR-6260 evaluations.
- The time of the maximum stresses from the thermal transients was included in the stress report. These were used to determine the average strain rates for the tensile transient loadings:
  - Plant unloading: 0.00757%/sec
  - Loss of flow: 0.00150%/sec
  - Loss of load: 0.00056%/sec
  - Loss of secondary pressure: 0.00292%/sec

For all other load set pairs, a value of 0.0001%/sec was assumed.

A second case was run that evaluated the expected number of cycles reported in Table 5-27 of NUREG/CR-6260. INEL had not removed this conservatism from their analysis because the usage factor with environmental effects was less than 1.0. No seismic stresses were included. This resulted in the following number of cycles being used:

- 2 hydrotest/cooldown cycles
- 101 cooldown and heatup cycles (even though two additional cooldown cycles were used with hydrotest)
- 92 reactor trip/unloading cycles
- 202 plant loading/unloading cycles (even though some of the unloading events were actually used with the reactor trips)

Table 4-4 summarizes the results of the pcPRAISE computations.

Both of the revised cases used the revised model with modified endurance limit uncertainty and the NUREG/CR-6717 modifications discussed in the previous section. The probability of initiation and leakage for the revised model was not much different from that shown in Tables 4-1 and 4-2. Use of actual cycles rendered both the probabilities of initiation and leakage negligible.

Table 4-4
Results for Old CE RPV Outlet Nozzle

Probability Case	Probability			
	NUREG/CR-6674	NUREG/CR-6717 + Endurance Limit Modified	Revised Model	Consideration of Actual Cycles
Initiation				
40 Years	0.591	1.1×10^{-4}	2.5×10^{-5}	$<10^{-6}$
60 Years	0.846	2.2×10^{-4}	7.8×10^{-4}	$<10^{-6}$
Leakage				
40 Years	0.071	$<10^{-6}$	$<10^{-6}$	$<10^{-6}$
60 Years	0.353	2.0×10^{-5}	1×10^{-6}	$<10^{-6}$

4.3 Evaluation of B&W RPV Outlet Nozzle

The design fatigue analysis for this component is given on page A.11 of NUREG/CR-6674, and is shown in Table 4-5:

Table 4-5
B&W RPV Outlet Nozzle Fatigue Analysis

Transient Pair	Amplitude, ksi (MPa)	Number in 40 Years	Usage
Heatup/Cooldown	37.96 (262)	240	0.049
Step load/Reactor trip	22.15 (153)	480	0.011
Plant loading/Unloading	17.24 (119)	48000	0.346
Other	16.69 (115)	9850	0 063
		Total	0.469

It is seen that plant loading/unloading dominates the usage, but all four transients have some usage contribution. Note that the stress amplitudes are lower than for the CE RPV outlet nozzle, and are in the range where the differences due to the standard deviation of the endurance strain amplitude can be appreciable.

Information from the stress report supplied to INEL was made available for this evaluation. The design stress analysis was conducted for a combination of pressure and thermal transient loadings at several locations along the nozzle bore. In the analysis, the piping bending stresses at the nozzle safe end regions were then conservatively added to each location, and were not adjusted for the change of section properties along the nozzle. As reported in NUREG/CR-6260, the piping stresses were reduced by a factor of 1/1.89 to account for the fact that the critical location was actually on the outside of the nozzle, and not on the inside exposed to the LWR environment. The following modifications were made to the pcPRAISE model input:

- The actual nozzle dimensions were used. The inside diameter of the nozzle was 36.62 inches (930 mm). The thickness at the location of the highest usage factor was 14.687 inches

(373 mm). The thickness at the safe end was 3.187 inches (81 mm). PNNL assumed a diameter of 24 inches (610 mm) and a thickness of 3 inches (76 mm). The thickness difference at the location of the highest usage factor is significant.

- The piping bending stress from the stress report was 27.6 ksi (190 MPa), conservatively added to the stress intensity range at all locations along the inside of the nozzle. This was reduced by the factor 1/1.89 per NUREG/CR-6260, and was further reduced to 2.602 ksi (18 MPa) to account for the nozzle section modulus increase. This was then varied around the circumference of the nozzle because its source was an applied piping bending moment. Note that an alternate case was run for the safe end location, where the stress reduction due to section modulus was not applied and, based on the stress report, the axial pressure stresses were larger and the thermal stresses were much lower.
- Evaluation of the strain rates showed that all were less than 0.001%/sec.
- The thermal stress distribution was assumed to have the same shape as used in NUREG/CR-6674.
- As suggested in NUREG/CR-6260, there are "probably no more than several hundred" of the plant loading/plant unloading transients, as compared to the 48,000 included in the design analysis (equivalent 3.28 times per day). It was conservatively estimated that there could be one per week, or 2080 in a 40-year life. Using the expected number of cycles from Table 5-44 of NUREG/CR-6260 for the heatup/cooldown of 155, the analysis was conducted with the following number of transients for 40 years:
  - Heatup/cooldown 155 cycles
  - Step load/reactor trip 480 cycles
  - Plant loading/unloading 2080 cycles
  - Other 9850 cycles

Table 4-6 summarizes the results for this component. Due to the removal of some of the conservatisms in the stress report, the probabilities of crack initiation and leakage are insignificant.

Table 4-6
Results for the B&W RPV Outlet Nozzle

Probability Case	Probability			
	NUREG/CR-6674	NUREG/CR-6717 + Endurance Limit Modified	Revised Analysis Nozzle Location	Revised Analysis Safe End Location
Initiation				
40 Years	0.774	6×10^{-6}	$<10^{-6}$	$<10^{-6}$
60 Years	0.899	1.4×10^{-5}	$<10^{-6}$	$<10^{-6}$
Leakage				
40 Years	0.183	2×10^{-6}	$<10^{-6}$	$<10^{-6}$
60 Years	0.544	3×10^{-6}	$<10^{-6}$	$<10^{-6}$

4.4 Evaluation of New GE Feedwater Nozzle Safe End

Stress reports were not available for this component; however, Table 5-109 of NUREG/CR-6260 provides the maximum temperature for each of the transient load set pairs used in the fatigue analysis. PNNL assumed a temperature of 590°F (310°C) in their evaluation. The analysis was re-run for the specific temperatures for each load set pair for the revised model including the NUREG/CR-6717 fatigue data curve fits and the modified endurance limit variation. The results are as shown in Table 4-7. Consideration of the actual temperatures reduced the probability of leakage at 60 years by more than two orders of magnitude compared to the original PNNL evaluation.

Table 4-7
Results for New GE Feedwater Safe End

Probability Case	Probability		
	NUREG/CR-6674	NUREG/CR-6717 + Endurance Limit Modified	Consideration of Actual Temperatures
Initiation			
40 Years	0.104	0.0139	0.0004
60 Years	0.253	0.0533	0.0024
Leakage			
40 Years	0 0013	0.0001	1×10^{-6}
60 Years	0.0147	0.0017	3.5×10^{-5}

4.5 Evaluation of New GE Feedwater Line Elbow

Stress reports were not available for this component; however, Table 5-123 of NUREG/CR-6260 provides the maximum temperature for each of the transient load set pairs used in the fatigue analysis. PNNL assumed a temperature of 590°F (310°C) in their evaluation. The analysis was re-run for the specific temperatures for each load set pair, with the revised model including the NUREG/CR-6717 fatigue data curve fits and the modified endurance limit variance. The results are as shown in Table 4-8. Consideration of the actual temperatures reduced the 60-year leakage probability by approximately two orders of magnitude as compared to that reported in NUREG/CR-6674.

Table 4-8
Results for New GE Feedwater Line Elbow

Probability Case	Probability		
	NUREG/CR-6674	NUREG/CR-6717 + Endurance Limit Modified	Consideration of Actual Temperatures
Initiation			
40 Years	0.159	0.140	0.0032
60 Years	0.365	0.434	0.0192
Leakage			
40 Years	0.0010	0.0003	2×10^{-6}
60 Years	0.0146	0.0107	1.8×10^{-4}

4.6 Evaluation of Old GE Feedwater Line RCIC Tee

Table 4-9 shows the fatigue analysis table for this component from NUREG/CR-6260. The analysis was conducted by INEL. It included considerable load set combinations, similar to what had been done for a similar BWR plant with an ASME Section III analysis. A large portion of the usage factor is associated with the assumed number of RCIC injection transients.

Stress reports were not available for this component; however, the RCIC tee location is far removed from the reactor vessel, so that it is not exposed to reactor temperatures. INEL reported that the maximum temperature for all transients was 392°F (200°C), whereas PNNL assumed a temperature of 590°F (310°C) in their evaluation. Therefore, the analysis was re-run for a temperature of 392°F (200°C) for each load set pair, with the revised model including the NUREG/CR-6717 fatigue data curve fits and the endurance limit modification.

In addition, PNNL assumed a 16-inch (406-mm) inside diameter and a thickness of one inch (25.4 mm) for the 20x20x8-inch (508x508x203-mm) tee (as reported by INEL). Typical BWR feedwater piping is Schedule 100, yielding a thickness of 1.281 inches (32.5 mm). Because the highest stresses in a tee are at the pipe-to-branch connection, the revised analysis was conducted with the inside diameter of the matching 8-inch (203 mm) pipe (assumed to be 8-inch Schedule 80 with an inside diameter of 7.44 inches [189 mm]) and with a thickness equal to the run pipe of 1.281 inches (32.5 mm) at the crotch of the tee.

Table 4-9
Fatigue Analysis for Old GE Feedwater Line RCIC Tee

Transient Pair	Amplitude, ksi (MPa)	Number in 40 years	Usage
Low-load set/RCIC initiation	121.95 (841)	10	0.286
Low-load set/RCIC and Reactor Water Clean-Up (RWCU) initiation	73.10 (504)	12	0.110
Low-load set/RCIC and RWCU initiation	70.78 (488)	423	3.555
Low-load set/Operating Basis Earthquake (OBE)	54.46 (375)	50	0.201
High-load set/RCIC and RWCU initiation	51.82 (357)	65	0 221
Low-load set/null	51.04 (352)	10	0.032
High-load set/null	46.88 (323)	32	0.073
High-load set/null	46.88 (323)	10	0.023
Low-load set/null	46.56 (321)	120	0.267
High-load set A/high-load set B	46.12 (318)	30	0.064
High-load set/low-load set	45.89 (316)	232	0.486
High-load set/high-load set	45.31 (312)	22	0.044
High-load set/high-load set	43.60 (301)	68	0.117
High-load set/RCIC initiation	42.58 (294)	50	0.078
High-load set/high-load set	42.25 (291)	284	0.430
Remaining 12 sets	—	1031	1.003
		Total	6.980

PNNL reported that the major contributor of stresses was due to $T_a - T_b$ (thermal discontinuity) stresses, so could not justify any increased strain rates for the component. To assess the effect of the thermal transient, a discontinuity analysis from 1.281 inches (32.5 mm) to 0.5 inches (12.7 mm) was conducted, assuming that the 8-inch (203 mm) pipe was Schedule 80 and that the transient was due to 500 gpm (1893 l/min) of 70°F (21°C) RCIC injection into a 392°F (200°C) feedwater line. The resulting thermal stress transient is shown in Figure 4-7. This transient was used to determine an effective strain rate for each transient, assuming the other contributors to the stress range were at the bounding low strain rate to produce maximum environmental effects. The revised analysis included the following changes to the loading input that had been assumed by PNNL:

- Low-Load Set/RCIC Initiation – Strain rate = 0.00144 %/sec
- Low-Load Set/RCIC and RWCU Initiation – Strain rate = 0.00204 %/sec
- Low-Load Set/RCIC and RWCU Initiation (#2) – Strain rate = 0.00201 %/sec
- Low-Load Set/RCIC and RWCU Initiation (#3) – Strain rate = 0.00355 %/sec

In addition, the number of seismic cycles associated with the low-load set/OBE load set pair was reduced to 5 from 50, while simultaneously increasing the high-load set/low-load set number of cycles by 45 to 277. This is justified because only a single cycle of a seismic event combines with another load set pair. Note that in many cases, INEL eliminated seismic cycles in the NUREG/CR-6260 evaluation.

The results are shown in Table 4-10. Whereas there was an increase in the probability of initiation and leakage for this component based on the model with the modified fatigue data curve fit of NUREG/CR-6717, the revised analysis decreased the leakage probability by approximately two orders of magnitude as compared with the original PNNL analysis. If actual loadings were available, it is expected that the leakage probability could be reduced significantly because piping stresses vary around the circumference of the pipe. INEL also admitted that an NB-3200 analysis could probably result in a significant decrease of the stresses for this component.

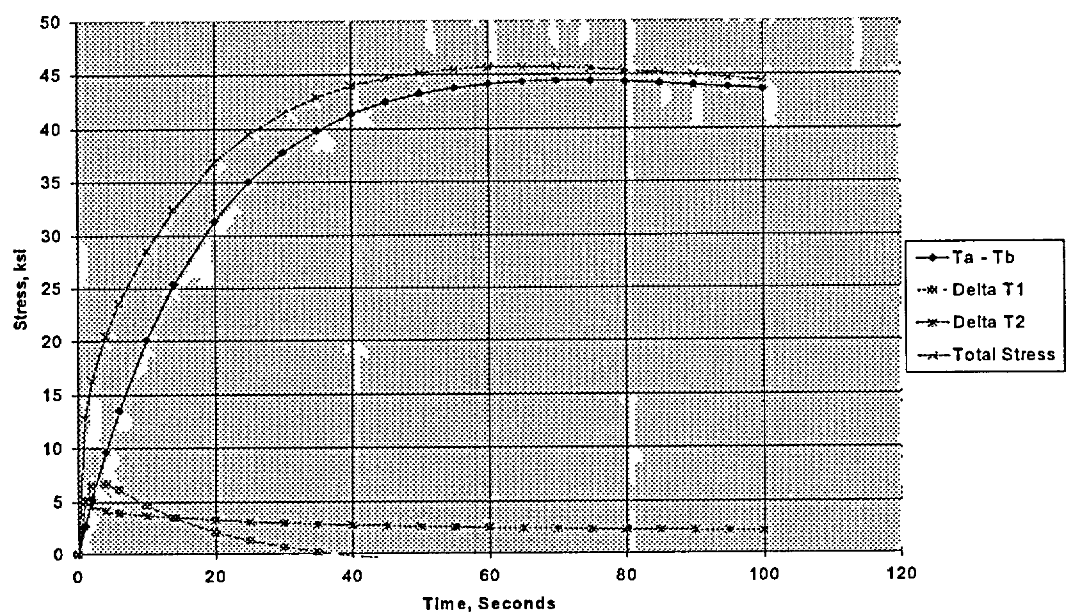

Figure 4-7
Stress Response for RCIC Injection

Table 4-10
Results for Old GE Feedwater Line RCIC Tee

Probability Case	Probability		
	NUREG/CR-6674	NUREG/CR-6717 + Endurance Limit Modified	Consideration of Actual Temperatures, OBE Cycles, and Strain Rates
Initiation			
40 Years	0.376	0.791	0.025
60 Years	0.782	0.981	0.108
Leakage			
40 Years	0.0030	0.0169	6×10^{-5}
60 Years	0.0592	0.2190	0.00139

4.7 Evaluation of New GE RHR Line Straight Pipe

Stress reports were not available for this component; however, NUREG/CR-6260 states that the maximum temperature for each load set pair was 551°F (288°C). The analysis was re-run for a maximum temperature of 551°F (288°C) for each load set pair, with the revised model including the NUREG/CR-6717 fatigue data curve fit and the modified endurance limit variation. The results are as shown in Table 4-11. Consideration of the actual temperatures reduced the probability of leakage at 60 years by slightly less than three orders of magnitude from the probabilities reported in the original PNNL evaluation.

Table 4-11
Results for New GE RHR Line Straight Pipe

Probability Case	Probability		
	NUREG/CR-6674	NUREG/CR-6717 + Endurance Limit Modified	Consideration of Actual Temperatures
Initiation			
40 Years	0.473	0.0005	4.7×10^{-4}
60 Years	0.671	0.0015	1.05×10^{-3}
Leakage			
40 Years	0.410	0.0004	0.0003
60 Years	0.621	0.0014	0.0008

4.8 Evaluation of Old GE RHR Feedwater Nozzle Bore

Stress reports were not available for this component; however, NUREG/CR-6260 provides actual minimum temperatures for each load set pair. The analysis was re-run with these temperatures for each load set pair, with the revised model including the NUREG/CR-6717 fatigue data curve fit and the modified endurance limit variation. The results are as shown in Table 4-12.

Consideration of the actual temperatures reduced the probability of leakage at 60 years slightly from the already low probabilities reported in the original PNNL evaluation.

Table 4-12
Results for Old GE Feedwater Nozzle Bore

Probability Case	Probability		
	NUREG/CR-6674	NUREG/CR-6717 + Endurance Limit Modified	Consideration of Actual Temperatures
Initiation			
40 Years	0.0727	0.0113	4.79×10^{-4}
60 Years	0.0242	0.040	2.75×10^{-3}
Leakage			
40 Years	1.0×10^{-5}	5×10^{-6}	$< 1 \times 10^{-6}$
60 Years	8.8×10^{-4}	1.39×10^{-4}	1.0×10^{-5}

4.9 Re-Evaluation of Core Damage Frequencies

NUREG/CR-6674 used failure probability results as determined in previous sections to estimate the core damage frequency (CDF). This value provides a measure of the risk contributed by failure of the component. The CDFs for the components analyzed by PNNL are included in Table 9.1 of NUREG/CR-6674. The methodology used by PNNL is described in Section 8 of NUREG/CR-6674, and a review of the equations in that section shows that the CDF is linearly proportional to the frequency of through-wall cracks (per year), which is denoted as F_{TWC}. Of the factors that enter into the CDF, the only factor changed in the current analysis is the value of F_{TWC}. Hence, the probability results used here can be combined with the PNNL results to obtain a revised value of the CDF. The following relation is used.

$$CDF_{revised} = \frac{F_{TWC(revised)}}{F_{TWC(PNNL)}} CDF_{PNNL}$$

The value of F_{TWC} is time-dependent, and is obtainable from the failure probability as a function of time, which is available from the pcPRAISE results. Denoting the leak probability as a function of time by $P_L(t)$, the value of $F_{TWC}(t)$ is given as:

$$F_{TWC}(t) = \frac{dP_L(t)/dt}{1 - P_L(t)}$$

This relation is given in Section 5.6 of NUREG/CR-6674. The derivative is obtained numerically from the pcPRAISE results. As described in NUREG/CR-6674, the derivative at 40 years is averaged over an 8-year time interval centered on 40 years, and the derivative at 60 years uses failure probabilities from 56–60 years. This procedure was also used herein, and a straight line was fit by linear least squares to the pcPRAISE values of $P(t)$ for the time intervals of interest, with the slope of this line being the derivative. This procedure is referred to as LSQ. It was not suitable in some instances, for example, where there was no change in the leak probability in the time interval of interest, even though the failure probabilities were not zero at 60 years. This could occur due to the finite number of Monte Carlo trials employed (typically 10^6). An alternative procedure was devised for this situation. All non-zero leak probabilities were considered, and the time for developing a leak was assumed to be log-normally distributed. The non-zero leak probabilities (as a function of time) were fitted by a straight line on log-normal probability scales using linear least squares. The slope and intercept of this line are related to the parameters of the log-normal distribution of leak times. The value of $F_{TWC}(t)$ at 40 and 60 years is then obtainable from the properties of a log-normal distribution and these values of the parameters. This procedure is referred to as the LN procedure.

Table 4-13 shows the results. For the ferritic components in this evaluation, the minimum core damage frequency at 60 years dropped from 1.22×10^{-7} to 7.5×10^{-11} per reactor year.

Table 4-13
Calculations of Core Damage Frequencies

	Results of Re-Evaluation					NUREG/CR-6674 Results (Table 9.1)				Re-Evaluated CDF	
	PL(40)	PL(60)	F_{TWC}(40)	F_{TWC}(60)	Method	F_{TWC}(40)	F_{TWC}(60)	CDF(40)	CDF(60)	CDF(40)	CDF (60)
B&W RPV Outlet Nozzle	0 00E-00	0 00E-00	0 0	0 0	Note 2	1 94E-02	3 35E-02	5 25E-08	9 03E-08	0 0	0 0
CE – New RPV Outlet Nozzle	0 00E+00	1.00E-06	0 0	0 0	Note 1	3 58E-04	2.57E-03	9 65E-10	6 93E-09	0 0	0.0
CE – New Safety Injection Nozzle	0 00E+00	1 00E-07	0 0	0 0	Note 1	3 75E-07	1 50E-06	1 88E-12	7 50E-12	0.0	0 0
CE – Old RPV Outlet Nozzle	0 00E+00	0 00E-00	0.0	0 0	Note 2	8 98E-03	2.27E-02	2.42E-08	6 13E-08	0 0	0.0
GE – New Feedwater Nozzle Safe End	1 00E-06	3 60E-05	1 50E-07	3.50E-06	LSQ	2 38E-04	1 23E-03	3 37E-11	1.84E-10	2 12E-14	5 23E-13
GE – New RHR Line Straight Pipe	3 00E-04	8 00E-04	2.20E-05	2 50E-05	LSQ	1.35E-02	2 25E-02	2.54E-11	2 03E-10	4.13E-14	2 26E-13
GE – New Feedwater Line Elbow	2 00E-05	1 80E-04	1 50E-06	2 00E-05	LSQ	1 69E-04	1 35E-03	3 04E-09	5 06E-09	2.70E-11	7 50E-11
GE – Old RPV Feedwater Nozzle Bore	0 00E+00	1 00E-05	2 05E-08	1.77E-06	LN	2 50E-06	9 76E-05	3.75E-14	1 46E-12	3 08E-16	2 65E-14
GE – Old Feedwater Line – RCIC Tee	6 00E-05	1 39E-03	8 50E-06	1 50E-04	LSQ	6 94E-04	5 54E-03	1 04E-10	8.30E-10	8.77E-12	2 25E-11
W – New RPV Outlet Nozzle	1 00E-06	8.00E-06	1 50E-07	1 00E-06	LSQ	3.17E-02	4.50E-02	8 57E-08	1.22E-07	4 06E-13	2.71E-12
W – Old RPV Inlet Nozzle	0 00E+00	2 00E-06	3 07E-08	2 14E-07	LN	7 53E-04	3 96E-03	2 03E-09	1 07E-08	8 28E-14	5 78E-13
W – Old RPV Outlet Nozzle	0 00E+00	1 00E-06	1.00E-07	0 0	note 1	1 56E-03	7.54E-03	4 21E-09	2 04E-08	2 70E-13	0.0

Note 1: Insufficient data for log-normal fit, only one failure in 10^6 Monte Carlo trials (LSQ used)
Note 2: No failures in Monte Carlo simulation

PL = Probability of leakage
F_{TWC} = Frequency of through-wall cracks, per year
CDF = Core damage frequency, per year
Numbers in parentheses indicate 40- or 60-year numbers

5 DISCUSSION

This report provides a re-assessment of the effects of environmental fatigue on the crack initiation and leakage probabilities of ferritic (carbon and low-alloy steel) components addressed in NUREG/CR-6260 and NUREG/CR-6674. The 25 ferritic component locations directly in contact with reactor water that were analyzed in NUREG/CR-6674 were assessed in terms of the probability of initiation and propagation of a fatigue crack. For 13 of these locations, the probability of crack initiation and leakage was extremely low (on the order of 10^{-6}) so that PNNL used a Latin Hypercube analysis to predict probabilities of through-wall cracking. These locations were not re-evaluated in this report due to their extremely low predicted probabilities of crack initiation and leakage.

The resulting probabilities of initiation and leakage from this re-evaluation are compared to results published in NUREG/CR-6674, as illustrated in Figures 5-1 and 5-2. Whereas the original PNNL study showed that there were many components with initiation probabilities greater than 0.1, Figure 5-1 shows that the revised analysis predicts only a single component (*old GE feedwater – RCIC tee*) with this high of a probability of initiation in an LWR environment. Similarly, Figure 5-2 shows that the original PNNL evaluation had many components with leakage probability greater than 0.001 and several approaching 1.0. The revised analysis predicts that the leakage probability of all but one component is less than 0.001. The exception is the *old GE feedwater – RCIC tee*, where further stress analysis would most certainly show that the leakage probability could be reduced.

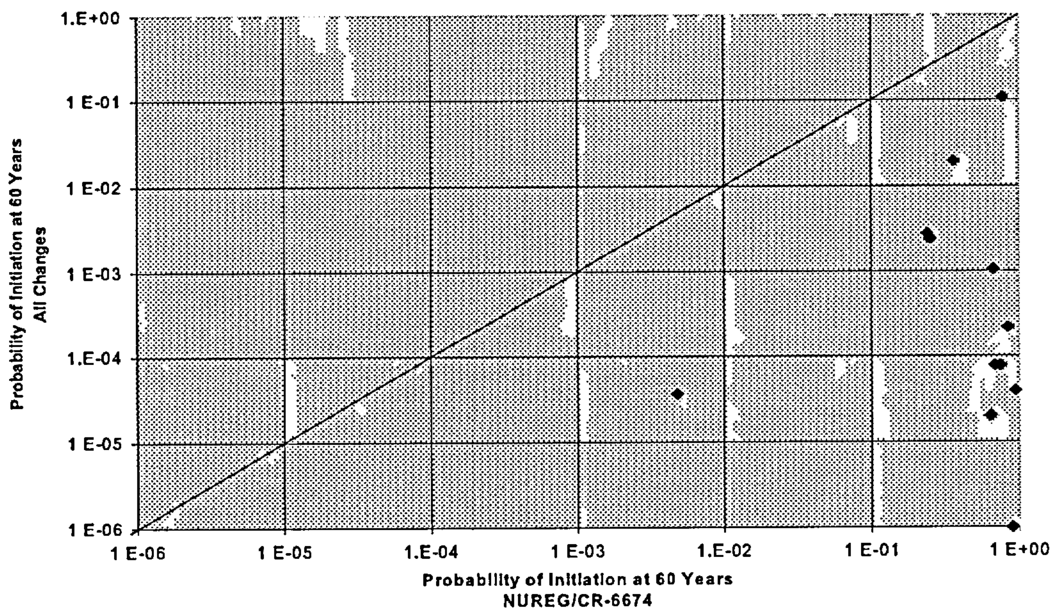

Figure 5-1
Final Comparison of Predicted Initiation Probabilities for 60 Years in LWR Environment

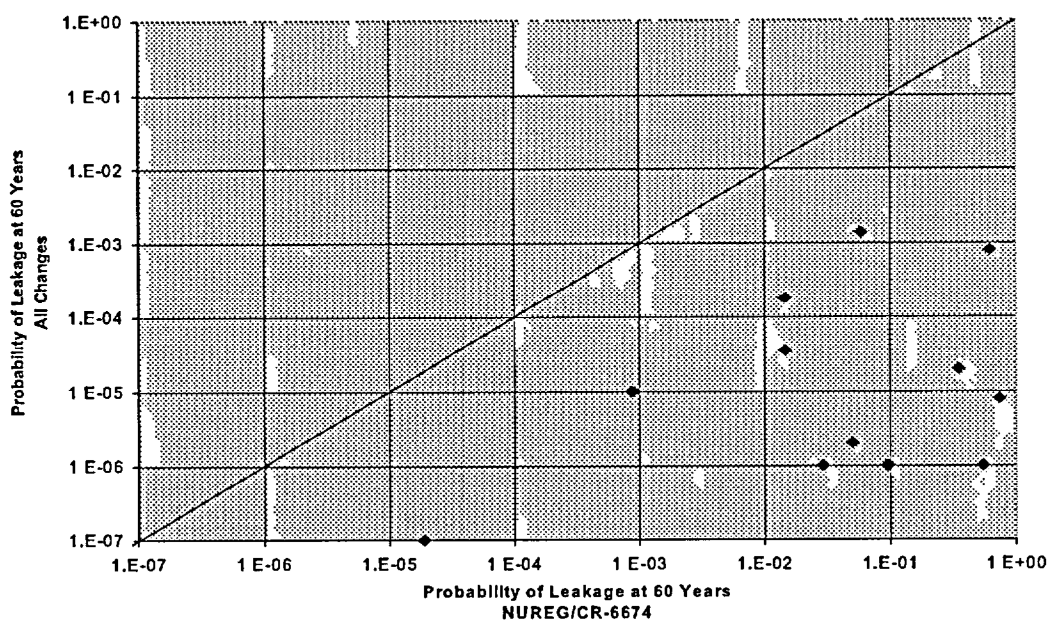

Figure 5-2
Final Comparison of Predicted Leakage Probabilities for 60 Years in LWR Environment

An important aspect of any activity that attempts to predict fatigue crack initiation and subsequent component leakage, and compares results over different time periods (for example, 40-year and 60-year), is the determination of results significance. Fatigue is a cycle-dependent (that is, time-dependent) mechanism. As operating time increases, a predicted increase in fatigue crack initiation and possible component leakage would be anticipated. However, it is important to ascertain if the absolute value of the predicted initiation and leakage, as well as the *change* in values from 40 years to 60 years is *significant*. This information is critical to concluding whether consideration of reactor water environmental effects during the license renewal period is warranted and, if so, whether current fatigue management philosophies are adequate. This implies that criteria be established to determine when the predicted initiation and leakage probabilities (including the change in these values from 40-year to 60-year) are to be considered significant.

Two measures of the potential for component failure were assessed. First, in terms of the existence of a fatigue crack, the cumulative probability of fatigue crack initiation for 40 and 60 years, as enhanced by reactor water environmental effects, was examined. In particular, if that cumulative probability is greater than 0.01 after 40 or 60 years, then there is a 1 in 100 chance that such a location would have a fatigue crack initiate in this operating period. That does not address subsequent crack detection during scheduled inservice examinations, but merely that the crack has a 1% chance of initiating.

The second measure of component failure was the potential for a through-wall leak. Here, the probability that a fatigue crack would propagate completely through the component wall thickness was examined. In particular, if the cumulative probability of through-wall cracking is greater than 0.001 in 40 or 60 years of operation, then there is one chance in 1000 that the crack will propagate completely through the wall in the operating period. Assuming that a plant has five ferritic locations with this probability of failure, then the implication is that, for 100 plants operating with these same components under the same conditions for the same operating period, one plant out of the population of 100 will have a 50% probability of a through-wall crack. In reality, only one or two out of the five assumed locations would have this high probability of leakage, so the 50% probability for a leak to occur in one plant out of the 100-plant population is very conservative.

This study did not consider any benefit from the following effects, which were not considered by PNNL or INEL in their studies:

- Many of the locations analyzed in this study are included in ASME Section XI inservice inspection programs. Although all of the specific locations exhibiting a high probability of crack initiation are not located at welds where the Section XI examinations are focused, it is expected that the scope of inspections performed would adequately detect initiated cracks and effectively manage the potential for leakage. Additionally, risk-informed inspection procedures are being implemented that specifically include candidate components evaluated in NUREG/CR-6260 in the inservice inspection program, if warranted.

- The stress input used in the PNNL study, and for many of the evaluations herein, took little credit for removing acknowledged conservatisms. There are numerous instances in NUREG/CR-6260 where INEL stated that detailed finite element analysis or accounting for actual transient severity would considerably reduce the calculated fatigue usage factors.
- The environmental conditions used by PNNL and this study used a very conservative assessment of environmental effects, including maximum temperature and most severe oxygen content for all load set pairs. Recent studies in the United States and Japan [9,10] indicate that there is an effective environmental factor that can take credit for an integrated effect of strain rate and possibly temperature in assessing the effect of light water reactor environment. Due to lack of information in existing stress reports and in NUREG/CR-6260, it was not possible to consider such effects. However, if these effects were considered, it is anticipated that the effect of reactor water environment on some of the analyzed carbon and low-alloy steel components would be further reduced.
- Recent industry studies show that the effects of environment on ferritic steel components might not be any more severe than the factor for moderate industrial environment that was included in developing the ASME fatigue curves. Industry studies have shown that the environmental factors should be reduced by a factor of three for ferritic components to account for this effect. This was not considered in the current re-evaluation.

The low crack initiation and leakage probabilities determined in this report could be significantly lower if the additional effects above were considered. In this case, the occurrence of fatigue-induced leakage due to design transients at the end of 60 years would be highly improbable.

Another way to look at the results of this evaluation is to compare the results to the air environment predictions for 40 years included in NUREG/CR-6674. These air environment results were considered by the NRC to be at low enough probabilities that no further action was required in the current licensed operation period for nuclear plants. In Figure 5-3, it is observed that there were a significant number of components where the initiation probability in air exceeded 0.1 in 40 years as reported in NUREG/CR-6674. For the current re-evaluation with LWR environment, only a single component exceeded a probability of 0.1 at 60 years, while the remaining components were at a significantly lower level. Similarly, in Figure 5-4, the PNNL study showed that most locations had a probability of leakage less than 0.001 in air after 40 years, except for one that exceeded a probability of 0.1. For the current evaluation, only one component slightly exceeded a probability of 0.001 after 60 years in the LWR environment. For both initiation and leakage, this study shows that the effects of LWR environment at 60 years are less than those published in NUREG/CR-6674 for 40 years with air environment.

Figure 5-5 compares the cumulative probability of leakage for 60 years, as determined in this study and in the original NUREG/CR-6674 analysis, to the cumulative probability of leakage for 40 years that was determined in the original NUREG/CR-6674 study. An increase in the predicted leakage from 40 to 60 years is evident from the original PNNL analysis (note that the square symbols lie above the 1:1 line). However, this increase is not due solely to the conservative consideration of environmental fatigue. Even without this consideration, an increase would be expected because fatigue is a time-related aging mechanism. (In the original analysis, the fatigue usage factor calculated for 40 years was multiplied by 1.5 to derive the predicted

60-year usage factor. This value was then used in the cumulative probability calculations.) The use of more realistic assumptions in the present study clearly demonstrates that the anticipated leakage at 60 years is less, in many cases by several orders of magnitude, than the leakage predicted to occur after 40 years in the NUREG/CR-6674 analysis. This is shown by the triangle symbols in Figure 5-5, which all lie below the 1:1 line and, in many cases, significantly below the 1:1 line. It is clear that the re-analysis performed in this study demonstrates that the effect of LWR environment is not significant and does not need to be considered in the extended operating period.

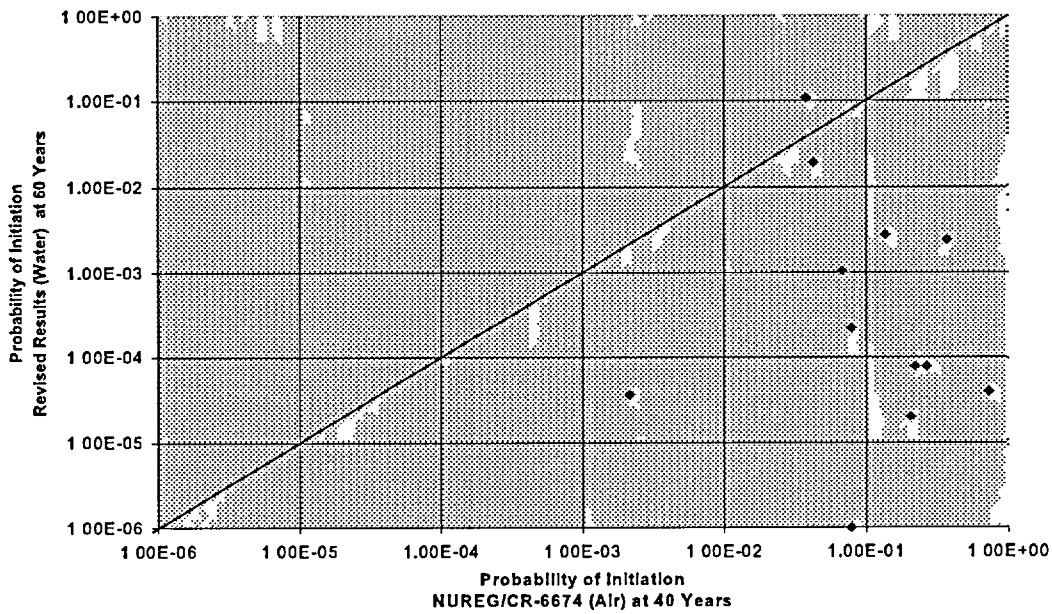

Figure 5-3
Comparison of Current Initiation Probabilities to Those for Air Environment From NUREG/CR-6674

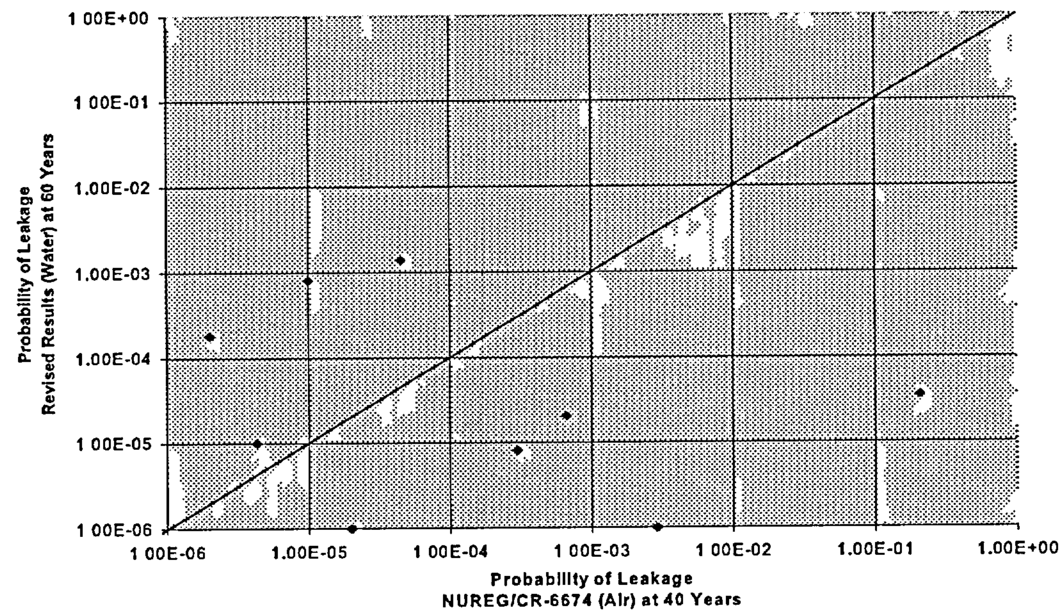

Figure 5-4
Comparison of Current Leakage Probabilities to Those for Air Environment From NUREG/CR-6674

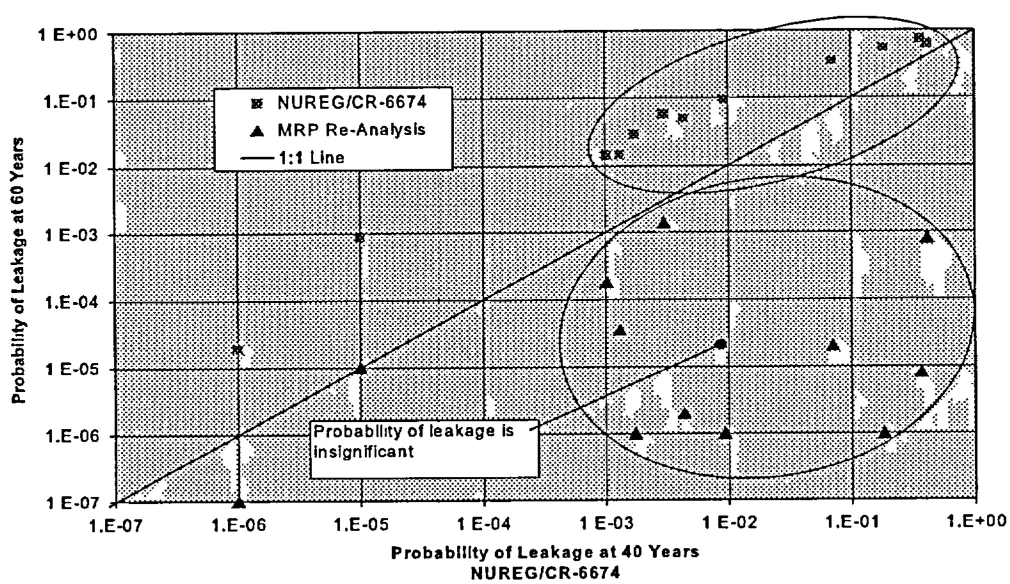

Figure 5-5
Cumulative Probability of Leakage at 60 Years Versus NUREG/CR-6674 Results at 40 Years

6
SUMMARY AND CONCLUSIONS

A re-evaluation was performed on the 40-year and 60-year probability of fatigue crack initiation and leakage for select components originally analyzed and reported in NUREG/CR-6674. This re-analysis clearly shows that there are many opportunities to reduce the conservatisms inherent in the analyses presented in NUREG/CR-6260 and NUREG/CR-6674.

The findings relative to the fatigue curve high-cycle end endurance limit variance are of special significance. By including a physically reasonable standard deviation of the endurance limit in the fatigue curves, the endurance limit alternating stress prediction agrees more with that expected from test data. This, in turn, eliminates the predicted initiation of fatigue cracks in those components that have a large number of relatively low stress transients.

The results of the latest fatigue data curve fits and recommendations in NUREG/CR-6717 were also evaluated. These show that there is generally a significant reduction in the probability of initiation and leakage with the latest curve fits as compared to that in NUREG/CR-6674. Two exceptions were noted, but utilization of temperature data published in NUREG/CR-6260 showed that leakage probabilities could be reduced by at least two orders of magnitude with no other change related to stresses or stress distributions. The initiation and leakage probabilities of several other components were reduced just by using the published maximum load set temperatures.

For the two components where specific re-evaluation of stresses was considered, the probability of leakage at the end of 60 years was reduced by at least four orders of magnitude. It is expected that this could also be accomplished at many other locations if stress report results were available.

Evaluation of the results of the revised analysis for LWR environments at 60 years against the initial results for air environment at 40 years shows that initiation and leakage probabilities are reduced. The probability of leakage at the levels predicted is so low that special evaluation of environmental effects for carbon and low-alloy steel components in a license renewal extended operating period is not warranted.

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