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December 13, 2000

Mr. Eugene V. Imbro, Chief
Mechanical and Civil Engineering Branch
Division of Engineering
U. S. Nuclear Regulatory Commission
Washington, DC 20006

Subject: Industry Guidance for Addressing Fatigue Environmental Effects in a License Renewal Application

Dear Mr. Imbro:

Enclosed for NRC staff review is an EPRI Technical Report entitled "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application." The objective of this guidance is to provide near term license renewal applicants with an approach for addressing fatigue environmental effects

We would like to work with the NRC staff to reach agreement on the guidance and ultimately include it as a reference in the Generic Aging Lessons Learned report. In our November 28 meeting with the NRC staff, they agreed to develop a schedule for reviewing and finalizing the guidance. It is our understanding the schedule will be available within 10 days of receipt of the document.

This guidance is extremely important to licensees that will submit renewal applications in 2001 and 2002 and we ask that the NRC staff consider a schedule that results in issuing the final document in the first quarter 2001.

Submittal of this document is a means of exchanging information with the NRC that is intended to support generic regulatory improvements. Therefore, we believe an exemption from any review fees is warranted based on the criteria in footnote 4 of 10 CFR Part 170.21.

XE04

Also, all correspondence on this issue should be sent to NEI with copies to Mr. John Carey of EPRI and Mr. Mike Robinson of Duke

Please contact me at (202) 739-8093 or by e-mail at djw@nei.org if you have any questions.

Sincerely,

Douglas J. Walters

Douglas J. Walters

C: Mr. Richard H. Wessman, NRC
Mr. Christopher I. Grimes, NRC

Enclosure

EPRI Technical Report

Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-xx)

TR-000000

December 8, 2000 Draft Rev. F

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REPORT SUMMARY

Background
Many utilities are currently embarking upon efforts to renew their operating licenses. One of the key areas of uncertainty relates to fatigue of pressure boundary components. Although the NRC has determined that fatigue is not a significant contributor to core damage frequency, they believe that the frequency of pipe leakage may increase significantly with operating time and have requested that license renewal applicants perform an assessment to determine the effects of reactor water coolant environment on fatigue, and, where appropriate, to manage this effect during the license renewal period. To-date, several utilities have addressed this request using different approaches.

Objective
The objectives of this report are to provide guidance for consideration of reactor coolant environmental effects and to minimize the amount of plant-specific work necessary to comply with NRC requirements for addressing this issue in a license renewal application.

Approach
Previous work by EPRI and utilities related to fatigue environmental effects and license renewal were reviewed. Reports on this subject, created by EPRI, NRC and NRC contractors were compiled. Recent license renewal applications, NRC Requests for Additional Information, and the commitments made by the first two license renewal applicants provided insight into NRC expectations. Given this, alternatives for addressing fatigue environmental effects were developed.

Results
A fatigue environmental effect license renewal approach has been developed that can be applied by any license renewal applicant, using a number of different paths. Based on a sampling approach, an assessment is conducted, using either fatigue environmental factors or demonstrated conservative design transients. Various methods are presented for fatigue management during the extended operating period.

EPRI Perspective
Utilities have committed significant resources to license renewal activities related to fatigue. Based on input from the first few applicants, NRC requirements for addressing fatigue environmental effects are continuing to change. These guidelines were developed to provide stability and assurance of NRC acceptance and include several approaches that may be taken to address fatigue environmental effects in a license renewal application. Use of the approaches provided in this document should limit the amount of effort necessary by individual license renewal applicants in addressing this requirement and putting activities in place for the extended operating period to manage fatigue reactor water environmental effects.

Keywords

Fatigue

License Renewal

Fatigue Environmental Effects

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EXECUTIVE SUMMARY

For about the last 10 years, the effects of light water reactor environment on fatigue have been the subject of research in both the United States and abroad. The conclusions from this research is that the reactor water temperature and oxygen content may have a significant effect on the fatigue life of carbon, low alloy and austenitic stainless steels. The degree of fatigue life degradation may be a function of the tensile strain rate during a transient, the specific material, the temperature and the oxygen content. There are certain threshold limits for these variables below which the effects are not significant. The effects of other than moderate environment effects were not considered in development of the ASME Code fatigue curves.

This issue has been evaluated by the Nuclear Regulatory Staff for several years. One of the major efforts was a program to evaluate the effects of reactor water environment for both early and late vintage plants designed by all US vendors. The results of this study, published in NUREG/CR-6260, showed that there were a few high usage factor locations in all reactor types, and that the effects of water reactor environment could cause fatigue usage factors to exceed the Code-required limit of 1.0. On the other hand, it was demonstrated that usage factors at many locations could be made to be acceptable by re-analysis.

Based on a risk study, reported in NUREG/CR-6674, NRC concluded that water reactor environmental effects were not a safety issue for the 60-year operating life, but that some limited assessment of its effect would be required in a license renewal extended operating period. Thus, for all license renewal submittals to date, there have been a round of questions, and utility commitments in some cases, to address the environmental effects on fatigue in the extended operating period.

This guideline offers methods for addressing environmental fatigue in a license renewal submittal. It requires that a sampling of the most affected fatigue sensitive locations be identified for evaluation and tracking in the extended operating period. For some locations, fairly simple assessments may be performed to show that there is adequate conservatism in the design transients to envelope the reactor water environmental effects. For other locations, detailed evaluations, such as conducted in NUREG/CR-6260 may be required. In the extended operating period, cycle counting or fatigue tracking are used for the sample of locations to show that Code limits are not exceeded. If they are, several approaches are identified for demonstrating acceptability for continued operations without repair or replacement of components.

Using the guidance provided herein, the amount of effort needed to justify individual license renewal submittals and respond to NRC questions should be minimized.

CONTENTS

LIST OF FIGURES

1 INTRODUCTION

The nuclear industry has discussed the issue of reactor water environment fatigue effects with the U. S. Nuclear Regulatory Commission (NRC) staff for several years. All of the license renewal applicants to-date have been required to commit to an approach to evaluate the effects of reactor water environment on specific Class 1 reactor coolant system components for the license renewal term in order to obtain approval for a renewed license.

The purpose of developing this guideline document for addressing reactor water environmental effects in a license renewal application is to define and justify several acceptable approaches that may be used to address this issue.

This report provides a discussion of the approaches that may be used for addressing reactor water environmental effects on fatigue of reactor coolant system components in the extended operating period (after 40 years). This report does not provide guidance on addressing fatigue as a Time Limiting Aging Analysis (TLAA) per 10CFR54.

Thus, the objectives of this report are as follows:

1. To provide guidance for consideration of fatigue reactor water environmental effects for license renewal applicants,

2. To define various approaches that can be used in the extended operating period to adequately manage the potential effects of reactor water environmental effects on fatigue, and

3. To minimize the amount of plant specific work necessary to comply with NRC requirements for considering reactor water environmental effects.

2 BACKGROUND

2.1 Research Results

NRC research in the area of reactor water environmental effects on fatigue began in the early 1990's. Based on testing both in Japan and in the U.S., fatigue life in a light water reactor (LWR) environment was determined to be adversely affected by oxygen content, strain amplitude, strain rate, temperature and sulfur content (for ferritic steels). Whereas LWR pressure boundary components are in contact with the reactor water at elevated temperatures, the fatigue curves in Section III of the ASME Boiler and Pressure Vessel Code were based on testing in air, primarily at room temperature. In 1993, a set of "interim" fatigue curves for carbon, low alloy and stainless steel were published in NUREG/CR-5999 [1].

To determine the effects of the environment for operating nuclear plants during the current 40-year licensing term and for an assumed 60-year extended period, Idaho National Engineering Laboratories (INEL) evaluated fatigue-sensitive component locations, documenting this study in NUREG/CR-6260 [2]. Using information from existing reactor component stress reports, supplemented by some additional evaluations, cumulative fatigue usage factors (CUFs) were calculated for plants designed by all four nuclear steam supply system (NSSS) vendors utilizing the interim fatigue curves provided in NUREG/CR-5999. The results showed that CUFs would exceed 1.0 at many locations, although the CUFs at some of these were shown to be less than 1.0 if conservatisms were removed from the evaluations.

Continued research led to changes to the fatigue curves utilized in deriving the results presented in NUREG/CR-6260. The latest proposed environmental fatigue correlations are presented in NUREG/CR-6583 for carbon and low alloy steels [3] and in NUREG/CR-5704 for austenitic stainless steels [4].

More recently, an evaluation was conducted to assess the implications of the LWR environment on component fatigue for a 60-year plant life. This study, based on the information in NUREG/CR-6260 and documented in NUREG/CR-6674 [5], concluded that the environmental effects of reactor water on fatigue curves had an insignificant contribution to core damage frequency. However, the frequency of pipe leakage was shown to increase in some cases.

2.2 License Renewal Environmental Fatigue Issue

The environmental fatigue issue for license renewal was finalized during the close-out of Generic Safety Issue 190 (GSI-190) [6] in December 1999. In a memorandum from NRC-RES

to NRC-NRR [7], it was concluded that environmental effects would have a negligible impact on core damage frequency, and as such, no generic regulatory action was required. However, since NUREG/CR-6674 [5] indicated that fatigue reactor coolant environmental effects would result in an increased frequency of pipe leakage, the NRC required that utilities that apply for license renewal address the effects of reactor water environment on fatigue usage in affected components.

2.3 Industry/EPRI Programs

Following the issuance of NUREG/CR-6260 [2], EPRI performed several studies to quantitatively address the issue of environmental fatigue during the license renewal period.

The initial efforts were focused on developing a simplified method for addressing environmental fatigue effects and evaluating more recent research results. The calculations reported in NUREG/CR-6260 were based on the interim fatigue design curves given in NUREG/CR-5999 [1]. The conservative approach in NUREG/CR-6260 and NUREG/CR-5999 penalized the component fatigue analysis, since later research identified that a combination of environmental conditions is required before reactor water environmental effects become pronounced. The strain rate must be sufficiently low and the strain range must be sufficiently high to cause continuing rupture of the passivation layer that protects the exposed surface of reactor components. Temperature, dissolved oxygen content, metal sulfur content, and water flow rate are additional variables to be considered. In order to take these parameters into consideration, EPRI and GE jointly developed a method, commonly called the F_{en} approach [8], that permits reactor water environmental effects to be applied selectively, as justified by evaluating the combination of effects that contribute to increased fatigue susceptibility.

The F_{en} approach was used in an EPRI project to evaluate fatigue-sensitive component locations in four types of nuclear power plants: an early-vintage Combustion Engineering (CE) PWR [9], an early-vintage Westinghouse PWR [10], and both late-vintage [11] and early-vintage [12] General Electric (GE) BWRs. Component locations similar to those evaluated in NUREG/CR-6260 were examined in these generic studies.

In the early-vintage Westinghouse PWR results [10], actual plant transient data (e.g., hot leg temperature, pressurizer water temperature) over three cycles of operation (1994, 1995, and 1996) were used to derive an effective environmental factor that could be applied to the design-basis CUF. The maximum effective F_{en} value (ratio of usage factor with reactor water environmental effects to that based on ASME Code fatigue curves) for any of the pressurizer and surge line locations (pressurizer shell, pressurizer surge nozzle, pressurizer spray nozzle, pressurizer water temperature instrument nozzle, RCS hot leg surge nozzle, and charging nozzle) was 1.91. This value is very low compared to the environmental multiplier of over seven from NUREG/CR-6260.

These findings were confirmed in the other PWR study of an early-vintage CE PWR [9], where the pressurizer surge line was studied in detail. This calculation provided another direct comparison with the same component location evaluated in NUREG/CR-6260. In this evaluation, the surge line elbow location, fabricated from austenitic stainless steel, has a design-

basis CUF of 0.705 calculated for 40 years of operation. NUREG/CR-6260 cites a value for the 40-year CUF of 8.07 when reactor water environmental effects are applied. This environmental CUF value is more than ten times the design-basis CUF. The EPRI evaluations [9] showed that the environmental multiplier was actually only about two based on actual plant transient monitoring data.

The NUREG/CR-6260 studies were conservative since they were based on the earlier NUREG/CR-5999 interim fatigue curves and had to be based on extremely conservative strain rates and other parameters. The use of actual plant data shows that usage factors are much lower than the CUFs calculated in the design basis analyses, and that the environmental factors are not as extreme as those presented in NUREG/CR-6260.

The NRC staff has not accepted the studies performed by EPRI [13], primarily because the environmental fatigue effects were based on data that was developed prior to the issuance of the latest reports by Argonne National Laboratory (ANL) [3,4]. The following issues were raised in a letter from NRC to the Nuclear Energy Institute [13]:

- The environmental fatigue correction factors developed in the EPRI studies were not based on the latest Argonne test report.
- The environmental factors develop in the EPRI studies were not based on a comparison environmental data at temperature to air data at room temperature.
- The NRC did not agree with the use of the factors of 4 (for carbon steel) and 2 (for stainless steel) to account for moderate environmental effects. Instead, the NRC staff believed that the maximum factors that could be used were 3 (for carbon steel) and 1.5 (for stainless steel).
- There was disagreement on the strain thresholds that were used.
- The NRC staff did not agree that cladding could be taken credit for in not considering environmental effects for the underlying carbon steel/low alloy steel materials, unless fatigue in the cladding was specifically addressed.
- The staff agreed with use of a weighted average strain rate for computing environmental effects as long as the maximum temperature of the transient was used.

To date, the industry has chosen not to pursue a formal response to the NRC regarding these areas of disagreement. Instead, the industry has worked with the initial license renewal applicants on the prototype resolutions to the issues. These prototype resolutions are a part of the foundation for this report.

Based on NRC review of more recent Japanese and ANL data, NRC believes that no credit should be given for moderate environmental effects [14].

The Pressure Vessel Research Council (PVRC) Steering Committee on Cyclic Life and Environmental Effects (CLEE) has reviewed published environmental fatigue test data and the F_{en} methodology. Based on this review, the most recent findings by ANL have been incorporated into the equations for the environmental factors. More importantly, it was concluded that the environmental factors could be reduced, by factors of 3.0 for carbon/low-alloy steel and 1.5 for stainless steel to accommodate moderate environmental effects included in the current ASME Code fatigue design curves. The PVRC recommendations have been forwarded

to the Board of Nuclear Codes and Standards (BNCS) [15]. The recommended evaluation procedure is included in Appendix B. Appendix C includes evaluations based on recent data that would support factors of 3.0 for carbon/low-alloy steel and 1.5 for stainless steel.

3
LICENSE RENEWAL APPROACH

3.1 Overview

This document describes how the technical issues associated with reactor water fatigue environmental effects evaluation may be addressed. To assess the effects of reactor water environment on fatigue life, a limited number of components will be assessed considering the effects of recent environmental fatigue data. These component locations serve as the leading indicators to assess the significance of environmental effects. For this limited number of components, the effects of the environment on fatigue life must be addressed and adequately managed in the extended operating period.

The process chosen to address environmental effects by the first four applicants for license renewal has varied. After a series of requests for additional information, the process that the NRC accepted for Calvert Cliffs and Oconee involved an analytical approach coupled with future planned refinements in their plant fatigue monitoring programs. There has been no acceptance of the approaches used by the other applicants, as these are still in the evaluation process. By developing guidelines for aging management of reactor water fatigue effects for license renewal and obtaining NRC concurrence, an acceptable approach for addressing this issue will be clearly documented for future license renewal applicants.

These guidelines provide a process to address environmental effects in the License Renewal Application. An aging management program is provided based on today's knowledge. The elements of that program may change in the future as more information becomes available. Attributes of the fatigue management activity are as follows:

Scope of Program: The program includes measures to mitigate fatigue cracking of reactor coolant pressure boundary components caused by reactor water environmental effects.

Preventive Actions: Tracking of operating transient cycles and/or maintaining usage factors less than unity, or assuring that fatigue cracks do not grow to the size allowed by ASME Section XI Appendix L, or other NRC-approval limits, assures that there is adequate margin against component leakage due to fatigue cracking.

Parameters Monitored/Inspected: The significant plant transients that cause fatigue damage or crack growth will be monitored. Alternately, more detailed fatigue analysis, or crack growth analysis combined with component ISI, can be used to show that the effects of transient operating cycles remain within established limits.

Detection of Aging Effects: For most locations, plant operating cycles or cumulative usage factors (CUFs) will be tracked against established fatigue limits. Where these limits are exceeded, component in-service inspection (ISI) at an interval sufficient to detect significant cracking may be used.

Monitoring and Trending: The program will monitor a sampling of locations expected to be most adversely affected by plant cycles and reactor water environment. Selection of the sample population will consider those locations identified in NUREG/CR-6260 as well as others.

Acceptance Criteria: Two alternate acceptance criteria are provided. For most locations, fatigue usage is kept below the ASME Section III Code-allowable limit. If this can not be demonstrated, an alternate approach is to show that any potential cracking is maintained below that allowed by ASME Section XI Appendix L, or other NRC-approved limit.

Corrective Actions: The program allows for alternate actions to keep the usage factors from exceeding Code limits, including more rigorous analysis (e.g., partial cycle counting, revised modern analysis, fatigue monitoring), converting to a flaw-tolerance-based approach (ASME Section XI, Appendix L or other NRC-approved methodology), or component repair/replacement.

Confirmation Process and Administrative Controls: Consistent with current requirements, site QA procedures, review and approval processes and administrative controls will be implemented in accordance with Appendix B of 10 CFR Part 50.

Operating Experience: Consistent with current practice, industry experience will continue to be reviewed. Applicable industry experience will be reviewed for applicability and additional inspections or other analytical programs will be undertaken to assure that unacceptable fatigue cracking does not occur, due to both anticipated and unanticipated transients.

3.2 Methods for Evaluation of Environmental Effects

There are several methods that can be used to assess the effects of reactor water environment on fatigue for each specific location to be considered. In this document, two primary methods are provided.

Figure 3-1 is a flowchart that shows an overview of the assessment approach.

- The first step is to identify the locations to be used in the assessment. This step is discussed in Section 3.2.1

- The second step is to select the method that will be used to manage the effects of environmental fatigue.

- Method 1 is based on demonstrating that the fatigue analysis for the design transients, when compared to an evaluation based on actual transients, will bound any environmental effects in the extended operating period. This method will apply to those locations where the design basis fatigue analysis can easily be shown to be very conservative with respect to transient definitions or number of design cycles. Further discussion is provided in Section 3.2.2.

- Method 2 includes an assessment of the actual expected fatigue usage factor including the influence of environmental effects. With this method, inherent conservatisms in design transients may be removed to arrive at realistic CUFs that include environmental effects. This approach is most applicable to locations where the design transients reflect actual operating conditions in the plant. Further discussion is provided in Section 3.2.3.

- The bottom of Figure 3-1 indicates that fatigue management occurs after the method is chosen for each location. This may be as simple as counting the accumulated cycles and showing that they remain less than utilized in the assessment. On the other hand, it may not be possible to show continued acceptance throughout the extended operating period such that additional actions are required. Such options are discussed in Section 3.3.

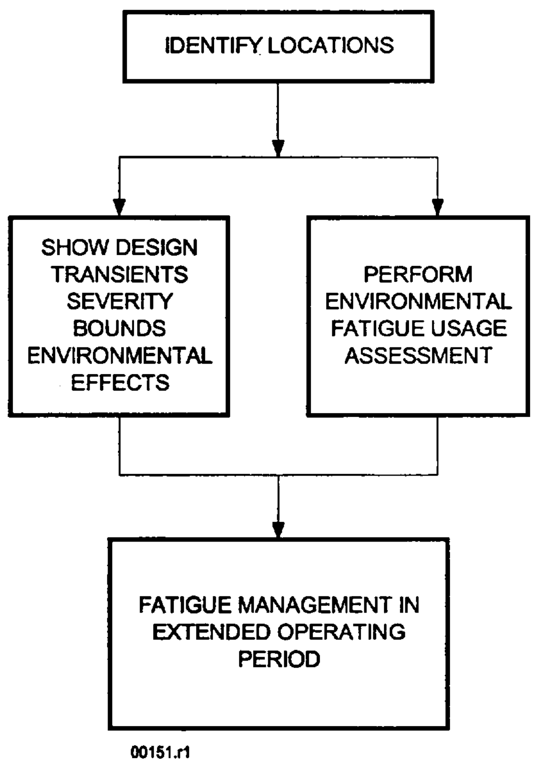

Figure 3-1
Overview of Methods for Fatigue Environmental Effects Assessment and Management

3.2.1 Identification of Locations for Assessment of Environmental Effects

A sampling of locations is chosen for the assessment of environmental effects. The purpose of identifying this set of locations is to focus the environmental assessment on just a few components that will serve as leading indicators of fatigue reactor water environmental effects. Figure 3-2 shows an overview of the approach identified for selecting locations.

For both PWR and BWR plants, the locations chosen in NUREG/CR-6260 were deemed to be representative of locations with relatively high usage factors for all plants. Although the locations may not have been those with the highest values of fatigue usage reported for the plants evaluated, they were considered representative enough that the effects of LWR environment on fatigue could be assessed. Thus, these locations should be considered in selecting the representative set of locations. Appendix A describes the locations.

The locations evaluated in NUREG/CR-6260 can be evaluated on a plant-unique basis. An evaluation in Appendix A shows that the environmental effects on some of the locations may not be significant. Similar plant unique evaluations may show that the NUREG/CR-6260 locations are also not significantly affected by fatigue, such that they need not be considered. Likewise, plant specific evaluations may identify other locations that are more affected. Thus, plant fatigue analyses should be reviewed to identify the specific locations where high usage factors were identified in the original design.

In original stress reports, high usage factors may have been reported in many cases that are unrealistically high, but met the ASME Code requirement of CUF < 1.0. In these cases, revised analysis may be conducted to derive a more realistic usage factor or to show that the actual usage factor is significantly less than reported.

In identifying the set of locations for the environmental assessment, it is important that a diverse set of locations be chosen with respect to component loading (including thermal transients), geometry, materials and reactor water environment. If high usage factors are presented for a number of locations that are similar in geometry, material, loading conditions and environment, the location with the highest expected CUF, considering environmental effects, should be chosen as the one to use in the environmental assessment. Similar to the approach taken in NUREG/CR-6260, the final set of locations chosen for the environmental assessment should include several different types of locations that are expected to have the highest CUFs, should be those most adversely affected by environmental effects and should include 6-10 component/system locations. The basis of location choice should be described in the individual plant license renewal application.

In conclusion, the following steps should be taken to identify the specific locations that are to be considered in the environmental assessment:

- Identify all Class 1 piping systems and major components. For the reactor vessel, there may be multiple locations to consider.

- For each system or component, identify the highest usage factor locations. By reasons of geometric discontinuities or local transient severity, there will generally be a few locations that have the highest usage factors when considering environmental effects. If the NUREG/CR-6260 locations are not included, add these locations, unless a specific plant evaluation is provided to show that the usage factor with environmental effects is lower than for other similar identified locations.

- From this list of locations, choose a set of 6-10 locations that are a representative sampling of locations with the highest expected usage factors when considering environmental effects. Considerations for excluding locations can include: (1) identification of excess conservatism in the transient grouping or other aspects of the design fatigue analysis, (2) locations that have similar loading conditions, geometry, material and reactor water environment to another selected location, or (3) an assessment of reactor water environmental fatigue affects shows that the expected usage factor with environmental effects is small (such as demonstrated in Appendix A).

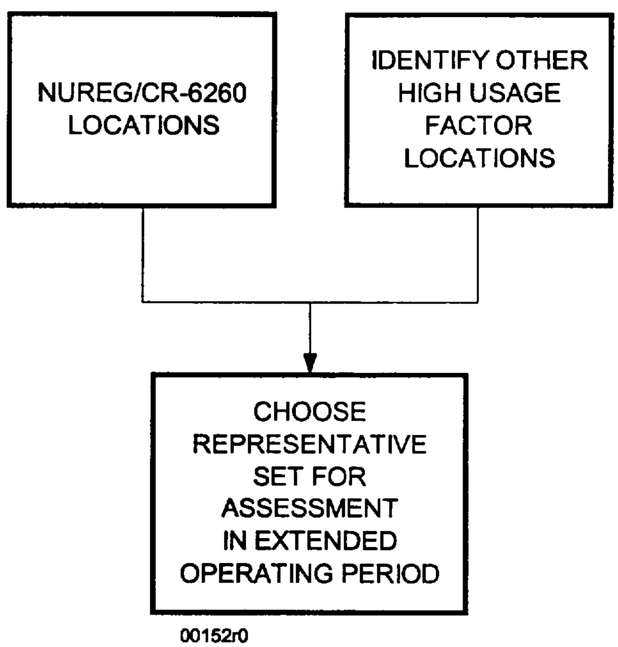

Figure 3-2
Identification of Component Locations for Fatigue Environmental Effects Assessment

3.2.2 Method 1: Design Basis Loading Assessment

With this assessment approach, the inherent conservatisms in the design basis loadings, considering both severity and number of the transients, are used to bound environmental effects. The influence of environmental effects is shown to be offset by the conservatism in design basis transients relative to actual plant transients. This assessment can be based on results of industry

studies and/or plant fatigue assessments using actual plant data. Figure 3-3 shows the approach for performing the assessment and managing fatigue in the extended operating period.

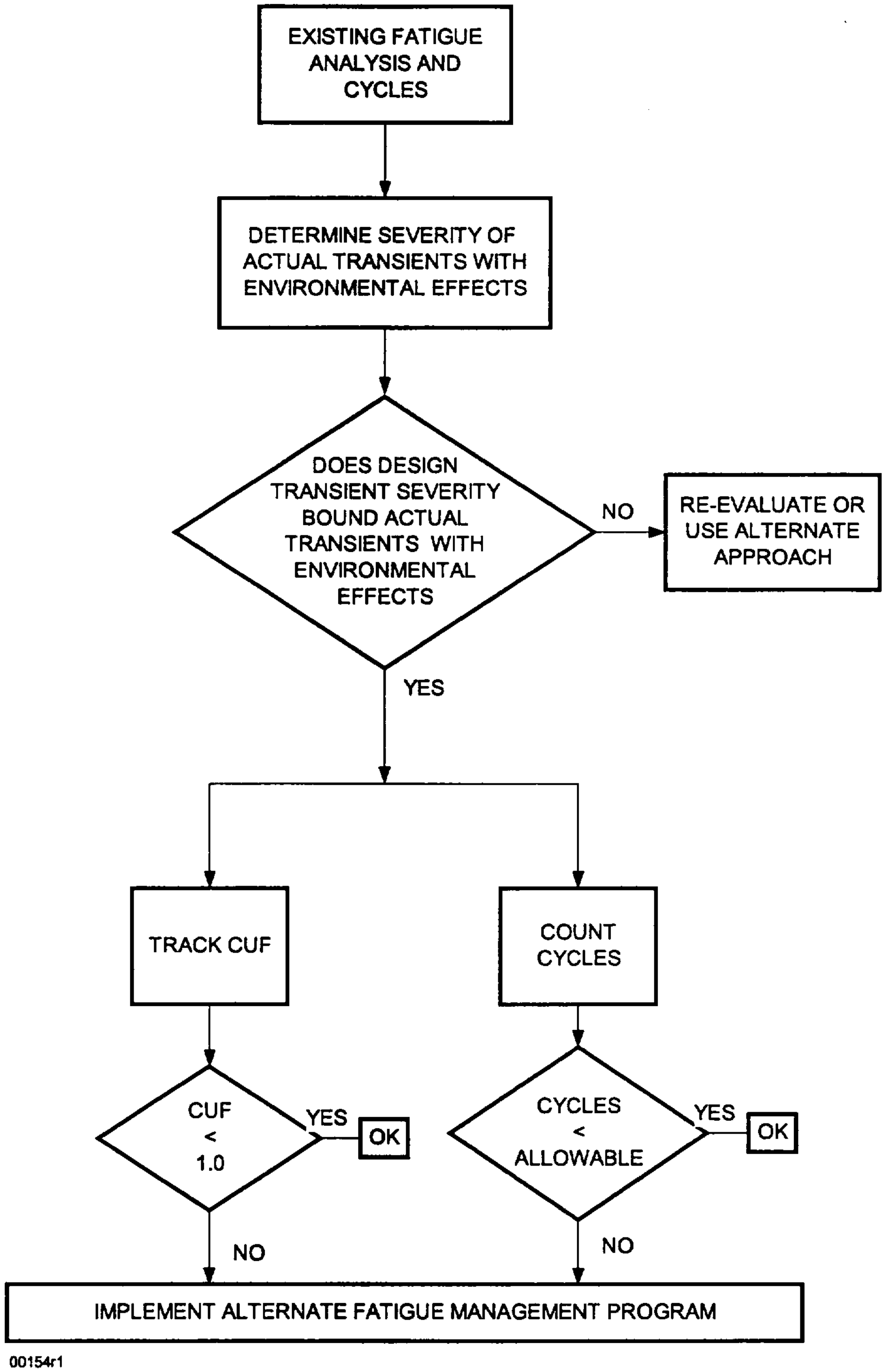

Figure 3-3
Fatigue Management if Design Transients Bound Fatigue Environmental Effects Effects

Determination of Existing Design Basis

Existing plant records must be reviewed to determine the cyclic loading specification (transient definition and number of cycles) and stress analysis for the location in question. In most cases, the loadings were conservatively defined before there was significant plant operating experience. As a result, the transient load definitions are based on step temperature changes with extremely conservative temperature ranges. When performing the component fatigue analysis, similar transients may have been grouped together to reduce the amount of effort expended in the stress analysis. Further, the number of actual cyclic loadings may exceed those considered in the fatigue evaluation. This review can arrive at a preliminary assessment as to whether sufficient conservatism exists in the existing fatigue analysis to bound environmental effects.

Demonstration That Design Transient Conservatisms Bound Actual Transients with Environmental Effects

CUFs with environmental effects include several increasing and decreasing factors relative to a design basis stress analysis to meet ASME Code requirements.

- Increase: There is some increase in fatigue usage due to environmental effects. This is a function of oxygen content, strain rate, strain amplitude, temperature, etc. The actual effect has been demonstrated in industry programs to be less than that derived in NUREG/CR-6260, where worst case parameters were chosen to determine environmental effects.

- Decrease: Actual transients may produce lower stresses than those considered in the design. Temperature and pressure ranges and rates of temperature change are almost always less severe. Because of the shape of fatigue curves, a small reduction in stress can result in a large reduction in CUF.

Plant data from actual operation can be taken for the transients that significantly contribute to usage factors. The actual transients can then be defined and can be compared to design transients. Transient thermal stress analysis can be conducted with both the design transients and actual transients to determine the contributions to usage factor. From this analysis, strain rates can be determined such that actual environmental factors can be determined (as compared to the bounding ones used in NUREG/CR-6260).

A more direct approach is to use results from plant fatigue monitoring based on actual plant transient data. This approach directly determines fatigue usage results based on actual plant data. For some representative operating period, the results from the fatigue monitoring program can be compared to the usage predicted based on design basis transients. Representative strain rates, strain ranges and temperatures during transients can be determined from the data to estimate actual environmental factors for the key transients. This was the approach used in an EPRI project to determine effective environmental factors [9, 10, 11].

In most plants, the rate of cycle accumulation has decreased significantly since the initial operating period. A comparison of the rate of cycle accumulation versus that considered in the

design analysis can be made. Where a significant difference is predicted to the end of the extended operating period, this can be taken into account as one of the conservatisms.

In some cases, especially if the evaluation is based on fatigue monitoring results, some bounding assumptions can be made such that specific environmental factors do not have to be developed for each load set pairing in the fatigue analysis.

Once all this information has been collected, the applicant can perform an evaluation to show that the design transients bound the actual transients with environmental effects considered. A way to accomplish this without resorting to CUF determination is by showing that the severity of the actual transients with environmental effects considered are bounded by the transient severity assumed in the component design. If the results of this effort are satisfactory, an applicant can monitor fatigue. Another option is to count cycles to ensure that the actual count remains less than that assumed in the design. As part of this option, an applicant may determine the number of cycles where CUF = 1.0. This number becomes the allowable number of cycles. A second option is to count cycles and compute a CUF using the design transient severity to ensure CUF remains below one (1.0). Either method is conservative.

If the above comparison is conducted using CUF values (design CUF and actual with environmental effects included in the CUF), then it is permissible to recalculate the design CUF using higher numbers of transients up to a set that results in design CUF = 1.0. This set then becomes the new "design" cycle limits. Similarly, it is permissible to use a number of projected actual cycles less than design. One of these variations may permit the comparison to be satisfactory. The allowable number of cycles for cycle counting is the lessor of the two.

Appendix D shows an application of the results from NUREG/CR-6260 and the Industry/EPRI programs to demonstrate that design basis transient definitions are conservative and more than compensate for environmental effects. For the example shown, simple cycle counting and CUF calculation based on design basis transient severity is shown to bound any potential environmental effects.

Consideration of Increased Cycles for Extended Period

If a revision to the fatigue analysis is to be performed, the applicant may wish to update the projected cycles. In most cases, it can be demonstrated that the number of expected cycles in the extended operating period will remain at or below those projected for the initial 40-year plant life. However, if more cycles are projected, an applicant should consider the significance of this in respect to the means of fatigue management selected. Before proceeding with the use of cycle counting, an applicant may choose to perform a revised fatigue analysis to confirm that the increased number of cycles will still result in CUF less than 1.0. An applicant may also choose to determine the number of cycles at which CUF would be expected to exceed one (1.0). These results can be used to determine the most appropriate method for managing fatigue at a given location.

Fatigue Analysis Re-evaluation

An applicant may not be able to show that the design transients bound the actual considering environmental effects. This may be due to the expected number of transients exceeding that assumed in the design, or to the transient severity, when the environment is considered, exceeding that used in the design. In this situation, a fatigue analysis may be revised. In this case, the applicant may update the fatigue analysis to determine the acceptance criteria to be used in the cycle counting approach to fatigue monitoring.

The amount of effort expended in conducting a revised fatigue analysis can vary. A simplified revised fatigue analysis could be performed using results from the existing fatigue analysis if sufficient detail is available. At the other end of the spectrum, a complete new analysis could be conducted. In any revised analysis, the design basis transients shall be utilized in establishing the design transient CUF. Actual transient behavior, combined with environmental effects, shall be considered in evaluating the CUF with environmental effects.

Fatigue Management Approach

As shown in Figure 3-3, the primary fatigue management approaches for the extended operating period consist of tracking either the CUF or number of the accumulated cycles. As previously discussed, the CUF is based on the design cycles. At such time that the CUF is projected to exceed 1.0, or the number of actual cycles is projected to exceed the allowable cycles, action must be taken such that the allowable limits will not be exceeded. If the cyclic or fatigue limits are expected to be exceeded during the license renewal period, further approaches to fatigue management would be required prior to reaching the limit, as described in Section 3.3.

3.2.3 Method 2: Fatigue Assessment Using Environmental Factors

With this assessment method, factors to account for environmental effects are incorporated into an updated fatigue evaluation for the location using the F_{en} approach outlined in Appendix B. Excess conservatism in both the loading definitions, number of cycles and the fatigue analyses may be considered. Figure 3-4 shows the approach for performing the assessment and managing fatigue in the extended operating period.

Determination of Existing Design Basis

Existing plant records must be reviewed to determine the cycling loading specification (transient definition and number of cycles) and stress analysis for the location in question. Review of the analysis may or may not show that excess conservatism exists.

Consideration of Increased Cycles for Extended Period

If a revision to the fatigue analysis is to be performed, the applicant may wish to update the projected cycles. In most cases, it can be demonstrated that the number of expected cycles in the extended operating period will remain at or below those projected for the initial 40-year plant life. However, if more cycles are projected, an applicant should consider the significance

of this in respect to the means of fatigue management selected. Before proceeding with the use of cycle counting, an applicant may choose to perform a revised fatigue analysis to confirm that the increased number of cycles will still result in CUF less than 1.0. An applicant may also choose to determine the number of cycles at which CUF would be expected to exceed one (1.0). These results can be used to determine the most appropriate method for managing fatigue at a given location.

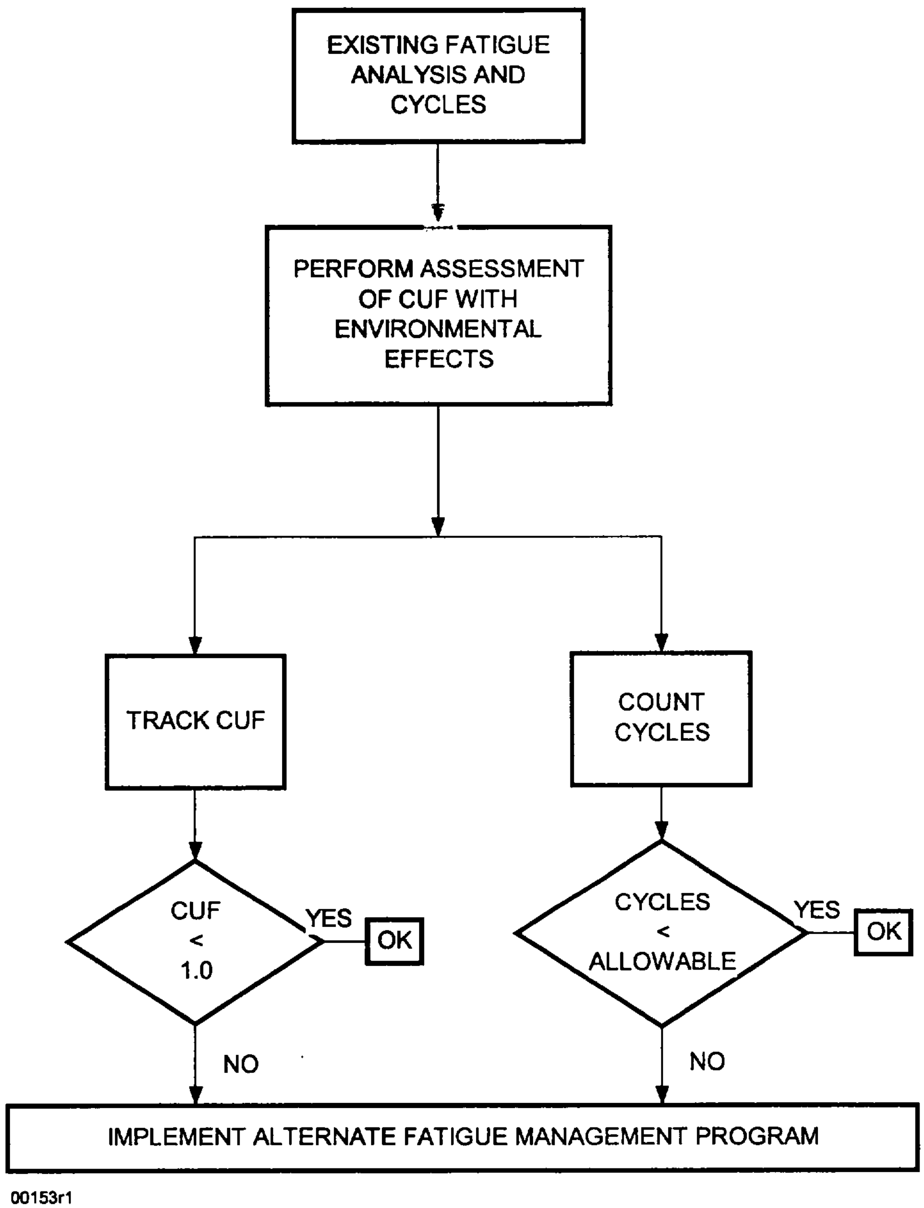

Figure 3-4
Fatigue Management if Environmental Assessment Conducted

Fatigue Assessment

A determination of CUF considering environmental effects is needed. This may be accomplished conservatively using information from design documentation or industry publications and bounding Fen factors, or it may require a more extensive approach.

A revised fatigue analysis may or may not be required. Possible reasons for updating the fatigue analysis could include:

- Excess conservatism in original fatigue analysis with respect to modeling, transient definition, or transient grouping

- For piping, use of an ASME Code Edition prior to 1977, Summer Addenda, which included the ΔT_1 term in Equation 10.

A simplified revised fatigue analysis may be performed using results from the existing fatigue analysis if sufficient detail is available. Alternately, a new complete analysis could be conducted to remove additional conservatisms. In the environment assessment, the environmental fatigue usage may be calculated with the following steps:

- For each load set pair in the fatigue analysis, determine an environmental factor F_{en}. This factor should be developed using the equations in NUREG/CR-6583 (for carbon or low alloy steel components) or NUREG/CR-5704 (for austenitic stainless steel components). Appendix B describes the latest procedure endorsed by the Pressure Vessel Research Council that includes the equations from these documents[15,16].

- The environmental factors may be calculated with consideration of temperature, oxygen and strain range thresholds. The average strain rate for the transient producing tension at the inside of the piping shall be used together with the maximum occurring just prior to or during the tensile stress producing transient of the load set pair. Reference 12 provides several examples using this approach.

- Using the F_{en} from the above step, determine an effective environmental multiplier (F'_{en}) by dividing F_{en} by a Z-factor that accounts for moderate environmental effects. PVRC recommends Z factors of 3.0 for ferritic steel components and 1.5 for austenitic components. (A detailed basis for these Z factors is contained in Appendix C.)

- The environmental partial fatigue usage for each load set pair is then determined by multiplying the original partial usage factor by F'_{en}. In no case shall the F'_{en} be less than 1.0.

- The usage factor is the sum of the partial usage factors calculated with consideration of environmental effects.

In many situations, the original design basis fatigue analysis may have been conducted in a very conservative manner, and may have been based on early versions of the ASME Code. There

may be some benefit in conducting a modern fatigue analysis. This revised analysis could take into account actual expected transients, less conservative assumptions or later versions of the ASME Code. For piping components, use of Code versions after the 1979 Edition is especially beneficial since the ΔT_1 term was removed from Equation 10 of NB-3650, reducing the need to apply conservative elastic plastic penalty factors. The re-analysis could also determine strain rate time histories that are not normally reported in existing component analysis, such that bounding environmental multipliers would not have to be used..

Fatigue Management Approach

As shown in Figure 3-4, the primary fatigue management approaches for the extended operating period consist of tracking either the CUF or number of accumulated cycles.

- For cycle counting, an updated allowable may be needed if the fatigue assessment determined the CUF to be larger than 1.0. One approach is to derive a reduced number of cycles that would limit the CUF to 1.0. On the other hand, if that CUF was shown to be less than 1.0, the allowable cycles may remain as assumed in the evaluation. So long as the number of cycles in the extended operating period remain within this allowed number of cycles, no further action is required.

- For CUF tracking, one approach would be to utilize a fatigue monitoring approach that accounts for the actual cyclic operating conditions for the location. This approach would track the CUF due to the actual cycle accumulation, and would take credit for the combined effects of all transients. Environmental factors would have to be factored into the monitoring approach. No further action is required as long as the computed usage factor remains less than 1.0

At such time that the CUF is projected to exceed 1.0, or the number of actual cycles is projected to exceed the allowable cycles, action must be taken such that the allowable limits will not be exceeded. If the cyclic or fatigue limits are expected to be exceeded during the license renewal period, further approaches to fatigue management would be required prior to reaching the limit, as described in Section 3.3.

3.3 Alternate Fatigue Management in the License Renewal Period

As identified in Section 3.2, results from cycle counting or fatigue monitoring may predict that established limits are exceeded during the extended operating period. If this occurs, there are several alternative approaches that may be used to justify continued operation with the affected component in service without having to perform repair or replacement. In addition, the fatigue management program may have to be expanded if plant-unique or industry experience shows that fatigue limits are exceeded or if cracking is discovered, due to either anticipated or unanticipated transients.

3.3.1 Reanalysis

Each of the methods for fatigue assessment/management described in Section 3.2 are based on fatigue analysis of the affected location. If allowable cycle or CUF limits are predicted to be exceeded during the extended operating period, then a revised analysis may be conducted to remove additional conservatisms. This is applicable to both the Method 1 and Method 2 assessment approaches (3.2.2 and 3.2.3). For Method 1, the re-analysis must be based on design transients, or actual transients with environmental effects, as applicable. For Method 2, excess conservatisms in both the analysis methods and the transient definitions may be considered. This analysis could take advantage of additional on-going industry research to better quantify reactor water environmental effects.

3.3.2 Partial Cycle Counting

For those locations that are primarily affected by a few significant cycle types, a cycle counting approach may be used to show that the cyclic limits or CUF with environmental effects does not exceed the appropriate limits in the extended operating period. When environmental factors are utilized (as described in 3.2.3), the cycle counting approach may take into account cycle severity and count partial cycles. This is a process where an analytical approach is used to show that cycles less severe than those considered in a design analysis may be counted as fractions of whole cycles. This analysis would not have to be completed until it became apparent that the environmentally affected usage factor limits would be exceeded.

3.3.3 Fatigue Monitoring

Fatigue monitoring to track CUF may be based on simple algorithms that convert cycles into CUF or may be based on more sophisticated systems that evaluate actual plant transient data. Fatigue monitoring, based on actual plant instrument data, calculates the CUF based on actual plant transient behavior. Its use is thus applicable when the fatigue monitoring assessment includes environmental effects as described in 3.2.3. It may also be used to form a basis for establishing actual transient severity as described in 3.2.2. The actual predicted CUF is generally much less than that resulting when using design transients. The algorithms in the fatigue monitoring system would have to incorporate environmental factors. This approach is described in more detail in References 9, 10 and 11.

3.3.4 Flaw Tolerance Evaluation and Inspection

When other methods cannot be used to show that the usage factor is less than 1.0, a flaw tolerance evaluation, coupled with inspection, may be used to justify continued operation. With this approach, it is assumed that a crack exists at the location where the CUF exceeds 1.0. Analysis is conducted to predict expected growth of the crack, including appropriate environmental effects. The crack size is chosen as that which might not be detected during an inspection of the location. An inspection interval is chosen based on the time for the assumed crack to grow to an allowable size. Figure 3-5 shows the approach for performing this assessment and managing fatigue in the extended operating period. An appropriate initial crack

size and aspect ratio and an appropriate crack growth law that considers environmental effects must be utilized. This can be accomplished by implementing ASME Section XI, Appendix L or other NRC-approved methodology.

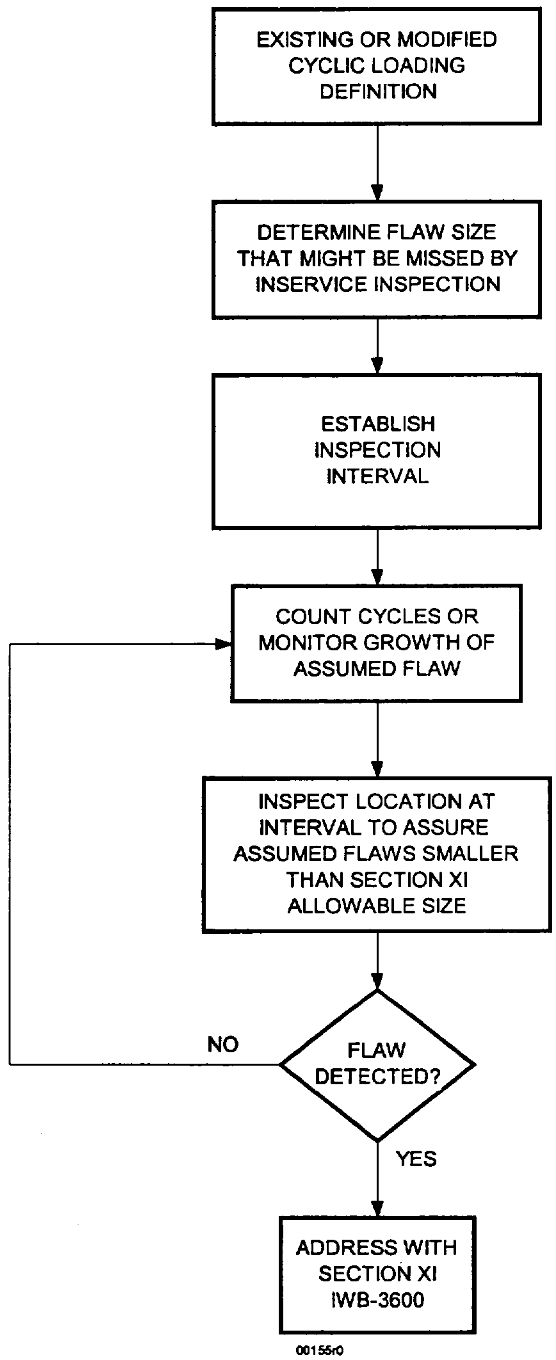

Figure 3-5
Fatigue Management Based on Flaw Tolerance

Determination of Expected Transient Loadings

Existing plant records must be reviewed to determine the cyclic loading specification (transient definition and number of cycles) and stress analysis for the location in question. Alternately, actual plant operating experience may be reviewed to define transients that are conservative but more accurately represent how the plant operates. A conservative definition of the number of plant operating cycles and transients that will occur during the extended operating period must be determined.

Initial Flaw Size Determination

The initial flaw size (depth and aspect ratio) shall be as defined in ASME Section XI, Appendix L or other NRC-approved methodology. (Work is ongoing to update the initial flaw size definition and to obtain regulatory acceptance.)

Crack Growth Evaluation

A revised stress analysis for the location may be required to determine the through-wall stress distribution for all significant transient and steady state conditions. Using an appropriate environmentally assisted fatigue crack growth law, fracture mechanics analysis must be conducted to predict the time to grow a crack to the Section XI allowable size. The allowable number of operating cycles and/or transients (or time) between inspections shall be determined per the requirements of ASME Section XI, Appendix L or other NRC-approved methodology.

Fatigue Management Approach

As shown in Figure 3-5, the fatigue management approach for the extended operating period consists of inspection of the location of the assumed flaw at an interval (or number of cycles) such that the assumed flaw does not exceed to the Section XI, Appendix L allowable flaw size, or such flaw size as allowed by other NRC-approved methodology. So long as no flaw is detected during inspection, the location is accepted for continued operation for another inspection interval. Before this approach can be used for fatigue management, the component must be examined to show that there is no detectable cracking. (If a flaw is ever detected, it must be dispositioned as required by ASME Section XI and related regulatory requirements.)

3.3.5 Modified Plant Operations

In some instances, actions may be taken to revise plant operations to reduce the transient severity or the rate of cycle accumulation. These effects will reduce the rate of usage accumulation (or crack growth rate) and can be taken into account in justifying extended operation.

3.3.6 Repair/Replacement

Although quite extreme, repair or replacement of affected locations may be economical, especially when combined with other component replacements. Prior component repairs or replacements may also be considered in establishing the current CUF at these locations.

3.3.7 Evaluation of Similar Components

If one of the fatigue management alternates above fails to show that limits can be satisfied, such that environmental fatigue usage limits are exceeded and repair/replacement or conversion to a flaw tolerance/inspection approach is required, an assessment of other similar locations shall be undertaken. The limiting location shall be added to the fatigue management program. The location chosen shall be one with similar loadings, geometry and materials.

3.4 Guidance for Plants with B31.1 Piping Systems

Many plants that were designed in the 1960's had piping systems that were designed in accordance with the rules of the ANSI B31.1 Power Piping Code. This Code did not require an explicit fatigue analysis. However, the effects of thermal expansion cycles were included. If the number of equivalent full range thermal expansion cycles was greater than 7,000, the allowable range of thermal expansion stress was reduced. There was no consideration of stresses due to through-wall thermal gradients, axial temperature gradients, or bi-metallic welds.

Although ANSI B31.1 and ASME Section III, Class 1 piping rules are fundamentally different, experience in operating plants has shown that piping systems designed to B31.1 are adequate. An evaluation of fatigue-sensitive B31.1 piping systems by EPRI [17] showed that there were only very limited locations in piping systems that exhibited high usage factors. In each case, these locations could be easily identified. It was concluded that high usage factors occurred only at locations that experienced significant thermal transients such as step temperature changes. In addition, the locations with high usage factors were always at structural or material discontinuities, such as pipe-to-valve or pipe-to-nozzle transition welds. The report also noted that the design features of B31.1 plants are essentially no different than those in more modern plants designed to ASME Section III, Class 1.

The high usage factor locations evaluated in NUREG/CR-6260 were primarily associated with piping systems discontinuities and occurred due to severe transients, except for PWR surge lines where a high number of stratification transients contributed to high usage factors.

The operation of B31.1 plants is also not different than that of plants designed to ASME Section III, Class 1. All have limitations on heatup/cooldown rates as required by ASME Section III/XI, and 10CFR50, Appendix G. The reactor vendors have provided feedback to plant operators to reduce the thermal fatigue challenges to components. Thus, the approach taken by an applicant with ANSI B31.1 piping systems need not be significantly different than that taken for a more modern plant:

- A sampling of fatigue sensitive locations can be taken, based on NUREG/CR-6260, possibly amplified by evaluations for a similar ASME Section III, Class 1 plant. For systems without specified design transients, a set of transients for tracking in the extended operating period must be established.

- Evaluations shall be undertaken to establish the usage factors at each of the locations. This may be based on similarity of geometry, materials, and transient cycles relative to other similarly-designed plants. Here, the information provided in NUREG/CR-6260 can be used. Alternately, an ASME Section III, Class 1 analysis can be conducted. This establishes the baseline fatigue usage, without environmental effects for the plant.

- Using this information, one of the approaches previously described for the ASME Section III, Class 1 plants can be used to evaluate and manage fatigue environmental effects.

3.5 Consideration of Industry Operating Experience

Consistent with current practice, industry experience with fatigue cracking will continue to be reviewed. The assessment of any fatigue cracking in the extended operating period will consider the effects of environment as a potential contributor. Monitoring of industry experience must consider fatigue cracking for both anticipated and unanticipated transients.

4 CONCLUSIONS

This report has developed several approaches that may be used by individual license renewal applicants to address the environmental effects on fatigue in a license renewal application. The approaches are geared to allow individual utilities to determine the optimum approach for their plants, allowing different approaches to be taken for different locations.

The overall approach taken for license renewal is to select a sampling of locations that might be affected by reactor water environmental effects. An assessment of the chosen locations is undertaken 1) to show that there is sufficient conservatism in the design basis transients to cover environmental effects, or 2) or to derive an expected fatigue usage factor including environmental effects. Then, either through tracking of reactor transient cycles or accumulated fatigue usage, utilities can determine if further steps must be taken to adequately manage fatigue environmental effects in the extended operating period.

A number of different methods are outlined for managing fatigue in the extended license renewal period should fatigue limits be exceeded. These include component reanalysis, fatigue monitoring, partial cycle counting, etc. Flaw tolerance evaluation as outlined in ASME Section XI, Appendix L, coupled with component inspection, is also included, although further work is underway by the Code to address regulatory concerns. Alternately, other NRC approved methodology could be used. Component repair/ replacement is also a possibility, but is recommended only where other approaches can not show acceptable results.

Consistent with current ASME Section XI philosophy for conducting additional examinations when flaws are found in service, the program includes expansion of the number of locations tracked if fatigue limits are exceeded in the extended operating period. In addition, utilities will continue to monitor operating plant fatigue experience, especially with respect to cracking that might indicate a strong contribution from fatigue environmental effects.

Using the guidance provided herein, the amount of effort needed to justify individual submittals and respond to NRC questions should be minimized.

5
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2. NUREG/CR-6260 (INEL-95/0045), "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995.

3. NUREG/CR-6583 (ANL-97/18), "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," March 1998.

4. NUREG/CR-5704 (ANL-98/31), "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.

5. NUREG/CR-6674 (PNNL-13227), "Fatigue Analysis of Components for 60-Year Plant Life," June 2000.

6. U. S. Nuclear Regulatory Commission, Generic Safety Issue 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life."

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15. Letter from Greg Hollinger (PVR) to J. H. Ferguson, Chairman Board of Nuclear Codes and Standards, October 31, 1999.

16. Mehta, H. S., "An Update on the Consideration of Reactor Water Effects in Code Fatigue Initiation Evaluations for Pressure Vessels and Piping," PVP-Vol. 410-2, p45-51, American Society of Mechanical Engineers, 2000.

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A
ASSESSMENT OF NUREG/CR-6260 RESULTS

1.0 INTRODUCTION

NUREG/CR-6260 [A1] documents a study by Idaho National Engineering Laboratories (INEL) in 1995 for the Nuclear Regulatory Commission (NRC) to assess the effects of light water reactor environment on fatigue of reactor coolant system components. At the time of the study, the "interim fatigue curves" from NUREG/CR-5999 [A2] were the only ones available; except that some revised interim fatigue curves for stainless steel were provided by Argonne National Laboratories (ANL) in 1994, as documented in NUREG/CR-6260. Following completion of this study, there have been numerous studies by industry, the Japanese, and ANL to improve the data, criteria, and methods for evaluation of fatigue environmental effects. The purpose of this Appendix is to summarize the NUREG/CR-6260 results and to assess the results relative to use as a baseline for evaluating fatigue environmental effects in license renewal. Appendix A, Section 2.0 discusses the "interim fatigue curves" and how they compare to the current data for reactor coolant components. Section 3.0 provides an evaluation of the specific reactor types evaluated in NUREG/CR-6260. The review shows that some of the locations can be excluded from consideration in license renewal evaluations. Section 4.0 provides conclusions reached from this evaluation.

2.0 EVALUATION OF NUREG/CR-6260 LOCATIONS USING LATEST ENVIRONMENTAL DATA

In NUREG/CR-6260, the fatigue curves were provided in the form of digitized points of stress amplitude versus number of cycles (S-N curves). The following curves were provided from NUREG/CR-5999:

Stainless Steel:	Single curve for all temperatures and strain rates that was also applicable to Alloy 600. This was updated for stainless steel after the issuance of NUREG/CR-5999 with an equation that also included strain rate dependency.
Carbon/Low Alloy Steel:	Low Oxygen water (single curve) High Oxygen 200°C (various strain rates) High Oxygen 250°C (various strain rates) High Oxygen 288°C (various strain rates)

Since equations were not provided, it is difficult to numerically compare these curves to that which would result using current ANL recommendations for carbon, low alloy and stainless steel. Figures were provided in NUREG/CR-6260 that plotted "Factor of Increase" for each curve relative to the ASME Code Fatigue curves. The "Factor of Increase" is equivalent to the common term F_{en} that is the ratio of fatigue usage with environmental effects divided by fatigue usage with air, or allowable cycles to fatigue crack initiation in air divided by allowable cycles with water reactor environmental effects. F_{en} equations are provided in the latest ANL reports for carbon and low alloy steel [A3] and stainless steel [A4]. The environmental correction factor (F_{en}) relative to room-temperature air for Types 304 and 316 stainless steel is given by:

$$F_{en}=\exp(0.935 - T^* \dot{\varepsilon}^* O^*).$$

where the constants for transformed temperature (T^*), strain rate ($\dot{\varepsilon}^*$), and dissolved oxygen (O^*) are defined as follows:

$T^* = 0$	$(T < 200°C)$
$T^* = 1$	$(T \geq. 200°C)$
$\dot{\varepsilon}^* = 0$	$(\dot{\varepsilon} > 0.4\%/sec)$
$\dot{\varepsilon}^* = \ell n(\dot{\varepsilon}/0.4)$	$(0.0004 \leq \dot{\varepsilon} \leq 0.4\%/sec)$
$\dot{\varepsilon}^* = \ell n(0.0004/0.4)$	$(\dot{\varepsilon} < 0.0004\%/sec)$
$0^* = 0.260$	$(DO < 0.05$ ppm$)$
$0^* = 0.172$	$(DO \geq 0.05$ ppm$)$

In the above,

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

The environmental correction factors relative to room-temperature air for carbon steel and alloy steel are given by:

$$F_{en} = \exp(0.585 - 0.00124\,T - 0.101 S^* T^* O^* \dot{\varepsilon}^*) \quad \text{(CS)}$$

$$F_{en} = \exp(0.929 - 0.00124\,T - 0.101 S^* O^* \dot{\varepsilon}^*) \quad \text{(LAS)}$$

where the transformed sulfur content (S^*), temperature (T^*), dissolved oxygen (O^*), and strain rate ($\dot{\varepsilon}^*$) are defined as follows:

$S^* = S$	$(0 < S \leq 0.015$ wt. %)
$S^* = 0.015$	$(S > 0.015$ wt. %)
$T^* = 0$	$(T < 150°C)$
$T^* = T - 150$	$(150 \leq T \leq 350°C)$
$0^* = 0$	(DO < 0.05 ppm)
$0^* = \ell n\,(DO/0.04)$	(0.05 ppm $\leq$ DO $\leq$ 0.5 ppm)
$0^* = \ell n\,(12.5)$	(DO > 0.5 ppm)
$\dot{\varepsilon}^* = 0$	($\dot{\varepsilon} > 1$ %/s)
$\dot{\varepsilon}^* = \ell n\,(\dot{\varepsilon})$	($0.001 \leq \dot{\varepsilon} \leq$ %/s)
$\dot{\varepsilon}^* = \ell n\,(0.001)$	($\dot{\varepsilon} < 0.001$ %/s)

In the above, the temperature (T), dissolved oxygen (DO) and strain rate ($\dot{\varepsilon}$) are as previously defined. The weight percent sulfur is S.

Using these evaluations, the current recommended fatigue life correction factors can be compared to that used in NUREG/CR-6260.

Figure A-1 compares the environmental factors for NUREG/CR-6260 and NUREG/CR-5704 for stainless steel. It is observed that the correction factor in NUREG/CR-6260 was dependent upon the alternating stress amplitude, but was not affected by oxygen content or strain rate.

PWR reactors generally operate at low oxygen. Thus, except in the region near S_a = 65ksi or for high strain rate transients, it would be expected that environmental effects would be more severe than reported in NUREG/CR-6260. Since strain rate is generally not available from stress reports utilized in NUREG/CR-6260, the increase is expected to be approximately 15.4/11 ~ 1.4 or a 40 percent increase. The factor of 11 is an estimated mean value of the "Factor of Increase".

For BWR reactors, the oxygen level is generally much higher. Therefore, the actual environmental factors would be lower than in NUREG/CR-6260 by approximately 8.4/11 ~ 0.75 or 25 percent less.

Figure A-2 shows a similar comparison for carbon and low-alloy steels in a low-oxygen environment as might be expected for PWRs. In this case, the environmental factors for low alloy steel are comparable except for high and low stress amplitudes. Thus, the NUREG/CR-6260 values should be about what would be expected with the new environmental data. For carbon steel, the expected environmental factors would be lower by approximately 1.5/2.1 ≈ 0.75 or 25 percent less.

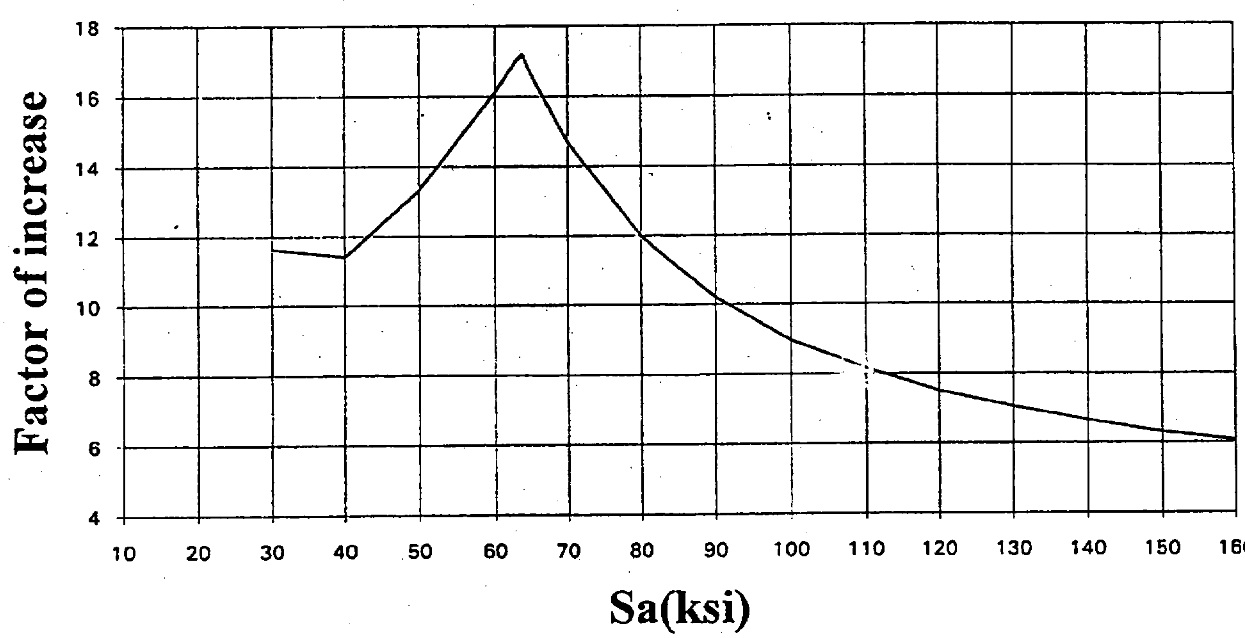
Figures A-3 to A-5 show comparisons for carbon/low-alloy steels in high oxygen water. In these cases, the current low temperature (200°C) environmental factors appear to be less than that evaluated in NUREG/CR-6260. The high-temperature (288°C) environmental factor in

Figure A-5 would be higher by approximately 58/45 ≈ 1.3 or 30 percent for carbon steel and 82/45 ≈ 1.8 or 80 percent for low alloy steel. These comparisons would be applicable to BWR components/environments.

In NUREG/CR-6260, revised interim fatigue curves for stainless steel were provided that had a strain rate effect. Comparisons between the NUREG/CR-6260 basic, the revised interim and the NUREG/CR-5704 environmental factors is shown in Figure A-6.

In summary, Table A-1 shows a comparison of environmental factor increase or decrease based on current data as compared to that of NUREG/CR-6260. The comparison is approximate, and actual differences would have to be determined based on alternating stresses, strain rates (if available), temperature, dissolved oxygen, etc.

NUREG/CR-6260 Figure 3-6:



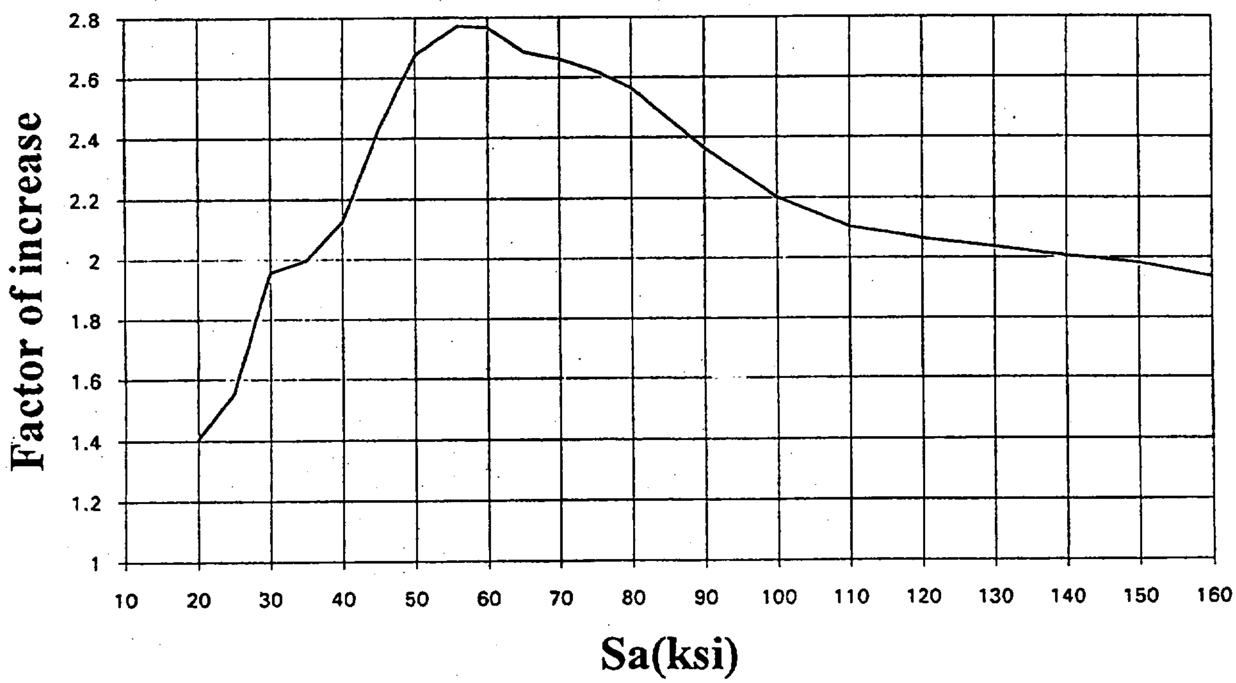
Note: Mean "Factor of Increase" estimated to be ≈ 11

NUREG/CR-5704 Eq. 13:

$\dot{\varepsilon}$, %/sec	DO, ppm	F_{en}
≤ 0.0004	< 0.05	15.4
0.004	< 0.05	8.4
0.04	< 0.05	4.6
0.4	< 0.05	2.5
≤ 0.0004	≥ 0.05	8.4
0.004	≥ 0.05	5.6
0.04	≥ 0.05	3.8
0.4	≥ 0.05	2.5

Figure A-1. Environmental Correction Factor in NUREG/CR-6260 Compared to NUREG/CR-5704 Recommendation for Stainless Steel

NUREG/CR-6260 Figure 3-7 (DO < 0.1 ppm):



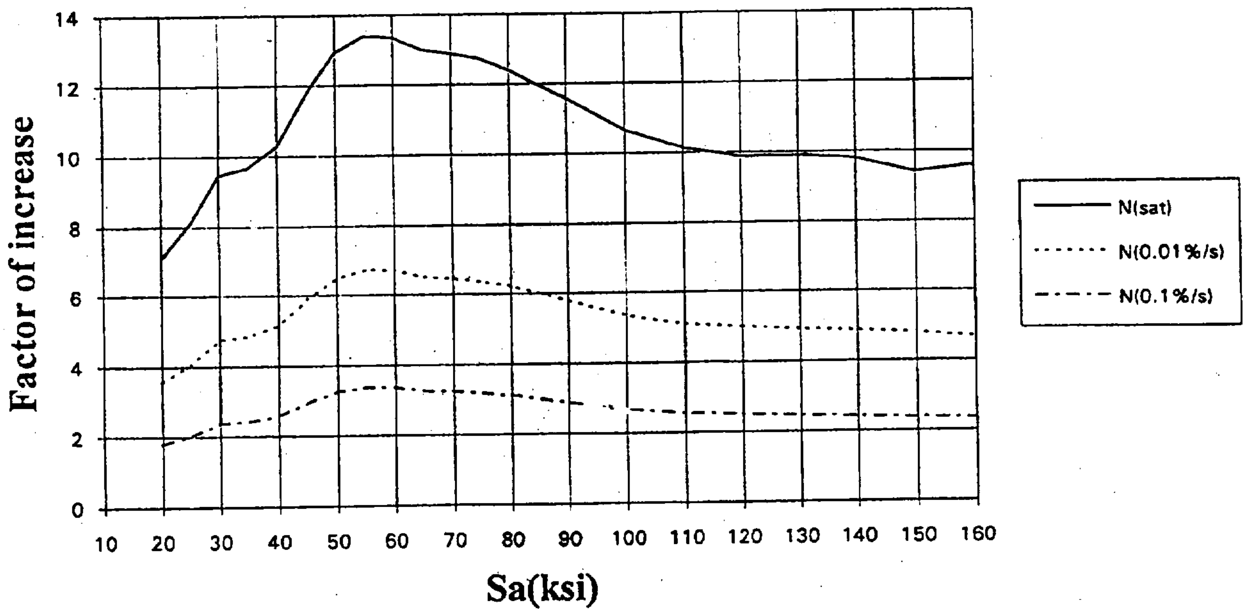
Note: Mean "Factor of Increase" estimated to be ≈ 2.1

NUREG/CR-6583 Equation 6.5a/6.5b (DO ≤ 0.05 ppm):

Temp., °C	Material	F_{en}
200	CS	1.7
250	CS	1.6
288	CS	1.5
200	LAS	2.4
250	LAS	2.3
288	LAS	2.2

Figure A-2. Environmental Correction Factor in NUREG/CR-6260 Compared to NUREG-6583 Recommendation for Carbon/Low-Alloy Steel in Low-Oxygen Environment

NUREG/CR-6260 Figure 3-8 (DO > 0.1 ppm):

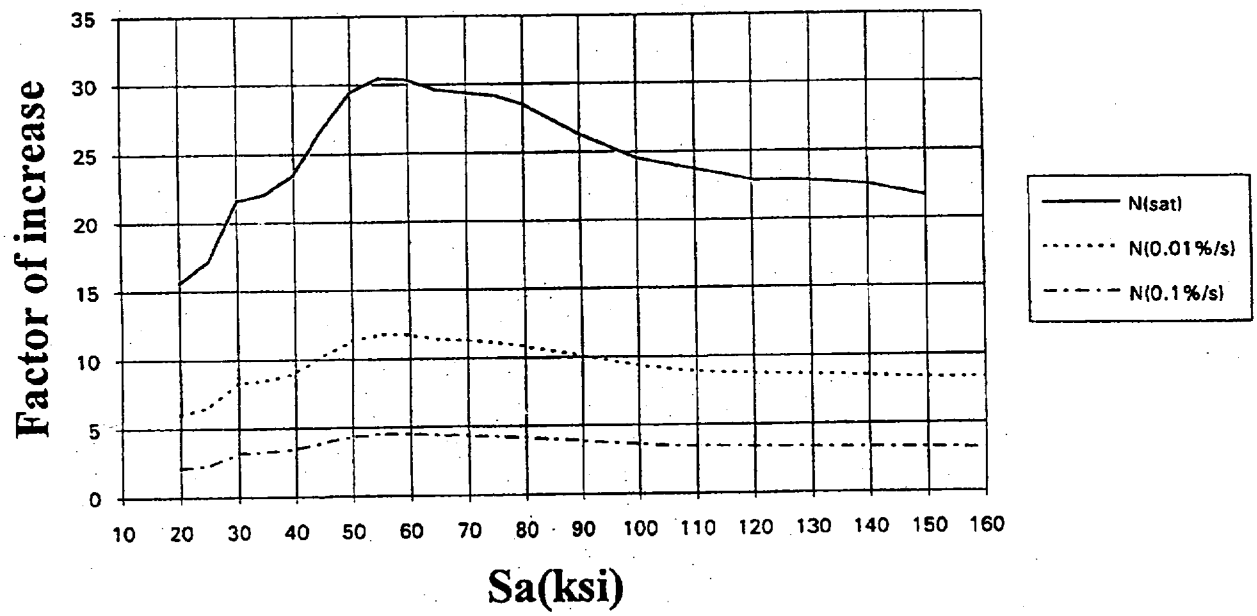


NUREG/CR-6583 Equation 6.5a/6.5b (DO ≥ 0.5 ppm):

$\dot{\varepsilon}$, %/sec	Material	F_{en}
≤ 0.001	CS	6.3
0.01	CS	4.1
0.1	CS	2.6
≤ 0.001	LAS	8.9
0.01	LAS	5.7
0.1	LAS	3.7

Figure A-3. Environmental Correction Factor in NUREG/CR-6260 Compared to NUREG/CR-6583 Recommendation for Carbon and Low-Alloy Steel in High Oxygen Environment at 200°C

NUREG/CR-6260 Figure 3-9 (DO > 0.1 ppm):

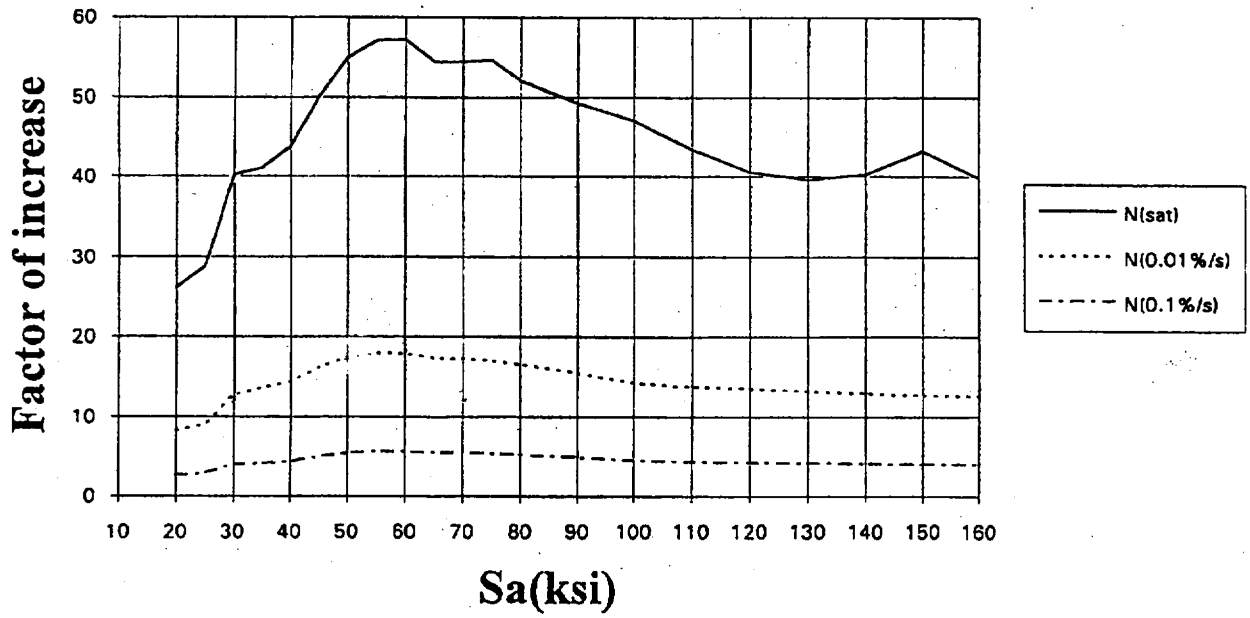


NUREG/CR-6583 Equation 6.5a/6.5b (DO ≥ 0.5 ppm):

$\dot{\varepsilon}$, %/sec	Material	F_{en}
≤ 0.001	CS	27.3
0.01	CS	9.23
0.1	CS	3.8
≤ 0.001	LAS	31.4
0.01	LAS	13.0
0.1	LAS	5.4

Figure A-4. Environmental Correction Factor in NUREG/CR-6260 Compared to NUREG/CR-6583 Recommendation for Carbon and Low-Alloy Steel in High Oxygen Environment a 250°C.

NUREG/CR-6260 Figure 3-10 (DO > 0.1 ppm):



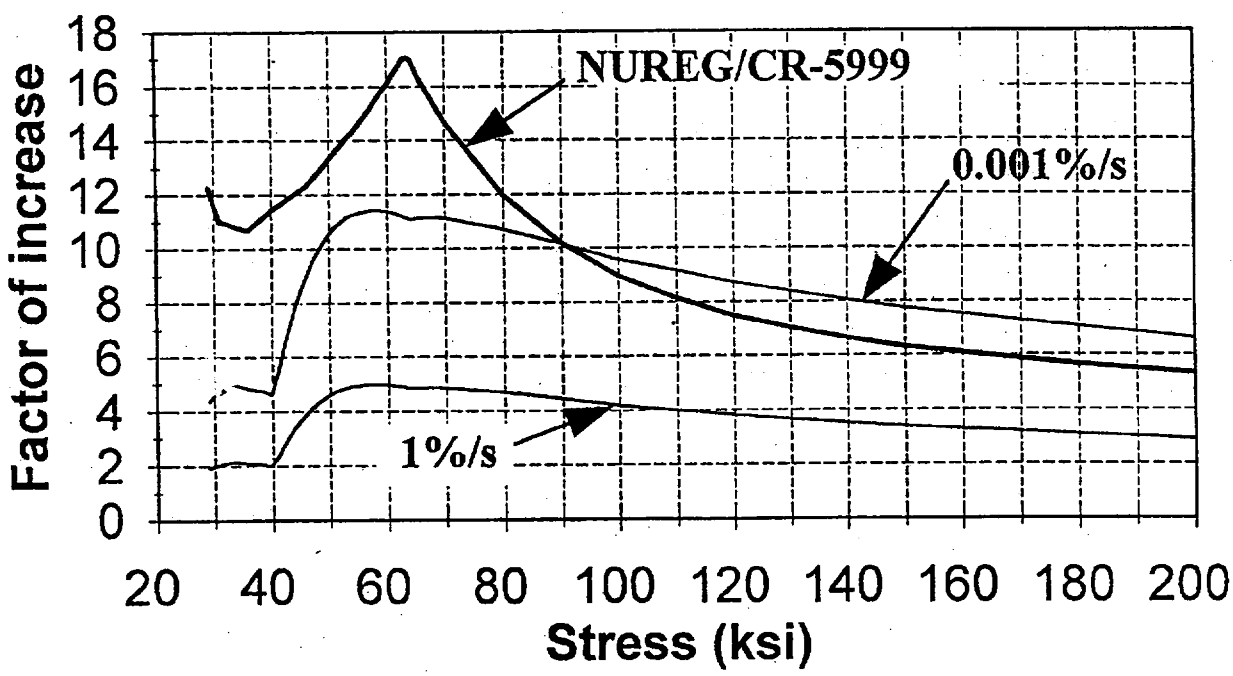
Note: Mean "Factor of Increase" estimated to be ≈ 45 for N(Sat)

NUREG/CR-6583 Equation 6.5a/6.5b (DO ≥ 0.5 ppm)

$\dot{\varepsilon}$, %/sec	Material	F_{en}
≤ 0.001	CS	58.1
0.01	CS	17.2
0.1	CS	5.1
≤ 0.001	LAS	81.9
0.01	LAS	24.3
0.1	LAS	7.2

Figure A-5. Environmental Correction Factor in NUREG/CR-6260 Compared to NUREG/CR-6583 Recommendation for Carbon and Low-Alloy Steel in High Oxygen Environment at 288°C

NUREG/CR-6260 Figure 3-19:



Note: Curves for 1%/s and 0.001 %/s were revised curves considered in NUREG-6260

NUREG/CR-5704 Eq. 13:

$\dot{\varepsilon}$, %/sec	DO, ppm	F_{en}
≤ 0.0004	< 0.05	15.4
0.001	< 0.05	12.1
0.004	< 0.05	8.4
0.04	< 0.05	4.6
0.4	< 0.05	2.5
≤ 0.0004	≥ 0.05	8.4
0.004	≥ 0.05	5.6
0.04	≥ 0.05	3.8
0.1	≥ 0.05	3.6
0.4	≥ 0.05	2.5

Figure A-6. Fatigue Life Correction Factor In NUREG/CR-6260 Compared to NUREG/CR-5704 Recommendation for Stainless Steel Curves

Table A-1

Approximate Increase or Decrease in NUREG/CR-6260 Usage Factors Accounting for NUREG/CR-5704 and NUREG/CR–6583 Environmental Factors

	Change, Percent		
Environment	Carbon Steel	Low-Alloy Steel	Stainless Steel
BWR (high 0_2)	+30	+80	-25
PWR (low 0_2)	-25	0	+40

3.0 EVALUATION OF NUREG/CR-6260 CUF SUMMARIES

The following evaluates the usage factor summaries provided in NUREG/CR-6260 for each of the reactor types addressed. The objective is to provide some insight into the relative importance of the locations as candidates for fatigue management in the extended operating period.

In evaluating the NUREG/CR-6260 summary tables, some additional information was found from the text description. This has been reported where significant. The 60-year extrapolations are not shown herein, since in most applications, the 60-year design cycles are the same as those expected for 40 years. On the other hand, if the 40-year CUF predictions in NUREG/CR-6260 were based on expected cycles, it would be appropriate to mentally compare these to a limit of CUF = 0.666 (at 40-years) to compare with a limit of CUF = 1.0 (at 60 years), or alternately multiply the 40-year CUF by a factor of 1.5.

In the previous section, the assessment of the effect of more recent environmental fatigue testing was approximated because the environmental factor was stress dependent. In the following, the NUREG/CR-6260 CUFs are re-evaluated to estimate the effect of later data. In these assessments, the environmental CUFs less than 0.5 can be assumed to require no additional assessment in a license renewal application. This assumption provides for a factor of two uncertainty in the assessment.

Thus, in calculating an estimated 60 year usage factor accounting for new data, the following formula is used:

$$CUF_{new} = F_{fen} \times F_{60} \times CUF_{old}$$

where:

CUF_{old} = usage factor from NUREG/CR-6260

F_{fen} = factor relating new environmental data fatigue curve to that used in NUREG/CR-6260

F_{60} = factor to extrapolate to 60 years

= 1.0 if CUF_{old} based on design cycles

= 1.5 CUF_{old} based on expected cycles at 40 years

3.1 Newer Vintage Combustion Engineering Plant

The summary of the NUREG/CR-6260 environmental evaluation is shown in Table A-2. The locations evaluated were either stainless steel or low-alloy steel. Since the plant is a PWR, low oxygen would be expected. From Table A-1, the reported usage factors are assumed to be unchanged for the low-alloy steel locations and increased by about 40 percent for the stainless steel locations.

Table A-3 shows either the design usage factors or the 60-year expected usage factors, corrected for environmental effects.

Table A-3
Revised Estimate of NUREG/CR-6260 CUFs for New Vintage Combustion Engineering Plant

Location	CUF_{new}
Vessel Head/Shell*	Design CUF_{old} = 1.0 x 1.0 x 0.014 = 0.014
Vessel Inlet Nozzle*	Design CUF_{old} = 1.0 x 1.0 x 0.475 = 0.475
Vessel Outlet Nozzle	Expected CUF_{old} = 1.0 x 1.5 x 0.472 = 0.71
Surge Line Elbow	Expected CUF_{old} = 1.4 x 1.5 x 3.476 = 7.30
Charging Nozzle*	Design CUF_{old} = 1.0 x 1.0 0.104 = 0.104
Charging Safe-End	Expected CUF_{old} = 1.4 x 1.5 x 0.774 = 1.62
SI Nozzle	Expected CUF_{old} = 1.0 x 1.5 x .475 = 0.72
SI Safe-end	Expected CUF_{old} = 1.4 x 1.5 x 0.387 = 0.81
SDC Elbow	Expected CUF_{old} = 1.4 x 1.5 x 0.502 = 1.05

* Locations with CUF low enough that further evaluation not required.

Table A-2
Summary of Newer Vintage Combustion Engineering Plant CUFs (40 Year Life)

Component	Location	Material	Design CUF	NUREG/CR-5999 CUF			Revised SS Curve
				Based on Design Stresses	Conservative Assumptions Removed	Based on Expected Cycles	
Reactor vessel	Lower head / shell	LAS	0.007	0.014	(1)	(1)	N/A
	Inlet nozzle	LAS	0.182	0.475	(1)	(1)	N/A
	Outlet nozzle	LAS	0.377 $(0.334)^2$	0.835	(1)	0.472	N/A
Surge line	Elbow	Stainless steel	0.981	8.684	(1)	3.476	2.597
Charging nozzle	Nozzle	LAS	0.050	0.104	(1)	(1)	N/A
	Safe end	Stainless steel	0.778	4.193	2.556	0.774	0.502
Safety injection nozzle	Nozzle	LAS	0.898	2.101	(1)	0.457	N/A
	Safe end	Stainless steel	0.360	3.215	1.609	0.387	0.286
Shutdown cooling line	Elbow	Stainless steel	0.894	6.100	2.030	0.502	0.487

Note (1): No additional calculations were performed.
(2): Outer surface CUF = 0.377, inner surface CUF = 0.334

3.2 Older Vintage Combustion Engineering Plant

The summary of the NUREG/CR-6260 environmental evaluation is shown in Table A-4. The locations evaluated were either stainless steel or low-alloy steel. Since the plant is a PWR, low oxygen would be expected. From Table A-1, the reported usage factors will remain unchanged for the low-alloy steel locations and increased by about 40 percent for the stainless steel locations.

Table A-5 shows either the design usage factors or the 60-year expected usage factors, corrected for environmental effects.

Table A-5
Revised Estimate of NUREG/CR-6260 CUFs for Older Vintage Combustion Engineering Plant

Location	CUF_{old}
Vessel lower head/Shell*	Design $CUF_{new} = 1.0 \times 1.0 \times 0.013 = 0.013$
Vessel Inlet Nozzle*	Design $CUF_{new} = 1.0 \times 1.0 \times 0.172 = 0.17$
Vessel Outlet Nozzle	Design $CUF_{new} = 1.0 \times 1.0 \times 0.554 = 0.55$
Surge Line Elbow	Expected $CUF_{new} = 1.4 \times 1.5 \times 1.345 = 2.82$
Charging Nozzle	Expected $CUF_{new} = 1.4 \times 1.5 \times 0.666 = 1.40$
SI Nozzle	Expected $CUF_{new} = 1.4 \times 1.5 \times 0.414 = 0.87$
SDC Inlet*	Design $CUF_{new} = 1.4 \times 1.0 \times 0.139 = 0.20$

* Locations with CUF low enough that further evaluation not required.

Table A-4
Summary of Older Vintage Combustion Engineering Plant CUFs (40 Year Life)

Component	Location	Material	Design CUF	NUREG/CR-5999 CUF			Revised SS Curve
				Based on Design Stresses	Conservative Assumptions Removed	Based on Expected Cycles	
Reactor vessel	Lower head to shell juncture	LAS	0.008	0.013	(1)	(1)	N/A
	Inlet nozzle	LAS	0.073	0.172	(1)	(1)	N/A
	Outlet nozzle	LAS	0.284	0.554	(1)	(1)	N/A
Surge line	Elbow	Stainless steel	0.705	8.070	(1)	1.345	0.661
Charging nozzle	Nozzle	Stainless steel	0.266	3.918	(1)	0.666	0.562
Safety injection nozzle	Nozzle	Stainless steel	0.088	1.320	(1)	0.414	0.317
Shutdown cooling line	Inlet transition	Stainless steel	0.014(2)	0.139	(1)	(1)	0.084

Note (1): No additional calculations were performed.
Note (2): Estimated by INEL. CUF calculation not required by licensing basis.

3.3 B&W 177 Fuel Element Assembly Plant

The summary of the NUREG/CR-6260 environmental evaluation is shown in Table A-6. The locations evaluated were either stainless steel, low-alloy steel, carbon steel or Alloy 600. Since the plant is a PWR, low oxygen would be expected. From Table A-1, the reported usage factors will remain unchanged for the low-alloy steel locations and increased by about 40 percent for the stainless steel locations. The carbon steel location usage factor will be reduced by 25 percent. It will be assumed that the Alloy 600 location remains applicable.

Table A-7 shows either the design usage factors or the 60-year expected usage factors, corrected for environmental effects.

Table A-7
Revised Estimate of NUREG/CR-6260 CUFs for B&W 177 Fuel Element Assembly Plant

Location	CUF_{new}
Vessel at Support*	Design CUF_{old} = 1.0 x 1.0 x 0.223 = 0.22
Lower Head Penetration	Expected CUF_{old} = 1.0 x 1.5 x 0.742 = 1.11
Outlet Nozzle*	Design CUF_{old} = 1.0 x 1.0 x 0.469 = 0.47
Surge Line Nozzle*	Expected CUF_{old} = 0.75 x 1.5 x 0.47 = 0.50
Surge Line Elbow	Expected CUF_{old} = 1.4 x 1.5 x 2.005 = 4.21
HPI/Makeup Nozzle	Expected CUF_{old} = 1.4 x 1.5 x 1.263 = 2.65
Core Flood Nozzle	Design CUF_{old} = 1.0 x 1.0 x 0.632 = 0.63
Decay Heat Tee	Expected CUF_{old} = 1.4 x 1.5 x 0.61 = 1.28

* Locations with CUF low enough that further evaluation not required.

Table A-6
Summary of B&W 177 Fuel Assembly Plant CUFs (40 Year Life)

Component	Location	Material	Design CUF	NUREG/CR-5999 CUF			Revised SS Curve
				Based on Design Stresses	Conservative Assumptions Removed	Based on Expected Cycles	
Reactor vessel	Near support skirt juncture	LAS	0.120	0.223	(1)	(1)	N/A
	Lower head penetration weld	Ni-Cr-Fe	0.097	1.466	(1)	0.742	0.546
	Outlet nozzle	LAS	0.900	2.148	0.469	(1)	N/A
Surge line	Hot leg nozzle	Carbon steel	0.592	1.092	(1)	0.470	N/A
	Pipe elbow	Stainless steel	0.490	4.656(2)	(1)	2.005	1.338
Makeup/ HPI nozzle	Safe end	Stainless steel	0.740	3.977	(1)	1.263	1.051
Core flood nozzle	Nozzle	LAS	0.345	0.632	(1)	(1)	N/A
Decay heat removal line (3)	Reducing tee	Stainless Steel	3.310	14.209	1.296	0.610	0.530

Note (1): No additional calculations were performed.
Note (2): Based on multiplier from other four PWR plant surge lines.
Note (3): From alternate B&W 177 fuel assembly plant.

3.4 Newer Vintage Westinghouse Plant

The summary of the NUREG/CR-6260 environmental evaluation is shown in Table A-8. The locations evaluated were either stainless steel or low-alloy steel. Since the plant is a PWR, low oxygen would be expected. From Table A-1, the reported usage factors should remain unchanged for the low-alloy steel locations and increased by about 40 percent for the stainless steel locations.

Table A-9 shows either the design usage factors or the 60-year expected usage factors, corrected for environmental effects.

Table A-9
Revised Estimate of NUREG/CR-6260 CUFs for Newer Vintage Westinghouse Plant

Location	CUF_{old}
Vessel Head Junction*	Design CUF_{new} = 1.0 x 1.0 x 0.018 = 0.02
Vessel Inlet Nozzle*	Design CUF_{new} = 1.0 x 1.0 x 0.29 = 0.29
Vessel Outlet Nozzle	Design CUF_{new} = = 1.0 x 1.0 x 0.658 = 0.68
Surge Line Nozzle	Expected CUF_{new} = = 1.4 x 1.5 x 2.458 = 5.16
Charging Nozzle	Design CUF_{new} = 1.4 x 1.0 x 4.859 = 6.80
SI Nozzle	Expected CUF_{new} = 1.4 x 1.5 x 1.511 = 3.17
RHR Inlet	Expected CUF_{new} = 1.4 x 1.5 x 2.371 = 4.98

* Locations with CUF low enough that further evaluation not required.

Table A-8
Summary of Newer Vintage Westinghouse Plant CUFs (40 Year Life)

Component	Location	Material	Design CUF	NUREG/CR-5999 CUF			Revised SS Curve
				Based on Design Stresses	Conservative Assumptions Removed	Based on Expected Cycles	
Reactor vessel	Lower head to shell juncture	LAS	0.012	0.018	(1)	(1)	N/A
	Inlet nozzle	LAS	0.110	0.290	(1)	(1)	N/A
	Outlet nozzle	LAS	0.398	0.658	(1)	(1)	N/A
Surge line	Hot leg nozzle	Stainless steel	0.743	7.562	(1)	2.458	1.734
Charging nozzle	Nozzle	Stainless steel	0.829	5.188	4.859	(1)	3.373
Safety injection nozzle	Nozzle	Stainless steel	0.966	4.874	4.145	1.511	1.460
Residual heat removal line	Inlet transition	Stainless steel	0.896(2)	5.727	(1)	2.371	2.733

Note (1): No additional calculations were performed.
(2): Without stratification, CUF = 0.243

3.5 Older Vintage Westinghouse Plant

The summary of the NUREG/CR-6260 environmental evaluation is shown in Table A-10 The locations evaluated were either stainless steel or low-alloy steel. Since the plant is a PWR, low oxygen would be expected. From Table A-1, the reported usage factors should remain unchanged for the low-alloy steel locations and increased by about 40 percent for the stainless steel locations.

Table A-11 shows either the design usage factors or the 60-year expected usage factors, corrected for environmental effects.

Table A-11
Revised Estimate of NUREG/CR-6260 CUFs for Older Vintage Westinghouse Plant

Location	CUF_{new}
Vessel Support Weld	Design CUF_{old} = 1.0 x 1.0 x 0.891 = 0.89
Vessel Inlet Nozzle	Design CUF_{old} = 1.0 x 1.0 x 0.496 = 0.50
Vessel Outlet Nozzle	Expected CUF_{old} = 1.0 x 1.5 x 0.347 = 0.52
Surge Line Safe-End	Expected CUF_{old} = 1.4 x 1.5 x 5.86 = 12.31
Charging Nozzle Inlet*	Design CUF_{old} = 1.4 x 1.0 x 0.349 = 0.49
SI Nozzle Weld	Design CUF_{old} = 1.4 x 1.0 x 0.416 = 0.58
RHR Tee*	Design CUF_{old} = 1.4 x 1.0 x 0.286 = 0.40

* Locations with CUF low enough that further evaluation not required.

Table A-10
Summary of Older Vintage Westinghouse Plant CUFs (40 Year Life)

Component	Location	Material	Design CUF	NUREG/CR-5999 CUF			Revised SS Curve
				Based on Design Stresses	Conservative Assumptions Removed	Based on Expected Cycles	
Reactor vessel	Core support guide weld	LAS	0.290	0.891	(1)	(1)	N/A
	Inlet nozzle	LAS	0.208 (3) (0.135)	0.496 (3) (0.302)	(1)	(1)	N/A
	Outlet nozzle	LAS	0.431 (0.193)	1.161 (0.499)	(1)	0.347	N/A
Surge line	Hot leg nozzle safe end	Stainless steel	0.900	6.814	(1)	5.860	4.248
Charging nozzle	Nozzle inlet	Stainless steel	0.030 (2)	0.349	(1)	(1)	0.319
Safety injection nozzle	Nozzle-to-pipe weld	Stainless steel	0.046 (2)	0.416	(1)	0.410	0.327
Residual heat removal line	Tee	Stainless steel	0.022 (2)	0.286	(1)	(1)	0.205

Note (1): No additional calculations were performed.
Note (2): Estimated by INEL. CUF calculation not required by licensing basis.
Note (3) Outside surface; numbers in parentheses are inside surface

3.6 Newer Vintage GE Plant

The summary of the NUREG/CR-6260 environmental evaluation is shown in Table A-12. The locations evaluated were either Alloy 600, stainless steel, carbon steel or low-alloy steel. Since the plant is a BWR, high oxygen would be expected. From Table A-1, the reported usage factors should be increased by about 80 percent for the low-alloy steel locations, increased by about 30 percent for carbon steel locations and decreased by about 25 percent for the stainless steel locations. It will be assumed that the environmental effects for the Alloy 600 locations remain applicable.

Table A-13 shows either the design usage factors or the 60-year expected usage factors, corrected for environmental effects:

Table A-13
Revised Estimate of NUREG/CR-6260 CUFs for Newer Vintage GE Plant

Location	CUF_{old}
Vessel at CRDM Penetration	Design CUF_{new} = 1.8 x 1.0 x 0.628 = 1.13
CRDM Weld*	Design CUF_{new} = 1.0 x 1.0 x 0.474 = 0.47
FW Thermal Sleeve	Expected CUF_{new} = 1.0 x 1.5 x 8.322 = 12.48
FW Safe-End	Design CUF_{new} = 1.3 x 1.0 x 1.085 = 1.41
Recirc. Tee	Design CUF_{new} = 0.75 x 1.0 x 0.83 = 0.62
CS Thermal Sleeve	Expected CUF_{new} = 1.5 x 1.0 x 0.517 = 0.77
CS Safe-End	Design CUF_{new} = 1.3 x 1.0 x 0.436 = 0.57
RHR Pipe	Design CUF_{new} = 1.3 x 1.0 x 11.26 = 14.63
FW Elbow	Design CUF_{new} = 1.3 x 1.0 x 3.688 = 4.79

* Locations with CUF low enough that further evaluation not required.

Table A-12
Summary of Newer Vintage GE Plant CUFs (40 Year Life)

Component	Location	Material	Design CUF	NUREG/CR-5999 CUF			Revised SS Curve
				Based on Design Stresses	Conservative Assumptions Removed	Based on Anticipated Cycles	
Reactor vessel	Near CRDM penetration	LAS	0.200	11.702	0.628	(1)	N/A
	CRDM penetration weld	Ni-Cr-Fe	0.407	2.716	0.474	(1)	0.359
Surge line	Thermal sleeve	Ni-Cr-Fe	0.795	5.141	(1)	8.322	6.471
	Safe end	Carbon steel	0.301	1.730	1.085	1.881	N/A
Recircu-lation system	Tee on suction pipe	Stainless steel	0.298	2.154	0.830	(1)	0.746
Core spray line	Nozzle thermal sleeve	Ni-Cr-Fe	0.165	0.943	(1)	0.517	0.637
	Safe end extension	Carbon steel	0.050	0.675	0.436	(1)	N/A
RHR line	Straight pipe	Carbon steel	0.407	11.260 (2)	(1)	(1)	N/A
Feedwater line	Elbow	Carbon steel	0.435	3.746	3.688	(1)	N/A

Note (1): No additional calculations were performed.
Note (2): Heavily influenced by thermal stratification transient and insufficient information to determine strain rate.

3.7 Older Vintage GE Plant

The summary of the NUREG/CR-6260 environmental evaluation is shown in Table A-14. The locations evaluated were either Alloy 600, stainless steel, carbon steel or low-alloy steel. Since the plant is a BWR, high oxygen would be expected. From Table A-1, the reported usage factors should be increased by about 80 percent for the low-alloy steel locations, increased by about 30 percent for carbon steel locations and decreased by about 25 percent for the stainless steel locations.

Table A-15 shows either the design usage factors or the 60-year expected usage factors, corrected for environmental effects.

Table A-15
Revised Estimate of NUREG/CR-6260 CUFs for Older Vintage GE Plant

Location	CUF_{new}
Vessel lower Head Penetration*	Design CUF_{old} = 1.8 x 0.079 = 0.14
FW Nozzle Bore	Expected CUF_{old} = 1.8 x 1.5 x 3.168 = 7.13
Recirc. RHR Tee	Expected CUF_{old} = 0.75 x 1.5 x 3.898 = 4.38
CS Nozzle	Expected CUF_{old} = 1.8 x 1.5 x 0.52 =1.4
CS Safe-End	Expected CUF_{old} = 0.75 x 1.5 x 2.305 = 2.59
RHR Transition	Expected CUF_{old} = 0.75 x 1.5 x 0.523 = 0.59
FW RCIC Tee	Expected CUF_{old} = 1.3 x 1.5 x 6.98 =13.60

* Locations with CUF low enough that further evaluation not required.

Table A-14
Summary of Older Vintage GE Plant CUFs (40 Year Life)

Component	Location	Material	Design CUF	NUREG/CR-5999 CUF			Revised SS Curve
				Based on Design Stresses	Conservative Assumptions Removed	Based on Expected Cycles	
Reactor vessel	Lower head to shell penetration	LAS	0.032	2.063	0.079	(1)	N/A
Feedwater nozzle	Bore	LAS	0.700	9.859	(1)	3.168	N/A
Recirculation system (3)	RHR return line tee	Stainless steel	0.397 / 0.526(2)	2.901	(1)	3.898	3.256
Core spray line	Nozzle	LAS	0.023	0.441	(1)	0.520	N/A
	Safe end	Stainless steel	0.182	1.778	(1)	2.305	1.772
RHR line (3)	Tapered transition	Stainless steel	0.032 / 0.045(2)	0.366	(1)	0.523	0.478
Feedwater line (3)	RCIC tee	Carbon Steel	0.427 / 0.584(2)	5.016	(1)	6.980	N/A

Note (1): No additional calculations were performed.
Note (2): CUFs based on representative design basis and anticipated number of cycles, respectively.
Note (3): Estimated by INEL using ASME Code NB-3600 techniques. CUF calculation not required by licensing basis.

4.0 CONCLUSIONS

This evaluation of the NUREG/CR-6260 results has provided an update of the expected usage factors at the end of 60 years that address the differences between the environmental correction factors in the NUREG/CR-5999 and those available from later fatigue test evaluation. Based on this evaluation, a specific set of locations have been identified for exclusion from further consideration:

Newer Vintage CE	Vessel Head/Shell Vessel Inlet Nozzle
Older Vintage CE:	Vessel Lower Head/Shell Vessel Inlet Nozzle SDC Inlet
B&W:	Vessel at Support Outlet Nozzle Surge Line Nozzle
Newer Vintage Westinghouse	Vessel Head Junction Vessel Inlet Nozzle
Older Vintage Westinghouse	Charging Nozzle Inlet RHR Tee
Newer Vintage GE	CRDM Weld
Older Vintage GE	Vessel Lower Head Penetration

Each applicant should review the plant specific evaluations of these locations to determine if differences in geometry and design/expected loadings would require those locations excluded herein to be considered. In addition, plant specific or other industry evaluations could possibly show that other locations can be excluded from further consideration.

5.0 REFERENCES

1. NUREG/CR-6260 (INEL-95/0045), "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995.

2. NUREG/CR-5999 (ANL-93/3), "Interim Fatigue Design Curves for Carbon, Low-Alloy, and Austenitic Stainless Steels in LWR Environments," April 1993.

3. NUREG/CR-6583 (ANL-97/18), "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," March 1998.

4. NUREG/CR-5704 (ANL-98/31), "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999

B
PVRC RECOMMENDATIONS FOR EVALUATING REACTOR WATER ENVIRONMENTAL FATIGUE EFFECTS

The steering committee on Cyclic Life and Environmental Effects (CLEE) of the Pressure Vessel Research Council (PVRC) has been studying the effects of reactor water environment on fatigue for several years. They have endorsed a revised form of the EPRI/GE methodology for fatigue evaluation methodology [1].

A Non-mandatory Code Appendix has been forwarded to the Board of Nuclear Codes and Standards (BNCS) [2]. The CLEE position and the rational for the latest methodology are discussed in a recent Pressure Vessel and Piping Conference paper [3]. The proposed Non-Mandatory Code Appendix as recommended by PVRC is reproduced in this Appendix. This approach is recommended for performing assessments of environmental effects on usage factors for reactor components.

REFERENCES

1. "An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Fatigue Evaluations," TR-105759, EPRI, Palo Alto, CA, December 1995.

2. Letter from Greg Hollinger (PVR) to J. H. Ferguson, Chairman Board of Nuclear Codes and Standards, October 31, 1999.

3. Mehta, H. S., "An Update on the Consideration of Reactor Water Effects in Code Fatigue Initiation Evaluations for Pressure Vessels and Piping," PVP-Vol. 410-2, p45-51, American Society of Mechanical Engineers, 2000.

NONMANDATORY APPENDIX XX

FATIGUE EVALUATIONS INCLUDING ENVIRONMENTAL EFFECTS

ARTICLE X-1000

SCOPE

This Appendix provides methods for performing fatigue usage factor evaluations of reactor coolant system and primary pressure boundary components when the effects of reactor water on fatigue initiation life are judged to be significant.

X-1100 ENVIRONMENTAL FATIGUE CORRECTION

The evaluation method uses as its input the partial fatigue usage factors U_1, U_2, U_3, ...U_n, determined in Class I fatigue evaluations. In Class I design by analysis procedure, the partial fatigue usage factors are calculated for each type of stress cycle in paragraph NB-3222.4(e)(5). For Class I piping products designed using NB-3600 procedure, Paragraph NB-3653 provides the procedure for the calculation of partial fatigue usage factors for each of the load set pairs.

The cumulative fatigue usage factor, U_{en}, considering the environmental effects is calculated as the following:

$$U_{en} = U_1 \bullet F_{en,1} + U_2 \bullet F_{en,2} + U_3 \bullet F_{en,3} \ldots U_i \bullet F_{en,i} \ldots + U_n \bullet F_{en,n}$$

where, $F_{en,i}$ is the effective environmental fatigue correction factor for the ith stress cycle (NB-3200) or load set pair (NB-3600).

X-1200 ENVIRONMENTAL FACTOR DEFINITION

X-1210 The nominal values of environmental fatigue correction factors are to be calculated using the expressions below.

Carbon Steel

$$Fen,nom = [\exp (0.559 - 0.101S^*T^*O^*\varepsilon'^*)] \quad (1)$$

Low Alloy Steel

$$Fen,nom = [\exp (0.903 - 0.101S^*T^*O^*\varepsilon'^*)] \quad (2)$$

Stainless Steels (wrought and cast)

$$F_{en,nom} = \exp [0.935 - T^*O^*\varepsilon'^*)] \qquad (3)$$

X-1260 The effective environmental fatigue correction factor, F_{en}, is obtained by dividing the nominal value calculated in X-1210 with a material-specific factor which accounts for moderate environmental fatigue effects already included in the S-N curves of Figures I-9.1 and I-9.2.

$$F_{en} = F_{en,nom}/Z, \text{ but no less than } 1.0$$

Where, Z = 3.0 for carbon and low alloy steels and 1.5 for wrought and cast stainless steels.

X-1300 EVALUATION PROCEDURES

For some types of stress cycles or load set pairs any one or more than one environmental parameters are below the threshold value for significant environmental fatigue effects. The value of the environmental fatigue correction factor, F_{en} for such types of stress cycles or load set pairs shall be equal to 1.0. Article X-2000 provides procedure for threshold criteria evaluation.

The procedures for the evaluation of F_{en} factors for design by analysis and for Class I piping products fatigue evaluations are provided in X-3000.

X-1400 NOMENCLATURE

The symbols adopted in this Appendix are defined as follows:

E = Young's Modulus, psi

F_{en} = Effective environmental correction factor applied to fatigue usage calculated using Code fatigue curves

$F_{en}(\tau)$ = Environmental correction factor calculated at a specific instant in time, τ.

$F_{en,int}$ = Environmental correction factor based on integrated approach.

DO = Dissolved oxygen content of water (ppm)

O* = Transformed oxygen content

S = Sulfur content of carbon and low-alloy steels, weight %

S* = Transformed sulfur content

S_{alt} = Alternating stress amplitude, psi

S_{range} = Range of stress intensity associated with a transient cycle, psi

T = Temperature (°C)

T^* = Transformed temperature.

T_a = Average temperature on side 'a' during a temperature transient

T_b = Average temperature on side 'b' during a temperature transient

T_c = Sum of $|T_a - T_b|$, $|\Delta T_1|$ and $|\Delta T_2|$ for temperature transient producing compressive stresses at the component surface in contact with fluid

T_m = Metal temperature during a temperature transient at surface in contact with fluid

T_t = Sum of $|T_a - T_b|$, $|\Delta T_1|$ and $|\Delta T_2|$ for temperature transient producing tensile stresses at the component surface in contact with water

ΔT_1 = Linear temperature gradient through a component wall during a temperature transient

ΔT_2 = Nonlinear temperature gradient through a component wall during a temperature transient

t_t = Elapsed time between the start of temperature transient and the time when T_t is reached, seconds

$t_{T,th}$ = Elapsed time between the start of decreasing temperature transient and the time when metal surface in contact with fluid reaches threshold temperature, seconds

U_{en} = Cumulative fatigue usage factor including the environmental effects

U_i = Cumulative fatigue usage factor for load set pair 'i' obtained by using Code fatigue curves

ε_i = Strain range for load set pair *i*, %

ε' = Strain rate, %/second

ε'^* = Transformed strain rate

$\varepsilon'^*(\tau)$ = Transformed strain rate at elapsed time equal to τ

ARTICLE X-2000

ENVIRONMENTAL FATIGUE THRESHOLD CONSIDERATIONS

X-2000 SCOPE

This Article provides procedure for screening out types of stress cycles or load set pairs for which any one or more than one environmental parameters are below the threshold value for significant environmental fatigue effects. The value of the environmental fatigue correction factor, F_{en} for such types of stress cycles or load set pairs shall be equal to 1.0.

X-2100 STRAIN AMPLITUDE THRESHOLDS

X-2110 The strain amplitude threshold for carbon and low alloy steels is 0.07%. F_{en} values shall be used at strain amplitudes equal to or exceeding 0.08%. A linear interpolation may be used to calculate F_{en} values for strain amplitudes between 0.07% and 0.08%.

X-2120 The strain amplitude threshold for wrought and cast stainless steels is 0.10%. F_{en} values shall be used at strain amplitudes equal to or exceeding 0.11%. A linear interpolation may be used to calculate F_{en} values for strain amplitudes between 0.10% and 0.11%.

X-2130 Calculate the strain amplitude, ε_i associated with a type of stress cycle or load set pair 'i' by multiplying the alternating stress intensity $S_{alt\,i}$ by 100 and dividing by the modulus of elasticity E. The value of E shall be obtained from the applicable design fatigue curves of Figs. I-9.0.

X-2140 If the value of ε_i calculated in X-2130 for a load set pair is less than or equal to appropriate value from X-2110 or X-2120, that load set pair satisfies the threshold criterion and the value of $F_{en\,i}$ is 1.0. No further evaluation with respect to other threshold values need be made for this load set pair.

X-2200 STRAIN RATE THRESHOLD

The strain rate threshold is 1.0%/second for carbon and low alloy steels, and 0.4%/second for wrought and cast stainless steels. A load set pair involving only the seismic loading satisfies the strain rate threshold criterion for strain rate and the value of $F_{en\,i}$ is 1.0. No further evaluation with respect to other threshold values need be made for this type of stress cycle or load set pair.

If the strain rate associated with the tensile stress load set for any other load set pair exceeds the threshold value, F_{en} is 1.0 for that load set pair.

X-2300 TEMPERATURE THRESHOLD

X-2310 The temperature threshold for carbon and low alloy steels is 150°C.

X-2320 The temperature threshold for wrought and cast stainless steels is 180°C.

X-2330 Define the effective temperature, T associated with a type of stress cycle or load set pair 'i' as equal to the higher of the highest temperatures in the two transients or load sets constituting the type of stress cycle or load set pair.

X-2340 If the temperature calculated in step (b) is less than or equal to the threshold value, the stress cycle or load set pair satisfies the threshold criterion for temperature and the value of $F_{en\,i}$ is 1.0.

X-2400 DISSOLVED OXYGEN THRESHOLD

This is applicable only to carbon and low alloy steels.

(a) Define the effective dissolved oxygen content, DO associated with a type of stress cycle or load set pair 'i' as equal to the higher of the highest oxygen content in the two transients or load sets constituting the type of stress cycle or load set pair.

(b) If the value of DO determined in step (a) for a type of stress cycle or load set pair is less than or equal to 0.05 ppm, that type of stress cycle or load set pair satisfies the threshold criterion and the value of $F_{en\,i}$ is 1.0.

ARTICLE X-3000

ENVIRONMENTAL FACTOR EVALUATION

X-3100 SCOPE

This Article provides procedure for calculating the F_{en} factors for types of stress cycles (NB-3200) or load set pairs (NB-3600). Only the types of stress cycles or load set pairs that do not meet the threshold criteria of X-2000 need to be considered for F_{en} calculation.

X-3200 EVALUATION PROCEDURE FOR DESIGN BY ANALYSIS

X-3210 Determination of Transformed Strain Rate

X-3211 The strain rate (%/sec) for a stress cycle is determined as the following:

$$\varepsilon' = S_{range\,i} \bullet 100/E \bullet t_{max}$$

where, $S_{range\,i}$ is the stress difference range for cycle'i' as determined in NB-3224.4(e)(5) and the t_{max} is the time in seconds when the stress difference reaches a maximum from the start of the temperature transient. This calculation is performed only for the step down temperature transient or other tensile stress producing cycle in the stress cycles constituting a pair.

X-3212 The transformed strain rate ε'^* for carbon and low alloy steels is obtained as the following:

$$\varepsilon'^* = 0 \qquad (\varepsilon' > 1\%/sec)$$

$$\varepsilon'^* = \ln(\varepsilon') \qquad (0.001 \le \varepsilon' \le 1\%/sec)$$

$\varepsilon'^* = \ln(0.001)$ ($\varepsilon' < 0.001\%/sec$)

X-3213 The transformed strain rate ε'^* for stainless steels is obtained as the following:

$\varepsilon'^* = 0$ ($\varepsilon' > 0.4\%/sec$)

$\varepsilon'^* = \ln(\varepsilon'/0.4)$ ($0.0004 \leq \varepsilon' \leq 0.4\%/sec$)

$\varepsilon'^* = \ln(0.0004/0.4)$ ($\varepsilon' < 0.0004\%/sec$)

X-3220 Determination of Transformed Temperature

X-3221 The temperature, T associated with a stress cycle 'i' is equal to the higher of the highest metal temperatures in the two transients constituting the stress cycle or load set pair.

X-3222 The transformed temperature T* for carbon and low alloy steels is obtained as the following:

$T^* = 0.0$ ($T < 150°C$)

$T^* = T-150$ ($T > 150°C$)

X-3223 The transformed temperatures T* for stainless steels are obtained as the following:

$T^* = 0.0$ ($T \leq 180°C$)

$T^* = (T-180)/40$ ($180°C < T < 220°C$)

$T^* = 1.0$ ($T > 220°C$)

X-3230 Determination of Transformed DO

X-3231 For carbon and low alloy steels, the effective dissolved oxygen content, DO associated with a load set pair 'i' is equal to the higher of the highest oxygen level in the two transients constituting the load set. The transformed DO, O* is obtained as follows:

$O^* = 0$ ($DO < 0.05$ ppm)

$O^* = \ln(DO/0.04)$ (0.05 ppm $\leq DO \leq 0.5$ ppm)

$O^* = \ln(12.5)$ ($DO > 0.5$ ppm)

X-3232 For wrought stainless steels, the effective dissolved oxygen content, DO associated with a load set pair 'i' is equal to the lower of the oxygen level in the two transients constituting the load set. The transformed DO, O* is obtained as follows:

$O^* = 0.260$ (DO < 0.05 ppm)

$O^* = 0.172$ (DO $\geq$ 0.05 ppm)

X-3233 For cast stainless steels, $O^* = 0.260$

X-3240 Determination of Transformed Sulfur for Carbon & Low Alloy Steels

The sulfur content S in terms of weight percent might be obtained from the certified material test report or an equivalent source. If the sulfur content is unknown, then its value shall be assumed as 0.015%. The transformed sulfur, S* is obtained as the following:

$S^* = S$ (0<S<0.015 wt%)

$S^* = 0.015$ (S>0.015 wt%)

X-3250 Determination of F_{en}

The environmental correction factor $F_{en\,i}$ for a type of stress cycle and the cumulative fatigue usage factor shall be calculated using equations given in X-1200.

X-3260 Determination of F_{en} Based on Damage Approach

Procedure similar to that described in X-3660 may be used to remove some of the conservatism built into the $F_{en\,i}$ determined in
X-3250.

X-3600 EVALUATION PROCEDURE FOR PIPING

The procedures in this Article use the input information and the partial fatigue usage results from the NB-3650 fatigue evaluation. The example of specific load set information needed is: internal pressure, the three moment components, |Ta-Tb|, ΔT1 and ΔT2. When the detailed results of one-dimensional transient heat transfer analyses are available in the form of time history of |Ta-Tb|, ΔT1 and ΔT2, such results may be used to reduce conservatisms in the calculated values of environmental correction factor.

X-3610 Determination of Strain Rate

The strain rate (%/sec) for a load set pair 'i' is determined as the following:

$$\varepsilon_i' = 200\bullet S_{alt\,i}\bullet[T_t/(T_t + T_c)]/(E\bullet t_t)$$

where, $S_{alt\,i}$ is the alternating stress intensity for load set pair 'i' calculated in NB-3653.3. This calculation is performed only for the step down temperature transient in a load set pair.

The transformed strain rate $\varepsilon_i'^*$ shall be obtained as described in X-3210.

X-3620 Determination of Transformed Temperatures

The transformed temperatures shall be obtained as described in X-3220.

X-3630 Determination of Transformed DO

The transformed DO shall be obtained as described in X-3230.

X-3640 Determination of Transformed Sulfur for Carbon and Low Alloy Steels

The transformed sulfur shall be obtained as described in X-3240.

X-3650 Determination of F_{en}

The environmental correction factor $F_{en\,i}$ shall be calculated using equations given in X-1200.

X-3660 Determination of F_{en} Based on Integrated Approach

When the results of detailed transient analyses are available to predict strain rate, such results may be used to reduce conservatisms in the calculated values of F_{en}. The following expression or equivalent shall be used:

$$F_{en,int} = (1/t_{T,th}) \int_0^{t_{T,th}} [F_{en}(\tau)]d\tau$$

The preceding value of F_{en} may be used in lieu of the F_{en} value calculated in X-3650. $F_{en}(\tau)$ is the appropriate environmental factor derived from X-1200, with time dependent properties/factors for the time in the transient where the temperature exceeds the threshold value.

C
MODERATE ENVIRONMENTAL EFFECTS

C-1. Introduction

One of the critical issues related to evaluation of reactor water environmental effects on component fatigue life is the credit to be taken for "moderate environmental effects." The original consideration of this sub-issue began with the development of fatigue design by analysis rules in the ASME Code in the 1960s [1]. Reference 1 states that:

"The design fatigue curves are based on strain-controlled fatigue tests of small polished specimens. A best-fit to the experimental data was obtained by applying the method of least squares to the logarithms of the experimental values. The design stress values were obtained from the best-fit curves by applying a factor of two on stress or a factor of twenty on cycles, whichever was more conservative at each point. 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."

The term "environment" in this statement has been interpreted by many, including the Cyclic Life and Environmental Effects (CLEE) Steering Committee of the Pressure Vessel Research Council (PVRC), to mean "moderate environmental effects."

At issue is the portion of the factor of 20 (at the low cycle end of the fatigue design curve) inherent in the ASME Code Section III explicit fatigue design curves that can be attributed to moderate effects of environment. Reference 2 provided the PVRC technical position on moderate environmental effects for carbon and low-alloy steels. Their analysis of the collected data in air showed a factor of about 4 to account for temperature effects and for data scatter, leaving a factor of 4 on the ASME mean air data as a reasonable "working" definition of moderate environmental effects. The PVRC CLEE also observed that the appropriate portion for austenitic stainless steels is about a factor of 2, out of the ASME Code factor of 20 at the low-cycle end of the fatigue design curve.

A number of technical arguments support these moderate environmental effects factors. First, Chopra and Shack [3] observed that "Because carbon and low-alloy steels and austenitic SSs develop a corrosion scale in LWR environments, the effect of surface finish may not be significant, i.e., the effects of surface roughness are included in environmentally assisted decrease in fatigue life in LWR coolant environments. In water, the subfactor on life to account for surface finish effects may be as low as 1.5 or may be eliminated completely; a factor of 1.5 on strain and 7 on cycles is adequate to account for the uncertainties that arise from material and loading variability. Therefore, the factor of 20 on life that is used in developing the design fatigue curves includes, as a safety margin, a factor of 3 or 4 on life that may be used to account

for the effects of environment on the fatigue life of these steels." This same factor of between 3 and 4 was observed as the actual safety margins in the PVRC fatigue tests on large-scale vessels reported in Reference 1.

Another argument in support of a moderate environmental effects factor of 3 to 4 is provided by the characteristics of equations used to fit the laboratory-simulated environmental fatigue data. These equations do not revert to the equations used to fit laboratory air data; instead, even for testing conditions such that simulated reactor water environmental effects are minimal, the equations contain an "environmental shift" much greater than 1. For example, the equation that fits reactor water environmental fatigue data for austenitic stainless steels predicts an asymptotic environmental shift of 2.55, even for temperatures below the environmental threshold.

The above reasoning supports a moderate environmental effects factor of 4 for carbon and low-alloy steels, and a factor of 2.5 for austenitic stainless steels. It should be pointed out, however, that, in their most recent publications, the PVRC (see Appendix B) has reduced the recommended moderate environmental effects factor, or Z factor, from 4 to 3 for carbon and low-alloy steel, and from 2 to 1.5 for austenitic stainless steel.

These moderate environmental factors were used in a number of industry generic studies [3, 4, 5, and 6] that were submitted to the NRC staff for review relative to potential closure of Generic Safety Issue 190. When the PVRC recommended values were used as an adjustment to environmental fatigue calculations, the industry studies found that cumulative usage factors (CUFs) could be shown to remain below 1.0 for 60 years of operation. Discussions between the industry and the NRC staff during the review of the generic studies have indicated a disagreement on this critical issue.

At first, the NRC staff agreed with the modified PVRC recommendation [7] for a moderate environmental factor of 3 (instead of 4) for carbon and low-alloy steels, and a factor of 1.5 (instead of 2) for austenitic stainless steels. More recently, however, the NRC staff has stipulated that no moderate environmental effects factor greater than 1.0 can be credited at all, because of the presumably greater data scatter for laboratory-simulated reactor water environmental effects, relative to the scatter in the air data [8]. While the scatter in the fatigue test data in air showed a scatter factor of about ± 2, the evaluations of scatter in the fatigue test data in simulated reactor water environments have claimed a scatter factor of about ± 5. The NRC staff relied heavily in this judgment on the arguments presented in NUREG/CR-6583 [9]. This reference argued that the size effect portion of the factor of 20 is about 1.4, the surface finish factor is between 2.0 and 3.0, and potential errors in the application of Miner's Rule (loading history) introduces a factor of 1.5 to 2.5. With a data scatter factor of 2.5, the total adjustment ranges between 10.0 and 26.0. Any increase in data scatter beyond a factor of about 2.0 would cause the adjustment to be well above the available factor of 20 on cyclic life at the low-cycle end of the fatigue curves.

Recent evaluations of data from Japanese fatigue testing programs have supported this argument. For example, Tsutsumi et al. [10] have analyzed an extensive set of data on austenitic stainless steels, and have also argued that the data exhibit increased variability. This increased variability, if true, would not permit any allocation of the ASME Code factor of 20 to be assigned to

moderate environmental effects, since surface finish, specimen size effects, and other considerations account for the residual factor of 4. A similar argument on data variability has been made by Higuchi [11] for carbon and low-alloy steels. However, the data evaluations seem to be based on a logical inconsistency – the evaluation of data variability also includes varying environmental effects; i.e., the data scatter assessment includes environmental effects variability that should be separated statistically from data scatter assessment within a particular water environment data set.

In the following sections, the data for both carbon and low-alloy steels, and for austenitic stainless steels, are reanalyzed with a clear separation between data sets at different environmental conditions, with an intent to separate data scatter within an essentially homogeneous environmental data set from the effects of environmental variability. Section C.2 discusses data scatter evaluations for austenitic stainless steels and Section C.3 discusses data scatter evaluation for carbon and low-alloy steel. Finally, Section C.4 draws conclusions about the portion of the factor of 20 at the low-cycle end of the fatigue design curves that can be attributed to moderate environmental effects.

C.2 Austenitic Stainless Steel Data Evaluation

The data from Table 4 of Reference 10 have been reanalyzed with a clear separation between environmentally homogeneous data sets; i.e., those test data obtained at different environmental test conditions. The data set separation begins by dividing the population of data points into a set for which the testing strain rates were relatively high (0.4 %/sec), and a set for which the test strain rate were relatively low (e.g., 0.01 %/sec, 0.04 %/sec, 0.001 %/sec, etc.). Essentially, the first data set contains data points for which the effects of simulated reactor water environments appear to be "moderate," and the second data set contains data points for which the effects of simulated reactor water environments appear not to be moderate. The lack of moderation may vary, but both the first and second data sets are analyzed as different environmental effects populations. The assessment of the first population gives an estimate of the data scatter within the moderate environmental effects data set, while the assessment of the second population provides an estimate of the difference between moderate and immoderate environmental effects. For convenience and simplicity of assessment, the first (moderate) and second (immoderate) data sets are further separated into subsets, based on testing strain amplitude, test temperature, and material type.

For example, all of the data for the austenitic stainless steels, including 304 and 316 stainless steel, cast material, weld metal, etc., with a strain amplitude in the neighborhood of 0.3 % and tested at a temperature of 360°C (680°F), 325°C (617°F), or 300°C (572°F) were evaluated in two populations. These included data with strain amplitudes of 0.28 %, 0.285 %, 0.29 %, 0.295 %, 0.3 %, 0.305 %, and 0.31 %. Data for all levels of dissolved oxygen (DO) were included. The data sets included weld metal and sensitized material. The two data sets consisted of 43 values for cyclic life. The moderate environmental effects population consisted of 21 points, for which the test strain rates were relatively high (0.4 %/sec). These data are listed in Table C-1. The immoderate environmental effects population consisted of 22 points, for which the test strain rates were relatively low. These data are not shown here, but can be found in Table 4 of Reference 10.

For the first subset (relatively high strain rate), the average fatigue life was 9,286 cycles, with an estimated standard deviation of 2,614 cycles. The highest value measured was 15,142 cycles (a weld value) and the lowest value measured was 4,125 cycles (also a weld value). For the second subset (relatively low strain rate), the average of the 22 data points was 4,717 cycles, with an estimated standard deviation of 2,490 cycles. The highest value measured was 10,684 cycles (a weld value) and the lowest value measured was 1,854 cycles.

There is a considerable difference in the average fatigue strength in these two populations – about a factor of two. This difference represents moderate versus immoderate environmental effects. However, by separating the two populations, using the ratio of estimated standard deviation to mean value as a measure, the ratio is 28 % for the relatively high strain rate data. Had the weld data been excluded, the ratio drops to less than 15 %. The ratio in the relatively low strain data populations is much greater, 53 %, about a factor of 2. The major concern is the variability in the data for moderate environmental effects, which can be seen to be much less than the factor of ± 2 that is attributed to the air fatigue data. Of course, if the two populations (all 43 data points) are analyzed together, as has been reported, the ratio is 49 %, very nearly the same ratio as for the low strain rate data population. The combination of the mean value ratio between the two populations and the increase in the ratio of standard deviation to mean value between the two populations gives the factor of ± 5 that has been observed in the literature.

A similar exercise was carried out for a strain amplitude near 0.6 %. The relatively high strain rate population consisted of data points at or near a strain amplitude of 0.6 % (0.58 %, 0.585 %, 0.59 %, 0.595 %, 0.6 % 0.605 %, and 0.61 %). Weld data, cast material data, all dissolved oxygen data, sensitized material, and aged material data were included in the population, which consisted of 22 data points. These data points are listed in Table C-2. The behavior of 304 stainless steel differed somewhat from that for 316 stainless steel, but the materials were included in the same population. Weld data showed considerable variability, but was also included in the population. The total set of 22 data points at relatively high strain rates gave an average fatigue life of 1,656 cycles with an estimated standard deviation of 493 cycles. The highest measured life was 2,381 cycles (a weld data point), while the lowest measured life was 666 cycles (also a weld data point). Removing the weld data from the data set reduces the standard deviation by a factor of 2.

The relatively low strain rate population consisted of an additional 39 data points, with much greater variability. The highest value was 1,365 cycles and the lowest value was 80 cycles, with a mean value of 538 cycles. Note that the difference between the moderate environmental effects mean value and the immoderate environmental effects mean value drops by a factor of 3. When the two populations were analyzed together, the mean value was found to drop to 941 cycles, with an estimated standard deviation of 662 cycles.

Again, using the ratio of standard deviation to mean value as the measure, the ratio was found to be 0.3 for the moderate environmental effects population, well below the factor of about 2 attributed to the scatter in air fatigue data. to the ratio for the combined moderate and immoderate environmental effects populations increased to 0.7, more than a factor of 2. Again, the combination of the mean value ratio and the increase in the ratio of standard deviation to mean value encompasses the factor of ± 5 in data scatter that has been observed in the literature.

Data at strain amplitudes at or near 0.2 % and 0.4 % were also examined. However, these data sets are too sparse to provide results with high confidence.

C.3 Carbon and Low-Alloy Steel Data Evaluation

Laboratory data for carbon and low-alloy steel under simulated reactor water environmental conditions have been collected by PVRC as a part of their studies on cyclic life and environmental effects. These data were provided to EPRI for the data scatter assessment. Again, the data populations were separated into two sets – those obtained at relatively high strain rate (e.g., 0.4 %/sec), referred to as the moderate environmental effects population, and those obtained at relatively low strain rates, referred to as the immoderate environmental effects population. Again, this type of separation of data populations permits variability due to data scatter within a population to be isolated from variability caused by stronger reactor water environmental influence.

As with the austenitic stainless steel data, the two populations were further subdivided into subsets with relatively homogeneous testing parameters. For example, data points at reactor operating temperatures (288°C, 300°C, etc.) and at approximately the same strain amplitude (e.g., 0.6 % strain) were grouped together. Much more data was available for low-alloy steels than for carbon steels so that, in some cases, the findings are limited to those applicable to low-alloy steels. The influence of dissolved oxygen (DO) was not found to distort the statistical evaluation, with the exception of the very highest DO levels (8 ppm). The term high DO is used to describe the 8 ppm data, while data at all other DO levels is described as low DO data. Weld data largely fit into the general populations, with the single exception of data from a single Japanese investigator.

As an example, 31 data points were found in the PVRC data base for low-alloy (e.g., SA-533B) and carbon (e.g., SA-106B) steels obtained at a relatively high strain rate (0.4 %/sec), at operating temperature (e.g., 288°C), and 0.6 % strain amplitude. Weld data and both high DO and low DO data are included. Of the 31 data points, 13 were included in the high DO subset and 18 were included in the low DO subset. Six of the 31 data points were for carbon steel, and were included in the high DO subset. These data are listed in Table C-3. The low DO mean value was found to be 2,378 cycles with an estimated standard deviation of 1,055 cycles. The high DO subset had a mean value of 1,693 cycles with a standard deviation of 419 cycles. The ratio of the standard deviation to the mean value is 0.44 for the low DO population and only 0.25 for the high DO population.

The PVRC data base contained 10 data points at a strain amplitude of 0.5 %, all from various Japanese investigators on low-alloy steels. The Japanese data included two high DO data points that fit into the general population. These data points are listed in Table C-4. The mean of the Japanese data was 2,908 cycles with a standard deviation of 818 cycles. The ratio is only 0.28.

The PVRC data base contained 15 data points at 0.4 % strain amplitude and moderately high strain rates. All of the data points were for low-alloy steel and all came from Japanese investigators. Weld data fit into the general population, with the exception of three outliers from the same investigator. These data points were included in the data evaluation. High DO data fit

into the general population with no exceptions. These data points are listed in Table C-5. The data mean for the 15 data points was 5,352 cycles, with a standard deviation of 2,229 cycles. The ratio is 0.42, reflecting the inclusion of two potential outliers and the combined low DO and high DO populations. Excluding the three weld metal outliers from the one investigator gives a revised mean of 6,082 cycles with a standard deviation of 1,617 cycles, thereby reducing the ratio to 0.27.

The PVRC data base contained 15 data points at 0.3 % strain amplitude and relatively high strain rates. All of the data points came from Japanese investigators. Three data points were for carbon steel, with 12 data points for low-alloy steel. High DO data, including those for carbon steel piping material, fit into the general population, as did all of the weld data. One data point for low-alloy steel at high DO appeared to be an outlier, but was included in the analysis set. These data points are listed in Table C-6. The data mean was found to be 16,195 cycles with a standard deviation of 7,629 cycles. The ratio for this more inclusive population is 0.47. Note that the data scatter in all of these calculations is less than or equal to the data scatter attributed to air environments.

C.4 Summary

From the examination of these data, it is concluded that the observations of large data scatter for simulated reactor water environmental testing are not warranted. Only by mixing data sets that represent moderate and immoderate environmental effects can large data scatter effects be observed. This relatively low data variability was observed by separating laboratory test data into two populations – a population containing test data obtained at relatively high strain rate (e.g., 0.4 %/sec) and a population containing test data obtained at relatively slow strain rates (e.g., 0.004 %/sec). The implication of this statistical separation is that the relatively high strain rate population exhibits moderate environmental effects, while the relatively slow strain rate population, in general, exhibits a reduction in fatigue life that is greater than moderate.

The statistical analysis of the separated populations shows that:

- Data variability for the relatively high strain rate population is much less than has been reported in the literature when the statistical analyses are based on combined populations. The ratio of the standard deviations to the mean values for both austenitic stainless steels and carbon/low-alloy steels tested at relatively high strain rates is in the range of 0.2 to 0.5, even when data populations are enlarged to include weld data.

- Very high dissolved oxygen levels do not compromise the relatively high strain rate data variance for austenitic stainless steels and for low-alloy steels. This is true for carbon steels, in general

- The addition of weld metal fatigue data increased the ratio of standard deviation to mean value in the relatively high strain rate population by about a factor of 2, from a ratio of about 0.25 to about 0.50. In many cases, the most extreme measured values

within a strain amplitude population, on both the low side and on the high side, were weld metal data points.

- The ratio of standard deviation to mean value in the relatively slow strain rate population was much greater, by a factor of 2 or more. Combining the ratio of mean values between moderate and immoderate effects populations with this increase in the ratio of standard deviation to mean value gives the estimated data scatter in the literature of ± 5.

The findings from this analysis support the recommendations of PVRC that moderate environmental effects factors of 3 for carbon and low-alloy steel, and 1.5 for austenitic stainless steels, are conservative. Greater moderate environmental effects factors can be justified.

Table C-1

Austenitic Stainless Steel (0.3 % Strain Amplitude)

Material	DO (ppb)	Temperature (°C)	Strain Amplitude (%)	Tensile Strain Rate (%/sec)	Cycles to Failure
316	5	325	0.3	0.4	8799
316	5	300	0.285	0.4	6391
316	8000	325	0.29	0.4	8761
316 (Pre-strained)	5	325	0.3	0.4	8760
316 (Forging)	5	325	0.3	0.4	10754
316 (Sensitized)	5	325	0.3	0.4	7428
316 (Weld Metal)	5	325	0.31	0.4	4125*
316 (Weld Metal)	5	325	0.3	0.4	6184
316 (Weld Metal)	5	325	0.3	0.4	15142
304	5	325	0.29	0.4	8798
304	5	300	0.305	0.4	7020
304	5	360	0.28	0.4	10326
304	8000	325	0.29	0.4	10242
304	5	325	0.3	0.4	7928
304 (Sensitized)	5	325	0.3	0.4	9879
308 (Weld Metal)	5	325	0.29	0.4	7954
SCS14A	5	325	0.295	0.4	9242
SCS14A (Aged)	5	325	0.3	0.4	11795
CF8M (Aged)	5	325	0.3	0.4	13327
SCS14A	5	325	0.3	0.4	10154
SCS14A	5	325	0.3	0.4	12000

* Potential Outlier

Table C-2

Austenitic Stainless Steel (0.6 % Strain Amplitude)

Material	DO (ppb)	Temperature (°C)	Strain Amplitude (%)	Tensile Strain Rate (%/sec)	Cycles to Failure
316	5	325	0.59	0.4	2070
316	5	300	0.605	0.4	1916
316	8000	325	0.6	0.4	2027
316 (Pre-strained)	5	325	0.59	0.4	2238
316 (Forging)	5	325	0.6	0.4	1572
316 (Sensitized)	5	325	0.61	0.4	2009
316	5	325	0.58	0.4	2089
316 (Weld Metal)	5	325	0.61	0.4	666
316	5	325	0.6	0.4	2460
316 (Weld Metal)	5	325	0.61	0.4	1075
316 (Weld Metal)	5	325	0.6	0.4	1922
304	5	325	0.59	0.4	1344
304	5	300	0.585	0.4	1189
304	5	360	0.58	0.4	1172
304	8000	325	0.59	0.4	988
304	5	325	0.6	0.4	1411
304 (Sensitized)	5	325	0.6	0.4	1318
308 (Weld Metal)	5	325	0.6	0.4	2381
SCS14A (Aged)	5	325	0.6	0.4	1380
CF8M (Aged)	5	325	0.6	0.4	2136
SCS14A	5	325	0.59	0.4	1606
SCS14A	5	325	0.6	0.4	1461

Table C-3

Carbon Steel/Low-Alloy Steel (0.6 % Strain Amplitude)

Material	DO	Temperature (°C)	Strain Amplitude (%)	Cycles to Failure	Investigator
533B LAS	High	290	0.6	1600	Higuchi
533B LAS	High	290	0.6	1690	Higuchi
533B LAS	High	290	0.6	1640	Higuchi
508-3 LAS	Low	250	0.58	3040	Kasai
508-3 LAS	Low	290	0.605	2284	Kasai
508-3 LAS	Low	250	0.58	4210	Kasai
508-3 LAS	Low	290	0.59	2810	Kasai
508-3 LAS	High	290	0.585	2120	Kasai
508-3 LAS	High	290	0.575	2372	Kasai
533B LAS	Low	288	0.6	1728	Nakao
533B LAS	Low	288	0.6	1692	Nakao
533B LAS	Low	288	0.6	1276	Nakao
508-3 LAS	High	290	0.593	783	Endou
508-3 LAS	Low	250	0.584	1695	Endou
508-3 LAS	Low	290	0.587	1899	Endou
508-3 LAS	High	288	0.6	1660	Higuchi
508-3 LAS	High	288	0.6	1920	Higuchi
508-3 LAS	High	288	0.6	1250	Higuchi
508-3 LAS	Low	288	0.6	3540	Higuchi
508-3 LAS	Low	288	0.6	3625	Higuchi
508-3 LAS	Low	288	0.6	3435	Higuchi
533B LAS Weld	High	290	0.6	1810	Higuchi
533B LAS Weld	High	290	0.6	1774	Higuchi
533B LAS Weld	Low	288	0.6	960	Nakao
533B LAS Weld	Low	288	0.6	1091	Nakao

Table C-4

Carbon Steel/Low-Alloy Steel (0.5 % Strain Amplitude)

Material	DO	Temperature (°C)	Strain Amplitude (%)	Cycles to Failure	Investigator
533B LAS	High	290	0.5	3348	Higuchi
533B LAS	High	290	0.5	3550	Higuchi
533B LAS	Low	288	0.5	1965	Nakao
508-3 LAS	Low	288	0.498	4022	Nagata
508-2 LAS	Low	288	0.5	2875	Nakao
533B LAS Weld	Low	288	0.5	1888	Nakao
533B LAS Weld	Low	288	0.5	1898	Nakao
333B-3 CS	Low	288	0.5	3426	Higuchi

Table C-5

Carbon Steel/Low-Alloy Steel (0.4 % Strain Amplitude)

Material	DO	Temperature (°C)	Strain Amplitude (%)	Cycles to Failure	Investigator
533B LAS	High	290	0.4	9400	Higuchi
533B LAS	High	290	0.4	6340	Higuchi
508-3 LAS	Low	250	0.395	8573	Kasai
533B LAS	Low	288	0.408	6353	Nagata
533B LAS	Low	288	0.4	8528	Nakao
533B LAS	Low	288	0.4	5700	Nakao
533B LAS	Low	288	0.4	6900	Nakao
533B LAS	Low	288	0.4	4030	Nakao
333B-3 CS	Low	288	0.4	15550*	Higuchi
508-3 LAS	High	290	0.404	1911*	Endou
508-3 LAS	High	288	0.4	5702	Higuchi
533B LAS Weld	High	290	0.4	5610	Higuchi
533B LAS Weld	High	290	0.4	5855	Higuchi
533B LAS Weld	Low	288	0.4	2670	Nakao
533B LAS Weld	Low	288	0.4	2708	Nakao
508-1 LAS	High	300	0.4	1600*	Kitigawa

* Potential Outliers

Table C-6

Carbon Steel/Low-Alloy Steel (0.3 % Strain Amplitude)

Material	DO	Temperature (°C)	Strain Amplitude (%)	Cycles to Failure	Investigator
533B LAS	High	290	0.3	32080	Higuchi
533B LAS	High	290	0.3	28700	Higuchi
533B LAS	Low	288	0.3	14760	Nakao
508-3 LAS	High	288	0.3	8080	Higuchi
533B LAS Weld	High	290	0.3	18500	Higuchi
533B LAS Weld	High	290	0.3	14800	Higuchi
508-3 LAS	Low	288	0.285	26020	Nakao
533B LAS	Low	288	0.28	26730	Nakao
508-3 LAS	Low	288	0.298	29000	Nagata
533B LAS Weld	Low	288	0.3	13840	Nakao
533B LAS Weld	Low	288	0.3	18730	Nakao
333B-2 CS	High	290	0.3	8460	Higuchi
333B-2 CS	Low	288	0.3	10860	Higuchi
333B-2 CS	Low	288	0.3	23840	Higuchi
508-1 LAS	High	300	0.3	2200*	Kitigawa

* Potential Outlier

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D
DEMONSTRATING TRANSIENT SEVERITY BOUNDS ENVIRONMENTAL EFFECTS

This appendix shows an example of the methodology for demonstrating that transient severity bounds any additional effects due to reactor water environmental effects. This example is that from a Request for Additional information (RAI) related to the Plant Hatch License Renewal Application.[1]

RAI 4.2-2:

Section 4.2.2 of the LRA contains a discussion of Generic Safety Issue (GSI) 190, "Fatigue Evaluation of Metal Components For 60-year Plant Life." GSI-190 addresses the effect of the reactor water environment on the fatigue life of metal components. The discussion in Section 4.2.2 indicates that EPRI license renewal fatigue studies have demonstrated that sufficient conservatism exists in the design transient definitions to compensate for potential reactor water environmental effects. The staff does not agree with the contention that the EPRI fatigue studies have demonstrated that sufficient conservatism exists in the design transient definitions to compensate for potential reactor water environmental effects. The staff identified several technical concerns regarding the EPRI studies. The staff technical concerns are contained in an August 6, 1999, letter to NEI. Although these concerns involved the EPRI procedure and its application to PWRs, the technical concerns regarding the application of the Argonne National Laboratory (ANL) statistical correlations and strain threshold values are also relevant to BWRs. In addition to the concerns referenced above, the staff has additional concerns regarding the applicability of the EPRI BWR studies to Plant Hatch. EPRI Report TR-107943, "Environmental Fatigue Evaluations of Representative BWR Components," addressed a BWR-6 plant and EPRI Report TR-110356, "Evaluation of Environmental Thermal Fatigue Effects on Selected Components in a Boiling Water Reactor Plant," used plant transient data from a newer vintage BWR-4 plant. The applicability of the EPRI fatigue studies to Plant Hatch has not been demonstrated. Provide the following additional information regarding resolution of the environmental fatigue issue:

a. Indicate whether the staff comments provided in the staff's August 6, 1999, letter to NEI, which are applicable to Hatch, have been considered in the assessment of the environmental fatigue issue at Plant Hatch. Discuss how the applicable staff comments were considered in the evaluation of environmental fatigue.

[1] Letter from Lewis Sumner, Jr. (Southern Co.) to U.S. Nuclear Regulatory Commission, "Edwin I Hatch Nuclear Plant, Response to License Renewal Requests for Additional Information", Docket Nos. 50-321 and 50-366, October 10, 2000.

b. Discuss the applicability of the component fatigue assessments in the EPRI Reports TR-107943 and TR-110356 to components in Hatch Units 1 & 2. The discussion should include a comparison of design transients, operating cycles and fabrication details for each component. Also include a comparison of the hydrogen water chemistry used at Hatch with the hydrogen water chemistry considered in the EPRI reports.

c. The staff assessed the impact of reactor water environment on fatigue life at high fatigue usage locations and presented the results in NUREG/CR-6260, "Application of NUREG/CR-5999, 'Interim Fatigue Curves to Selected Nuclear Power Plant Components'," March 1995. Formulas currently acceptable to the staff for calculating the environmental correction factors for carbon and low alloy steels are contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and those for austenitic stainless steels are contained in NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design of Austenitic Stainless Steels." Provide an assessment of the 6 locations identified in NUREG/CR-6260 for an older vintage BWR-4 considering the applicable environmental fatigue correlations provided in NUREG/CR-6583 and NUREG/CR-5704 reports for Hatch Units 1 and 2.

RESPONSE TO RAI 4.2-2:

a. The staff comments provided in the August 6, 1999, NRC letter to NEI have been considered in the assessment of the environmental fatigue issue at Plant Hatch through the margins present by considering design basis severity of thermal transients.

Primarily, the NRC concerns presented in the August 6, 1999 letter are associated with the more recent laboratory fatigue data in simulated LWR reactor water environments that have been generated by ANL since the time of the EPRI generic studies. These data have resulted in improved environmental correction factor correlations, which are documented in NUREG/CR-6583 (for carbon/low alloy steel) and NUREG/CR-5704 (for stainless steel). The improved correlations were not available at the time the EPRI generic studies were performed.

For carbon and low-alloy steels, the correlations published in NUREG/CR-6583 do not differ substantially from the correlations used in the EPRI generic studies.

However, the change in strain threshold may have a significant effect, and that effect has been evaluated, as follows.

A recalculation was performed based on one of the examples contained in EPRI Report No. TR-105759, "An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and PipingFatigue Evaluations," December 1995, for a BWR carbon steel feedwater piping location with a design-basis fatigue usage factor of 0.1409 for 40 years. An alternating stress threshold of 30 ksi (approximating the alternating strain threshold of 0.10%) was used initially to adjust the incremental fatigue usage for eight out of thirty-one load pairs, giving an additional

(environmental) fatigue usage of 0.0477, for a 40-year adjusted total of 0.1886. The overall environmental multiplier (Fen) in this case was 1.34 (1.68 for the eight affected load pairs).

Reducing the alternating stress threshold to 21 ksi (approximating the revised alternating strain threshold of 0.07%) would require an environmental adjustment for six additional load pairs. Assuming that the Fen multiplier of 1.68 would continue to apply for the fourteen affected load pairs, the estimate for the adjusted fatigue usage factor would be

$$0.1409 - 0.0803 + 1.68\,(0.0803) = 0.1955.$$

The overall Fen multiplier increases only to 1.39.

Because the additional load pairs that would have to be included contribute relatively small increments to the total CUF, the change in the strain range threshold does not cause a significant impact on the calculated fatigue usage. Therefore, the results of the EPRI generic studies provide a reasonable estimate of the impact of potential environmental fatigue effects for carbon/low alloy steel components, and are considered to remain valid.

For austenitic stainless steels, the data are more penalizing than the data used in the EPRI generic studies.

For the case of relatively low temperature (< 200°C), a low (bounding) strain rate, and either high or low dissolved oxygen, the environmental shift is 2.55. For relatively high temperature (> 200°C), low dissolved oxygen, and a low (bounding) strain rate, the environmental shift may be as high as 15.35, although there is a reduction above 250°C where the environmental factor decreases to about 3.20 at 340°C. These factors are higher than those obtained from the relationships used in the EPRI generic studies. As a result, further evaluation was performed as described below.

For most of the component locations evaluated in the EPRI generic studies, these most recent data do not pose a problem for the demonstration that the 60-year CUF is less than 1.0, including reactor water environmental effects. Again, a significant benefit accrues to the Fen approach in this regard, since most of the penalizing thermal transients in the BWR environment lie below the threshold temperature of 200°C. Therefore, the environmental shift is relatively low, provided that separate multipliers are used for the portions of the transient that are above and below 200°C. However, for the most fatigue-sensitive PWR locations, (e.g., surge line elbows), the environmentally-adjusted CUF increases over that calculated in the EPRI generic studies by a factor of about two.

Therefore, a reasonable approach to accounting for the more recent laboratory data for stainless steel material is to conservatively apply a factor of 2.0 to the EPRI generic study results. This is considered to be very conservative for the BWR.

The CUF results from the most applicable EPRI generic study (EPRI TR-110356, see Item (b) below) are shown in Table 1, with modifications to account for the more recent

data in NUREG/CR-6583 and NUREG/CR-5704. The design basis fatigue usage for each location is also shown for comparison. The results in Table 1 clearly demonstrate that the conservatism of design basis transient definitions overwhelms all environmental effects. The CUF for all locations, including environmental effects and projected to 60 years, is at least a factor of 12.9 below the original design basis CUF.

These results indicate that tracking CUF based on design basis transient definitions, such as the Plant Hatch CCTLP does, provides conservative estimates of CUF for the license renewal period.

b. The most applicable evaluation for Plant Hatch with respect to the EPRI generic studies is EPRI Report No. TR-110356, "Evaluation of Environmental Thermal Fatigue Effects on Selected Components in a Boiling Water Reactor." The other two EPRI reports (EPRI Report Nos. TR-107515 and TR-107943) have limited direct applicability to Plant Hatch, but were referenced in the Plant Hatch application for completeness, since the EPRI studies built off the results of each other. It was therefore considered necessary to reference the main EPRI study (EPRI Report No. TR-107515), along with both follow-on studies performed for BWRs, to provide a comprehensive reference source.

Nevertheless, focusing on EPRI Report No. TR-110356, those results are considered directly applicable to Plant Hatch. First, the results documented in that report apply to a BWR-4 that is identical to the Plant Hatch design. Therefore, the Class 1 systems associated with the plants are the same, which defines the characteristics of the thermal transients in these systems. As a result, the design basis transient definitions associated with the plants are very similar. This is demonstrated in Table 2, where the design basis transient definitions for both plants are compared.

The BWR-4 evaluated in EPRI Report No. TR-110356 did not consider hydrogen water chemistry (HWC), as evidenced by the plots of dissolved oxygen in that report. Both units at Plant Hatch have implemented HWC. The maximum effect of the change in dissolved oxygen as a result of HWC implementation is adequately addressed via the conservative factors described under the response to Item (a) above.

There are only two issues relevant to fabrication details and the associated effects of reactor water environment on fatigue. First, the sulfur content, where applicable, was conservatively assumed to be at a maximum level in EPRI TR-110356. Second, the material type (i.e., stainless or carbon/low alloy steel) is similar between the two plants, and was considered appropriately in all fatigue evaluations. In fact, material types between most BWRs are very similar, as evidenced by the comparison shown in Table 3 between Plant Hatch and the older vintage BWR-4 evaluated in NUREG/CR-6260. Therefore, fabrication details are not considered to have any effect on the application of the results in EPRI Report No. TR-110356 to Plant Hatch.

c. The locations investigated in NUREG/CR-6260 for the older vintage BWR are listed in Table 3. Also shown in Table 3 are the equivalent locations where CUF is monitored via

the Plant Hatch CCTLP, and the projected 60-year CUF for each location based on plant operation to-date.

Table 3 demonstrates that all BWR locations from NUREG/CR-6260 were evaluated for Plant Hatch. All of these locations are either bounded by locations monitored via the Plant Hatch CCTLP or the design 40-year CUF is below the 0.10 threshold for monitoring by the program. The projected CUFs for all monitored locations remain within the allowable value of 1.0 for the license renewal period. Furthermore, the Plant Hatch CCTLP includes several other locations (nine total, five on Unit 1 and four on Unit 2), beyond those evaluated in NUREG/CR-6260, thereby providing a more comprehensive CUF assessment.

Note that the Plant Hatch RHR suction piping that Table 3 credits for monitoring two of the NUREG/CR-6260 locations is being removed from the CCTLP because a newer stress analysis shows the 40-year CUF for that location below the 0.10 threshold for monitoring.

As discussed in the response to Item (a) above, the appropriate correlations from NUREG/CR-6583 and NUREG/CR-5704 have been accounted for via the conservatism in design basis transient definitions.

Table 1
Revised Fatigue Usage Results for BWR (Including Environmental Effects)

Case No.	Location	Projected 60 Year Usage Factor from TR-110356 (with F_{en})	Correction Factor to Account for NUREG/CR-6583 or NUREG/CR-5704	Revised 60 Year Usage Factor (with F_{en}) [1]	Design Basis Fatigue Usage [2]	Margin [3]
1	1 = CRD Penetration	0.034	2.0	0.068	0.875	12.9
	2 = FW Loop A Safe End	0.009	2.0	0.018	0.471	26.2
	3 = FW Loop A Nozzle Forging	0.001	1.0	0.001	< 0.1	~100
	4 = FW Loop B Safe End	0.009	2.0	0.018	0.471	26.2
	5 = FW Loop B Nozzle Forging	0.001	1.0	0.001	< 0.1	~100
2	1 = CRD Penetration	0.013	2.0	0.026	0.875	33.7
	2 = FW Loop A Safe End	0.009	2.0	0.018	0.471	26.2
	3 = FW Loop A Nozzle Forging	0.001	1.0	0.001	< 0.1	~100
	4 = FW Loop B Safe End	0.009	2.0	0.018	0.471	26.2
	5 = FW Loop B Nozzle Forging	0.001	1.0	0.001	< 0.1	~100
3	1 = CRD Penetration	0.016	2.0	0.032	0.875	27.3
	2 = FW Loop A Safe End	0.009	2.0	0.018	0.471	26.2
	3 = FW Loop A Nozzle Forging	0.001	1.0	0.001	< 0.1	~100
	4 = FW Loop B Safe End	0.009	2.0	0.018	0.471	26.2
	5 = FW Loop B Nozzle Forging	0.001	1.0	0.001	< 0.1	~100

Notes:
1. The "Revised 60-Year Usage Factor" is equal to the "Projected 60-Year Usage Factor from TR-110356" multiplied by the "Correction Factor to Account for NUREG/CR-6583 or NUREG/CR-5704."
2. As documented in the governing design basis fatigue analysis report.
3. The "Margin" is equal to the "Design Basis Fatigue Usage" divided by the "Revised 60-Year Usage Factor."

Table 2
Design Basis Plant Transient Comparison for the BWR-4 in EPRI Report No. TR-110356 vs. Plant Hatch

Transient	BWR-4 No. of Cycles	Hatch Unit 1 No. of Cycles	Hatch Unit 2 No. of Cycles
Boltup	123	123	123
Design Hydrostatic Test	130	130	130
Startup	117	120	117
Turbine Roll & Increase to Rated Power	not specified	120	not specified
Daily Reduction to 75% Power	10,000	10,000	10,000
Weekly Reduction to 50% Power	2,000	2,000	2,000
Rod Pattern Change (Rod Worth Test)	400	400	400
Loss of Feedwater Heaters, Turbine Trip with 100% Steam Bypass, Unit 1 = Turbine Trip at 25% Power	10	10	10
Loss of Feedwater Heaters, Partial Feedwater Heater Bypass	70	70	70
SCRAM, Turbine Generator Trip, Feedwater On, Isolation Valves Stay Open	40	40	40
SCRAM, All Other	140	147	140
Rated Power Normal Operation	not specified	not specified	not specified
Reduction to 0% Power	111	118	111
Hot Standby	111	118	111
Shutdown/Vessel Flooding	111	118	111
Unbolt	123	123	123
Refueling	not specified	not specified	not specified
Pre-Operational Blowdown	10	0	10
Loss of Feedwater Pumps, Isolation Valves Close	5	10	5
Reactor Over Pressure with Delayed SCRAM, Feedwater Stays On, Isolation Valves Stay Open	1	1	1
Single Relief or Safety Valve Blowdown	8	2	8
Automatic Blowdown	1	0	1
Improper Start of Cold Recirculation Loop	1	5	1
Sudden Start of Pump in Cold Recirculation Loop	1	5	1
Improper Startup with Recirculation Pumps Off & Drain Shut Off	1	0	1
Pipe Rupture and Blowdown	1	0	not specified
Natural Circulation Startup	3	0	3
Loss of AC Power, Natural Circulation Restart	5	0	5
Code Hydrostatic Test	0	3	3

Table 3
Locations Evaluated in NUREG/CR-6260 for
Older Vintage General Electric Plant (BWR-4) vs. Plant Hatch

NUREG/CR-6260 Location	NUREG/CR-6260 Material	Addressed by Plant Hatch CCTLP?	Plant Hatch Material	Projected 60-Year CUF for Plant Hatch (1)
Reactor Vessel (Lower Head to Shell Transition)	SA-302 Low Alloy Steel	YES (2)	SA-533 Grade B Class 1 Low Alloy Steel	U1 = 0.0669 U2 = 0.0513
Feedwater Nozzle (Bore)	SA-508 Low Alloy Steel	YES	SA-508 Class 2 Low Alloy Steel	U1 = 0.1663 U2 = 0.3643
Recirculation System (RHR Return Line Tee)	SA-358 Type 304 Stainless Steel	YES (3)	SA-358 Type 316NG Class 1 Stainless Steel	U1 < 0.1500 (7) U2 < 0.1500 (7)
Core Spray System (Nozzle)	SA-302 Grade B Low Alloy Steel	YES (4)	SA-508 Class 2 Low Alloy Steel	U1 = 0.4796 U2 = 0.2983
Core Spray System (Safe End)	SA-376 Type 316 Stainless Steel	YES (5)	SA-182 Type F304 Stainless Steel	U1 = 0.1605 U2 < 0.1500 (7)
Residual Heat Removal Line (Tapered Transition)	SA-358 Type 304 Stainless Steel	YES (3)	SA-358 Type 316NG Class 1	U1 < 0.1500 (7) U2 < 0.1500 (7)
Feedwater Line (RCIC Tee)	SA-106 Grade B Carbon Steel	YES (6)	SA-106 Grade B Carbon Steel	U1 = 0.5607 U2 = 0.7435 (8)

Notes:

1. Based on actual transient counts through 12/31/1999.
2. The limiting location in the RPV shell is monitored for both units at Plant Hatch, which is considered to adequately represent the NUREG/CR-6260 location.
3. The limiting location in the Unit 2 RHR suction piping, at the elbow near the recirculation suction tee, was monitored in the Plant Hatch CCTLP, and was considered to adequately represent the NUREG/CR-6260 location. Newer stress analysis shows the 40-year CUF < 0.10, so this location will no longer be monitored. The 40-year design CUF < 0.10 for the Unit 1 RHR suction piping and was never monitored by the CCTLP. Therefore, the CCTLP addresses this location for both units by determining the CUF is below the threshold for monitoring.
4. The RPV recirculation inlet nozzle, which bounds the core spray nozzle at Plant Hatch, is monitored for both units in the Plant Hatch CCTLP. This is considered to adequately represent the NUREG/CR-6260 location.
5. The limiting location in the Unit 1 core spray piping system is monitored in the Plant Hatch CCTLP, and is considered to adequately represent the NUREG/CR- 6260 location. The 40-year design CUF < 0.10 for the Unit 2 core spray piping system.
6. The limiting location in the feedwater piping system is monitored for both units in the Plant Hatch CCTLP, which is considered to adequately represent the NUREG/CR-6260 location. On Unit 1, the monitored piping includes the HPCI, RCIC, and RWCU Class 1 piping connected to the feedwater line.
7. The 40-year design CUF is less than 0.10 for this location so it is not monitored.
8. The RCIC Tee on Unit 2 is in the Class 2 portion of the system. The CUF given is for the bounding location on the Class 1 portion of the feedwater line.