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Washington, DC 20555-0001

**SUBJECT:** Revision 1 to Interim Staff Guidance (ISG)-11: "Environmental Assisted Fatigue for Carbon/Low-Alloy Steel"

**PROJECT NUMBER: 690**

Enclosed is ISG-11, Revision 1, "Environmental Assisted Fatigue for Carbon/Low-Alloy Steel." This enclosure provides the information needed to resolve NRC staff issues related to this ISG.

Previous steps in addressing this issue were as follows:

- January 17, 2003, NEI submitted an ISG document for staff review entitled, "Interim Staff Guidance (ISG)-11: Environmental Assisted Fatigue for Carbon/Low-Alloy Steel."

- June 30, 2003, NRC forwarded to NEI a request for additional information (RAI) on the proposed ISG-11.

- The RAI was subsequently discussed during a public meeting held July 24, 2003, and a telecon held August 11, 2003, with members of the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Fatigue Issue Task Group (ITG).

- September 4, 2003, NEI submitted a response to the NRC RAI, "Response to NRC Request for Additional Information on Proposed Interim Staff Guidance (ISG)-11: Environmental Assisted Fatigue for Carbon/Low-Alloy Steel." In the submittal letter it was indicated that the proposed ISG-11 would be revised as a result of the additional analyses performed in response to the RAI and forwarded to the NRC.

The overall conclusion of the ISG remains that 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 for carbon and low-alloy steel components for either PWR or BWR components.

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Rec'd 11/19/03

We look forward to early staff acceptance of ISG-11 Revision 1. If you have any questions, please contact me (202-739-8080 or am@nei.org) or Fred Emerson (202-739-8086 or fae@nei.org).

Sincerely,

Alex Marion

Alexander Marion

Enclosure

## INTERIM STAFF GUIDANCE-11, Revision 1

## ENVIRONMENTAL ASSISTED FATIGUE FOR CARBON/LOW-ALLOY STEEL

### EXECUTIVE SUMMARY

Recommendations are provided below in the form of Interim Staff Guidance (ISG) that modify 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 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), and supplemented with additional analyses performed in response to questions received from the NRC staff are summarized here.

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 both 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 the probabilities of through-wall cracking at 60 years downward to less than 0.001 except for one component with a probability of leakage of 0.006.*** 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 is 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 or updated 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*, as supplemented by NEI Memorandum, A. Marion, to P.T. Kuo, Program Director, Office of Nuclear Reactor Regulation, *Response to NRC Request for Additional Information on Proposed Interim Staff Guidance (ISG)-11: Environmental Assisted Fatigue for Carbon/Low-Alloy Steel*, September 4, 2003."

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.<br><br>***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 and the additional analyses performed. 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,*** **as supplemented by NEI Memorandum, A. Marion, to P.T. Kuo, Program Director, Office of Nuclear Reactor Regulation,** ***Response to NRC Request for Additional Information on Proposed Interim Staff Guidance (ISG)-11: Environmental Assisted Fatigue for Carbon/Low-Alloy Steel,*** **September 4, 2003".**

ATTACHMENT 1

INTERIM STAFF GUIDANCE-11
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 Summary
- Section 7.0 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 EPRI in 1982 [7], based on carbon steel piping component tests performed by the General Electric Company 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. The results of this review raise concern regarding the applicability of laboratory environmental fatigue data to operating environments and provides a persuasive body of information in support of the probabilistic re-calculations discussed in Section 5.0. However, the technical basis for ISG-11, Rev. 1 does not rely upon this information for resolution of the environmental fatigue issue for carbon and low-alloy steel locations.

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.

In addition to the effort documented in MRP-74, additional analyses were performed in response to an NRC Request for Additional Information (RAI) issued on June 30, 2003 [20]. The MRP response to the RAI was forwarded to the NRC on September 4, 2003 [21]. Relevant results from these additional analyses are incorporated herein where appropriate, and are referred to as "revised analysis."

## 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°F was used in NUREG/CR-6674. Assumed through-wall stress distributions were also used in NUREG/CR-6674. The evaluations documented in MRP-74 used more realistic alternatives for these assumptions, and also updated the probabilistic calculations to incorporate more recent laboratory fatigue data [22,23,24].

The MRP-74 report was the principal element of the technical basis supporting ISG-11 and was submitted to the NRC in January 2003 [25]. In response to the initial submittal of ISG-11 and the MRP-74 report, several questions were provided by the Staff [20]. Based on these questions, the pcPRAISE [26] model for the probabilistic representation of fatigue curves was revised to assure that size/geometry and surface finish effects, and mean stress effects were addressed in a manner consistent with that described in NUREG/CR-6335 [27]. The relevant results of the revised analysis are incorporated into this Revision 1 of ISG-11. Details of the revised analysis are available in the response to the NRC RAI [21].

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 all having 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 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 and compare the original and realistic endurance limit variation to existing environmental data [21]. For example, Figure 1 shows the probabilistic fatigue curve for carbon steel in air compared to the air data, while Figure 2 shows the same data compared to the modified 10-percent endurance limit variation used in the revised analysis. Other similar comparisons are shown in Figures 3 to 8. Although there are insufficient data to mathematically perform regression analysis to determine the revised endurance limit, it is clear that there are essentially no data below the 5 percent lower bound probability quantile at the high-cycle end of the curves with the modified approach.

In addition to the above modifications, 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. For several components, the daily loading/unloading cycles considered in the original stress analysis were reduced to once per week, still very conservative compared to the base-loading approach used at all nuclear plants in the United States.

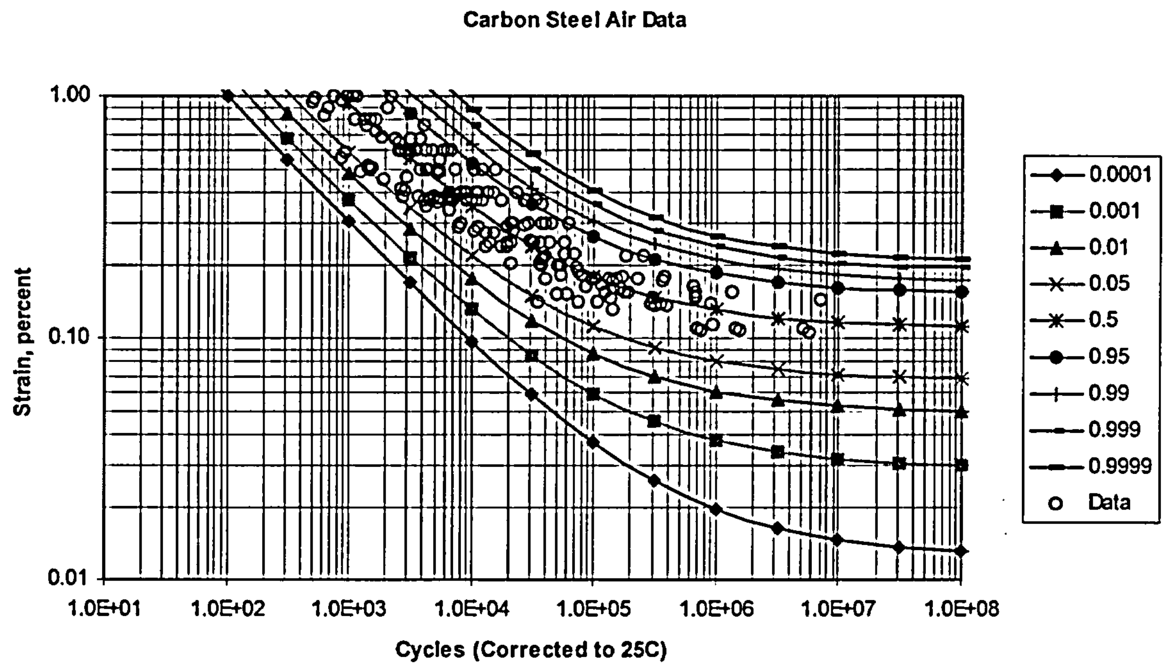


**Figure 1. ANL Data for Carbon Steel in Air – ANL Endurance Limit Variation**

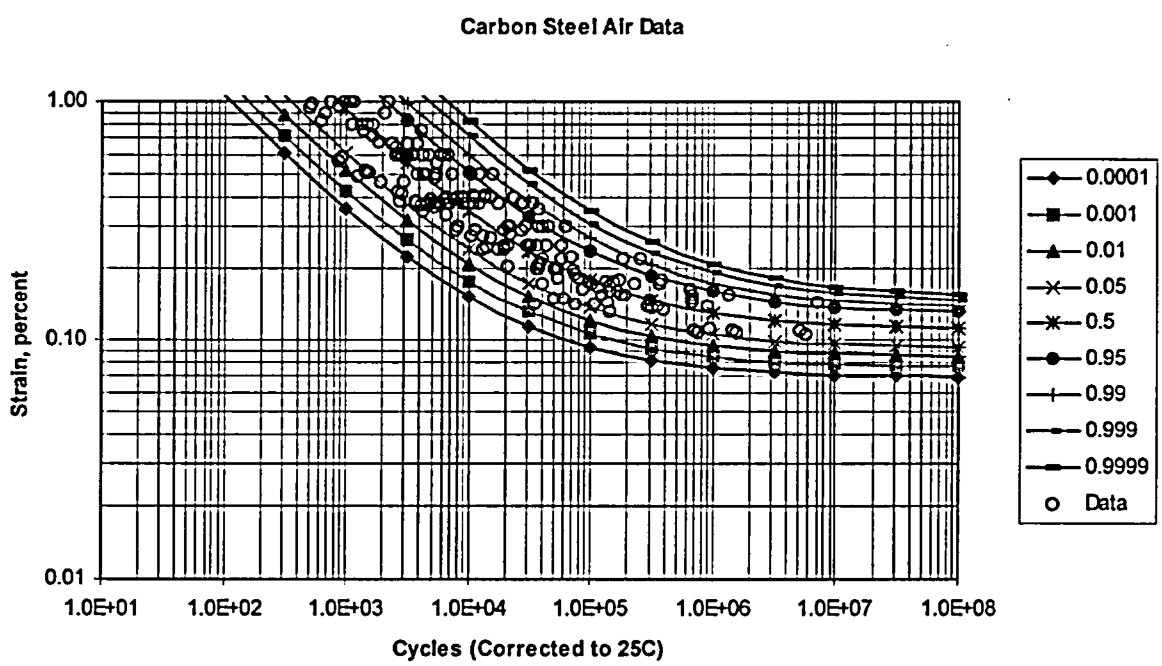


**Figure 2. ANL Data for Carbon Steel in Air – 10 Percent Endurance Limit Variation**

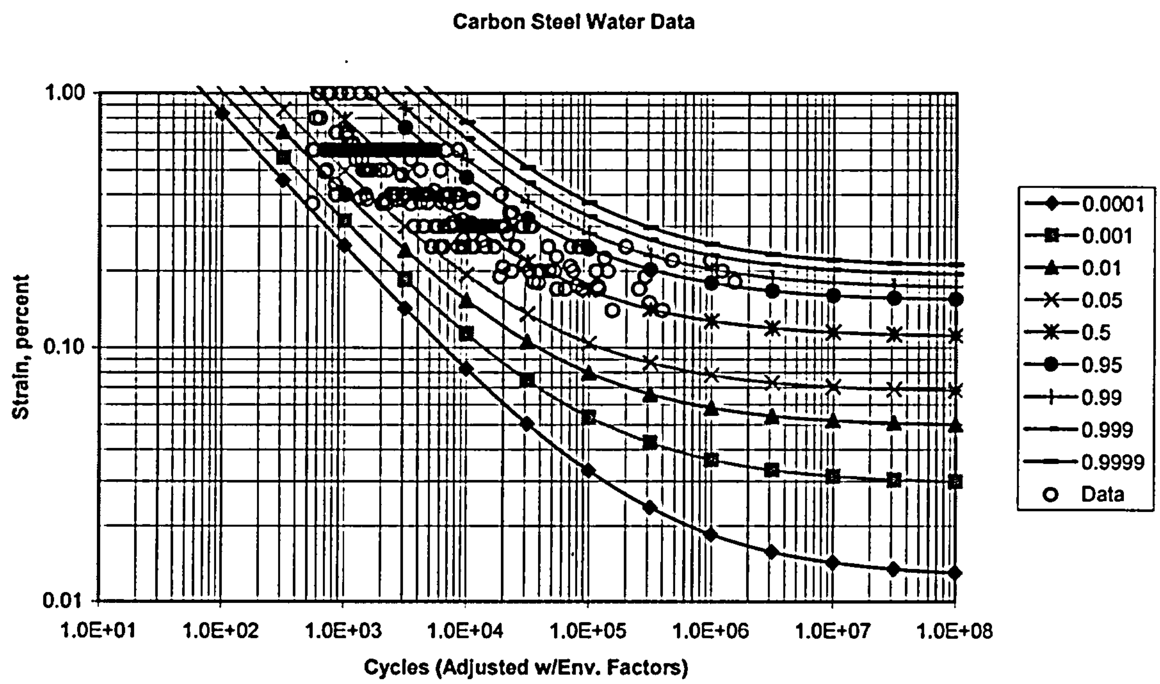


**Figure 3. ANL Data for Carbon Steel in Water – ANL Endurance Limit Variation**

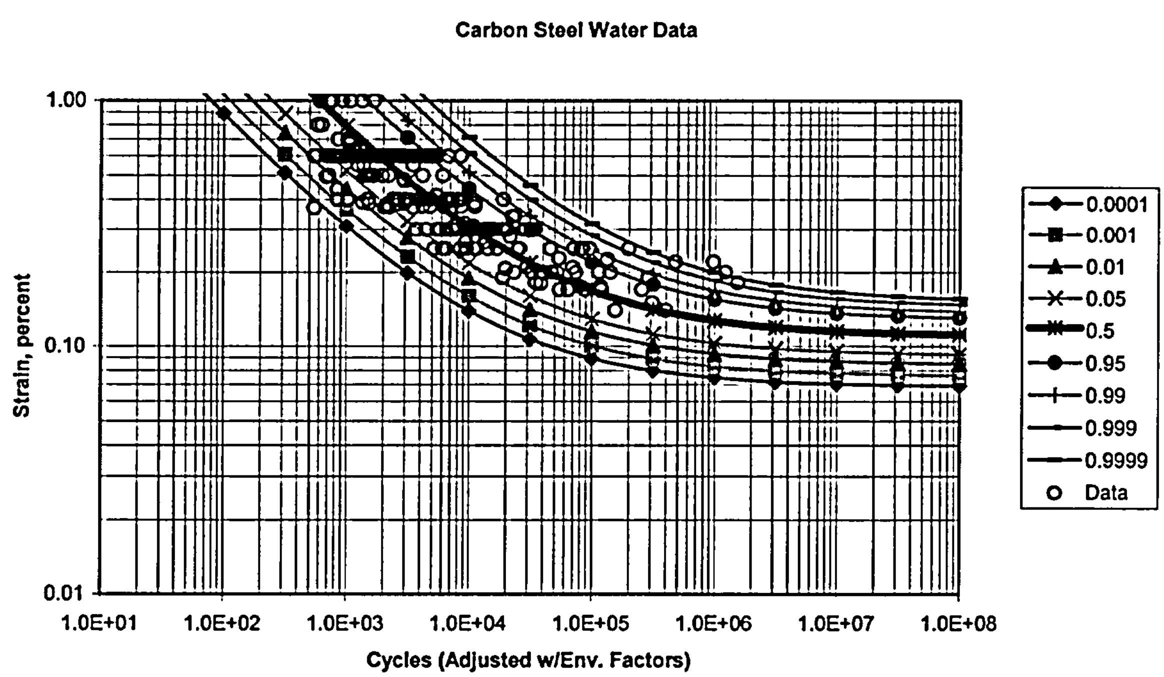


**Figure 4. ANL Data for Carbon Steel in Water – 10 Percent Endurance Limit Variation**

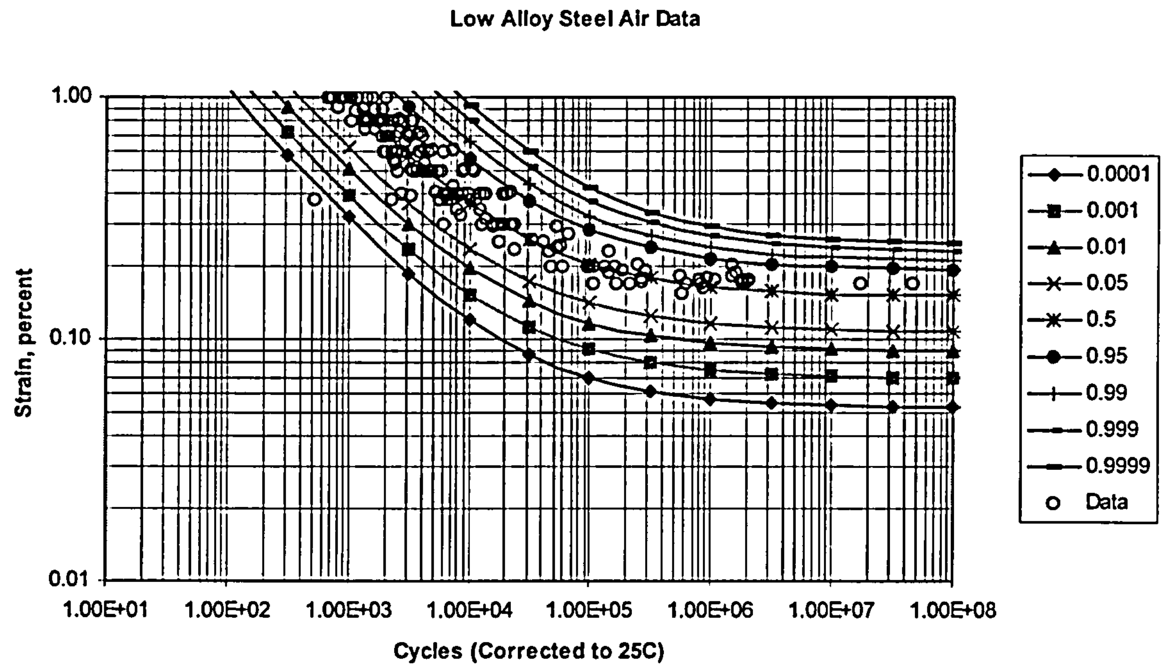


**Figure 5. ANL Data for Low Alloy Steel in Air – ANL Endurance Limit Variation**

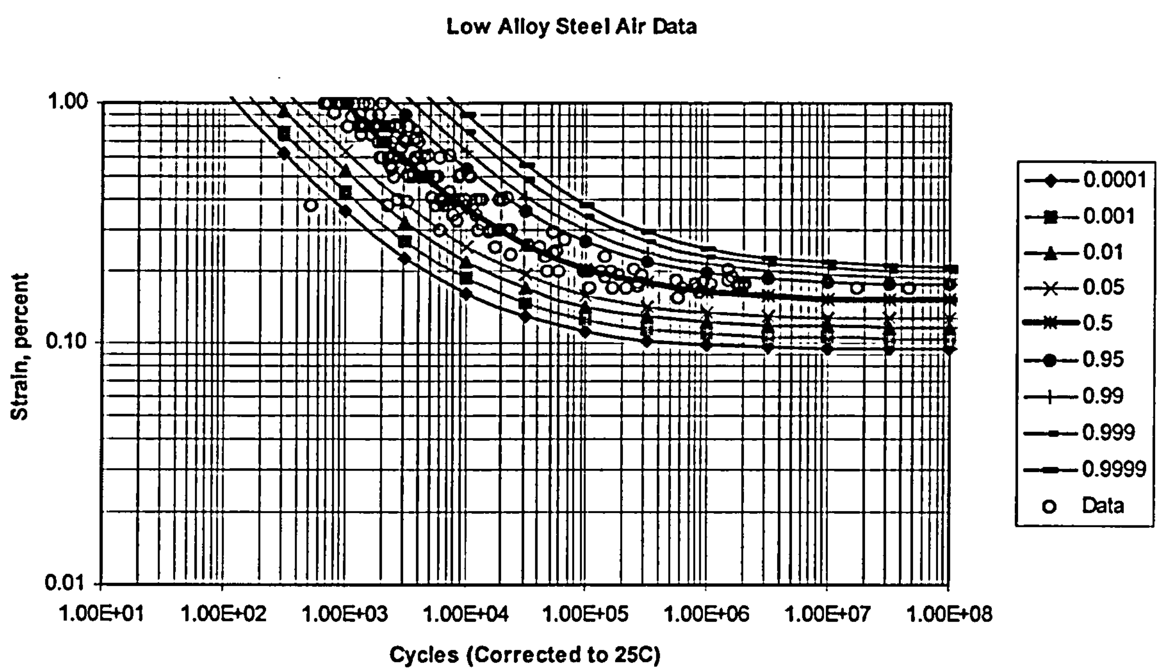


**Figure 6. ANL Data for Low Alloy Steel in Air – 10 Percent Endurance Limit Variation**

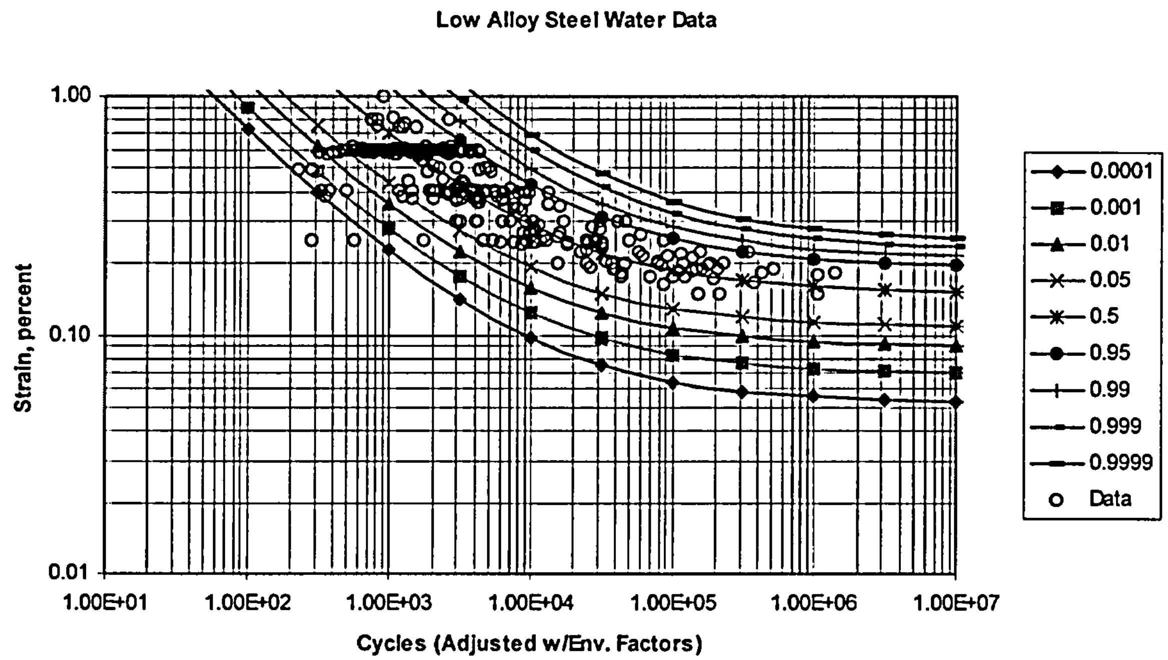


**Figure 7. ANL Data for Low Alloy Steel in Water – ANL Endurance Limit Variation**

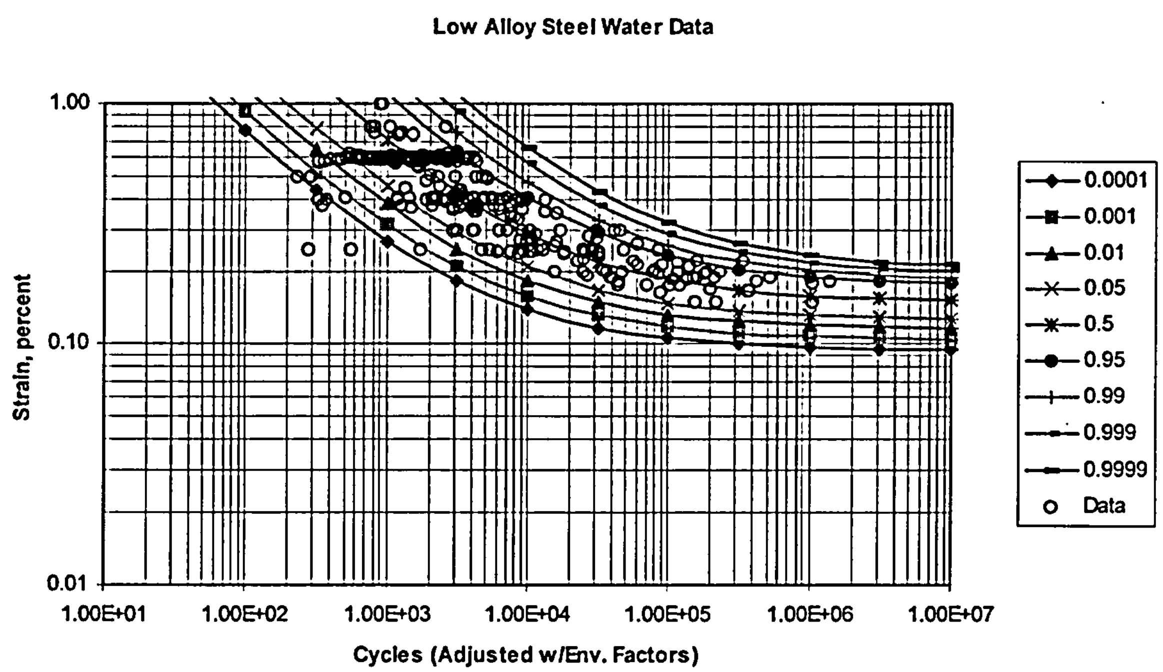


**Figure 8. Data for Low Alloy Steel in Water – 10 Percent Endurance Limit Variations**

Table 1 compares the 40-year and 60-year calculated results for cumulative probability of leakage from NUREG/CR-6674 and the revised analysis [21]. Only one location, the RHR

straight pipe in the feedwater line of a newer vintage BWR, has a 60-year cumulative through-wall leakage probability above the threshold value of 0.001 (0.00634, see Table 1). However, there was no opportunity to evaluate any reduction in conservatisms for this component both in NUREG/CR-6260 and the revised analysis since there were a very high number of cycles defined and the transients were not well identified. If the conditions of the analysis were actually known, the probability of leakage could be reduced consistent with other components.

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

| Component | 40 Year Life | | 60 Year Life | |
|---|---|---|---|---|
| | NUREG/CR- 6674 | Revised Analysis | NUREG/CR- 6674 | Revised Analysis |
| B&W RPV OUTLET NOZZLE | 1.83E-01 | < 1.00E-05 | 5.44E-01 | 1.00E-05 |
| CE-NEW RPV OUTLET NOZZLE | 1.74E-03 | <1.00E-05 | 2.90E-02 | 4.00E-05 |
| CE-NEW SAFETY INJECTION NOZZLE | 1.00E-06 | <1.00E-05 | 1.90E-05 | <1.00E-05 |
| CE-OLD RPV OUTLET NOZZLE | 7.05E-02 | <1.00E-05 | 3.53E-01 | <1.00E-05 |
| GE-NEW FEEDWATER NOZZLE SAFE END | 1.31E-03 | <1.00E-05 | 1.47E-02 | <1.00E-05 |
| GE-NEW RHR LINE STRAIGHT PIPE | 4.10E-01 | 1.47E-03 | 6.21E-01 | 6.34E-03 |
| GE-NEW FEEDWATER LINE ELBOW | 1.03E-03 | <1.00E-05 | 1.46E-02 | <1.00E-05 |
| GE-OLD RPV FEEDWATER NOZZLE BORE | 1.00E-05 | <1.00E-05 | 8.80E-04 | 1.05E-04 |
| GE-OLD FEEDWATER LINE – RCIC TEE | 2.99E-03 | <1.00E-05 | 5.92E-02 | 2.00E-05 |
| W-NEW RPV OUTLET NOZZLE | 3.65E-01 | 4.00E-05 | 7.42E-01 | 2.90E-04 |
| W-OLD RPV INLET NOZZLE | 4.38E-03 | <1.00E-05 | 5.04E-02 | 1.00E-05 |
| W-OLD RPV OUTLET NOZZLE | 9.33E-03 | 1.00E-05 | 9.60E-02 | 9.00E-05 |

Figures 9 and 10 illustrate the principal results from this study. Figure 9 compares the cumulative probability of initiation at 40 and 60 years from NUREG/CR-6674 to that from the revised analysis [21]. A decrease in cumulative probability of initiation of up to approximately four orders of magnitude is evident from the EPRI MRP re-analysis. Figure 10 compares the cumulative probability of leakage at 40 and 60 years from NUREG/CR-6674 to that from the revised analysis. Note that only revised leakage probability values greater or equal to $10^{-5}$ are shown. All other results are lower than $10^{-5}$, in many cases significantly lower. 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.

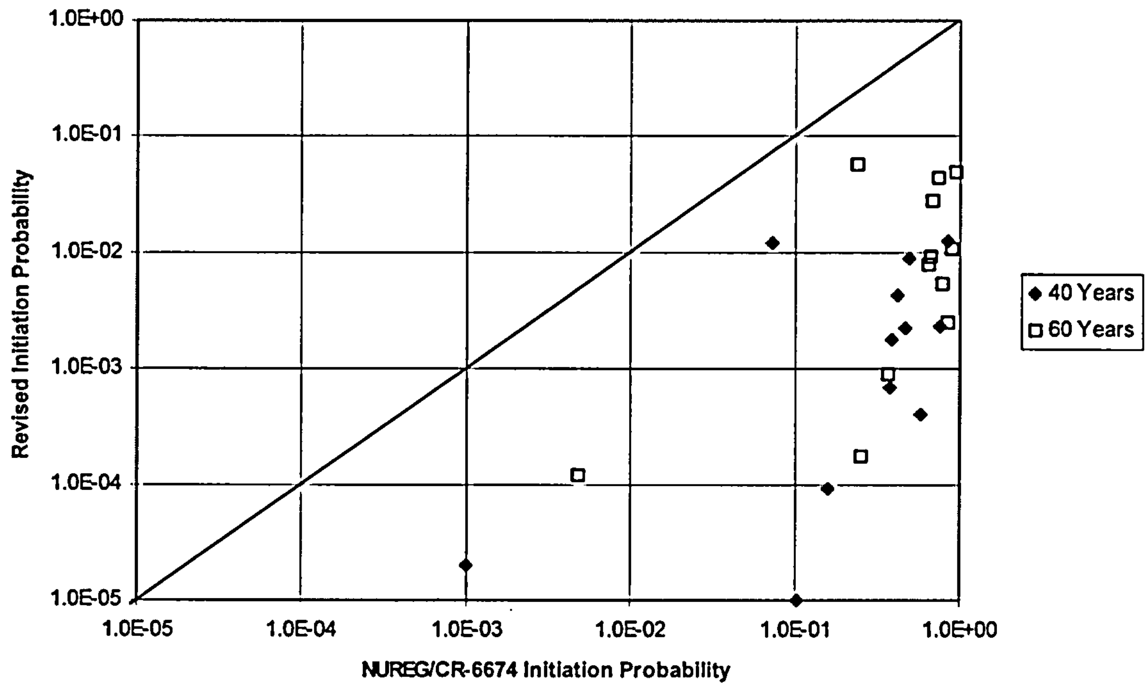


**Figure 9. Comparison of Initiation Probabilities – Revised Analysis versus NUREG/CR-6674**

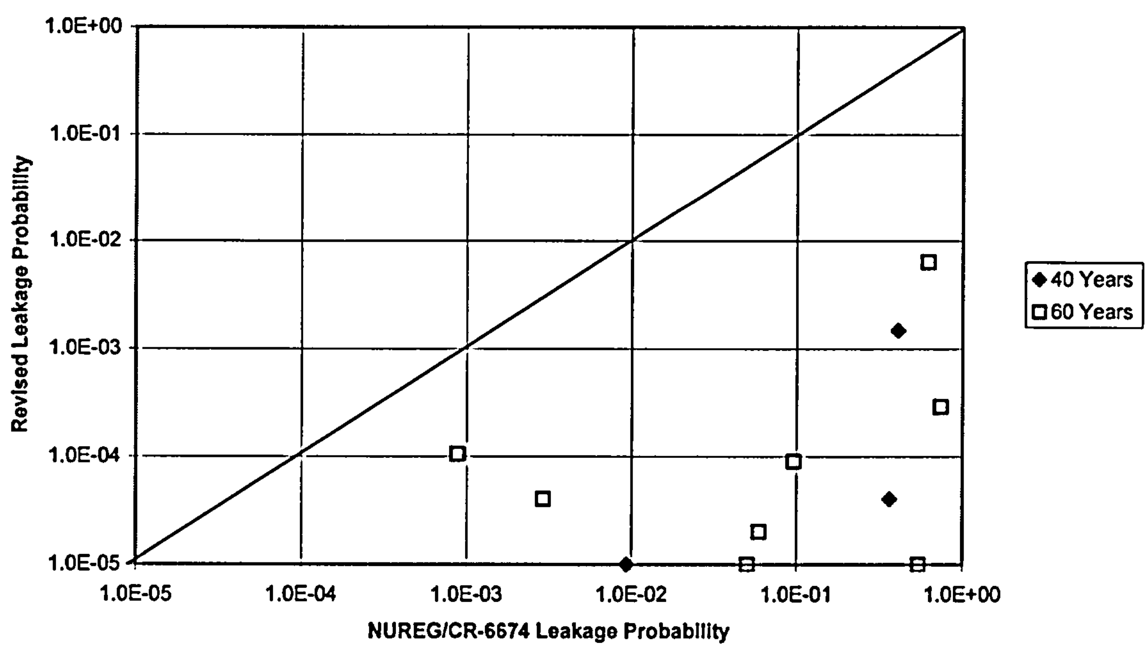


**Figure 10. Comparison of Leakage Probabilities – Revised Analysis versus NUREG/CR-6674**

This conclusion is further illustrated in Figures 11 and 12. Figure 11 provides a comparison between the cumulative probabilities of initiation at both 40 and 60 years reported in NUREG/CR-6674 and the revised analysis [21]. It is evident that the consideration of more realistic assumptions in the analysis reduces the cumulative probability of initiation, in most cases significantly. Figure 12 provides the same comparison for cumulative probabilities of leakage. The reduction in cumulative probability of leakage is more pronounced when realistic assumptions are considered.

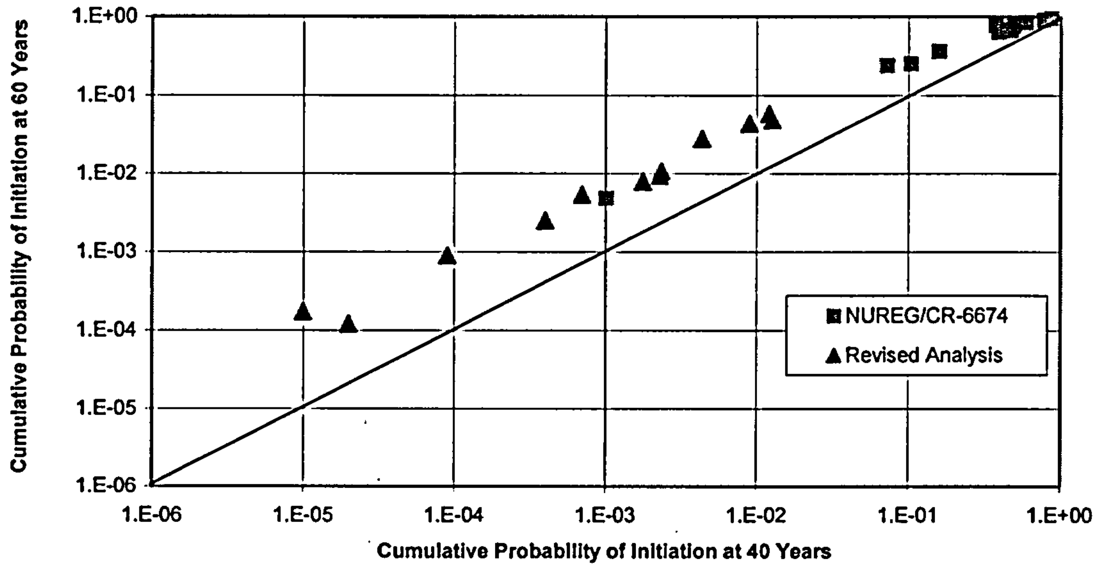


Figure 11. Cumulative Probability of Initiation at 60 Years Versus 40 Years

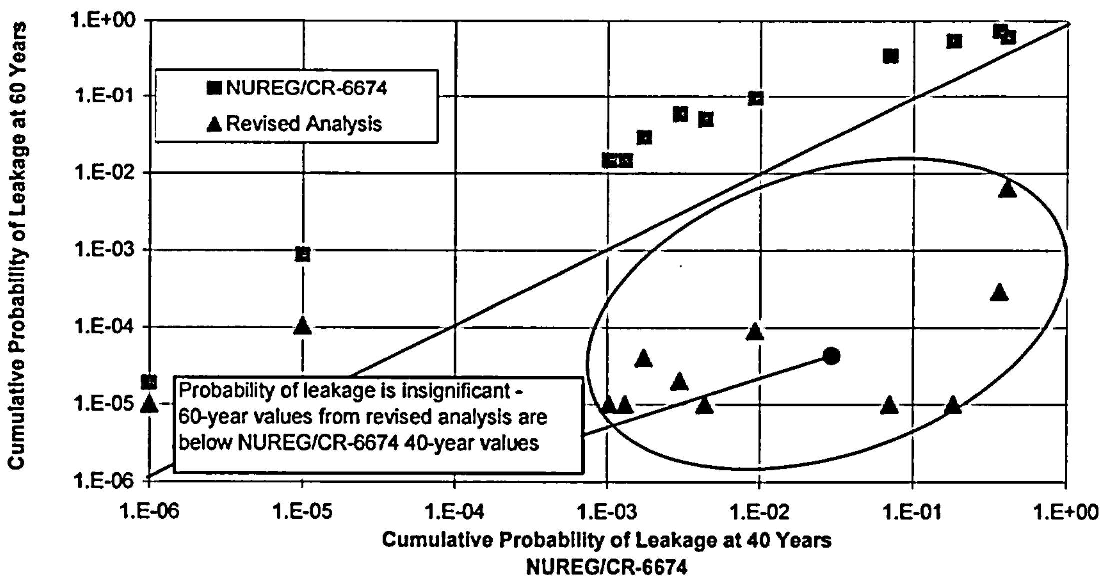


Figure 12. Cumulative Probability of Leakage at 60 Years Versus 40 Years

Figure 12 compares the probability of leakage for 60 years, as determined in the revised analysis [21] 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 the revised analysis 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 12 (for those components with a leakage probability greater than $10^{-3}$ as originally predicted by the NUREG/CR-6674 analysis) 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 11 and 12 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 13 compares the 60-year initiation probabilities for carbon/low-alloy steel components in environment from the revised analysis with results from the NUREG/CR-6674 study in air. In all cases except one, the re-analysis indicates that consideration of reactor water environment results in initiation probabilities at 60 years that are lower, in many cases significantly, than predicted in the NUREG/CR-6674 study in air.

Figure 14 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.

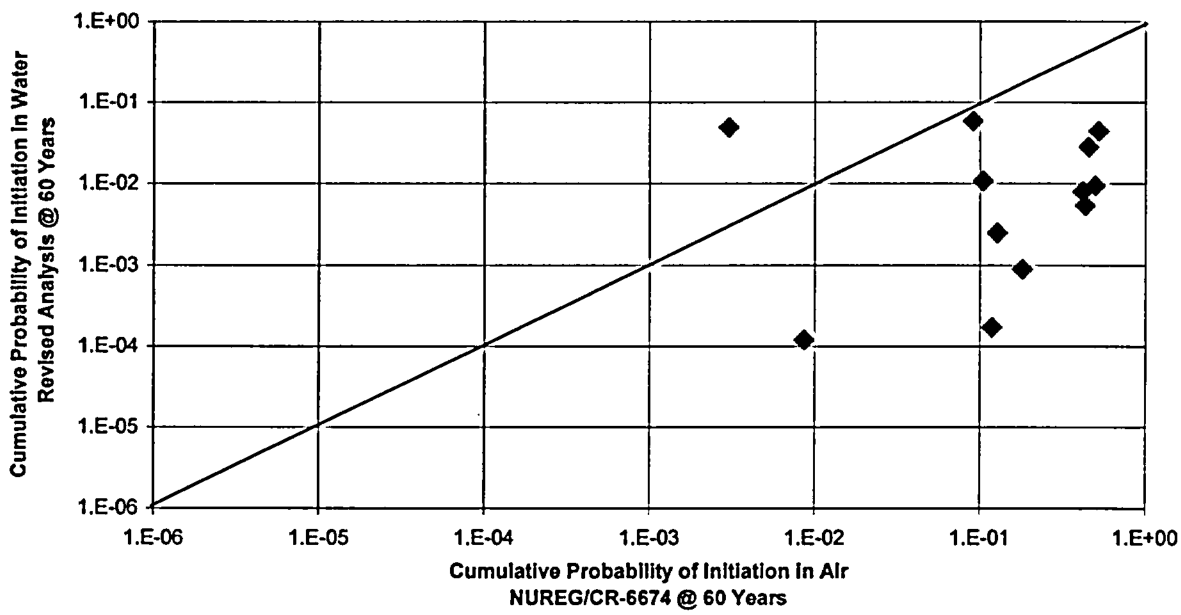


**Figure 13. Comparison of 60-Year Cumulative Probabilities of Initiation in Environment for Revised Analysis and NUREG/CR-6674 in Air.**

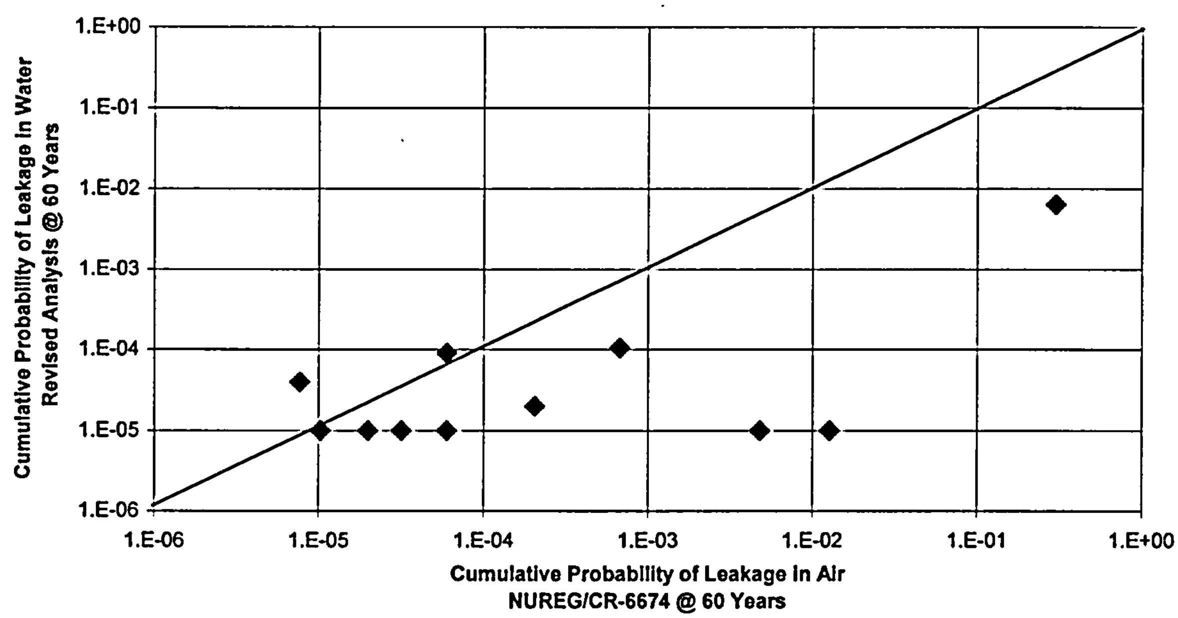


**Figure 14. Comparison of 60-Year Cumulative Probabilities of Leakage in Environment for the Revised Analysis and NUREG/CR-6674 in Air.**

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 $1.22 \times 10^{-7}$. In the revised analysis [21] the maximum 60-year CDF reported was $1.1 \times 10^{-10}$, representing over three 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 as supplemented by the revised analysis [21] 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 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 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.

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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