UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and 50-286-LR
ENTERGY NUCLEAR OPERATIONS, INC.	)	ASLBP No. 07-858-03-LR-BD01
(Indian Point Nuclear Generating Units 2 and 3)	)	

SUPPORTING ATTACHMENTS TO APPLICANT'S ANSWER TO NEW AND AMENDED CONTENTION NEW YORK STATE 26B/RIVERKEEPER TC-1B (METAL FATIGUE)

Filed on October 4, 2010

LIST OF ATTACHMENTS

Attachment	Description
1	Excerpt from NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, Rev. 1 (Sept. 2005)
2	Excerpt from Vol. 2 of NUREG-1801, *Generic Aging Lessons Learned (GALL) Report* – Tabulation of Results, Rev. 1 (Sept. 2005)
3	Excerpt from NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (Feb. 1995)
4	Excerpt from NUREG/CR-6909, *Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials* (Feb. 2007)
5	NL-10-082, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program" (Aug. 9, 2010)
6	Excerpts from EPRI, MRP-47, *Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* (Rev. 1, Sept. 2005)
7	NL-08-021, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application Amendment 2," (Jan. 22, 2008)
8	SAND94-0187, *Evaluation of Conservatisms and Environmental Effects in ASME Code, Section III, Class 1 Fatigue Analysis*, at iii (Aug. 1994) **(available for purchase from the National Technical Information Service (NTIS) at http://www.ntis.gov/)**
9	Excerpt from NUREG-1916, Vol. 2, *Safety Evaluation Report Related to the License Renewal of Shearon Harris Nuclear Power Plant, Unit 1* (Nov. 2008)
10	Letter from Thomas J. Natale, Harris Nuclear Plant, to NRC Document Control Desk, "Shearon Harris Nuclear Power Plant, Unit No. 1, Docket No. 50-400/License No. NPF-63, [LRA] Amendment 2: Changes Resulting from Responses to Site audit Questions Regarding Time-Limited Aging Analyses," encl. 3, at 67-93 (Aug. 31, 2007) (Harris Nuclear Plant License Renewal Audit Question and Response Database)
11	Excerpt from Memorandum from Peter Wen, Sr., ACRS Staff Engineer, to ACRS Members, "Certification of the Minutes of the ACRS Plant License

LIST OF ATTACHMENTS

Attachment	Description
	Renewal Subcommittee Meeting Regarding Shearon Harris Nuclear Power Plant on May 7, 2008 – Rockville Maryland" (July 1, 2008).
12	Westinghouse FENOC-08-109, Letter from K. Blanchard to C. Custer, FENOC, "FirstEnergy Nuclear Operating Company, Beaver Valley Unit 1 and 2, Responses to NRC RAIs Regarding Pressurizer Surge Line Environmental Fatigue" (Rev. 1, June 25, 2008)
13	American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III , Rules for Construction of Nuclear Facility Components, Division 1, Subarticle NB-3200 (Design by Analysis) (1998 Edition) **(available for purchase from the ASME at http://www.asme.org/)**
14	Excerpt from NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* (Apr. 1999)
15	Excerpt from NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels* (Mar. 1998)
16	R&D Status Report from January/February 1983 issue of the EPRI Journal
17	NL-08-084, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex," (May 16, 2008)

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 1

NUREG-1800, Rev. 1

Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants

Manuscript Completed: September 2005
Date Published: September 2005

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

3.0 INTRODUCTION TO STAFF REVIEW OF AGING MANAGEMENT

The NRC project manager (PM) responsible for the safety review of the license renewal application (LRA) is responsible for assigning to appropriate NRC Office of Nuclear Reactor Regulation (NRR) divisions the review or audit of aging management reviews (AMRs) or aging management programs (AMPs) identified in the applicant's LRA. The PM should document to which organization each AMR or AMP is assigned. The assigned AMRs and AMPs should be reviewed per the criteria described in Sections 3.1 through 3.6 of this standard review plan (SRP-LR, NUREG-1800) for review of license renewal applications, as directed by the scope of each of these sections.

The NRC divisions that are usually assigned responsibility for the review of AMRs and AMPs are the Division of Engineering (DE), Division of System Safety Analysis (DSSA), and the Division of Regulatory Improvement Program (DRIP) License Renewal and Environmental Impacts Program (RLEP). Typically, the PM will assign DRIP/RLEP to review the AMRs and AMPs that the LRA identifies as being consistent with the GALL Report or NRC-approved precedents. As common exceptions to this assignment, the PM will assign to DE those AMRs and AMPs that address issues identified as emerging technical issues. Usually, AMRs and AMPs that are not in one of the aforementioned categories are assigned to DE.

Review of the AMPs requires assessment of ten program elements as defined in this SRP-LR. The NRC divisions assigned the AMP should review the ten program elements to verify their technical adequacy. For three of the ten program elements (corrective actions, confirmation process, and administrative controls) the NRC division responsible for quality assurance should verify that the applicant has documented a commitment in the FSAR Supplement to expand the scope of its 10 CFR Part 50, Appendix B program to address the associated program elements for each AMP. If the applicant chooses alternate means of addressing these three program elements (e.g., use of a process other than the applicant's 10 CFR Part 50, Appendix B program), the NRC divisions assigned to review the AMP should request that the Division responsible for quality assurance review the applicant's proposal on a case-by-case basis.

3.0.1 Background on the Types of Reviews

10 CFR 54.21(a)(3) requires that the LRA must demonstrate, for systems, structures, and components (SSCs) identified in the scope of license renewal and subject to an AMR pursuant to 10 CRF 54.21(a)(1), that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. This AMR consists of identifying the material, environment, aging effects, and the AMP(s) credited for managing the aging effects.

Sections 3.1 through 3.6 of this SRP-LR describe how the AMRs and AMPs are reviewed. One method that the applicant may use to conduct its AMRs is to satisfy the NUREG-1801 (GALL Report) recommendations. The applicant may choose to use methodology other than that in the GALL Report to demonstrate compliance with 10 CFR 54.21(a)(3).

As stated in the GALL Report:

> The GALL Report is a technical basis document to the SRP-LR, which provides the staff with guidance in reviewing a license renewal application. The GALL Report should be treated in the same manner as an approved topical report that is generically applicable. An applicant may reference the GALL Report in a license renewal application to

demonstrate that the programs at the applicant's facility correspond to those reviewed and approved in the GALL Report and that no further staff review is required, as described in the next paragraph. If the material presented in the GALL Report is applicable to the applicant's facility, the staff should find the applicant's reference to the GALL Report acceptable. In making this determination, the staff should consider whether the applicant has identified specific programs described and evaluated in the GALL Report. The staff, however, should not conduct a re-review of the substance of the matters described in the GALL Report. Rather, the staff should ensure that the applicant verifies that the approvals set forth in the GALL Report for generic programs apply to the applicant's programs. The focus of the staff review should be on augmented programs for license renewal. The staff should also review information that is not addressed in the GALL Report or is otherwise different from that in the GALL Report.

If an applicant takes credit for a program in the GALL Report, it is incumbent on the applicant to ensure that the plant program contains all the elements of the referenced GALL Report program. In addition, the conditions at the plant must be bounded by the conditions for which the GALL Report program was evaluated. The above verifications must be documented on-site in an auditable form. The applicant should include a certification in the license renewal application that the verifications have been completed and are documented on-site in an auditable form.

The GALL Report contains one acceptable way to manage aging effects for license renewal. An applicant may propose alternatives for staff review in its plant-specific license renewal application. Use of the GALL Report is not required, but its use should facilitate both preparation of a license renewal application by an applicant and timely, uniform review by the NRC staff.

In addition, the GALL Report does not address scoping of structures and components for license renewal. Scoping is plant-specific, and the results depend on the plant design and current licensing basis. The inclusion of a certain structure or component in the GALL Report does not mean that this particular structure or component is within the scope of license renewal for all plants. Conversely, the omission of a certain structure or component in the GALL Report does not mean that this particular structure or component is not within the scope of license renewal for any plants.

The GALL Report contains an evaluation of a large number of structures and components that may be in the scope of a typical LRA. The evaluation results documented in the GALL Report indicate that many existing, typical generic aging management programs are adequate to manage aging effects for particular structures or components for license renewal without change. The GALL Report also contains recommendations on specific areas for which generic existing programs should be augmented (require further evaluation) for license renewal and documents the technical basis for each such determination. In addition, the GALL Report identifies certain SSCs that may or may not be subject to particular aging effects, and for which industry groups are developing generic aging management programs or investigating whether aging management is warranted. To the extent the ultimate generic resolution of such an issue will need NRC review and approval for plant-specific implementation, as indicated in a plant-specific FSAR supplement, and reflected in the SER associated with a particular LR application, an amendment pursuant to 10 CFR 50.90 will be necessary.

In this SRP-LR, Subsection 3.X.2 (where X denotes number 1-6) presents the acceptance criteria describing methods to determine whether the applicant has met the requirements of NRC's regulations in 10 CFR 54.21. Subsection 3.X.3 presents the review procedures to be followed. Some rows (line-items) in the AMR tables (in Chapters II through VIII of the GALL Report, Vol. II) establish the need to perform "further evaluations." The acceptance criteria for satisfying these "further evaluations" are found in Subsections 3.X.2.2. The related review procedures are provided in Subsections 3.X.3.2.

In Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," the NRC has endorsed an acceptable methodology for applicants to structure license renewal applications. Using the guidance described in the aforementioned Regulatory Guide, the applicant documents in the LRA whether its AMR line-item is consistent or not consistent with the GALL Report.

A portion of the AMR includes the assessment of the AMPs in the GALL Report. The applicant may choose to use an AMP that is consistent with the GALL Report AMP, or may choose a plant-specific AMP.

If a GALL Report AMP is selected to manage aging, the applicant may take one or more exceptions to specific GALL Report AMP program elements. However, any deviation or exception to the GALL Report AMP should be described and justified. Exceptions are portions of the GALL Report AMP that the applicant does not intend to implement.

In some cases, an applicant may choose an existing plant program that does not currently meet all the program elements defined in the GALL Report AMP. If this is the situation, the applicant may make a commitment to augment the existing program to satisfy the GALL Report AMP element prior to the period of extended operation. This commitment is an AMP enhancement.

Enhancements are revisions or additions to existing aging management programs that the applicant commits to implement prior to the period of extended operation. Enhancements include, but are not limited to, those activities needed to ensure consistency with the GALL Report recommendations. Enhancements may expand, but not reduce, the scope of an AMP.

An audit and review is conducted at the applicant's facility to evaluate those AMRs or AMPs that the applicant claims to be consistent with the GALL Report. An audit also includes technical assessments of exceptions or enhancements to the GALL Report AMP program elements. Reviews are performed to address those AMRs or AMPs related to emergent issues, stated to be not consistent with the GALL Report, or based on an NRC-approved precedent (e.g., AMRs and AMPs addressed in an NRC SER of a previous LRA). As a result of the criteria established in 10 CFR Part 54, and the guidance provided in SRP-LR, GALL Report, Regulatory Guide 1.188, and the applicant's exceptions and/or enhancements to a GALL Report AMP, the following types of AMRs and AMPs should be audited or reviewed by the NRC staff.

AMRs

- AMR results consistent with the GALL Report
- AMR results for which further evaluation is recommended by the GALL Report
- AMR results not consistent with or not addressed in the GALL Report

AMPs

- Consistent with GALL Report AMPs

- Plant-specific AMPs

FSAR Supplement

- Each LRA AMP will provide an FSAR Supplement which defines changes to the FSAR that will be made as a condition of a renewed license. This FSAR Supplement defines the aging management programs the applicant is crediting to satisfy 10 CFR 54.21(a)(3).
- The FSAR Supplement should also contain a commitment to implement the LRA AMP enhancement prior to the period of extended operation.

3.0.2 Applications with approved Extended Power Uprates

Extended power uprates (EPU) are licensing actions that some licensees have recently requested the NRC staff to approve.. This can affect aging management. In a NRC staff letter to the Advisory Committee on Reactor Safeguards, dated October 26, 2004, (ADAMS Accession ML042790085), the NRC Executive Director for Operation states that, "All license renewal applications with an approved EPU will be required to perform an operating experience review and its impact on [aging] management programs for structures, and components before entering the period of extended operation." One way for an applicant with an approved EPU to satisfy this criterion is to document its commitment to perform an operating experience review and its impact on aging management programs for systems, structures, and components (SSCs) before entering the period of extended operation as part of its license renewal application. Such licensee commitments should be documented in the NRC staff's SER written in support of issuing a renewed license. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date. EPU impact on SSCs should be part of the license renewal review. If necessary, the PM will assign a responsible group to address EPU.

4.3 METAL FATIGUE ANALYSIS

Review Responsibilities

Primary - Branch responsible for the TLAA issues
Secondary - None

4.3.1 Areas of Review

A metal component subjected to cyclic loading at loads less than the static design load may fail because of fatigue. Metal fatigue of components may have been evaluated based on an assumed number of transients or cycles for the current operating term. The validity of such metal fatigue analysis is reviewed for the period of extended operation.

The metal fatigue analysis review includes, as appropriate, a review of in service flaw growth analyses, reactor vessel underclad cracking analysis, reactor vessel internals fatigue analysis, postulated high energy line break, leak-before-break, RCP flywheel, and metal bellows.

4.3.1.1 Time-Limited Aging Analysis

Metal components may be designed or analyzed based on requirements in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code or the American National Standards Institute (ANSI) guidance. These codes contain explicit metal fatigue or cyclic considerations based on TLAAs.

4.3.1.1.1 ASME Section III, Class 1

ASME Class 1 components, which include core support structures, are analyzed for metal fatigue. ASME Section III (Ref. 1) requires a fatigue analysis for Class 1 components that considers all transient loads based on the anticipated number of transients. A Section III Class 1 fatigue analysis requires the calculation of the "cumulative usage factor" (CUF) based on the fatigue properties of the materials and the expected fatigue service of the component. The ASME Code limits the CUF to a value of less than or equal to one for acceptable fatigue design. The fatigue resistance of these components during the period of extended operation is an area of review.

4.3.1.1.2 ANSI B31.1

ANSI B31.1 (Ref. 2) applies only to piping. It does not call for an explicit fatigue analysis. It specifies allowable stress levels based on the number of anticipated thermal cycles. The specific allowable stress reductions due to thermal cycles are listed in Table 4.3-1. For example, the allowable stress would be reduced by a factor of 1.0, i.e., no reduction, for piping that is not expected to experience more than 7,000 thermal cycles during plant service, but would be reduced to half of the maximum allowable static stress for 100,000 or more thermal cycles. The fatigue resistance of these components during the period of extended operation is an area of review.

4.3.1.1.3 Other Evaluations Based on CUF

The codes also contain metal fatigue analysis criteria based on a CUF calculation [the 1969 edition of ANSI B31.7 (Ref. 3) for Class 1 piping, ASME NC-3200 vessels, ASME NE-3200

Class MC components, and metal bellows designed to ASME NC-3649.4(e)(3), ND-3649.4(e)(3), or NE-3366.2(e)(3)]. For these components, the discussion relating to ASME Section III, Class 1 in Subsection 4.3.1.1.1 of this review plan section applies.

4.3.1.1.4 ASME Section III, Class 2 and 3

ASME Section III, Class 2 and 3 piping cyclic design requirements are similar to the guidance in ANSI B31.1. The discussion relating to B31.1 in Subsection 4.3.1.1.2 of this review plan section applies.

4.3.1.2 Generic Safety Issue

The fatigue design criteria for nuclear power plant components have changed as the industry consensus codes and standards have developed. The fatigue design criteria for a specific component depend on the version of the design code that applied to that component, i.e., the code of record. There is a concern that the effects of the reactor coolant environment on the fatigue life of components were not adequately addressed by the code of record.

The NRC has decided that the adequacy of the code of record relating to metal fatigue is a potential safety issue to be addressed by the current regulatory process for operating reactors (Refs. 4 and 5). The effects of fatigue for the initial 40-year reactor license period were studied and resolved under Generic Safety Issue (GSI)-78, "Monitoring of Fatigue Transient Limits for reactor coolant system," and GSI-166, "Adequacy of Fatigue Life of Metal Components" (Ref. 6). GSI-78 addressed whether fatigue monitoring was necessary at operating plants. As part of the resolution of GSI-166, an assessment was made of the significance of the more recent fatigue test data on the fatigue life of a sample of components in plants where Code fatigue design analysis had been performed. The efforts on fatigue life estimation and ongoing issues under GSI-78 and GSI-166 for 40-year plant life were addressed separately under a staff generic task action plan (Refs. 7 and 8). The staff documented its completion of the fatigue action plan in SECY-95-245 (Ref. 9).

SECY-95-245 was based on a study described in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components" (Ref. 10). In NUREG/CR-6260, sample locations with high fatigue usage were evaluated. Conservatisms in the original fatigue calculations, such as actual cycles versus assumed cycles, were removed, and the fatigue usage was recalculated using a fatigue curve considering the effects of the environment. The staff found that most of the locations would have a CUF of less than the ASME Code limit of 1.0 for 40 years. On the basis of the component assessments, supplemented by a 40-year risk study, the staff concluded that a backfit of the environmental fatigue data to operating plants could not be justified. However, because the staff was less certain that sufficient excessive conservatisms in the original fatigue calculations could be removed to account for an additional 20 years of operation for renewal, the staff recommended in SECY-95-245 that the samples in NUREG/CR-6260 should be evaluated considering environmental effects for license renewal. GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life," was established to address the residual concerns of GSI-78 and GSI-166 regarding the environmental effects on fatigue of pressure boundary components for 60 years of plant operation.

The scope of GSI-190 included design basis fatigue transients. It studied the probability of fatigue failure and its effect on core damage frequency (CDF) of selected metal components for 60-year plant life. The results showed that some components have cumulative probabilities of crack initiation and through-wall growth that approach one within the 40- to 60-year period. The maximum failure rate (through-wall cracks per year) was in the range of 10^{-2} per year, and those failures were generally associated with high cumulative usage factor locations and components with thinner walls, i.e., pipes more vulnerable to through-wall cracks. In most cases, the leakage from these through-wall cracks is small and not likely to lead to core damage. It was concluded that no generic regulatory action is necessary and that GSI-190 is resolved based on results of probabilistic analyses and sensitivity studies, interactions with the industry (NEI and EPRI), and different approaches available to licensees to manage the effects of aging (Refs. 11 and 12).

However, the calculations supporting resolution of this issue, which included consideration of environmental effects, indicate the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, the staff concluded that licensees are to address the effects of coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

The applicant's consideration of the effects of coolant environment on component fatigue life for license renewal is an area of review.

4.3.1.3 FSAR Supplement

Detailed information on the evaluation of TLAAs is contained in the renewal application. A summary description of the evaluation of TLAAs for the period of extended operation is contained in the applicant's FSAR supplement. The FSAR supplement is an area of review.

4.3.2 Acceptance Criteria

The acceptance criteria for the areas of review described in Subsection 4.3.1 of this review plan section delineate acceptable methods for meeting the requirements of the NRC's regulations in 10 CFR 54.21(c)(1).

4.3.2.1 Time-Limited Aging Analysis

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following:

(i) the analyses remain valid for the period of extended operation,

(ii) the analyses have been projected to the end of the extended period of operation, or

(iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Specific acceptance criteria for metal fatigue are:

4.3.2.1.1 ASME Section III, Class 1

For components designed or analyzed to ASME Class 1 requirements, the acceptance criteria, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.2.1.1.1 10 CFR 54.21(c)(1)(i)

The existing CUF calculations remain valid because the number of assumed transients would not be exceeded during the period of extended operation.

4.3.2.1.1.2 10 CFR 54.21(c)(1)(ii)

The CUF calculations have been reevaluated based on an increased number of assumed transients to bound the period of extended operation. The resulting CUF remains less than or equal to unity for the period of extended operation.

4.3.2.1.1.3 10 CFR 54.21(c)(1)(iii)

In Chapter X of the GALL report (Ref. 13), the staff has evaluated a program for monitoring and tracking the number of critical thermal and pressure transients for the selected reactor coolant system components. The staff has determined that this program is an acceptable aging management program to address metal fatigue of the reactor coolant system components according to 10 CFR 54.21(c)(1)(iii). The GALL report may be referenced in a license renewal application and should be treated in the same manner as an approved topical report. In referencing the GALL report, the applicant should indicate that the material referenced is applicable to the specific plant involved and should provide the information necessary to adopt the finding of program acceptability as described and evaluated in the report. The applicant should also verify that the approvals set forth in the GALL report for the generic program apply to the applicant's program.

4.3.2.1.2 ANSI B31.1

For piping designed or analyzed to B31.1, the acceptance criteria, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.2.1.2.1 10 CFR 54.21(c)(1)(i)

The existing fatigue strength reduction factors remain valid because the number of cycles would not be exceeded during the period of extended operation.

4.3.2.1.2.2 10 CFR 54.21(c)(1)(ii)

The fatigue strength reduction factors have been reevaluated based on an increased number of assumed thermal cycles and the stress reduction factors (e.g., Table 4.3-1) given in the applicant's code of record to bound the period of extended operation. The adjusted fatigue strength reduction factors are such that the component design basis remains valid during the period of extended operation.

4.3.2.1.2.3 10 CFR 54.21(c)(1)(iii)

The effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The component could be replaced and the allowable stresses for the replacement will be sufficient as specified by the code during the period of extended operation.

Alternative acceptance criteria under 10 CFR 54.21(c)(1)(iii) have yet to be developed. They will be evaluated on a case-by-case basis to ensure that the aging effects will be managed such that the intended functions(s) will be maintained during the period of extended operation.

4.3.2.1.3 Other Evaluations Based on CUF

The acceptance criteria in Subsection 4.3.2.1.1 of this review plan section apply.

4.3.2.1.4 ASME Section III, Class 2 and 3

The acceptance criteria in Subsection 4.3.2.1.2 of this review plan section apply.

4.3.2.2 Generic Safety Issue

The staff recommendation for the closure of GSI-190 is contained in a December 26, 1999 memorandum from Ashok Thadani to William Travers (Ref. 11). The staff recommended that licensees address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. One method acceptable to the staff for satisfying this recommendation is to assess the impact of the reactor coolant environment on a sample of critical components. These critical components should include, as a minimum, those selected in NUREG/CR-6260 (Ref. 10). The sample of critical components can be evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses. Formulas for calculating the environmental life correction factors for carbon and low-alloy steels are contained in NUREG/CR-6583 (Ref. 14) and those for austenitic SSs are contained in NUREG/CR-5704 (Ref. 15).

4.3.2.3 FSAR Supplement

The specific criterion for meeting 10 CFR 54.21(d) is:

> The summary description of the evaluation of TLAAs for the period of extended operation in the FSAR supplement is appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the TLAAs regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21(c)(1).

4.3.3 Review Procedures

For each area of review described in Subsection 4.3.1, the following review procedures should be followed:

4.3.3.1 Time-Limited Aging Analysis

4.3.3.1.1 ASME Section III, Class 1

For components designed or analyzed to ASME Class 1 requirements, the review procedures, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.3.1.1.1 10 CFR 54.21(c)(1)(i)

The operating transient experience and a list of the assumed transients used in the existing CUF calculations for the current operating term are reviewed to ensure that the number of assumed transients would not be exceeded during the period of extended operation.

4.3.3.1.1.2 10 CFR 54.21(c)(1)(ii)

The operating transient experience and a list of the increased number of assumed transients projected to the end of the period of extended operation are reviewed to ensure that the transient projection is adequate. The revised CUF calculations based on the projected number of assumed transients are reviewed to ensure that the CUF remains less than or equal to one at the end of the period of extended operation.

The code of record should be used for the reevaluation, or the applicant may update to a later code edition pursuant to 10 CFR 50.55a. In the latter case, the reviewer verifies that the requirements in 10 CFR 50.55a are met.

4.3.3.1.1.3 10 CFR 54.21(c)(1)(iii)

The applicant may reference the GALL report in its license renewal application, as appropriate. The review should verify that the applicant has stated that the report is applicable to its plant with respect to its program that monitors and tracks the number of critical thermal and pressure transients for the selected reactor coolant system components. The reviewer verifies that the applicant has identified the appropriate program as described and evaluated in the GALL report. The reviewer also ensures that the applicant has stated that its program contains the same program elements that the staff evaluated and relied upon in approving the corresponding generic program in the GALL report. No further staff evaluation is necessary.

4.3.3.1.2 ANSI B31.1

For piping designed or analyzed to ANSI B31.1 guidance, the review procedures, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.3.1.2.1 10 CFR 54.21(c)(1)(i)

The operating cyclic experience and a list of the assumed thermal cycles used in the existing allowable stress determination are reviewed to ensure that the number of assumed thermal cycles would not be exceeded during the period of extended operation.

4.3.3.1.2.2 10 CFR 54.21(c)(1)(ii)

The operating cyclic experience and a list of the increased number of assumed thermal cycles projected to the end of the period of extended operation are reviewed to ensure that the thermal cycle projection is adequate. The revised allowable stresses based on the projected number of assumed thermal cycles and the stress reduction factors given in the applicant's code of record are reviewed to ensure that they remain sufficient as specified by the code during the period of extended operation. Typical stress reduction factors based on thermal cycles are given in Table 4.3-1.

The code of record should be used for the reevaluation, or the applicant may use the criteria of 10 CFR 50.55a. In the latter case, the reviewer verifies that the requirements in 10 CFR 50.55a are met.

4.3.3.1.2.3 10 CFR 54.21(c)(1)(iii)

The applicant's proposed program to ensure that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation is reviewed. If the applicant proposed a component replacement before it exceeds the assumed thermal cycles, the reviewer verifies that the allowable stresses for the replacement will remain sufficient as specified by the code during the period of extended operation. Other applicant-proposed programs will be reviewed on a case-by-case basis.

4.3.3.1.3 Other Evaluations Based on CUF

The review procedures in Subsection 4.3.3.1.1 of this review plan section apply.

4.3.3.1.4 ASME Section III, Class 2 and 3

The review procedures in Subsection 4.3.3.1.2 of this review plan section apply.

4.3.3.2 Generic Safety Issue

The reviewer verifies that the applicant has addressed the staff recommendation for the closure of GSI-190 contained in a December 26, 1999 memorandum from Ashok Thadani to William Travers (Ref. 11). The reviewer verifies that the applicant has addressed the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. If an applicant has chosen to assess the impact of the reactor coolant environment on a sample of critical components, the reviewer verifies the following:

1. The critical components include, as a minimum, those selected in NUREG/CR-6260 (Ref. 10).

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

3. Formulas for calculating the environmental life correction factors are those contained in NUREG/CR–6583 (Ref. 14) for carbon and low-alloy steels, and in NUREG/CR-5704 (Ref. 15) for austenitic SSs, or an approved technical equivalent.

4.3.3.3 FSAR Supplement

The reviewer verifies that the applicant has provided information, to be included in the FSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA. Table 4.3-2 contains examples of acceptable FSAR supplement information for this TLAA. The reviewer verifies that the applicant has provided a FSAR supplement with information equivalent to that in Table 4.3-2.

The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license condition, until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59. If the applicant updates the FSAR to include the final FSAR supplement before the license is renewed, no condition will be necessary.

As noted in Table 4.3-2, an applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities, including enhancements and commitments to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

4.3.4 Evaluation Findings

The reviewer determines whether the applicant has provided sufficient information to satisfy the provisions of this section and whether the staff's evaluation supports conclusions of the following type, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), to be included in the staff's safety evaluation report:

> On the basis of its review, as discussed above, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the metal fatigue TLAA, [choose which is appropriate] (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The staff also concludes that the FSAR Supplement contains an appropriate summary description of the metal fatigue TLAA evaluation for the period of extended operation as reflected in the license condition.

4.3.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

4.3.6 References

1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers.

2. ANSI/ASME B31.1, "Power Piping," American National Standards Institute.

3. ANSI/ASME B31.7-1969, "Nuclear Power Piping," American National Standards Institute.

4. SECY-93-049, "Implementation of 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses for Nuclear Power Plants,'" March 1, 1993.

5. Staff Requirements Memorandum from Samuel J. Chilk, dated June 28, 1993.

6. NUREG-0933, "A Prioritization of Generic Safety Issues," Supplement 20, July 1996.

7. Letter from William T. Russell of NRC to William Rasin of the Nuclear Management and Resources Council, dated July 30, 1993.

8. SECY-94-191, "Fatigue Design of Metal Components," July 26, 1994.

9. SECY-95-245, "Completion of The Fatigue Action Plan," September 25, 1995.

10. NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995.

11. Letter from Ashok C. Thadani of the Office of Nuclear Regulatory Research to William D. Travers, Executive Director of Operations, dated December 26, 1999.

12. NUREG/CR-6674, "Fatigue Analysis of Components for 60-Year Plant Life," June 2000.

13. NUREG-1801, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, March 2001.

14. NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low–Alloy Steels," March 1998.

15. NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.

Table 4.3-1. Stress Range Reduction Factors

Number of Equivalent Full Temperature Cycles	Stress Range Reduction Factor
7,000 and less	1.0
7,000 to 14,000	0.9
14,000 to 22,000	0.8
22,000 to 45,000	0.7
45,000 to 100,000	0.6
100,000 and over	0.5

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

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

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

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

Table A.1-1. Elements of an Aging Management Program for License Renewal

Element	Description
1. Scope of program	Scope of program should include the specific structures and components subject to an AMR for license renewal.
2. Preventive actions	Preventive actions should prevent or mitigate aging degradation.
3. Parameters monitored or inspected	Parameters monitored or inspected should be linked to the degradation of the particular structure or component intended function(s).
4. Detection of aging effects	Detection of aging effects should occur before there is a loss of structure or component intended function(s). This includes aspects such as method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one-time inspections to ensure timely detection of aging effects.
5. Monitoring and trending	Monitoring and trending should provide predictability of the extent of degradation, and timely corrective or mitigative actions.
6. Acceptance criteria	Acceptance criteria, against which the need for corrective action will be evaluated, should ensure that the structure or component intended function(s) are maintained under all CLB design conditions during the period of extended operation.
7. Corrective actions	Corrective actions, including root cause determination and prevention of recurrence, should be timely.
8. Confirmation process	Confirmation process should ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.
9. Administrative controls	Administrative controls should provide a formal review and approval process.
10. Operating experience	Operating experience of the aging management program, including past corrective actions resulting in program enhancements or additional programs, should provide objective evidence to support the conclusion that the effects of aging will be managed adequately so that the structure and component intended function(s) will be maintained during the period of extended operation.

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 2

NUREG-1801, Vol. 2, Rev. 1

Generic Aging Lessons Learned (GALL) Report

Tabulation of Results

Manuscript Completed: September 2005
Date Published: September 2005

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

X.M1 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

Program Description

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

The AMP addresses the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant. Examples of critical components are identified in NUREG/CR-6260. The sample of critical components can be evaluated by applying environmental life correction factors to the existing ASME Code fatigue analyses. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels.

As evaluated below, this is an acceptable option for managing metal fatigue for the reactor coolant pressure boundary, considering environmental effects. Thus, no further evaluation is recommended for license renewal if the applicant selects this option under 10 CFR 54.21(c)(1)(iii) to evaluate metal fatigue for the reactor coolant pressure boundary.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes preventive measures to mitigate fatigue cracking of metal components of the reactor coolant pressure boundary caused by anticipated cyclic strains in the material.

2. ***Preventive Actions:*** Maintaining the fatigue usage factor below the design code limit and considering the effect of the reactor water environment, as described under the program description, will provide adequate margin against fatigue cracking of reactor coolant system components due to anticipated cyclic strains.

3. ***Parameters Monitored/Inspected:*** The program monitors all plant transients that cause cyclic strains, which are significant contributors to the fatigue usage factor. The number of plant transients that cause significant fatigue usage for each critical reactor coolant pressure boundary component is to be monitored. Alternatively, more detailed local monitoring of the plant transient may be used to compute the actual fatigue usage for each transient.

4. ***Detection of Aging Effects:*** The program provides for periodic update of the fatigue usage calculations.

5. ***Monitoring and Trending:*** The program monitors a sample of high fatigue usage locations. This sample is to include the locations identified in NUREG/CR-6260, as minimum, or propose alternatives based on plant configuration.

6. ***Acceptance Criteria:*** The acceptance criteria involves maintaining the fatigue usage below the design code limit considering environmental fatigue effects as described under the program description.

7. ***Corrective Actions:*** The program provides for corrective actions to prevent the usage factor from exceeding the design code limit during the period of extended operation.

Acceptable corrective actions include repair of the component, replacement of the component, and a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded during the extended period of operation. For programs that monitor a sample of high fatigue usage locations, corrective actions include a review of additional affected reactor coolant pressure boundary locations. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of Appendix B to 10 CFR Part 50. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls..

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** The program reviews industry experience regarding fatigue cracking. Applicable experience with fatigue cracking is to be considered in selecting the monitored locations.

References

NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels,* U.S. Nuclear Regulatory Commission, April 1999.

NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components,* U.S. Nuclear Regulatory Commission, March 1995.

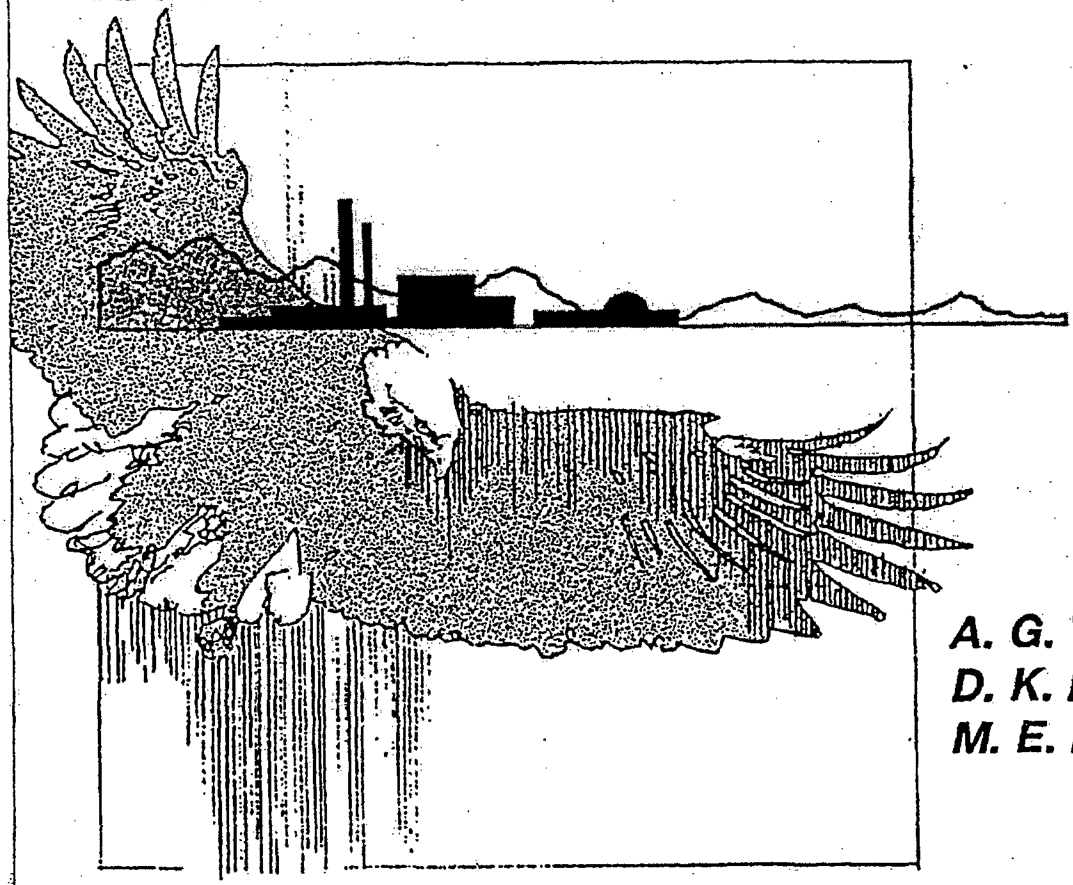
NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels,* U.S. Nuclear Regulatory Commission, March 1998.

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 3

IDAHO NATIONAL ENGINEERING LABORATORY

NUREG/CR-6260
INEL-95/0045



February 1995

A. G. Ware
D. K. Morton
M. E. Nitzel

Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components

Idaho National Engineering Laboratory

Lockheed
Idaho Technologies Company

Work performed under
DOE Contract
No. DE-AC07-94ID13223

NUREG/CR-6260
INEL-95/0045
Distribution Category: R5

Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components

A. G. Ware
D. K. Morton
M. E. Nitzel

Manuscript Completed February 1995

Idaho National Engineering Laboratory
Lockheed Idaho Technologies Company
Idaho Falls, Idaho 83415

Prepared for the
Division of Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Idaho Operations Office
Contract DE-AC07-94ID13223
FIN J2081

ABSTRACT

Recent test data indicate that the effects of the light water reactor (LWR) environment could significantly reduce the fatigue resistance of materials used in the reactor coolant pressure boundary components of operating nuclear power plants. Argonne National Laboratory has developed interim fatigue curves based on test data simulating LWR conditions, and published them in NUREG/CR-5999. In order to assess the significance of these interim fatigue curves, fatigue evaluations of a sample of the components in the reactor coolant pressure boundary of LWRs were performed. The sample consists of components from facilities designed by each of the four U.S. nuclear steam supply system vendors. For each facility, six locations were studied, including two locations on the reactor pressure vessel. In addition, there are older vintage plants where components of the reactor coolant pressure boundary were designed to codes that did not require an explicit fatigue analysis of the components. In order to assess the fatigue resistance of the older vintage plants, an evaluation was also conducted on selected components of three of these plants. This report discusses the insights gained from the application of the interim fatigue curves to components of seven operating nuclear power plants.

4. APPROACH

4.1 Selection of Components for Analysis

The components chosen for the evaluation of the five PWR plants [B&W, Combustion Engineering (one older vintage and one newer vintage), and Westinghouse (one older vintage and one newer vintage)] are as follows:

1. Reactor vessel shell and lower head.

2. Reactor vessel inlet and outlet nozzles.

3. Pressurizer surge line (including hot leg and pressurizer nozzles).

4. Reactor coolant piping charging system nozzle.

5. Reactor coolant piping safety injection nozzle.

6. Residual heat removal (RHR) system Class 1 piping.

The terminology used above is for Westinghouse plants. The first three components are the same for Combustion Engineering and B&W plants, but the latter three components for the three PWR nuclear steam supply system (NSSS) vendors are different either simply in name or in the routing of the piping. For cases where there is no direct one-for-one correspondence, the location that most nearly corresponded to the Westinghouse component was chosen. These locations are described in Section 5.

The components chosen for the evaluation of the two BWR plants [General Electric (one older vintage and one newer vintage)] are as follows:

1. Reactor vessel shell and lower head.

2. Reactor vessel feedwater nozzle.

3. Reactor recirculation piping (including inlet and outlet nozzles).

4. Core spray line reactor vessel nozzle and associated Class 1 piping.

5. RHR Class 1 piping.

6. Feedwater line Class 1 piping.

For both PWR and BWR plants, these components are not necessarily the locations with the highest design CUFs in the plant, but were chosen to give a representative overview of components that had higher CUFs and/or were important from a risk perspective. For example, the reactor vessel shell and lower head was chosen for its risk importance.

4.2 Application of NUREG/CR-5999 Fatigue Curves

NUREG/CR-5999 includes one fatigue curve for stainless steel, but several curves for carbon/low-alloy steels which are based on the sulfur content of the steel and the oxygen level in the coolant. For the five PWR plants, the curves for high-sulfur steel and a low-oxygen environment (typical for PWRs) were used. For the two BWR plants, the curves for high-sulfur steel and a high-oxygen environment were used. The high-oxygen (greater than 100 ppm) environment considered in the selected curves is consistent with the water chemistry in BWRs without hydrogen water chemistry. Neither of the two BWR plants evaluated have used hydrogen water chemistry.

4.2.1 Interior and Exterior Surfaces.

The highest CUFs for components in the seven plants evaluated in this fatigue assessment study generally occur on the interior surfaces which experience the full effects of thermal shocks from fluid temperature changes. In a few cases the highest CUF was found to occur on the exterior surface (because of stress concentration effects), and in other cases no differentiation between interior and exterior surfaces was made in the licensee's calculations. Since it is expected that the interior

5.5 Older Vintage Westinghouse Plant

A comparison of the design CUFs from the licensee's design basis calculations and CUFs using the NUREG/CR-5999 interim fatigue curves was carried out for the locations of highest design CUF for the six components listed below:

1. Reactor vessel shell and lower head
2. Reactor vessel inlet and outlet nozzles
3. Pressurizer surge line (including hot leg and pressurizer nozzles)
4. Reactor coolant piping charging system nozzle (representative design basis fatigue calculation performed by INEL)
5. Reactor coolant piping safety injection nozzle (representative design basis fatigue calculation performed by INEL)
6. Residual Heat Removal system Class 1 piping (representative design basis fatigue calculation performed by INEL).

As of late 1993, the plant has been operated approximately 20 of the 40 years currently approved in its operating license. Table 5-83 shows the design basis cycles for transients that are important from a fatigue standpoint for the six components that were evaluated. The numbers of transients to date have been extrapolated to 40 years by multiplying by 40/20.

The results of a generic Westinghouse plant study of thermal stratification in surge lines was included in the licensee's fatigue analysis of the surge line. There were no plant specific data to remove conservatism assumptions for this particular plant.

5.5.1 Reactor Vessel Shell and Lower Head. The highest CUF on the lower shell and head is 0.290 for the inside surface of the lower head near the shell-to-head transition, where core support guides are welded to the interior of the shell. The SA-302 Grade B head is protected from the coolant by a layer of stainless steel and Alloy 600 cladding. No fatigue analysis is performed for the cladding.

5.5.1.1 NUREG/CR-5999 CUF Based on Licensee's Design Calculation Stresses. The licensee's CUF calculations used the ASME Code, Section III, 1965 edition, through Summer 1966 addenda.

The effect of the NUREG/CR-5999 interim fatigue curve is shown in Table 5-84. As previously discussed, the results shown in Table 5-84 assume that the coolant is in contact with the low-alloy steel base metal underneath the cladding. The S_{alt} values were adjusted for the effect of the modulus of elasticity by multiplying by 30/27, the ratio of the modulus of elasticity on the fatigue curve in the current edition of the Code to the value at 500°F for SA-302 Grade B low-alloy steel. The 1965 Code edition did not require an adjustment for the effect of the modulus of elasticity.

Table 5-83. Number of selected design basis cycles compared to anticipated number of cycles over 40-year license life.

Transient	Design basis cycles	Anticipated cycles for 40 years
Heatup/cooldown	200	172
Reactor trip	400	426
Hydrotest	5	2
5% power change	14500	512
10% power change (up/down)	2000/7000	42/86
50% power change	200	136

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 4

NUREG/CR-6909
ANL-06/08

Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials

Final Report

Argonne National Laboratory

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

NUREG/CR-6909
ANL-06/08

Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials

Final Report

Manuscript Completed: November 2006
Date Published: February 2007

Prepared by
O. K. Chopra and W. J. Shack

Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

H. J. Gonzalez, NRC Project Manager

Prepared for
Division of Fuel, Engineering and Radiological Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
NRC Job Code N6187

Abstract

The ASME Boiler and Pressure Vessel Code provides rules for the design of Class 1 components of nuclear power plants. Figures I–9.1 through I–9.6 of Appendix I to Section III of the Code specify design curves for applicable structural materials. However, the effects of light water reactor (LWR) coolant environments are not explicitly addressed by the Code design curves. The existing fatigue strain–vs.–life (ε–N) data illustrate potentially significant effects of LWR coolant environments on the fatigue resistance of pressure vessel and piping steels. Under certain environmental and loading conditions, fatigue lives in water relative to those in air can be a factor of ≈12 lower for austenitic stainless steels, ≈3 lower for Ni-Cr-Fe alloys, and ≈17 lower for carbon and low-alloy steels. This report summarizes the work performed at Argonne National Laboratory on the fatigue of piping and pressure vessel steels in LWR environments. The existing fatigue ε–N data have been evaluated to identify the various material, environmental, and loading parameters that influence fatigue crack initiation, and to establish the effects of key parameters on the fatigue life of these steels. Fatigue life models are presented for estimating fatigue life as a function of material, loading, and environmental conditions. The environmental fatigue correction factor for incorporating the effects of LWR environments into ASME Section III fatigue evaluations is described. The report also presents a critical review of the ASME Code fatigue design margins of 2 on stress (or strain) and 20 on life and assesses the possible conservatism in the current choice of design margins.

This page is intentionally left blank.

Foreword

This report summarizes, reviews, and quantifies the effects of the light-water reactor (LWR) environment on the fatigue life of reactor materials, including carbon steels, low-alloy steels, nickel-chromium-iron (Ni-Cr-Fe) alloys, and austenitic stainless steels. The primary purpose of this report is to provide the background and technical bases to support Regulatory Guide 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors."

Previously published related reports include NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," issued April 1999; NUREG/CR-6717, "Environmental Effects on Fatigue Crack Initiation in Piping and Pressure Vessel Steels," issued May 2001; NUREG/CR-6787, "Mechanism and Estimation of Fatigue Crack Initiation in Austenitic Stainless Steels in LWR Environments," issued August 2002; NUREG/CR-6815, "Review of the Margins for ASME Code Fatigue Design Curve – Effects of Surface Roughness and Material Variability," issued September 2003; and NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," issued February 1998. This report provides a review of the existing fatigue ε-N data for carbon steels, low-alloy steels, Ni-Cr-Fe alloys, and austenitic stainless steels to define the potential effects of key material, loading, and environmental parameters on the fatigue life of the steels. By drawing upon a larger database than was used in earlier published reports, the U.S. Nuclear Regulatory Commission (NRC) has been able to update the Argonne National Laboratory (ANL) fatigue life models used to estimate the fatigue curves as a function of those parameters. In addition, this report presents a procedure for incorporating environmental effects into fatigue evaluations. The database described in this report (and its predecessors) reinforces the position espoused by the NRC that a guideline for incorporating the LWR environmental effects in the fatigue life evaluations should be developed and that the design curves for the fatigue life of pressure boundary and internal components fabricated from stainless steel should be revised. Toward that end, this report proposes a method for establishing reference curves and environmental correction factors for use in evaluating the fatigue life of reactor components exposed to LWR coolants and operational experience.

Data described in this review have been used to define fatigue design curves in air that are consistent with the existing fatigue data. Specifically, the published data indicate that the existing code curves are nonconservative for austenitic stainless steels (e.g., Types 304, 316, and 316NG). Regulatory Guide 1.207 endorses the new stainless steel fatigue design curves presented herein for incorporation in fatigue analyses for new reactors. However, because of significant conservatism in quantifying other plant-related variables (such as cyclic behavior, including stress and loading rates) involved in cumulative fatigue life calculations, the design of the current fleet of reactors is satisfactory.

Brian W. Sheron, Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Executive Summary

Section III, Subsection NB, of the ASME Boiler and Pressure Vessel Code contains rules for the design of Class 1 components of nuclear power plants. Figures I–9.1 through I–9.6 of Appendix I to Section III specify the Code design fatigue curves for applicable structural materials. However, Section III, Subsection NB–3121 of the Code states that the effects of the coolant environment on fatigue resistance of a material were not intended to be addressed in these design curves. Therefore, the effects of environment on the fatigue resistance of materials used in operating pressurized water reactor (PWR) and boiling water reactor (BWR) plants, whose primary–coolant pressure boundary components were designed in accordance with the Code, are uncertain.

The current Section–III design fatigue curves of the ASME Code were based primarily on strain–controlled fatigue tests of small polished specimens at room temperature in air. Best–fit curves to the experimental test data were first adjusted to account for the effects of mean stress and then lowered by a factor of 2 on stress and 20 on cycles (whichever was more conservative) to obtain the design fatigue curves. These factors are not safety margins but rather adjustment factors that must be applied to experimental data to obtain estimates of the lives of components. Recent fatigue–strain–vs.–life (ε–N) data obtained in the U.S. and Japan demonstrate that light water reactor (LWR) environments can have potentially significant effects on the fatigue resistance of materials. Specimen lives obtained from tests in simulated LWR environments can be much shorter than those obtained from corresponding tests in air.

This report reviews the existing fatigue ε–N data for carbon and low–alloy steels, wrought and cast austenitic stainless steels (SSs), and nickel-chromium-iron (Ni-Cr-Fe) alloys in air and LWR environments. The effects of various material, loading, and environmental parameters on the fatigue lives of these steels are summarized. The results indicate that in air, the ASME mean curve for low–alloy steels is in good agreement with the available experimental data, and the curve for carbon steels is somewhat conservative. However, in air, the ASME mean curve for SSs is not consistent with the experimental data at strain amplitudes <0.5% or stress amplitudes <975 MPa (<141 ksi); the ASME mean curve is nonconservative. The results also indicate that the fatigue data for Ni-Cr-Fe alloys are not consistent with the current ASME Code mean curve for austenitic SSs.

The fatigue lives of carbon and low–alloy steels, austenitic SSs, and Ni-Cr-Fe alloys are decreased in LWR environments. The reduction depends on some key material, loading, and environmental parameters. The fatigue data are consistent with the much larger database on enhancement of crack growth rates in these materials in LWR environments. The key parameters that influence fatigue life in these environments, e.g., temperature, dissolved–oxygen (DO) level in water, strain rate, strain (or stress) amplitude, and, for carbon and low–alloy steels, S content of the steel, have been identified. Also, the range of the values of these parameters within which environmental effects are significant has been clearly defined. If these critical loading and environmental conditions exist during reactor operation, then environmental effects will be significant and need to be included in the ASME Code fatigue evaluations.

Fatigue life models developed earlier to predict fatigue lives of small smooth specimens of carbon and low–alloy steels, wrought and cast austenitic SSs, and Ni-Cr-Fe alloys as a function of material, loading, and environmental parameters have been updated/revised by drawing upon a larger fatigue ε–N database. The functional form and bounding values of these parameters were based on experimental observations and data trends. An approach that can be used to incorporate the effects of LWR coolant environments into the ASME Code fatigue evaluations, based on the environmental fatigue correction factor, F_{en}, is discussed. The fatigue usage for a specific stress cycle of load set pair based on the Code fatigue design curves is multiplied by the correction factor to account for environmental effects.

The report also presents a critical review of the ASME Code fatigue design margins of 2 on stress and 20 on life and assesses the possible conservatism in the current choice of design margins. Although these factors were intended to be somewhat conservative, they should not be considered safety margins. These factors cover the effects of variables that can influence fatigue life but were not investigated in the experimental data that were used to obtain the fatigue design curves. Data available in the literature have been reviewed to evaluate the margins on cycles and stress that are needed to account for such differences and uncertainties. Monte Carlo simulations were performed to determine the margin on cycles needed to obtain a fatigue design curve that would provide a somewhat conservative estimate of the number of cycles to initiate a fatigue crack in reactor components. The results suggest that for both carbon and low–alloy steels and austenitic SSs, the current ASME Code requirements of a factor of 20 on cycle to account for the effects of material variability and data scatter, as well as size, surface finish, and loading history in low cycle fatigue, contain at least a factor of 1.7 conservatism. Thus, to reduce this conservatism, fatigue design curves have been developed from the ANL fatigue life model by first correcting for mean stress effects, and then reducing the mean–stress adjusted curve by a factor of 2 on stress or 12 on cycles, whichever is more conservative. These design curves are consistent with the existing fatigue ε–N data. A detailed procedure for incorporating environmental effects into fatigue evaluations is presented.

7 Margins in ASME Code Fatigue Design Curves

Conservatism in the ASME Code fatigue evaluations may arise from (a) the fatigue evaluation procedures and/or (b) the fatigue design curves. The overall conservatism in ASME Code fatigue evaluations has been demonstrated in fatigue tests on components.[120,121] Mayfield et al.[120] have shown that, in air, the margins on the number of cycles to failure for elbows and tees were 40–310 and 104–510, respectively, for austenitic SS and 118–2500 and 123–1700, respectively, for carbon steel. The margins for girth butt welds were significantly lower, 6–77 for SS and 14–128 for carbon steel. Data obtained by Heald and Kiss[121] on 26 piping components at room temperature and 288°C showed that the design margin for cracking exceeds 20, and for most of the components, it is >100. In these tests, fatigue life was expressed as the number of cycles for the crack to penetrate through the wall, which ranged in thickness from 6 to 18 mm. Consequently, depending on wall thickness, the actual margins to form a 3–mm crack may be lower by a factor of more than 2.

Deardorff and Smith[122] discussed the types and extent of conservatism present in the ASME Section III fatigue evaluation procedures and the effects of LWR environments on fatigue margins. The sources of conservatism in the procedures include the use of design transients that are significantly more severe than those experienced in service, conservative grouping of transients, and use of simplified elastic–plastic analyses that lead to higher stresses. The authors estimated that the ratio of the CUFs computed with the mean experimental curve for test specimen data in air and more accurate values of the stress to the CUFs computed with the Code fatigue design curve were ≈60 and 90, respectively, for PWR and BWR nozzles. The reductions in these margins due to environmental effects were estimated to be factors of 5.2 and 4.6 for PWR and BWR nozzles, respectively. Thus, Deardorff and Smith[122] argue that, after accounting for environmental effects, factors of 12 and 20 on life for PWR and BWR nozzles, respectively, account for uncertainties due to material variability, surface finish, size, mean stress, and loading sequence.

However, other studies on piping and components indicate that the Code fatigue design procedures do not always ensure large margins of safety.[123,124] Southwest Research Institute performed fatigue tests in room–temperature water on 0.91–m–diameter carbon and low–alloy steel vessels.[123] In the low–cycle regime, ≈5–mm–deep cracks were initiated slightly above (a factor of <2) the number of cycles predicted by the ASME Code design curve (Fig. 62a). Battelle–Columbus conducted tests on 203–mm or 914–mm carbon steel pipe welds at room temperature in an inert environment, and Oak Ridge National Laboratory (ORNL) performed four–point bend tests on 406–mm–diameter Type 304 SS pipe removed from the C–reactor at the Savannah River site.[124] The results showed that the number of cycles to produce a leak was lower, and in some cases significantly lower, than that expected from the ASME Code fatigue design curves (Fig. 62a and b). The most striking results are for the ORNL "tie–in" and flawed "test" weld; these specimens cracked completely through the 12.7–mm–thick wall in a life 6 or 7 times shorter than expected from the Code curve. Note that the Battelle and ORNL results represent a through–wall crack; the number of cycles to initiate a 3–mm crack may be a factor of 2 lower.

Much of the margin in the current evaluations arises from design procedures (e.g., stress analysis rules and cycle counting) that, as discussed by Deardorff and Smith,[122] are quite conservative. However, the ASME Code permits new and improved approaches to fatigue evaluations (e.g., finite–element analyses, fatigue monitoring, and improved K_e factors) that can significantly decrease the conservatism in the current fatigue evaluation procedures.

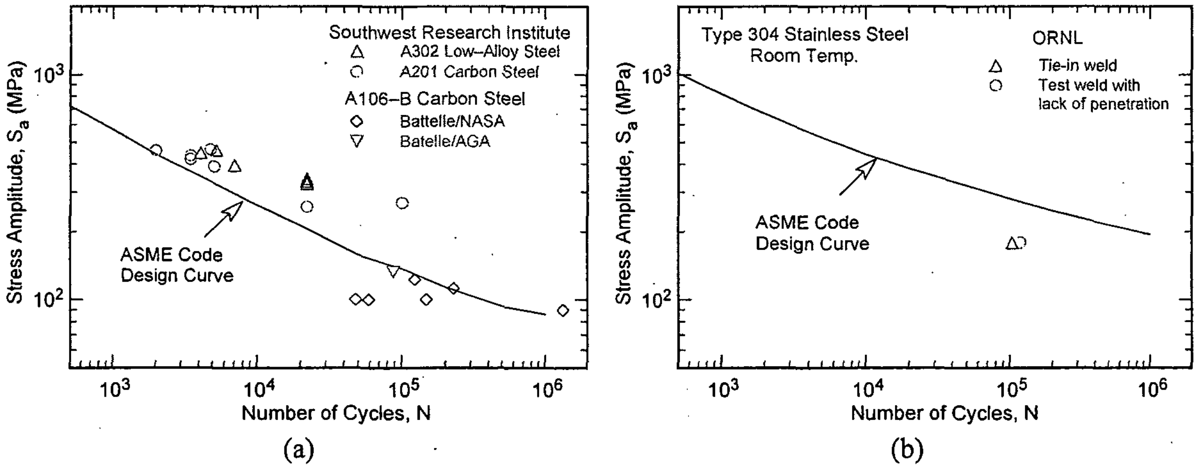


Figure 62. Fatigue data for (a) carbon and low–alloy steel and (b) Type 304 stainless steel components (Refs. 123,124).

The factors of 2 on stress and 20 on cycles used in the Code were intended to cover the effects of variables that can influence fatigue life but were not investigated in the tests that provided the data for the curves. It is not clear whether the particular values of 2 and 20 include possible conservatism. A study sponsored by the PVRC to assess the margins of 2 and 20 in fatigue design curves concluded that these margins should not be changed.[125]

The variables that can affect fatigue life in air and LWR environments can be broadly classified into three groups:

(a) Material
- (i) Composition
- (ii) Metallurgy: grain size, inclusions, orientation within a forging or plate
- (iii) Processing: cold work, heat treatment
- (iv) Size and geometry
- (v) Surface finish: fabrication surface condition
- (vi) Surface preparation: surface work hardening

(b) Loading
- (i) Strain rate: rise time
- (ii) Sequence: linear damage summation or Miner's rule
- (iii) Mean stress
- (iv) Biaxial effects: constraints

(c) Environment
- (i) Water chemistry: DO, lithium hydroxide, boric acid concentrations
- (ii) Temperature
- (iii) Flow rate

The existing fatigue ε–N database covers an adequate range of material parameters (i–iii), a loading parameter (i), and the environment parameters (i–ii); therefore, the variability and uncertainty in fatigue life due to these parameters have been incorporated into the model. The existing data are most likely conservative with respect to the effects of surface preparation because the fatigue ε–N data are obtained for specimens that are free of surface cold work. Fabrication procedures for fatigue test specimens generally follow American Society for Testing and Materials (ASTM) guidelines, which require that the final polishing of the specimens avoid surface work–hardening. Biaxial effects are covered by design procedures and need not be considered in the fatigue design curves.

As discussed earlier, under the conditions typical of operating BWRs, environmental effects on the fatigue life are a factor of ≈2 lower at high flow rates (7 m/s) than those at very low flow rates (0.3 m/s or lower) for carbon and low–alloy steels and are independent of flow rate for austenitic SSs.[19,20] However, because of the uncertainties in the flow conditions at or near the locations of crack initiation, the beneficial effect of flow rate on the fatigue life of carbon and low-alloy steels is presently not included in fatigue evaluations.

Thus, the contributions of four groups of variables, namely, material variability and data scatter, specimen size and geometry, surface finish, and loading sequence (Miner's rule), must be considered in developing fatigue design curves that are applicable to components.

7.1 Material Variability and Data Scatter

The effects of material variability and data scatter must be included to ensure that the design curves not only describe the available test data well, but also adequately describe the fatigue lives of the much larger number of heats of material that are found in the field. The effects of material variability and data scatter have been evaluated for the various materials by considering the best–fit curves determined from tests on individual heats of materials or loading conditions as samples of the much larger population of heats of materials and service conditions of interest. The fatigue behavior of each of the heats or loading conditions is characterized by the value of the constant A in Eq. 6. The values of A for the various data sets are ordered, and median ranks are used to estimate the cumulative distribution of A for the population. The distributions were fit to lognormal curves. The median value of A and standard deviation for each sample, as well as the number of data sets in the sample, are listed in Table 11. The 95/95 value of the margin on the median value to account for material variability and data scatter vary from 2.1 to 2.8 for the various samples. These margins applied to the mean value of life determined from the ANL fatigue life models provide 95% confidence that the fatigue life of 95 percentile of the materials and loading conditions of interest will be greater than the resultant value.

Table 11. The median value of A and standard deviation for the various fatigue ε–N data sets used to evaluate material variability and data scatter.

	Air Environment			Water Environment		
	Median Value of A	Standard Deviation	Number of Data Sets	Median Value of A	Standard Deviation	Number of Data Sets
Carbon Steel	6.583	0.477	17	5.951	0.376	33
Low-Alloy Steel	6.449	0.375	32	5.747	0.484	26
Stainless Steel	6.891	0.417	51	6.328	0.462	36

7.2 Size and Geometry

The effect of specimen size on the fatigue life was reviewed in earlier reports.[6,39] Various studies conclude that "size effect" is not a significant parameter in the design curve margins when the fatigue curve is based on data from axial strain control rather than bending tests. No intrinsic size effect has been observed for smooth specimens tested in axial loading or plain bending. However, a size effect does occur in specimens tested in rotating bending; the fatigue endurance limit decreases by ≈25% if the specimen size is increased from 2 to 16 mm but does not decrease further with larger sizes. Also, some effect of size and geometry has been observed on small–scale–vessel tests conducted at the Ecole

Polytechnique in conjunction with the large–size–pressure–vessel tests carried out by the Southwest Research Institute.[123] The tests at the Ecole Polytechnique were conducted in room–temperature water on 19–mm–thick shells with ≈305–mm inner diameter nozzles and made of machined bar stock. The results indicate that the fatigue lives determined from tests on the small–scale–vessel are 30–50% lower than those obtained from tests on small, smooth fatigue specimen. However, the difference in fatigue lives in these tests cannot be attributed to specimen size alone, it is due to the effects of both size and surface finish.

During cyclic loading, cracks generally form at surface irregularities either already in existence or produced by slip bands, grain boundaries, second phase particles, etc. In smooth specimens, formation of surface cracks is affected by the specimen size; crack initiation is easier in larger specimens because of the increased surface area and, therefore, increased number of sites for crack initiation. Specimen size is not likely to influence crack initiation in specimens with rough surfaces because cracks initiate at existing irregularities on the rough surface. As discussed in the next section, surface roughness has a large effect on fatigue life. Consequently, for rough surfaces, the effect of specimen size may not be considered in the margin of 20 on life. However, conservatively, a factor of 1.2–1.4 on life may be used to incorporate size effects on fatigue life in the low–cycle regime.

7.3 Surface Finish

The effect of surface finish must be considered to account for the difference in fatigue life expected in actual components with industrial–grade surface finish compared to the smooth polished surface of a test specimen. Fatigue life is sensitive to surface finish; cracks can initiate at surface irregularities that are normal to the stress axis. The height, spacing, shape, and distribution of surface irregularities are important for crack initiation. The effect of surface finish on crack initiation is expressed by Eq. 12 in terms of the RMS value of surface roughness (R_q).

The roughness of machined surfaces or natural finishes can range from ≈0.8 to 6.0 µm. Typical surface finish for various machining processes is in the range of 0.2–1.6 µm for cylindrical grinding, 0.4–3.0 µm for surface grinding, 0.8–3.0 µm for finish turning, and drilling and 1.6–4.0 µm for milling. For fabrication processes, it is in the range of 0.8–3.0 µm for extrusion and 1.6–4.0 µm for cold rolling. Thus, from Eq. 12, the fatigue life of components with such rough surfaces may be a factor of 2–3.5 lower than that of a smooth specimen.

Limited data in LWR environments on specimens that were intentionally roughened indicate that the effects of surface roughness on fatigue life is the same in air and water environments for austenitic SSs, but are insignificant in water for carbon and low-alloy steels. Thus, in LWR environments, a factor of 2.0–3.5 on life may also be used to account for the effects of surface finish on the fatigue life of austenitic SSs, but the factor may be lower for carbon and low-alloy steels, e.g., a factor of 2 may be used for carbon and low-alloy steels.

7.4 Loading Sequence

The effects of variable amplitude loading of smooth specimens were also reviewed in an earlier report.[39] In a variable loading sequence, the presence of a few cycles at high strain amplitude causes the fatigue life at smaller strain amplitude to be significantly lower than that at constant–amplitude loading, i.e., the fatigue limit of the material is lower under variable loading histories.

As discussed in Section 2, fatigue life has conventionally been divided into two stages: initiation, expressed as the cycles required to form microstructurally small cracks (MSCs) on the surface, and propagation, expressed as cycles required to propagate these MSCs to engineering size. The transition from initiation to propagation stage strongly depends on applied stress amplitude; at stress levels above the fatigue limit, the transition from initiation to propagation stage occurs at crack depths in the range of 150 to 250 μm. However, under constant loading at stress levels below the fatigue limit of the material (e.g., $\Delta\sigma_1$ in Fig. 1), although microcracks ≈10 μm can form quite early in life, they do not grow to an engineering size. Under the variable loading conditions encountered during service of power plants, cracks created by growth of MSCs at high stresses ($\Delta\sigma_3$ in Fig. 1) to depths larger than the transition crack depth can then grow to an engineering size even at stress levels below the fatigue limit.

Studies on fatigue damage in Type 304 SS under complex loading histories[126] indicate that the loading sequence of decreasing strain levels (i.e., high strain level followed by low strain level) is more damaging than that of increasing strain levels. The fatigue life of the steel at low strain levels decreased by a factor of 2–4 under a decreasing–strain sequence. In another study, the fatigue limit of medium carbon steels was lowered even after low–stress high–cycle fatigue; the higher the stress, the greater the decrease in fatigue threshold.[127] A recent study on Type 316NG and Ti-stabilized Type 316 SS on strain-controlled tests in air and PWR environment with constant or variable strain amplitude reported a factor of 3 or more decrease in fatigue life under variable amplitude compared with constant amplitude.[128] Although the strain spectrum used in the study was not intended to be representative of real transients, it represents a generic case and demonstrates the effect of loading sequence on fatigue life.

Because variable loading histories primarily influence fatigue life at low strain levels, the mean fatigue ε–N curves are lowered to account for damaging cycles that occur below the constant–amplitude fatigue limit of the material. However, conservatively, a factor of 1.2–2.0 on life may be used to incorporate the possible effects of load histories on fatigue life in the low–cycle regime.

7.5 Fatigue Design Curve Margins Summarized

The ASME Code fatigue design curves are currently obtained from the mean data curves by first adjusting for the effects of mean stress, and then reducing the life at each point of the adjusted curve by a factor of 2 on strain and 20 on life, whichever is more conservative. The factors on strain are needed primarily to account for the variation in the fatigue limit of the material caused by material variability, component size, surface finish, and load history. Because these variables affect life through their influence on the growth of short cracks (<100 μm), the adjustment on strain to account for such variations is typically not cumulative, i.e., the portion of the life can only be reduced by a finite amount. Thus, it is controlled by the variable that has the largest effect on life. In relating the fatigue lives of laboratory test specimens to those of actual reactor components, the factor of 2 on strain that is currently being used to develop the Code design curves is adequate to account for the uncertainties associated with material variability, component size, surface finish, and load history.

The factors on life are needed to account for variations in fatigue life in the low–cycle regime. Based on the discussions presented above the effects of various material, loading, and environmental parameters on fatigue life may be summarized as follows:

(a) The results presented in Table 11 may be used to determine the margins that need to be applied to the mean value of life to ensure that the resultant value of life would bound a specific percentile (e.g., 95 percentile) of the materials and loading conditions of interest.

(b) For rough surfaces, specimen size is not likely to influence fatigue life, and therefore, the effect of specimen size need not be considered in the margin of 20 on life. However, conservatively, a factor of 1.2–1.4 on life may be used to incorporate size effects on fatigue life.

(c) Limited data indicate that, for carbon and low-alloy steels, the effects of surface roughness on fatigue life are insignificant in LWR environments. A factor of 2 on life may be used for carbon and low-alloy steels in water environments instead of the 2.0–3.5 used for carbon and low–alloy steels in air and for austenitic SSs in both air and water environments.

(d) Variable loading histories primarily influence fatigue life at low strain levels, i.e., in the high–cycle regime, and the mean fatigue ε–N curves are lowered by a factor of 2 on strain to account for damaging cycles that occur below the constant–strain fatigue limit of the material. Conservatively, a factor of 1.2–2.0 on life may be used to incorporate the possible effects of load histories on fatigue life in the low–cycle regime.

The subfactors that are needed to account for the effects of the various material, loading, and environmental parameters on fatigue life are summarized in Table 12. The total adjustment on life may vary from 6 to 27. Because the maximum value represents a relatively poor heat of material and assumes the maximum effects of size, surface finish, and loading history, the maximum value of 27 is likely to be quite conservative. A value of 20 is currently being used to develop the Code design curves from the mean-data curves.

Table 12. Factors on life applied to mean fatigue ε–N curve to account for the effects of various material, loading, and environmental parameters.

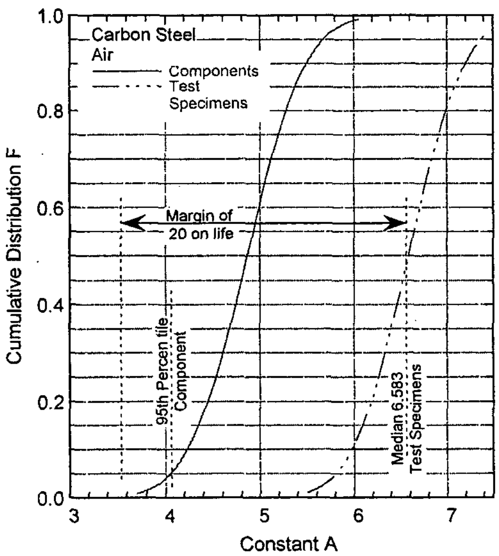
Parameter	Section III Criterion Document	Present Report
Material Variability and Data Scatter (minimum to mean)	2.0	2.1–2.8
Size Effect	2.5	1.2–1.4
Surface Finish, etc.	4.0	2.0–3.5*
Loading History	–	1.2–2.0
Total Adjustment	20	6.0–27.4

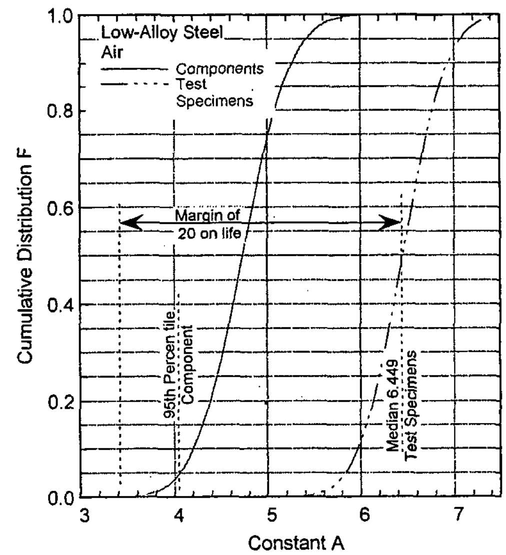
*A factor of 2 on life may be used for carbon and low-alloy steels in LWR environments.

To determine the most appropriate value for the design margin on life, Monte Carlo simulations were performed using the material variability and data scatter results given in Table 11, and the margins needed to account for the effects of size, surface finish, and loading history listed in Table 12. A lognormal distribution was also assumed for the effects of size, surface finish, and loading history, and the minimum and maximum values of the adjustment factors, e.g., 1.2–1.4 for size, 2.0–3.5 for surface finish, and 1.2–2.0 for loading history, were assumed to represent the 5th and 95th percentile, respectively. The cumulative distribution of the values of A in the fatigue ε–N curve for test specimens and the adjusted curve that represents the behavior of actual components is shown in Fig. 63 for carbon and low–alloy steels and austenitic SSs.

The results indicate that, relative to the specimen curve, the median value of constant A for the component curve decreased by a factor of 5.6 to account for the effects of size, surface finish, and loading history, and the standard deviation of heat-to-heat variation of the component curve increased by 6–10%. The margin that has to be applied to the mean data curve for test specimens to obtain a component curve that would bound 95% of the population, is 11.0–12.7 for the various materials; the values are given in

Table 13. An average value of 12 on life may be used for developing fatigue design curves from the mean data curve. The choice of bounding the 95th percentile of the population for a design curve is somewhat arbitrary. It is done with the understanding that the design curve controls fatigue initiation, not failure. The choice also recognizes that there are conservatisms implied in the choice of log normal distributions, which have an infinite tail, and in the identification of what in many cases are bounding values of the effects as 95th percentile values.





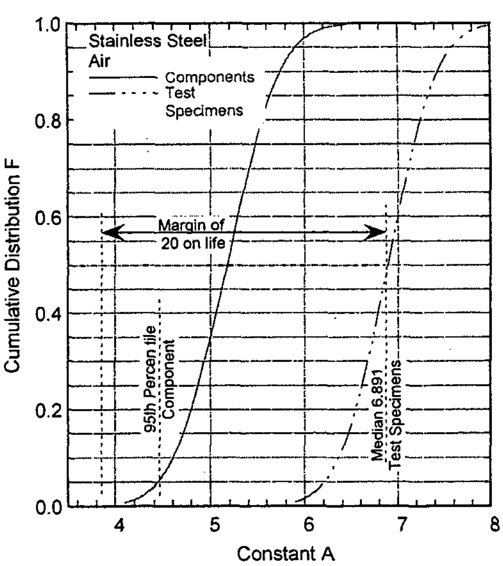


Figure 63.
Estimated cumulative distribution of parameter A in the ANL models that represent the fatigue life of test specimens and actual components in air.

Table 13. Margin applied to the mean values of fatigue life to bound 95% of the population.

Material	Air Environment
Carbon Steels	12.6
Low–Alloy Steels	11.0
Austenitic Stainless Steels	11.6

These results suggest that for all materials, the current ASME Code requirements of a factor of 20 on cycles to account for the effects of material variability and data scatter, as well as specimen size, surface finish, and loading history, contain at least a factor of 1.7 conservatism (i.e., $20/12 \approx 1.7$). Thus, to reduce this conservatism, fatigue design curves may be obtained from the mean data curve by first correcting for mean stress effects using the modified Goodman relationship, and then reducing the mean–stress adjusted curve by a factor of 2 on stress or 12 on cycles, whichever is more conservative. Fatigue design curves have been developed from the ANL fatigue life models using this procedure; the curves for carbon and low–alloy steels are presented in Section 4.1.10 and for wrought and cast austenitic SSs in Section 5.1.8.

8 Summary

The existing fatigue ε–N data for carbon and low–alloy steels, wrought and cast austenitic SSs, and Ni-Cr-Fe alloys have been evaluated to define the effects of key material, loading, and environmental parameters on the fatigue lives of these steels. The fatigue lives of these materials are decreased in LWR environments; the magnitude of the reduction depends on temperature, strain rate, DO level in water, and, for carbon and low–alloy steels, the S content of the steel. For all steels, environmental effects on fatigue life are significant only when critical parameters (temperature, strain rate, DO level, and strain amplitude) meet certain threshold values. Environmental effects are moderate, e.g., less than a factor of 2 decrease in life, when any one of the threshold conditions is not satisfied. The threshold values of the critical parameters and the effects of other parameters (such as water conductivity, water flow rate, and material heat treatment) on the fatigue life of the steels are summarized.

In air, the fatigue life of carbon and low–alloy steels depends on steel type, temperature, orientation, and strain rate. The fatigue life of carbon steels is a factor of ≈1.5 lower than that of low–alloy steels. For both steels, fatigue life decreases with increase in temperature. Some heats of carbon and low–alloy steels exhibit effects of strain rate and orientation. For these heats, fatigue life decreases with decreasing strain rate. Also, based on the distribution and morphology of sulfides, the fatigue properties in the transverse orientation may be inferior to those in the rolling orientation. The data indicate significant heat–to–heat variation; at 288°C, the fatigue life of carbon and low–alloy steels may vary by up to a factor of 3 above or below the mean value. Fatigue life is very sensitive to surface finish; the fatigue life of specimens with rough surfaces may be up to a factor of 3 lower than that of smooth specimens. The results also indicate that in room-temperature air, the ASME mean curve for low–alloy steels is in good agreement with the available experimental data, and the curve for carbon steels is somewhat conservative.

The fatigue lives of both carbon and low–alloy steels are decreased in LWR environments; the reduction depends on temperature, strain rate, DO level in water, and S content of the steel. The fatigue life is decreased significantly when four conditions are satisfied simultaneously, viz., the strain amplitude, temperature, and DO in water are above certain minimum levels, and the strain rate is below a threshold value. The S content in the steel is also important; its effect on life depends on the DO level in water.

Although the microstructures and cyclic–hardening behavior of carbon and low–alloy steels differ significantly, environmental degradation of the fatigue life of these steels is very similar. For both steels, only a moderate decrease in life (by a factor of <2) is observed when any one of the threshold conditions is not satisfied, e.g., low–DO PWR environment, temperatures <150°C, or vibratory fatigue. The existing fatigue S–N data have been reviewed to establish the critical parameters that influence fatigue life and define their threshold and limiting values within which environmental effects are significant.

In air, the fatigue lives of Types 304 and 316 SS are comparable; those of Type 316NG are superior to those of Types 304 and 316 SS at high strain amplitudes. The fatigue lives of austenitic SSs in air are independent of temperature in the range from room temperature to 427°C. Also, variation in strain rate in the range of 0.4–0.008%/s has no effect on the fatigue lives of SSs at temperatures up to 400°C. The fatigue ε–N behavior of cast SSs is similar to that of wrought austenitic SSs. The results indicate that the ASME mean-data curve for SSs is not consistent with the experimental data at strain amplitudes <0.5% or stress amplitudes <975 MPa (<141 ksi); the ASME mean curve predicts significantly longer lives than those observed experimentally.

The fatigue lives of cast and wrought austenitic SSs decrease in LWR environments compared to those in air. The decrease depends on strain rate, DO level in water, and temperature. A minimum threshold strain is required for an environmentally assisted decrease in the fatigue life of SSs, and this strain appears to be independent of material type (weld or base metal) and temperature in the range of 250–325°C. Environmental effects on fatigue life occur primarily during the tensile–loading cycle and at strain levels greater than the threshold value. Strain rate and temperature have a strong effect on fatigue life in LWR environments. Fatigue life decreases with decreasing strain rate below 0.4%/s; the effect saturates at 0.0004%/s. Similarly, the fatigue ε–N data suggest a threshold temperature of 150°C; in the range of 150–325°C, the logarithm of life decreases linearly with temperature.

The effect of DO level may be different for different steels. In low–DO water (i.e., <0.01 ppm DO) the fatigue lives of all wrought and cast austenitic SSs are decreased significantly; composition or heat treatment of the steel has little or no effect on fatigue life. However, in high–DO water, the environmental effects on fatigue life appear to be influenced by the composition and heat treatment of the steel; the effect of high-DO water on the fatigue lives of different compositions and heat treatment of SSs is not well established. Limited data indicate that for a high–C Type 304 SS, environmental effects are significant only for sensitized steel. For a low–C Type 316NG SS, some effect of environment was observed even for mill–annealed steel (nonsensitized steel) in high–DO water, although the effect was smaller than that observed in low–DO water. Limited fatigue ε–N data indicate that the fatigue lives of cast SSs are approximately the same in low– and high–DO water and are comparable to those observed for wrought SSs in low–DO water. In the present report, environmental effects on the fatigue lives of wrought and cast austenitic SSs are considered to be the same in high-DO and low-DO environments.

The fatigue ε–N data for Ni-Cr-Fe alloys indicate that although the data for Alloy 690 are very limited, the fatigue lives of Alloy 690 are comparable to those of Alloy 600. Also, the fatigue lives of the Ni-Cr-Fe alloy welds are comparable to those of the wrought Alloys 600 and 690 in the low–cycle regime, i.e., $<10^5$ cycles, and are slightly superior to the lives of wrought materials in the high–cycle regime. The fatigue data for Ni-Cr-Fe alloys in LWR environments are very limited; the effects of key loading and environmental parameters on fatigue life are similar to those for austenitic SSs. For example, the fatigue life of these steels decreases logarithmically with decreasing strain rate. Also, the effects of environment are greater in the low-DO PWR water than the high-DO BWR water. The existing data are inadequate to determine accurately the functional form for the effect of temperature on fatigue life.

Fatigue life models developed earlier to predict fatigue lives of small smooth specimens of carbon and low–alloy steels and wrought and cast austenitic SSs as a function of material, loading, and environmental parameters have been updated/revised using a larger fatigue ε–N database. The functional form and bounding values of these parameters were based on experimental observations and data trends. The models are applicable for predicted fatigue lives $\leq 10^6$ cycles. The ANL fatigue life model proposed in the present report for austenitic SSs in air is also recommended for predicting the fatigue lives of small smooth specimens of Ni-Cr-Fe alloys.

An approach, based on the environmental fatigue correction factor, is discussed to incorporate the effects of LWR coolant environments into the ASME Code fatigue evaluations. To incorporate environmental effects into a Section III fatigue evaluation, the fatigue usage for a specific stress cycle of load set pair based on the current Code fatigue design curves is multiplied by the correction factor.

The report also presents a critical review of the ASME Code fatigue design margins of 2 on stress and 20 on life and assesses the possible conservatism in the current choice of design margins. These factors cover the effects of variables that can influence fatigue life but were not investigated in the tests that provided the data for the design curves. Although these factors were intended to be somewhat conservative, they should not be considered safety margins because they were intended to account for variables that are known to affect fatigue life. Data available in the literature have been reviewed to evaluate the margins on cycles and stress that are needed to account for the differences and uncertainties. Monte Carlo simulations were performed to determine the margin on cycles needed to obtain a fatigue design curve that would provide a somewhat conservative estimate of the number of cycles to initiate a fatigue crack in reactor components. The results suggest that for both carbon and low–alloy steels and austenitic SSs, the current ASME Code requirements of a factor of 20 on cycles to account for the effects of material variability and data scatter, as well as size, surface finish, and loading history, contain at least a factor of 1.7 conservatism. Thus, to reduce this conservatism, fatigue design curves have been developed from the ANL model by first correcting for mean stress effects, and then reducing the mean–stress adjusted curve by a factor of 2 on stress and 12 on cycles, whichever is more conservative. A detailed procedure for incorporating environmental effects into fatigue evaluations is also presented in Appendix A.

This page is intentionally left blank.

References

1. Langer, B. F., "Design of Pressure Vessels for Low–Cycle Fatigue," ASME J. Basic Eng. 84, 389–402, 1962.

2. "Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Sections III and VIII, Division 2," The American Society of Mechanical Engineers, New York, 1969.

3. Cooper, W. E., "The Initial Scope and Intent of the Section III Fatigue Design Procedure," Welding Research Council, Inc., Technical Information from Workshop on Cyclic Life and Environmental Effects in Nuclear Applications, Clearwater, Florida, Jan. 22-21, 1992.

4. Chopra, O. K., and W. J. Shack, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low–Alloy Steels," NUREG/CR–6583, ANL–97/18, March 1998.

5. Gavenda, D. J., P. R. Luebbers, and O. K. Chopra, "Crack Initiation and Crack Growth Behavior of Carbon and Low–Alloy Steels," Fatigue and Fracture 1, Vol. 350, S. Rahman, K. K. Yoon, S. Bhandari, R. Warke, and J. M. Bloom, eds., American Society of Mechanical Engineers, New York, pp. 243–255, 1997.

6. Chopra, O. K., and W. J. Shack, "Environmental Effects on Fatigue Crack Initiation in Piping and Pressure Vessel Steels," NUREG/CR–6717, ANL–00/27, May 2001.

7. Chopra, O. K., "Mechanisms and Estimation of Fatigue Crack Initiation in Austenitic Stainless Steels in LWR Environments," NUREG/CR–6787, ANL–01/25, Aug. 2002.

8. Hale, D. A., S. A. Wilson, E. Kiss, and A. J. Gianuzzi, "Low–Cycle Fatigue Evaluation of Primary Piping Materials in a BWR Environment," GEAP–20244, U.S. Nuclear Regulatory Commission, Sept. 1977.

9. Hale, D. A., S. A. Wilson, J. N. Kass, and E. Kiss, "Low Cycle Fatigue Behavior of Commercial Piping Materials in a BWR Environment," J. Eng. Mater. Technol. 103, 15–25, 1981.

10. Ranganath, S., J. N. Kass, and J. D. Heald, "Fatigue Behavior of Carbon Steel Components in High–Temperature Water Environments," BWR Environmental Cracking Margins for Carbon Steel Piping, EPRI NP–2406, Appendix 3, Electric Power Research Institute, Palo Alto, CA, May 1982.

11. Ranganath, S., J. N. Kass, and J. D. Heald, "Fatigue Behavior of Carbon Steel Components in High–Temperature Water Environments," Low–Cycle Fatigue and Life Prediction, ASTM STP 770, C. Amzallag, B. N. Leis, and P. Rabbe, eds., American Society for Testing and Materials, Philadelphia, pp. 436–459, 1982.

12. Nagata, N., S. Sato, and Y. Katada, "Low–Cycle Fatigue Behavior of Pressure Vessel Steels in High–Temperature Pressurized Water," ISIJ Intl. 31 (1), 106–114, 1991.

13. Higuchi, M., and K. Iida, "Fatigue Strength Correction Factors for Carbon and Low–Alloy Steels in Oxygen–Containing High–Temperature Water," Nucl. Eng. Des. 129, 293–306, 1991.

14. Katada, Y., N. Nagata, and S. Sato, "Effect of Dissolved Oxygen Concentration on Fatigue Crack Growth Behavior of A533 B Steel in High Temperature Water," ISIJ Intl. 33 (8), 877–883, 1993.

15. Kanasaki, H., M. Hayashi, K. Iida, and Y. Asada, "Effects of Temperature Change on Fatigue Life of Carbon Steel in High Temperature Water," Fatigue and Crack Growth: Environmental Effects, Modeling Studies, and Design Considerations, PVP Vol. 306, S. Yukawa, ed., American Society of Mechanical Engineers, New York, pp. 117–122, 1995.

16. Nakao, G., H. Kanasaki, M. Higuchi, K. Iida, and Y. Asada, "Effects of Temperature and Dissolved Oxygen Content on Fatigue Life of Carbon and Low–Alloy Steels in LWR Water Environment," Fatigue and Crack Growth: Environmental Effects, Modeling Studies, and Design Considerations, PVP Vol. 306, S. Yukawa, ed., American Society of Mechanical Engineers, New York, pp. 123–128, 1995.

17. Higuchi, M., K. Iida, and Y. Asada, "Effects of Strain Rate Change on Fatigue Life of Carbon Steel in High–Temperature Water," Fatigue and Crack Growth: Environmental Effects, Modeling Studies, and Design Considerations, PVP Vol. 306, S. Yukawa, ed., American Society of Mechanical Engineers, New York, pp. 111–116, 1995; also Proc. of Symp. on Effects of the Environment on the Initiation of Crack Growth, ASTM STP 1298, American Society for Testing and Materials, Philadelphia, 1997.

18. Higuchi, M., K. Iida, and K. Sakaguchi, "Effects of Strain Rate Fluctuation and Strain Holding on Fatigue Life Reduction for LWR Structural Steels in Simulated PWR Water," Pressure Vessel and Piping Codes and Standards, PVP Vol. 419, M. D. Rana, ed., American Society of Mechanical Engineers, New York, pp. 143–152, 2001.

19. Hirano, A., M. Yamamoto, K. Sakaguchi, T. Shoji, and K. Iida, "Effects of Water Flow Rate on Fatigue Life of Ferritic and Austenitic Steels in Simulated LWR Environment," Pressure Vessel and Piping Codes and Standards – 2002, PVP Vol. 439, M. D. Rana, ed., American Society of Mechanical Engineers, New York, pp. 143–150, 2002.

20. Hirano, A., M. Yamamoto, K. Sakaguchi, and T. Shoji, "Effects of Water Flow Rate on Fatigue Life of Carbon and Stainless Steels in Simulated LWR Environment," Pressure Vessel and Piping Codes and Standards – 2004, PVP Vol. 480, American Society of Mechanical Engineers, New York, pp. 109–119, 2004.

21. Fujiwara, M., T. Endo, and H. Kanasaki, "Strain Rate Effects on the Low–Cycle Fatigue Strength of 304 Stainless Steel in High–Temperature Water Environment. Fatigue Life: Analysis and Prediction," Proc. Intl. Conf. and Exposition on Fatigue, Corrosion Cracking, Fracture Mechanics, and Failure Analysis, ASM, Metals Park, OH, pp. 309–313, 1986.

22. Mimaki, H., H. Kanasaki, I. Suzuki, M. Koyama, M. Akiyama, T. Okubo, and Y. Mishima, "Material Aging Research Program for PWR Plants," Aging Management Through Maintenance Management, PVP Vol. 332, I. T. Kisisel, ed., American Society of Mechanical Engineers, New York, pp. 97–105, 1996.

23. Kanasaki, H., R. Umehara, H. Mizuta, and T. Suyama, "Fatigue Lives of Stainless Steels in PWR Primary Water," Trans. 14th Intl. Conf. on Structural Mechanics in Reactor Technology (SMiRT 14), Lyon, France, pp. 473–483, 1997.

24. Kanasaki, H., R. Umehara, H. Mizuta, and T. Suyama, "Effects of Strain Rate and Temperature Change on the Fatigue Life of Stainless Steel in PWR Primary Water," Trans. 14th Intl. Conf. on Structural Mechanics in Reactor Technology (SMiRT 14), Lyon, France, pp. 485–493, 1997.

25. Higuchi, M., and K. Iida, "Reduction in Low–Cycle Fatigue Life of Austenitic Stainless Steels in High–Temperature Water," Pressure Vessel and Piping Codes and Standards, PVP Vol. 353, D. P. Jones, B. R. Newton, W. J. O'Donnell, R. Vecchio, G. A. Antaki, D. Bhavani, N. G. Cofie, and G. L. Hollinger, eds., American Society of Mechanical Engineers, New York, pp. 79–86, 1997.

26. Hayashi, M., "Thermal Fatigue Strength of Type 304 Stainless Steel in Simulated BWR Environment," Nucl. Eng. Des. 184, 135–144, 1998.

27. Hayashi, M., K. Enomoto, T. Saito, and T. Miyagawa, "Development of Thermal Fatigue Testing with BWR Water Environment and Thermal Fatigue Strength of Austenitic Stainless Steels," Nucl. Eng. Des. 184, 113–122, 1998.

28. Tsutsumi, K., H. Kanasaki, T. Umakoshi, T. Nakamura, S. Urata, H. Mizuta, and S. Nomoto, "Fatigue Life Reduction in PWR Water Environment for Stainless Steels," Assessment Methodologies for Preventing Failure: Service Experience and Environmental Considerations, PVP Vol. 410-2, R. Mohan, ed., American Society of Mechanical Engineers, New York, pp. 23–34, 2000.

29. Tsutsumi, K., T. Dodo, H. Kanasaki, S. Nomoto, Y. Minami, and T. Nakamura, "Fatigue Behavior of Stainless Steel under Conditions of Changing Strain Rate in PWR Primary Water," Pressure Vessel and Piping Codes and Standards, PVP Vol. 419, M. D. Rana, ed., American Society of Mechanical Engineers, New York, pp. 135–141, 2001.

30. Tsutsumi, K., M. Higuchi, K. Iida, and Y. Yamamoto, "The Modified Rate Approach to Evaluate Fatigue Life under Synchronously Changing Temperature and Strain Rate in Elevated Temperature Water," Pressure Vessel and Piping Codes and Standards – 2002, PVP Vol. 439, M. D. Rana, ed., American Society of Mechanical Engineers, New York, pp. 99–107, 2002.

31. Higuchi, M., T. Hirano, and K. Sakaguchi, "Evaluation of Fatigue Damage on Operating Plant Components in LWR Water," Pressure Vessel and Piping Codes and Standards – 2004, PVP Vol. 480, American Society of Mechanical Engineers, New York, pp. 129–138, 2004.

32. Nomura, Y., M. Higuchi, Y. Asada, and K. Sakaguchi, "The Modified Rate Approach Method to Evaluate Fatigue Life under Synchronously Changing Temperature and Strain Rate in Elevated Temperature Water in Austenitic Stainless Steels," Pressure Vessel and Piping Codes and Standards – 2004, PVP Vol. 480, American Society of Mechanical Engineers, New York, pp. 99–108, 2004.

33. Higuchi, M., K. Sakaguchi, A. Hirano, and Y. Nomura, "Revised and New Proposal of Environmental Fatigue Life Correction Factor (F_{en}) for Carbon and Low–Alloy Steels and Nickel Alloys in LWR Water Environments," Proc. of the 200 ASME Pressure Vessels and Piping Conf., July 23–27, 2006, Vancouver, BC, Canada, paper # PVP2006–ICPVT–93194.

34. Chopra, O. K., and W. J. Shack, "Evaluation of Effects of LWR Coolant Environments on Fatigue Life of Carbon and Low–Alloy Steels," Effects of the Environment on the Initiation of Crack Growth, ASTM STP 1298, W. A. Van Der Sluys, R. S. Piascik, and R. Zawierucha, eds., American Society for Testing and Materials, Philadelphia, pp. 247–266, 1997.

35. Chopra, O. K., and W. J. Shack, "Low–Cycle Fatigue of Piping and Pressure Vessel Steels in LWR Environments," Nucl. Eng. Des. 184, 49–76, 1998.

36. Chopra, O. K., and D. J. Gavenda, "Effects of LWR Coolant Environments on Fatigue Lives of Austenitic Stainless Steels," J. Pressure Vessel Technol. 120, 116–121, 1998.

37. Chopra, O. K., and J. L. Smith, "Estimation of Fatigue Strain–Life Curves for Austenitic Stainless Steels in Light Water Reactor Environments," Fatigue, Environmental Factors, and New Materials, PVP Vol. 374, H. S. Mehta, R. W. Swindeman, J. A. Todd, S. Yukawa, M. Zako, W. H. Bamford, M. Higuchi, E. Jones, H. Nickel, and S. Rahman, eds., American Society of Mechanical Engineers, New York, pp. 249–259, 1998.

38. Chopra, O. K., "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," NUREG/CR–5704, ANL–98/31, 1999.

39. Chopra, O. K., and W. J. Shack, "Review of the Margins for ASME Code Design Curves – Effects of Surface Roughness and Material Variability," NUREG/CR–6815, ANL–02/39, Sept. 2003.

40. Chopra, O. K., B. Alexandreanu, and W. J. Shack, "Effect of Material Heat Treatment on Fatigue Crack Initiation in Austenitic Stainless Steels in LWR Environments," NUREG/CR–6878, ANL–03/35, July 2005.

41. Terrell, J. B., "Fatigue Life Characterization of Smooth and Notched Piping Steel Specimens in 288°C Air Environments," NUREG/CR–5013, EM–2232 Materials Engineering Associates, Inc., Lanham, MD, May 1988.

42. Terrell, J. B., "Fatigue Strength of Smooth and Notched Specimens of ASME SA 106–B Steel in PWR Environments," NUREG/CR–5136, MEA–2289, Materials Engineering Associates, Inc., Lanham, MD, Sept. 1988.

43. Terrell, J. B., "Effect of Cyclic Frequency on the Fatigue Life of ASME SA–106–B Piping Steel in PWR Environments," J. Mater. Eng. 10, 193–203, 1988.

44. Lenz, E., N. Wieling, and H. Muenster, "Influence of Variation of Flow Rates and Temperature on the Cyclic Crack Growth Rate under BWR Conditions," Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, The Metallurgical Society, Warrendale, PA, 1988.

45. Garud, Y. S., S. R. Paterson, R, B, Dooley, R. S. Pathania, J. Hickling, and A. Bursik, "Corrosion Fatigue of Water Touched Pressure Retaining Components in Power Plants," EPRI TR–106696, Final Report, Electric Power Research Institute, Palo Alto, Nov. 1997.

46. Faidy, C., T. Le Courtois, E. de Fraguier, J–A Leduff, A. Lefrancois, and J. Dechelotte, "Thermal Fatigue in French RHR System," Int. Conf. on Fatigue of Reactor Components, Napa, CA, July 31–August 2, 2000.

47. Kussmaul, K., R. Rintamaa, J. Jansky, M. Kemppainen, and K. Törrönen, "On the Mechanism of Environmental Cracking Introduced by Cyclic Thermal Loading," in IAEA Specialists Meeting, Corrosion and Stress Corrosion of Steel Pressure Boundary Components and Steam Turbines, VTT Symp. 43, Espoo, Finland, pp. 195–243, 1983.

48. Hickling, J., "Strain Induced Corrosion Cracking of Low-Alloy Reactor Pressure Vessel Steels under BWR Conditions," Proc. 10th Intl. Symp. on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, F. P. Ford, S. M. Bruemmer, and G. S. Was, eds., The Minerals, Metals, and Materials Society, Warrendale, PA, CD-ROM, paper 0156, 2001.

49. Hickling, J., "Research and Service Experience with Environmentally Assisted Cracking of Low–Alloy Steel," Power Plant Chem., 7 (1), 4–15, 2005.

50. Iida, K., "A Review of Fatigue Failures in LWR Plants in Japan," Nucl. Eng. Des. 138, 297–312, 1992.

51. NRC IE Bulletin No. 79–13, "Cracking in Feedwater System Piping," U.S. Nuclear Regulatory Commission, Washington, DC, June 25, 1979.

52. NRC Information Notice 93–20, "Thermal Fatigue Cracking of Feedwater Piping to Steam Generators," U.S. Nuclear Regulatory Commission, Washington, DC, March 24, 1993.

53. Kussmaul, K., D. Blind, and J. Jansky, "Formation and Growth of Cracking in Feed Water Pipes and RPV Nozzles," Nucl. Eng. Des. 81, 105–119, 1984.

54. Gordon, B. M., D. E. Delwiche, and G. M. Gordon, "Service Experience of BWR Pressure Vessels," Performance and Evaluation of Light Water Reactor Pressure Vessels, PVP Vol.–119, American Society of Mechanical Engineers, New York, pp. 9–17, 1987.

55. Lenz, E., B. Stellwag, and N. Wieling, "The Influence of Strain–Induced Corrosion Cracking on the Crack Initiation in Low–Alloy Steels in HT–Water – A Relation Between Monotonic and Cyclic Crack Initiation Behavior," in IAEA Specialists Meeting Corrosion and Stress Corrosion of Steel Pressure Boundary Components and Steam Turbines, VTT Symp. 43, Espoo, Finland, pp. 243–267, 1983.

56. Hickling, J., and D. Blind, "Strain–Induced Corrosion Cracking of Low–Alloy Steels in LWR Systems – Case Histories and Identification of Conditions Leading to Susceptibility," Nucl. Eng. Des. 91, 305–330, 1986.

57. Hirschberg, P., A. F. Deardorff, and J. Carey, "Operating Experience Regarding Thermal Fatigue of Unisolable Piping Connected to PWR Reactor Coolant Systems," Int. Conf. on Fatigue of Reactor Components, Napa, CA, July 31–August 2, 2000.

58. NRC Information Notice 88–01, "Safety Injection Pipe Failure," U.S. Nuclear Regulatory Commission, Washington, DC (Jan. 27, 1988).

59. NRC Bulletin No. 88–08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," U.S. Nuclear Regulatory Commission, Washington, DC, June 22; Suppl. 1, June 24; Suppl. 2, Aug. 4, 1988; Suppl. 3, April 1989.

60. Sakai, T., "Leakage from CVCS Pipe of Regenerative Heat Exchanger Induced by High–Cycle Thermal Fatigue at Tsuruga Nuclear Power Station Unit 2," Int. Conf. on Fatigue of Reactor Components, Napa, CA, July 31–August 2, 2000.

61. Hoshino, T., T. Ueno, T. Aoki, and Y. Kutomi, "Leakage from CVCS Pipe of Regenerative Heat Exchanger Induced by High–Cycle Thermal Fatigue at Tsuruga Nuclear Power Station Unit 2," Proc. 8th Intl. Conf. on Nuclear Engineering, 1.01 Operational Experience/Root Cause Failure Analysis, Paper 8615, American Society of Mechanical Engineers, New York, 2000.

62. Stephan, J.–M., and J. C. Masson, "Auxiliary+ Feedwater Line Stratification and Coufast Simulation," Int. Conf. on Fatigue of Reactor Components, Napa, CA, July 31–August 2, 2000.

63. Kilian, R., J. Hickling, and R. Nickell, "Environmental Fatigue Testing of Stainless Steel Pipe Bends in Flowing, Simulated PWR Primary Water at 240°C," Third Intl. Conf. Fatigue of Reactor Components, MRP-151, Electric Power Research Institute, Palo Alto, CA, Aug. 2005.

64. NRC Bulletin No. 88–11, "Pressurizer Surge Line Thermal Stratification," U.S. Nuclear Regulatory Commission, Washington, DC, Dec. 20, 1988.

65. Majumdar, S., O. K. Chopra, and W. J. Shack, "Interim Fatigue Design Curves for Carbon, Low–Alloy, and Austenitic Stainless Steels in LWR Environments," NUREG/CR–5999, ANL–93/3, 1993.

66. Keisler, J., O. K. Chopra, and W. J. Shack, "Fatigue Strain–Life Behavior of Carbon and Low–Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments," NUREG/CR–6335, ANL–95/15, 1995.

67. Park, H. B., and O. K. Chopra, "A Fracture Mechanics Approach for Estimating Fatigue Crack Initiation in Carbon and Low-Alloy Steels in LWR Coolant Environment," Assessment Methodologies for Preventing Failure: Service Experience and Environmental Considerations, PVP Vol. 410-2, R. Mohan, ed., American Society of Mechanical Engineers, New York, pp. 3–11, 2000.

68. O'Donnell, T. P., and W. J. O'Donnell, "Stress Intensity Values in Conventional S-N Fatigue Specimens," Pressure Vessels and Piping Codes and Standard: Volume 1 – Current Applications, PVP Vol. 313–1, K. R. Rao and Y. Asada, eds., American Society of Mechanical Engineers, New York, pp. 191–192, 1995.

69. Amzallag, C., P. Rabbe, G. Gallet, and H.–P. Lieurade, "Influence des Conditions de Sollicitation Sur le Comportement en Fatigue Oligocyclique D'aciers Inoxydables Austénitiques," Memoires Scientifiques Revue Metallurgie Mars, pp. 161–173, 1978.

70. Solomon, H. D., C. Amzallag, A. J. Vallee, and R. E. De Lair, "Influence of Mean Stress on the Fatigue Behavior of 304L SS in Air and PWR Water," Proc. of the 2005 ASME Pressure Vessels and Piping Conf., July 17–21, 2005, Denver, CO, paper # PVP2005–71064.

71. Solomon, H. D., C. Amzallag, R. E. De Lair, and A. J. Vallee, "Strain Controlled Fatigue of Type 304L SS in Air and PWR Water," Proc. Third Intl. Conf. on Fatigue of Reactor Components, Seville, Spain, Oct. 3–6, 2004.

72. Jaske, C. E., and W. J. O'Donnell, "Fatigue Design Criteria for Pressure Vessel Alloys," Trans. ASME J. Pressure Vessel Technol. 99, 584–592, 1977.

73. Conway, J. B., R. H. Stentz, and J. T. Berling, "Fatigue, Tensile, and Relaxation Behavior of Stainless Steels," TID–26135, U.S. Atomic Energy Commission, Washington, DC, 1975.

74. Keller, D. L., "Progress on LMFBR Cladding, Structural, and Component Materials Studies During July, 1971 through June, 1972, Final Report," Task 32, Battelle–Columbus Laboratories, BMI–1928, 1977.

75. Jacko, R. J., "Fatigue Performance of Ni–Cr–Fe Alloy 600 under Typical PWR Steam Generator Conditions," EPRI NP–2957, Electric Power Research Institute, Palo Alto, CA, March 1983.

76. Dinerman, A. E., "Cyclic Strain Fatigue of Inconel at 75 to 600°F," KAPL–2084, Knolls Atomic Power Laboratory, Schenectady, NY, August 1960.

77. Mowbray, D. F., G. J. Sokol, and R. E. Savidge, "Fatigue Characteristics of Ni–Cr–Fe Alloys with Emphasis on Pressure–Vessel Cladding," KAPL–3108, Knolls Atomic Power Laboratory, Schenectady, NY, July 1965.

78. Van Der Sluys, W. A., B. A. Young, and D. Doyle, "Corrosion Fatigue Properties on Alloy 690 and Some Nickel–Based Weld Metals," Assessment Methodologies for Preventing Failure: Service Experience and Environmental Considerations, PVP Vol. 410-2, R. Mohan, ed., American Society of Mechanical Engineers, New York, pp. 85–91, 2000.

79. Iida, K., T. Bannai, M. Higuchi, K. Tsutsumi, and K. Sakaguchi, "Comparison of Japanese MITI Guideline and Other Methods for Evaluation of Environmental Fatigue Life Reduction," Pressure Vessel and Piping Codes and Standards, PVP Vol. 419, M. D. Rana, ed., American Society of Mechanical Engineers, New York, pp. 73–81, 2001.

80. Chopra, O. K., and W. J. Shack, "Overview of Fatigue Crack Initiation in Carbon and Low–Alloy Steels in Light Water Reactor Environments," J. Pressure Vessel Technol. 121, 49–60, 1999.

81. Higuchi, M., "Revised Proposal of Fatigue Life Correction Factor F_{en} for Carbon and Low Alloy Steels in LWR Water Environments," Assessment Methodologies for Preventing Failure: Service Experience and Environmental Considerations, PVP Vol. 410-2, R. Mohan, ed., American Society of Mechanical Engineers, New York, pp. 35–44, 2000.

82. Leax, T. R., "Statistical Models of Mean Stress and Water Environment Effects on the Fatigue Behavior of 304 Stainless Steel," Probabilistic and Environmental Aspects of Fracture and Fatigues, PVP Vol. 386, S. Rahman, ed., American Society of Mechanical Engineers, New York, pp. 229–239, 1999.

83. Ford, F. P., S. Ranganath, and D. Weinstein, "Environmentally Assisted Fatigue Crack Initiation in Low–Alloy Steels - A Review of the Literature and the ASME Code Design Requirements," EPRI Report TR-102765, Electroic Power Research Institute, Palo Alto, CA, Aug. 1993.

84. Ford, F. P., "Prediction of Corrosion Fatigue Initiation in Low–Alloy and Carbon Steel/Water Systems at 288°C," Proc. 6th Intl. Symp. on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, R. E. Gold and E. P. Simonen, eds., The Metallurgical Society, Warrendale, PA, pp. 9–17, 1993.

85. Mehta, H. S., and S. R. Gosselin, "Environmental Factor Approach to Account for Water Effects in Pressure Vessel and Piping Fatigue Evaluations," Nucl. Eng. Des. 181, 175–197, 1998.

86. Mehta, H. S., "An Update on the Consideration of Reactor Water Effects in Code Fatigue Initiation Evaluations for Pressure Vessels and Piping," Assessment Methodologies for Preventing Failure: Service Experience and Environmental Considerations, PVP Vol. 410-2, R. Mohan, ed., American Society of Mechanical Engineers, New York, pp. 45–51, 2000.

87. Van Der Sluys, W. A., and S. Yukawa, "Status of PVRC Evaluation of LWR Coolant Environmental Effects on the S–N Fatigue Properties of Pressure Boundary Materials," Fatigue and Crack Growth: Environmental Effects, Modeling Studies, and Design Considerations, PVP Vol. 306, S. Yukawa, ed., American Society of Mechanical Engineers, New York, pp. 47–58, 1995.

88. Van Der Sluys, W. A., "PVRC's Position on Environmental Effects on Fatigue Life in LWR Applications," Welding Research Council Bulletin 487, Welding Research Council, Inc., New York, Dec. 2003.

89. O'Donnell, W. J., W. J. O'Donnell, and T. P. O'Donnell, "Proposed New Fatigue Design Curves for Austenitic Stainless Steels, Alloy 600, and Alloy 800," Proc. of the 2005 ASME Pressure Vessels and Piping Conf., July 17–21, 2005, Denver, CO, paper # PVP2005–71409.

90. O'Donnell, W. J., W. J. O'Donnell, and T. P. O'Donnell, "Proposed New Fatigue Design Curves for Carbon and Low–Alloy Steels in High Temperature Water," Proc. of the 2005 ASME Pressure Vessels and Piping Conf., July 17–21, 2005, Denver, CO, paper # PVP2005–71410.

91. Abdel–Raouf, H., A. Plumtree, and T. H. Topper, "Effects of Temperature and Deformation Rate on Cyclic Strength and Fracture of Low–Carbon Steel," Cyclic Stress–Strain Behavior – Analysis, Experimentation, and Failure Prediction, ASTM STP 519, American Society for Testing and Materials, Philadelphia, pp. 28–57, 1973.

92. Lee, B. H., and I. S. Kim, "Dynamic Strain Aging in the High–Temperature Low–Cycle Fatigue of SA 508 Cl. 3 Forging Steel," J. Nucl. Mater. 226, 216–225, 1995.

93. Maiya, P. S., and D. E. Busch, "Effect of Surface Roughness on Low–Cycle Fatigue Behavior of Type 304 Stainless Steel," Met. Trans. 6A, 1761–1766, 1975.

94. Maiya, P. S., "Effect of Surface Roughness and Strain Range on Low–Cycle Fatigue Behavior of Type 304 Stainless Steel," Scripta Metall. 9, 1277–1282, 1975.

95. Stout, K. J., "Surface Roughness – Measurement, Interpretation, and Significance of Data," Mater. Eng. 2, 287–295, 1981.

96. Iida, K., "A Study of Surface Finish Effect Factor in ASME B & PV Code Section III," Pressure Vessel Technology, Vol. 2, L. Cengdian and R. W. Nichols, eds., Pergamon Press, New York, pp. 727–734, 1989.

97. Manjoine, M. J., and R. L. Johnson, "Fatigue Design Curves for Carbon and Low Alloy Steels up to 700°F (371°C)," Material Durability/Life Prediction Modeling: Materials for the 21st Century, PVP–Vol. 290, American Society of Mechanical Engineers, New York, 1994.

98. Johnson, L. G., "The Median Ranks of Sample Values in Their Population with an Application to Certain Fatigue Studies," Ind. Math. 2, 1–9, 1951.

99. Lipson, C., and N. J. Sheth, *Statistical Design and Analysis of Engineering Experiments*, McGraw Hill, New York, 1973.

100. Beck, J., and K. Arnold, *Parameter Estimation in Engineering and Science*, J. Wiley, New York, 1977.

101. Stambaugh, K. A., D. H. Leeson, F. V. Lawrence, C. Y. Hou, and G. Banas, "Reduction of S-N Curves for Ship Structural Details," Welding Research Council 398, January 1995.

102. Ford, F. P., and P. L. Andresen, "Stress Corrosion Cracking of Low–Alloy Pressure Vessel Steel in 288°C Water," Proc. 3rd Int. Atomic Energy Agency Specialists' Meeting on Subcritical Crack Growth, NUREG/CP–0112, Vol. 1, pp. 37–56, Aug. 1990.

103. Ford, F. P., "Overview of Collaborative Research into the Mechanisms of Environmentally Controlled Cracking in the Low Alloy Pressure Vessel Steel/Water System," Proc. 2nd Int. Atomic Energy Agency Specialists' Meeting on Subcritical Crack Growth, NUREG/CP–0067, MEA–2090, Vol. 2, pp. 3–71, April 1986.

104. Wire, G. L., and Y. Y. Li, "Initiation of Environmentally–Assisted Cracking in Low–Alloy Steels," Fatigue and Fracture Vol. 1, PVP Vol. 323, H. S. Mehta, ed., American Society of Mechanical Engineers, New York, pp. 269–289, 1996.

105. Pleune, T. T., and O. K. Chopra, "Artificial Neural Networks and Effects of Loading Conditions on Fatigue Life of Carbon and Low–Alloy Steels," Fatigue and Fracture Vol. 1, PVP Vol. 350, S. Rahman, K. K. Yoon, S. Bhandari, R. Warke, and J. M. Bloom, eds., American Society of Mechanical Engineers, New York, pp. 413–423, 1997.

106. Solomon, H. D., R. E. DeLair, and A. D. Unruh, "Crack Initiation in Low-Alloy Steel in High-Purity Water," Effects of the Environment on the Initiation of Crack Growth, ASTM STP 1298, W. A. Van Der Sluys, R. S. Piascik, and R. Zawierucha, eds., American Society for Testing and Materials, Philadelphia, pp. 135–149, 1997.

107. Solomon, H. D., R. E. DeLair, and E. Tolksdorf, "LCF Crack Initiation in WB36 in High-Temperature Water," Proc. 9th Intl. Symp. on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, F. P. Ford, S. M. Bruemmer, and G. S. Was, eds., The Minerals, Metals, and Materials Society, Warrendale, PA, pp. 865–872, 1999.

108. Cullen, W. H., M. Kemppainen, H. Hänninen, and K. Törrönen, "The Effects of Sulfur Chemistry and Flow Rate on Fatigue Crack Growth Rates in LWR Environments," NUREG/CR–4121, 1985.

109. Van Der Sluys, W. A., and R. H. Emanuelson, "Environmental Acceleration of Fatigue Crack Growth in Reactor Pressure Vessel Materials and Environments," Environmentally Assisted Cracking: Science and Engineering, ASTM STP 1049, W. B. Lisagor, T. W. Crooker, and B. N. Leis, eds., American Society for Testing and Materials, Philadelphia, PA, pp. 117–135, 1990.

110. Atkinson, J. D., J. Yu, and Z.–Y. Chen, "An Analysis of the Effects of Sulfur Content and Potential on Corrosion Fatigue Crack Growth in Reactor Pressure Vessel Steels," Corros. Sci. 38 (5), 755–765, 1996.

111. Auten, T. A., S. Z. Hayden, and R. H. Emanuelson, "Fatigue Crack Growth Rate Studies of Medium Sulfur Low Alloy Steels Tested in High Temperature Water," Proc. 6th Int. Symp. on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, R. E. Gold and E. P. Simonen, eds., The Metallurgical Society, Warrendale, PA, pp. 35–40, 1993.

112. Wire, G. L., T. R. Leax, and J. T. Kandra, "Mean Stress and Environmental Effects on Fatigue in Type 304 Stainless Steel," in Probabilistic and Environmental Aspects of Fracture and Fatigues, PVP Vol. 386, S. Rahman, ed., American Society of Mechanical Engineers, New York, pp. 213–228, 1999.

113. Diercks, D. R., "Development of Fatigue Design Curves for Pressure Vessel Alloys Using a Modified Langer Equation," Trans. ASME J. Pressure Vessel Technol. 101, 292–297, 1979.

114. Solomon, H. D., and C. Amzallag, "Comparison of Models Predicting the Fatigue Behavior of Austenitic Stainless Steels," Proc. of the 2005 ASME Pressure Vessels and Piping Conf., July 17–21, 2005, Denver, CO, paper # PVP2005–71063.

115. Kim, Y. J., "Characterization of the Oxide Film Formed on Type 316 Stainless Steel in 288°C Water in Cyclic Normal and Hydrogen Water Chemistries," Corrosion 51 (11), 849–860, 1995.

116. Kim, Y. J., "Analysis of Oxide Film Formed on Type 304 Stainless Steel in 288°C Water Containing Oxygen, Hydrogen, and Hydrogen Peroxide," Corrosion 55 (1), 81–88, 1999.

117. Chopra, O. K., "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR–4513, ANL–93/22, Aug. 1994.

118. Chopra, O. K., "Effect of Thermal Aging on Mechanical Properties of Cast Stainless Steels," in Proc. of the 2nd Int. Conf. on Heat–Resistant Materials, K. Natesan, P. Ganesan, and G. Lai, eds., ASM International, Materials Park, OH, pp. 479–485, 1995.

119. Higuchi, M., "Review and Consideration of Unsettled Problems on Evaluation of Fatigue Damage in LWR Water," Proc. of the 2005 ASME Pressure Vessels and Piping Conf., July 17–21, 2005, Denver, CO, paper # PVP2005–71306.

120. Mayfield, M. E., E. C. Rodabaugh, and R. J. Eiber, "A Comparison of Fatigue Test Data on Piping with the ASME Code Fatigue Evaluation Procedure," ASME Paper 79–PVP–92, American Society of Mechanical Engineers, New York, 1979.

121. Heald, J. D., and E. Kiss, "Low Cycle Fatigue of Nuclear Pipe Components," J. Pressure Vessel Technol. 74, PVP–5, 1–6, 1974.

122. Deardorff, A. F., and J. K. Smith, "Evaluation of Conservatisms and Environmental Effects in ASME Code, Section III, Class 1 Fatigue Analysis," SAND94–0187, prepared by Structural Integrity Associates, San Jose, CA, under contract to Sandia National Laboratories, Albuquerque, NM, 1994.

123. Kooistra, L. F., E. A. Lange, and A. G. Pickett, "Full–Size Pressure Vessel Testing and Its Application to Design," J. Eng. Power 86, 419–428, 1964.

124. Scott, P. M., and G. M. Wilkowski, "A Comparison of Recent Full–Scale Component Fatigue Data with the ASME Section III Fatigue Design Curves," in Fatigue and Crack Growth: Environmental Effects, Modeling Studies, and Design Considerations, PVP Vol. 306, S. Yukawa, ed., American Society of Mechanical Engineers, New York, pp. 129–138, 1995.

125. Hechmer, J., "Evaluation Methods for Fatigue - A PVRC Project," in Fatigue, Environmental Factors, and New Materials, PVP Vol. 374, H. S. Mehta, R. W. Swindeman, J. A. Todd, S. Yukawa, M. Zako, W. H. Bamford, M. Higuchi, E. Jones, H. Nickel, and S. Rahman, eds., American Society of Mechanical Engineers, New York, pp. 191–199, 1998.

126. Manjoine, M. J., "Fatigue Damage Models for Annealed Type 304 Stainless Steel under Complex Strain Histories," Trans. 6th Intl. Conf. on Structural Mechanics in Reactor Technology (SMiRT), Vol. L, 8/1, North–Holland Publishing Co., pp. 1–13, 1981.

127. Nian, L., and Du Bai–Ping, "The Effect of Low–Stress High–Cycle Fatigue on the Microstructure and Fatigue Threshold of a 40Cr Steel," Int. J. Fatigue 17 (1), 43–48, 1995.

128. Solin, J. P., "Fatigue of Stabilized SS and 316NG Alloy in PWR Environment," Proc. of the 2006 ASME Pressure Vessels and Piping Conf., July 23–27, 2006, Vancouver, BC, Canada, paper # PVP2006–ICPVT–93833.

This page is intentionally left blank.

APPENDIX A

Incorporating Environmental Effects into Fatigue Evaluations

A1 Scope

This Appendix provides the environmental fatigue correction factor (F_{en}) methodology that is considered acceptable for incorporating the effects of reactor coolant environments on fatigue usage factor evaluations of metal components for new reactor construction. The methodology for performing fatigue evaluations for the four major categories of structural materials, e.g., carbon steel, low-alloy steels, wrought and cast austenitic stainless steels, and Ni-Cr-Fe alloys, is described.

A2 Environmental Correction Factor (F_{en})

The effects of reactor coolant environments on the fatigue life of structural materials are expressed in terms of a nominal environmental fatigue correction factor, $F_{en,nom}$, which is defined as the ratio of fatigue life in air at room temperature ($N_{air,RT}$) to that in water at the service temperature (N_{water}):

$$F_{en,nom} = N_{air,RT}/N_{water}. \qquad (A.1)$$

The nominal environmental fatigue correction factor, $F_{en,nom}$, for carbon steels is expressed as

$$F_{en,nom} = \exp(0.632 - 0.101\, S^*\, T^*\, O^*\, \dot{\varepsilon}^*), \qquad (A.2)$$

and for low-alloy steels, it is expressed as

$$F_{en,nom} = \exp(0.702 - 0.101\, S^*\, T^*\, O^*\, \dot{\varepsilon}^*), \qquad (A.3)$$

where S^*, T^*, O^*, and $\dot{\varepsilon}^*$ are transformed S content, temperature, DO level, and strain rate, respectively, defined as:

$$\begin{aligned} S^* &= 0.001 && (S \le 0.001 \text{ wt.\%}) \\ S^* &= S && (S \le 0.015 \text{ wt.\%}) \\ S^* &= 0.015 && (S > 0.015 \text{ wt.\%}) \end{aligned} \qquad (A.4)$$

$$\begin{aligned} T^* &= 0 && (T < 150°C) \\ T^* &= T - 150 && (T = 150–350°C) \end{aligned} \qquad (A.5)$$

$$\begin{aligned} O^* &= 0 && (DO \le 0.04 \text{ ppm}) \\ O^* &= \ln(DO/0.04) && (0.04 \text{ ppm} < DO \le 0.5 \text{ ppm}) \\ O^* &= \ln(12.5) && (DO > 0.5 \text{ ppm}) \end{aligned} \qquad (A.6)$$

$$\begin{aligned} \dot{\varepsilon}^* &= 0 && (\dot{\varepsilon} > 1\%/s) \\ \dot{\varepsilon}^* &= \ln(\dot{\varepsilon}) && (0.001 \le \dot{\varepsilon} \le 1\%/s) \\ \dot{\varepsilon}^* &= \ln(0.001) && (\dot{\varepsilon} < 0.001\%/s). \end{aligned} \qquad (A.7)$$

For both carbon and low-alloy steels, a threshold value of 0.07% for strain amplitude (one-half the strain range for the cycle) is defined, below which environmental effects on the fatigue life of these steels do not occur. Thus,

$$F_{en,nom} = 1 \qquad (\varepsilon_a \leq 0.07\%). \tag{A.8}$$

For wrought and cast austenitic stainless steels,

$$F_{en,nom} = \exp(0.734 - T' O' \dot{\varepsilon}'). \tag{A.9}$$

where T', $\dot{\varepsilon}'$, and O' are transformed temperature, strain rate, and DO level, respectively, defined as:

$$\begin{aligned} T' &= 0 && (T < 150°C) \\ T' &= (T - 150)/175 && (150 \leq T < 325°C) \\ T' &= 1 && (T \geq 325°C) \end{aligned} \tag{A.10}$$

$$\begin{aligned} \dot{\varepsilon}' &= 0 && (\dot{\varepsilon} > 0.4\%/s) \\ \dot{\varepsilon}' &= \ln(\dot{\varepsilon}/0.4) && (0.0004 \leq \dot{\varepsilon} \leq 0.4\%/s) \\ \dot{\varepsilon}' &= \ln(0.0004/0.4) && (\dot{\varepsilon} < 0.0004\%/s) \end{aligned} \tag{A.11}$$

$$O' = 0.281 \qquad \text{(all DO levels).} \tag{A.12}$$

For wrought and cast austenitic stainless steels, a threshold value of 0.10% for strain amplitude (one-half the strain range for the cycle) is defined, below which environmental effects on the fatigue life of these steels do not occur. Thus,

$$F_{en,nom} = 1 \qquad (\varepsilon_a \leq 0.10\%). \tag{A.13}$$

For Ni-Cr-Fe alloys,

$$F_{en,nom} = \exp(-T' \dot{\varepsilon}' O'), \tag{A.14}$$

where T', $\dot{\varepsilon}'$, and O' are transformed temperature, strain rate, and DO, respectively, defined as:

$$\begin{aligned} T' &= T/325 && (T < 325°C) \\ T' &= 1 && (T \geq 325°C) \end{aligned} \tag{A.15}$$

$$\begin{aligned} \dot{\varepsilon}' &= 0 && (\dot{\varepsilon} > 5.0\%/s) \\ \dot{\varepsilon}' &= \ln(\dot{\varepsilon}/5.0) && (0.0004 \leq \dot{\varepsilon} \leq 5.0\%/s) \\ \dot{\varepsilon}' &= \ln(0.0004/5.0) && (\dot{\varepsilon} < 0.0004\%/s) \end{aligned} \tag{A.16}$$

$$\begin{aligned} O' &= 0.09 && \text{(NWC BWR water)} \\ O' &= 0.16 && \text{(PWR or HWC BWR water).} \end{aligned} \tag{A.17}$$

For Ni-Cr-Fe alloys, a threshold value of 0.10% for strain amplitude (one-half the strain range for the cycle) is defined, below which environmental effects on the fatigue life of these alloys do not occur. Thus,

$F_{en,nom} = 1 \qquad (\varepsilon_a \leq 0.10\%).$ (A.18)

A3 Fatigue Evaluation Procedure

The evaluation method uses as its input the partial fatigue usage factors U_1, U_2, U_3, ...U_n, determined in Class 1 fatigue evaluations. To incorporate environmental effects into the Section III fatigue evaluation, the partial fatigue usage factors for a specific stress cycle or load set pair, based on the current Code fatigue design curves, is multiplied by the environmental fatigue correction factor:

$U_{en.1} = U_1 \cdot F_{en,1}.$ (A.19)

In the Class 1 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 1 piping products designed using the 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 partial usage factors are obtained from the Code fatigue design curves provided they are consistent, or conservative, with respect to the existing fatigue ε–N data. For example, the Code fatigue design curve for austenitic SSs developed in the 1960s is not consistent with the existing fatigue database and, therefore, will yield nonconservative estimates of usage factors for most heats of austenitic SSs that are used in the construction of nuclear reactor components. Examples of calculating partial usage factors are as follows:

(1) For carbon and low–alloy steels with ultimate tensile strength ≤552 MPa (≤80 ksi), the partial fatigue usage factors are obtained from the ASME Code fatigue design curve, i.e., Fig. I–9.1 of the mandatory Appendix I to Section III of the ASME Code. As an alternative, to reduce conservatism in the current Code requirement of a factor of 20 on life, partial usage factors may be determined from the fatigue design curves that were developed from the ANL fatigue life model, i.e., Figs. A.1 and A.2 and Table A.1.

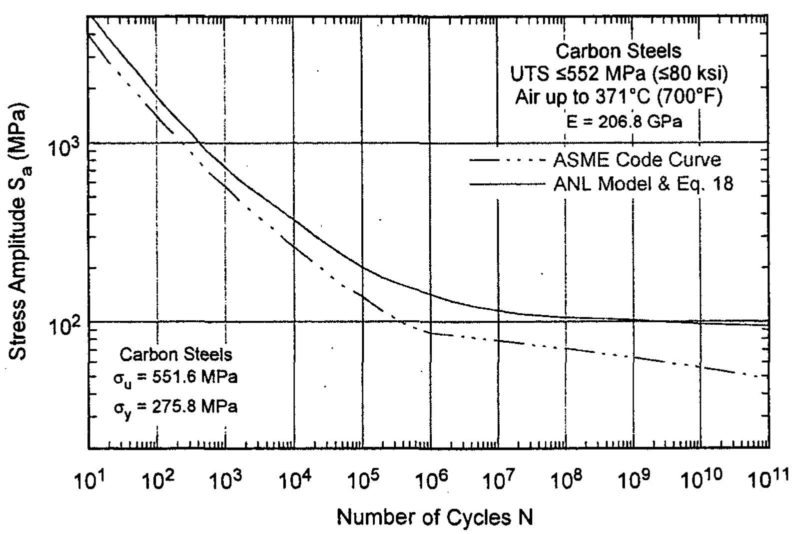


Figure A.1.
Fatigue design curve for carbon steels in air. The curve developed from the ANL model is based on factors of 12 on life and 2 on stress.

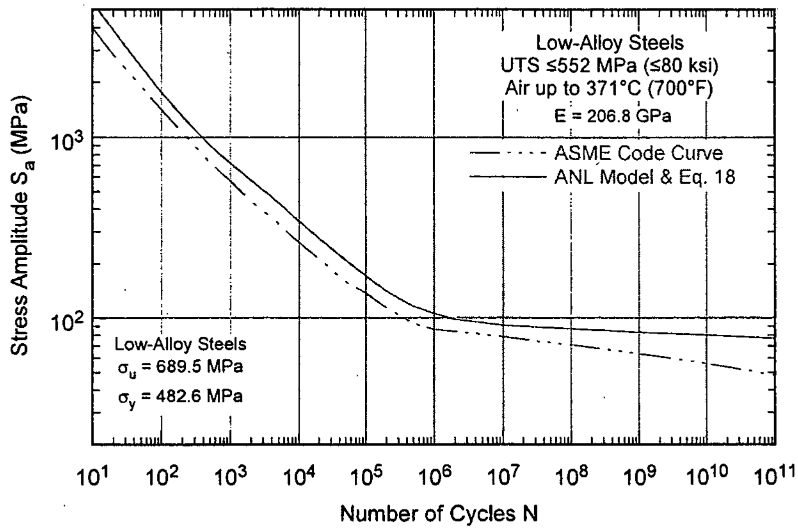


Figure A.2. Fatigue design curve for low-alloy steels in air. The curve developed from the ANL model is based on factors of 12 on life and 2 on stress.

Table A.1. Fatigue design curves for carbon and low-alloy steels and proposed extension to 10^{11} cycles.

	Stress Amplitude (MPa/ksi)				Stress Amplitude (MPa/ksi)		
Cycles	ASME Code Curve	Eqs. 15 & 18 Carbon Steel	Eqs. 16 & 18 Low-Alloy Steel	Cycles	ASME Code Curve	Eqs. 15 & 18 Carbon Steel	Eqs. 16 & 18 Low-Alloy Steel
1 E+01	3999 (580)	5355 (777)	5467 (793)	2 E+05	114 (16.5)	176 (25.5)	141 (20.5)
2 E+01	2827 (410)	3830 (556)	3880 (563)	5 E+05	93 (13.5)	154 (22.3)	116 (16.8)
5 E+01	1896 (275)	2510 (364)	2438 (354)	1 E+06	86 (12.5)	142 (20.6)	106 (15.4)
1 E+02	1413 (205)	1820 (264)	1760 (255)	2 E+06		130 (18.9)	98 (14.2)
2 E+02	1069 (155)	1355 (197)	1300 (189)	5 E+06		120 (17.4)	94 (13.6)
5 E+02	724 (105)	935 (136)	900 (131)	1 E+07	76.5 (11.1)	115 (16.7)	91 (13.2)
1 E+03	572 (83)	733 (106)	720 (104)	2 E+07		110 (16.0)	90 (13.1)
2 E+03	441 (64)	584 (84.7)	576 (83.5)	5 E+07		107 (15.5)	88 (12.8)
5 E+03	331 (48)	451 (65.4)	432 (62.7)	1 E+08	68.3 (9.9)	105 (15.2)	87 (12.6)
1 E+04	262 (38)	373 (54.1)	342 (49.6)	1 E+09	60.7 (8.8)	102 (14.8)	83 (12.0)
2 E+04	214 (31)	305 (44.2)	276 (40.0)	1 E+010	54.5 (7.9)	97 (14.1)	80 (11.6)
5 E+04	159 (23)	238 (34.5)	210 (30.5)	1 E+011	48.3 (7.0)	94 (13.6)	77 (11.2)
1 E+05	138 (20.0)	201 (29.2)	172 (24.9)				

(2) For wrought or cast austenitic SSs and Ni-Cr-Fe alloys, the partial fatigue usage factors are obtained from the new fatigue design curve proposed in the present report for austenitic SSs, i.e., Fig. A.3 and Table A.2.

The cumulative fatigue usage factor, U_{en}, considering the effects of reactor coolant environments is then calculated as the following:

$$U_{en} = U_1 \cdot F_{en,1} + U_2 \cdot F_{en,2} + U_3 \cdot F_{en,3} + U_i \cdot F_{en,i} \ldots + U_n \cdot F_{en,n}, \tag{A.20}$$

where $F_{en,i}$ is the nominal environmental fatigue correction factor for the "i"th stress cycle (NB-3200) or load set pair (NB-3600). Because environmental effects on fatigue life occur primarily during the tensile-loading cycle (i.e., up-ramp with increasing strain or stress), this calculation is performed only for the tensile stress producing portion of the stress cycle constituting a load pair. Also, the values for key parameters such as strain rate, temperature, dissolved oxygen in water, and for carbon and low–alloy steels S content, are needed to calculate F_{en} for each stress cycle or load set pair. As discussed in Sections 4 and 5 of this report, the following guidance may be used to determine these parameters:

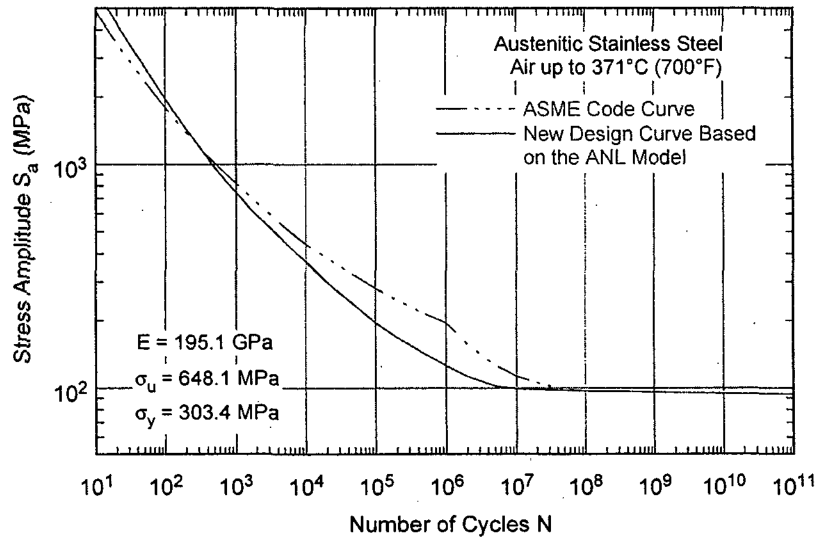


Figure A.3. Fatigue design curve for austenitic stainless steels in air.

Table A.2. The new and current Code fatigue design curves for austenitic stainless steels in air.

	Stress Amplitude (MPa/ksi)			Stress Amplitude (MPa/ksi)	
Cycles	New Design Curve	Current Design Curve	Cycles	New Design Curve	Current Design Curve
1 E+01	6000 (870)	4881 (708)	2 E+05	168 (24.4)	248 (35.9)
2 E+01	4300 (624)	3530 (512)	5 E+05	142 (20.6)	214 (31.0)
5 E+01	2748 (399)	2379 (345)	1 E+06	126 (18.3)	195 (28.3)
1 E+02	1978 (287)	1800 (261)	2 E+06	113 (16.4)	157 (22.8)
2 E+02	1440 (209)	1386 (201)	5 E+06	102 (14.8)	127 (18.4)
5 E+02	974 (141)	1020 (148)	1 E+07	99 (14.4)	113 (16.4)
1 E+03	745 (108)	820 (119)	2 E+07		105 (15.2)
2 E+03	590 (85.6)	669 (97.0)	5 E+07		98.6 (14.3)
5 E+03	450 (65.3)	524 (76.0)	1 E+08	97.1 (14.1)	97.1 (14.1)
1 E+04	368 (53.4)	441 (64.0)	1 E+09	95.8 (13.9)	95.8 (13.9)
2 E+04	300 (43.5)	383 (55.5)	1 E+10	94.4 (13.7)	94.4 (13.7)
5 E+04	235 (34.1)	319 (46.3)	1 E+11	93.7 (13.6)	93.7 (13.6)
1 E+05	196 (28.4)	281 (40.8)	2 E+10		

(1) An average strain rate for the transient always yields a conservative estimate of F_{en}. The lower bound or saturation strain rate of 0.001%/s for carbon and low–alloy steels or 0.0004%/s for austenitic SSs can be used to perform the most conservative evaluation.

(2) For the case of a constant strain rate and a linear temperature response, an average temperature (i.e., average of the maximum and minimum temperatures for the transients) may be used to calculate F_{en}. In general, the "average" temperature that should be used in the calculations should produce results that are consistent with the results that would be obtained using the modified rate approach described in Section 4.2.14 of this report. The maximum temperature can be used to perform the most conservative evaluation.

(3) The DO value is obtained from each transient constituting the stress cycle. For carbon and low-alloy steels, the dissolved oxygen content, DO, associated with a stress cycle is the highest oxygen level in the transient, and for austenitic stainless steels, it is the lowest oxygen level in the transient. A value of 0.4 ppm for carbon and low-alloy steels and 0.05 ppm for austenitic stainless steels can be used for the DO content to perform a conservative evaluation.

(4) 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 the maximum value specified in the procurement specification or the applicable construction Code.

The detailed procedures for incorporating environmental effects into the Code fatigue evaluations have been presented in several articles. The following two may be used for guidance:

(1) Mehta, H. S., "An Update on the Consideration of Reactor Water Effects in Code Fatigue Initiation Evaluations for Pressure Vessels and Piping," Assessment Methodologies for Preventing Failure: Service Experience and Environmental Considerations, PVP Vol. 410-2, R. Mohan, ed., American Society of Mechanical Engineers, New York, pp. 45–51, 2000.

(2) Nakamura, T., M. Higuchi, T. Kusunoki, and Y. Sugie, "JSME Codes on Environmental Fatigue Evaluation," Proc. of the 2006 ASME Pressure Vessels and Piping Conf., July 23–27, 2006, Vancouver, BC, Canada, paper # PVP2006–ICPVT11–93305.

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 5

Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, N.Y. 10511-0249
Tel (914) 788-2055

Fred Dacimo
Vice President
License Renewal

NL-10-082

August 9, 2010

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: License Renewal Application – Completion of Commitment #33
Regarding the Fatigue Monitoring Program
Indian Point Nuclear Generating Unit Nos. 2 and 3
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

REFERENCE

1. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application" (NL-07-039)
2. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Boundary Drawings (NL-07-040)
3. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Environmental Report References (NL-07-041)
4. Entergy Letter dated October 11, 2007, F. R, Dacimo to Document Control Desk, "License Renewal Application (LRA)" (NL-07-124)
5. Entergy Letter November 14, 2007, F. R, Dacimo to Document Control Desk, "Supplement to License Renewal Application (LRA) Environmental Report References" (NL-07-133)

Dear Sir or Madam:

In the referenced letters, Entergy Nuclear Operations, Inc. applied for renewal of the Indian Point Energy Center operating license. This letter contains information supporting the completion of commitment 33 to the License Renewal Application regarding the Fatigue Monitoring Program.

There are no new commitments identified in this submittal. If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-734-6710.

Sincerely,

FD/mb

Attachments:
1. Environmental Fatigue Evaluations
2. List of Regulatory Commitments

cc: Mr. S. J. Collins, Regional Administrator, NRC Region I
Mr. J. Boska, Senior Project Manager, NRC, NRR, DORL
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
Ms. Kimberly Green, NRC Safety Project Manager
NRC Resident Inspectors Office, Indian Point
Mr. Paul Eddy, NYS Dept. of Public Service
Mr. Francis J. Murray, Jr., President and CEO, NYSERDA

ATTACHMENT 1 TO NL-10-082

Environmental Fatigue Evaluations

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNITS 2 AND 3
DOCKET NOS. 50-247 & 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION
ENVIRONMENTAL FATIGUE EVALUATIONS

Environmental Fatigue Evaluation for Indian Point Unit 2 and Unit 3

Entergy has applied for renewed operating licenses for Indian Point Nuclear Generating Unit 2 and Unit 3 (IP2 and IP3). In the license renewal application, Entergy committed to address environmentally assisted fatigue. Entergy's commitment, amended by letter, NL-08-021, dated January 22, 2008, reads as follows.

> At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), under the Fatigue Monitoring Program, IP2 and IP3 will implement one or more of the following:
>
> (1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following:
>
> > I. For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), with existing fatigue analysis valid for the period of extended operation, use the existing CUF.
> >
> > 2. Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.
> >
> > 3. Representative CUF values from other plants, adjusted to or enveloping the IPEC plant specific external loads may be used if demonstrated applicable to IPEC.
> >
> > 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.
>
> (2) Consistent with the Fatigue Monitoring Program, Corrective Actions, repair or replace the affected locations before exceeding a CUF of 1.0.

Entergy has updated the fatigue usage calculations using refined fatigue analyses to determine CUFs when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs for the locations shown in LRA Table 4.3-13 and LRA Table 4.3-14.
The tables from the LRA are repeated as follows with the updated CUF values inserted.

The results of the refined analyses are shown in the tables as <u>underlined values.</u>

Table 4.3-13
IP2 Cumulative Usage Factors for NUREG/CR-6260 Limiting Locations

NUREG-6260 Generic Location		IP2 Plant-Specific Location	Material Type	CUF of Record	Per NUREG/CR-6583 or NUREG/CR-5704	
					F_{en}	Environmentally Adjusted CUF
1	Vessel shell and lower head	Bottom head to shell	LAS	0.004	2.45	0.01
2	Vessel inlet and outlet nozzles	Reactor vessel inlet nozzle	LAS	0.05	2.45	0.12
2	Vessel inlet and outlet nozzles	Reactor vessel outlet nozzle	LAS	0.281	2.45	0.69
3	Pressurizer surge line nozzles	Pressurizer surge line nozzle	LAS	~~0.264~~ 0.109	~~2.45~~ 1.74	~~0.646~~ 0.188
3	Pressurizer surge line piping	Surge line piping to safe end weld	SS	~~0.6~~ 0.062	~~15.35~~ 13.26	~~9.21~~ 0.822
4	RCS piping charging system nozzle	Charging system nozzle	SS	~~0.99~~ 0.0323	~~15.35~~ 8.7	~~15.20~~ 0.2809
5	RCS piping safety injection nozzle	NA	SS	~~NA[2]~~ 0.1083	~~15.35~~ 7.8975	~~NA[2]~~ 0.8553
6	RHR Class 1 piping	NA	SS	~~NA[2]~~ 0.0721	~~15.35~~ 13.08	~~NA[2]~~ 0.9434

Table 4.3-14
IP3 Cumulative Usage Factors for NUREG/CR-6260 Limiting Locations

	NUREG-6260 Location	IP3 Plant-Specific Location	Material Type	CUF of Record	Per NUREG/CR-6583 or NUREG/CR-5704	
					F_{en}	Environmentally Adjusted CUF
1	Vessel shell and lower head	Bottom head to shell	LAS	0.02	2.45	0.05
2	Vessel inlet and outlet nozzles	Reactor vessel inlet nozzle	LAS	0.049	2.45	0.12
2	Vessel inlet and outlet nozzles	Reactor vessel outlet nozzle	LAS	0.259	2.45	0.64
3	Pressurizer surge line nozzles	Pressurizer surge line nozzle	LAS	~~0.9612~~ 0.0903	~~2.45~~ 1.74	~~2.35~~ 0.157
3	Pressurizer surge line piping	Surge line piping to safe end weld	SS	~~0.6~~ 0.0411	~~15.35~~ 14.45	~~9.21~~ 0.594
4	RCS piping charging system nozzle	NA	SS	~~NA[2]~~ 0.1812	~~15.35~~ 3.98	~~NA[2]~~ 0.722
5	RCS piping safety injection nozzle	NA	SS	~~NA[2]~~ 0.1709	~~15.35~~ 5.0117	~~NA[2]~~ 0.8565
6	RHR Class 1 piping	NA	SS	~~NA[2]~~ 0.1279	~~15.35~~ 7.79	~~NA[2]~~ 0.9961

As described in LRA Section 4.3.3 and shown in Tables 4.3-13 and 4.3-14, the CUFs for the reactor vessel locations were not changed. The evaluation documented in the LRA used bounding Fens applied to the CUFs of record for the bottom head to shell region, the reactor vessel inlet nozzle and the reactor vessel outlet nozzle. The tables show the Cumulative Usage Factors are all below 1.0 for these three locations for both units.

The stress and fatigue evaluations for the remaining piping components listed in Table 4.3-13 and 4.3-14 were performed using standard methods of the ASME Code, Section III. Detailed stress models of the surge line hot leg nozzle, pressurizer surge nozzle, reactor coolant piping charging system nozzle, reactor coolant piping safety injection nozzle, and the RHR system class 1 piping locations were prepared. The analyst developed detailed stress history inputs for all transients considered in the evaluation, which were subsequently used to provide detailed inputs used in the EAF evaluations. The ASME Code evaluations were performed for the piping components to reduce conservatism in the analyses or because the current licensing basis (CLB) qualification was to the ANSI B31.1 Power Piping Code, which did not require a fatigue usage factor calculation. The evaluations were limited to the stress qualifications related to the fatigue requirements of the ASME Code. The primary stress qualifications documented in the CLB remain applicable for the components evaluated.

The evaluation of EAF was accomplished through the application of Fen factors, as described in NUREG/CR-5704 for the stainless steels in the pressurizer surge line, the reactor coolant piping charging and safety injection system nozzles, and the RHR system Class 1 piping. In addition, Fen factors for the pressurizer surge nozzle dissimilar metal weld also considered the carbon steel factors in NUREG/CR-6583. The Fen factors are calculated based on detailed inputs described in the applicable NUREG and are directly applied to the ASME Code fatigue results.

The refined fatigue analyses of the Indian Point Unit 2 and Unit 3 piping locations corresponding to the locations identified in NUREG/CR-6260 for older vintage Westinghouse plants demonstrate that cumulative fatigue usage factors including consideration of reactor water environmental effects are below a value of 1.0 for transients postulated for 60 years of operation.

The results of this evaluation resolve commitment number 33 in the NRC Safety Evaluation Report on license renewal for Indian Point Nuclear Generating Unit 2 and Unit 3.

ATTACHMENT 2 TO NL-10-082

List of Regulatory Commitments

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNITS 2 AND 3
DOCKET NOS. 50-247 & 50-286

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
5	Enhance the External Surfaces Monitoring Program for IP2 and IP3 to include periodic inspections of systems in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(1) and (a)(3). Inspections shall include areas surrounding the subject systems to identify hazards to those systems. Inspections of nearby systems that could impact the subject systems will include SSCs that are in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(2).	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.10 A.3.1.10 B.1.11
6	Enhance the Fatigue Monitoring Program for IP2 to monitor steady state cycles and feedwater cycles or perform an evaluation to determine monitoring is not required. Review the number of allowed events and resolve discrepancies between reference documents and monitoring procedures. Enhance the Fatigue Monitoring Program for IP3 to include all the transients identified. Assure all fatigue analysis transients are included with the lowest limiting numbers. Update the number of design transients accumulated to date.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.11 A.3.1.11 B.1.12, Audit Item 164
7	Enhance the Fire Protection Program to inspect external surfaces of the IP3 RCP oil collection systems for loss of material each refueling cycle. Enhance the Fire Protection Program to explicitly state that the IP2 and IP3 diesel fire pump engine sub-systems (including the fuel supply line) shall be observed while the pump is running. Acceptance criteria will be revised to verify that the diesel engine does not exhibit signs of degradation while running; such as fuel oil, lube oil, coolant, or exhaust gas leakage. Enhance the Fire Protection Program to specify that the IP2 and IP3 diesel fire pump engine carbon steel exhaust components are inspected for evidence of corrosion and cracking at least once each operating cycle. Enhance the Fire Protection Program for IP3 to visually inspect the cable spreading room, 480V switchgear room, and EDG room CO_2 fire suppression system for signs of degradation, such as corrosion and mechanical damage at least once every six months.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.12 A.3.1.12 B.1.13

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
33	At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), under the Fatigue Monitoring Program, IP2 and IP3 will implement one or more of the following: (1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following: 1. For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), with existing fatigue analysis valid for the period of extended operation, use the existing CUF. 2. Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component. 3. Representative CUF values from other plants, adjusted to or enveloping the IPEC plant specific external loads may be used if demonstrated applicable to IPEC. 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF. (2) Consistent with the Fatigue Monitoring Program, Corrective Actions, repair or replace the affected locations before exceeding a CUF of 1.0.	IP2: September 28, 2011 IP3: December 12, 2013 Complete	NL-07-039 NL-07-153 NL-08-021 NL-10-082	A.2.2.2.3 A.3.2.2.3 4.3.3 Audit item 146
34	IP2 SBO / Appendix R diesel generator will be installed and operational by April 30, 2008. This committed change to the facility meets the requirements of 10 CFR 50.59(c)(1) and, therefore, a license amendment pursuant to 10 CFR 50.90 is not required.	April 30, 2008 Complete	NL-07-078 NL-08-074	2.1.1.3.5

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 6

ELECTRIC POWER RESEARCH INSTITUTE

Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47, Revision 1)

Technical Report

Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47 Revision 1)

1012017

Final Report, September 2005

EPRI Project Manager
J. Carey

ELECTRIC POWER RESEARCH INSTITUTE
3420 Hillview Avenue, Palo Alto, California 94304-1395 • PO Box 10412, Palo Alto, California 94303-0813 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

THIS DOCUMENT WAS PREPARED BY THE ORGANIZATION(S) NAMED BELOW AS AN ACCOUNT OF WORK SPONSORED OR COSPONSORED BY THE ELECTRIC POWER RESEARCH INSTITUTE, INC. (EPRI). NEITHER EPRI, ANY MEMBER OF EPRI, ANY COSPONSOR, THE ORGANIZATION(S) BELOW, NOR ANY PERSON ACTING ON BEHALF OF ANY OF THEM:

(A) MAKES ANY WARRANTY OR REPRESENTATION WHATSOEVER, EXPRESS OR IMPLIED, (I) WITH RESPECT TO THE USE OF ANY INFORMATION, APPARATUS, METHOD, PROCESS, OR SIMILAR ITEM DISCLOSED IN THIS DOCUMENT, INCLUDING MERCHANTABILITY AND FITNESS FOR A PARTICULAR PURPOSE, OR (II) THAT SUCH USE DOES NOT INFRINGE ON OR INTERFERE WITH PRIVATELY OWNED RIGHTS, INCLUDING ANY PARTY'S INTELLECTUAL PROPERTY, OR (III) THAT THIS DOCUMENT IS SUITABLE TO ANY PARTICULAR USER'S CIRCUMSTANCE; OR

(B) ASSUMES RESPONSIBILITY FOR ANY DAMAGES OR OTHER LIABILITY WHATSOEVER (INCLUDING ANY CONSEQUENTIAL DAMAGES, EVEN IF EPRI OR ANY EPRI REPRESENTATIVE HAS BEEN ADVISED OF THE POSSIBILITY OF SUCH DAMAGES) RESULTING FROM YOUR SELECTION OR USE OF THIS DOCUMENT OR ANY INFORMATION, APPARATUS, METHOD, PROCESS, OR SIMILAR ITEM DISCLOSED IN THIS DOCUMENT.

ORGANIZATION(S) THAT PREPARED THIS DOCUMENT

Structural Integrity Associates, Inc.

ORDERING INFORMATION

Requests for copies of this report should be directed to EPRI Orders and Conferences, 1355 Willow Way, Suite 278, Concord, CA 94520, (800) 313-3774, press 2 or internally x5379, (925) 609-9169, (925) 609-1310 (fax).

Electric Power Research Institute and EPRI are registered service marks of the Electric Power Research Institute, Inc.

CITATIONS

This report was prepared by

Structural Integrity Associates, Inc.
6855 South Havana Street, Suite 350
Centennial, CO 80112

Authors:
Gary L. Stevens, P.E.

and:

J Michael Davis, P.E.
Duke Energy

This report describes research sponsored by the Electric Power Research Institute (EPRI).

The report is a corporate document that should be cited in the literature in the following manner:

Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47 Revision 1). EPRI, Palo Alto, CA: 2005. 1012017.

REPORT SUMMARY

For about the last 15 years, the effects of light water reactor environment on fatigue have been the subject of research in both the United States and abroad. Based on a risk study reported in NUREG/CR-6674, the NRC concluded that reactor water environmental effects were not a safety issue for a 60-year operating life, but that some limited assessment of its effect would be required for a license renewal extended operating period beyond 40 years. This guideline offers methods for addressing environmental fatigue in a license renewal submittal.

Background
Many utilities are currently embarking upon efforts to renew their operating licenses. One of the key areas of uncertainty in this process 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, manage this effect during the license renewal period. As the license renewal application process progressed starting in 1998, several utilities addressed this request using different approaches. In more recent years, a unified approach has emerged that has obtained regulator approval and allowed utilities to satisfactorily address this issue and obtain a renewed operating license for 60 years of plant operation.

Objectives

- To provide guidance for assessment and management of reactor coolant environmental effects
- To minimize the amount of plant-specific work necessary to comply with NRC requirements for addressing this issue in a license renewal application
- To provide "details of execution" for applying the environmental fatigue approach currently accepted by the NRC in the license renewal application process.

Approach
The project team reviewed previous work by EPRI and utilities related to fatigue environmental effects and license renewal including reports on this subject created by EPRI, NRC, and NRC contractors. Recent license renewal applications, NRC Requests for Additional Information, and the commitments made by the past license renewal applicants provided insight into NRC expectations. After evaluation of all this information, the project team developed alternatives for addressing fatigue environmental effects. This revision provides guidelines based on industry experience, consensus, and insight gained from more than six years of experience with this issue and the license renewal approval process.

Results

The report describes a fatigue environmental effect license renewal approach that can be applied by any license renewal applicant. It provides guidelines for performing environmental fatigue assessments using fatigue environmental factors from currently accepted F_{en} methodology.

EPRI Perspective

Utilities have committed significant resources to license renewal activities related to fatigue. Based on input from applicants to-date, NRC requirements for addressing fatigue environmental effects continued to change for the first few applicants, but more recently have become more unified. These guidelines were developed to provide stability, refined guidance, and assurance of NRC acceptance and include an approach that may be taken to address fatigue environmental effects in a license renewal application. Use of the approach 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 reactor water environmental effects on fatigue.

Keywords

Fatigue
License Renewal
Reactor Water Environmental Fatigue Effects

ABSTRACT

For about the last 15 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 are that the reactor water temperature and chemical composition (particularly oxygen content or ECP) can have a significant effect on the fatigue life of carbon, low alloy, and austenitic stainless steels. The degree of fatigue life reduction is a function of the tensile strain rate during a transient, the specific material, the temperature, and the water chemistry. The effects of other than moderate environment were not considered in the original development of the ASME Code Section III fatigue curves.

This issue has been studied by the Nuclear Regulatory Commission (NRC) for many 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 U.S. vendors. The results of that 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 reactor water environment could cause fatigue usage factors to exceed the ASME Code-required fatigue usage limit of 1.0. On the other hand, it was demonstrated that usage factors at many locations could be shown acceptable by refined analysis and/or fatigue monitoring of actual plant transients.

Based on a risk study reported in NUREG/CR-6674, the NRC concluded that reactor water environmental effects were not a safety issue for a 60-year operating life, but that some limited assessment of its effect would be required for a license renewal extended operating period beyond 40 years. Thus, for all license renewal submittals to-date, there have been formal questions raised on the topic of environmental fatigue and, in all cases, utility commitments to address the environmental effects on fatigue in the extended operating period. Many plants have already performed these commitments.

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. NUREG/CR-6260 locations are considered an appropriate sample for F_{en} evaluation as long as none exceed the acceptance criteria with environmental effects considered. If this occurs, the sampling is to be extended to other locations. For these locations, evaluations similar to those conducted in NUREG/CR-6260 are required. In the extended operating period, fatigue monitoring is used for the sample of locations to show that ASME Code limits are not exceeded. If these limits are exceeded, corrective actions are identified for demonstrating acceptability for continued operation.

Using the guidance provided herein, the amount of effort needed to justify individual license renewal submittals and respond to NRC questions should be minimized, and a more unified, consistent approach should be achieved throughout the industry. More importantly, this revision provides "details of execution" for applying the environmental fatigue approach currently accepted by the NRC in the license renewal application process.

9. ADMINISTRATIVE CONTROLS

   The Thermal Fatigue Licensing Basis Monitoring Guideline actions are implemented by plant work processes.

10. OPERATING EXPERIENCE

   Refer to Sections 1.1 and 2.5.2.3 of Reference [23] for a discussion of how operating experience becomes part of the Thermal Fatigue Licensing Basis Monitoring Guideline implementation.

3.2 Method for Evaluation of Environmental Effects

There are several methods that have been published to assess the effects of reactor water environment on fatigue for each specific location to be considered. In this document, guidance is provided for performing evaluations in accordance with NUREG/CR-6583 [3] for carbon and low alloy steels and NUREG/CR-5704 [4] for austenitic stainless steels, since these are the currently accepted methodologies for evaluating environmental fatigue effects. Other methods that have been published, including those currently being used in Japan, are documented in References [18] and [22].

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 perform an assessment of the effects of environmental fatigue on the locations identified in Step 1. This includes an assessment of the actual expected fatigue usage factor including the influence of environmental effects. 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 significantly envelope actual operating conditions in the plant. Further discussion is provided in Section 3.2.2. Specific guidance on performing such evaluation is provided in Section 4.0.
- The bottom of Figure 3-1 indicates that fatigue management occurs after the evaluation from Step 2 is performed for each location. This may be as simple as counting the accumulated cycles and showing that they remain less than or equal to the number of cycles utilized in the assessment performed in Step 2. 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. Refer also to Reference [23] for a discussion of cycle counting.

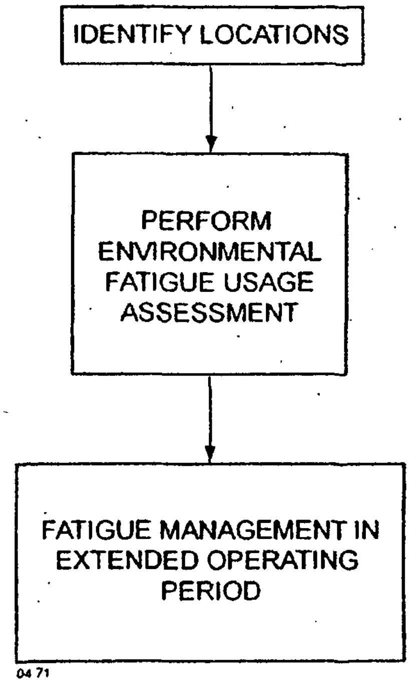


Figure 3-1
Overview of Fatigue Environmental Effects Assessment and Management

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 and evaluating locations.

For both PWR and BWR plants, the locations chosen in NUREG/CR-6260 [2] 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.

The locations evaluated in NUREG/CR-6260 [2] for the appropriate vendor/vintage plant should be evaluated on a plant-unique basis. For cases where acceptable fatigue results are demonstrated for these locations for 60 years of plant operation including environmental effects, additional evaluations or locations need not be considered. However, plant-unique evaluations may show that some of the NUREG/CR-6260 [2] locations do not remain within allowable limits for 60 years of plant operation when environmental effects are considered. In this situation, plant specific evaluations should expand the sampling of locations accordingly to include other locations where high usage factors might be a concern.

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

If necessary, in identifying the set of locations for the expanded 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 typical environmental fatigue multipliers, should be chosen as the bounding location to use in the environmental fatigue assessment. Similar to the approach taken in NUREG/CR-6260 [2], the final set of locations chosen for expanded environmental assessment should include several different types of locations that are expected to have the highest CUFs and should be those most adversely affected by environmental effects. 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 the locations evaluated in NUREG/CR-6260 [2] for the appropriate vintage/vendor plant.
- Perform a plant-unique environmental fatigue assessment for the NUREG/CR-6260 locations.
- If the CUF results for all locations above are less than or equal to the allowable (typically 1.0) for the 60-year operating life, the environmental assessment may be considered complete; additional evaluations or locations need not be considered.
- If the CUF results for any locations above are greater than the allowable for the 60-year operating life, expand the locations evaluated, considering the following:
  - Identify all Class 1 piping systems and major components. For the reactor pressure 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.
  - From the list of locations that results from the above steps, choose a set of 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, or (2) locations that have similar loading conditions, geometry, material, and reactor water environment compared to another selected location.

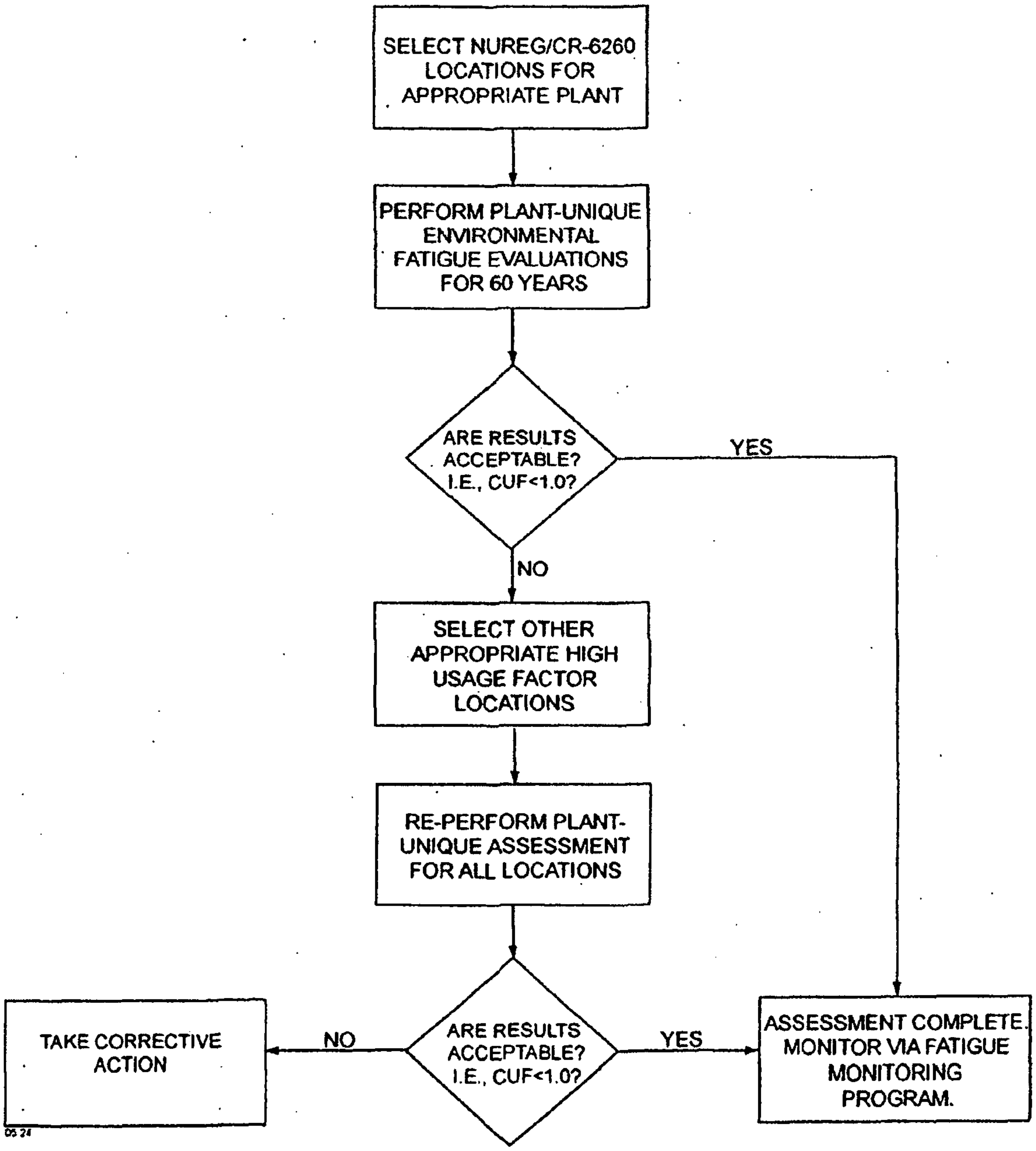


Figure 3-2
Identification of Component Locations and Fatigue Environmental Effects Assessment

3.2.2 Fatigue Assessment Using Environmental Factors

In performing an assessment of environmental fatigue effects, factors to account for environmental effects are incorporated into an updated fatigue evaluation for each selected location using the F_{en} approach documented in NUREG/CR-6583 [3] for carbon and low alloy steels and NUREG/CR-5704 [4] for austenitic stainless steels. Excess conservatism in the loading definitions, number of cycles, and the fatigue analyses may be considered. Figure 3-3 shows the approach for performing the assessment and managing fatigue in the extended operating period.

<u>Determination of Existing Licensing Basis</u>

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. Review of the analysis may or may not show that excess conservatism exists. Reference [23] provides guidance on reviewing the original design basis, the operating basis, and additions imposed by the regulatory oversight process, to determine the fatigue licensing basis events for which the component is required to be evaluated.

<u>Consideration of Increased Cycles for Extended Period</u>

As a part of the license renewal application process, the applicant must update the projected cycles to account for 60 years of plant operation. The first possible outcome is 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. In this case, the governing fatigue analyses will not require modification to account for the extended period of operation.

The second possibility is that more cycles are projected to occur for 60 years of plant operation than were postulated for the first 40 years. In this case, an applicant must address the increased cycle counts. One possible solution is to perform a revised fatigue analysis to confirm that the increased number of cycles will still result in a CUF less than or equal to the allowable. A second possibility is to determine the number of cycles at which the CUF would be expected to reach the allowable. This cycle quantity then becomes the allowable against which the actual operation is tracked. Section 3.3 discusses options to be employed if this lower allowable is projected to be exceeded.

<u>Fatigue Assessment</u>

Fatigue assessment includes the determination of CUF considering environmental effects. This may be accomplished conservatively using information from design documentation and bounding F_{en} factors from NUREG/CR-6583 [3] and NUREG/CR-5704 [4], or it may require a more extensive approach (as discussed in Section 4.0).

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, transient grouping and/or use of an early edition of the ASME Code.

- For piping, use of an ASME Code Edition prior to 1979 Summer Addenda, which included the ΔT_1 term in Equation (10) of NB-3650. Use of a later code reduces the need to apply conservative elastic-plastic penalty factors.
- Re-analysis may be needed to determine strain rate time histories possibly not reported in existing component analyses, such that bounding environmental multipliers (i.e., very low or "saturated" strain rates) would not have to be used.

A simplified revised fatigue analysis may be performed using results from the existing fatigue analysis, if sufficient detail is available. Alternatively, a new complete analysis could be conducted to remove additional conservatisms. Such an evaluation would not necessarily need the full pedigree of a certified ASME Code Section III analysis (i.e.. Certified Design Specification, etc.), but it should utilize all of the characteristic methods from Section III for computing CUF. In the environmental fatigue assessment, the environmental fatigue usage may be calculated using 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 [3] or NUREG/CR-5704 [4]. (Section 4.0 provides specific guidance on performing an F_{en} evaluation)
- 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.

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 accumulated cycles.

- For cycle counting, an updated allowable number of cycles may be needed if the fatigue assessment determined the CUF to be larger than allowable. One approach is to derive a reduced number of cycles that would limit the CUF to less than or equal to the allowable value (typically 1.0). On the other hand, if the assessed CUF was shown to be less than or equal to the allowable, the allowable number of cycles may remain as assumed in the evaluation, or increased appropriately. As long as the number of cycles in the extended operating period remains within this allowed number of cycles, no further action is required.
- For CUF tracking, one approach would be to utilize fatigue monitoring that accounts for the actual cyclic operating conditions for each 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 or applied to the CUF results of such monitoring. No further action is required as long as the computed usage factor remains less than or equal to the allowable value.

Prior to such time that the CUF is projected to exceed the allowable value, or the number of actual cycles is projected to exceed the allowable number of 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. Further details on guidelines for thermal fatigue monitoring and compliance/mitigation options are provided in Reference [23].

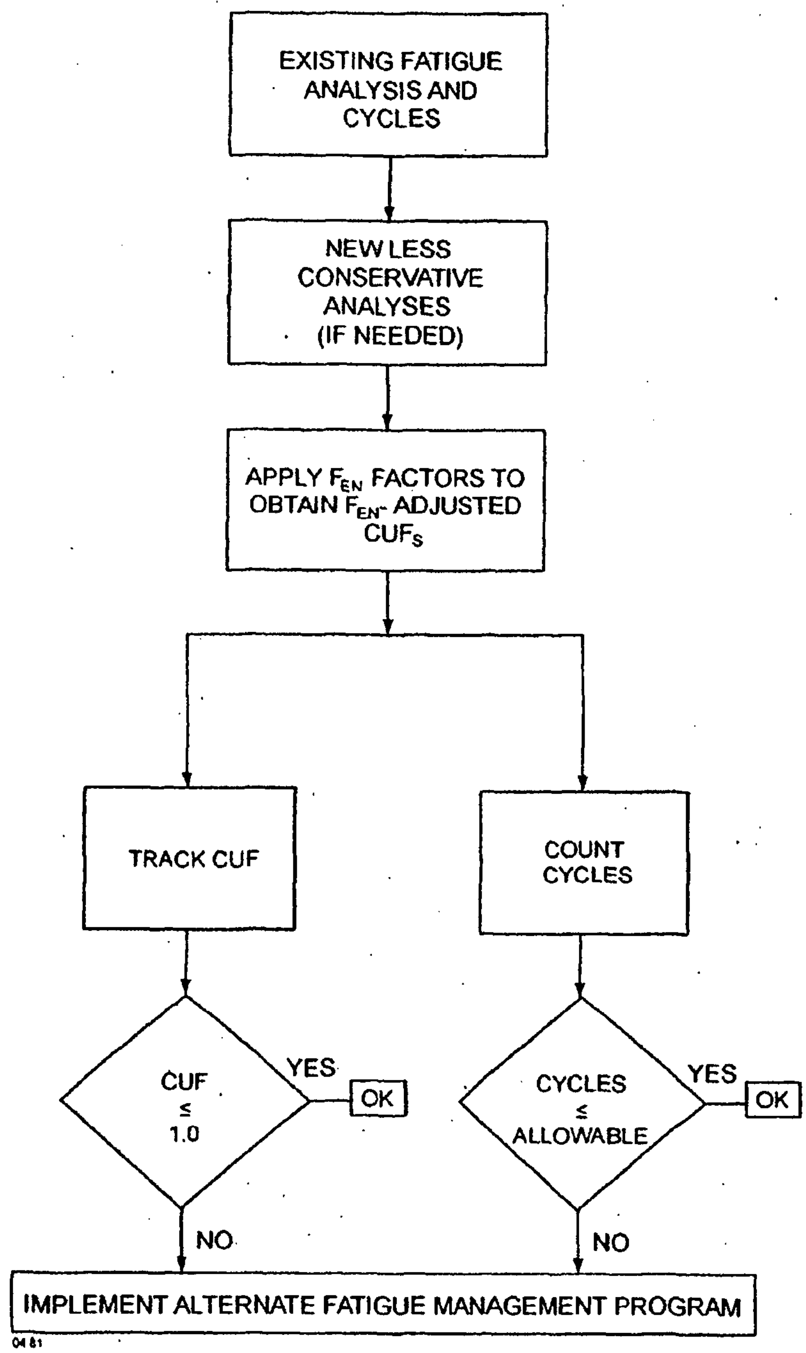


Figure 3-3
Fatigue Management if Environmental Assessment Conducted

A separate section follows for each parameter utilized in the F_{en} expressions, that is transformed sulfur content (S^*), transformed temperature (T^*), transformed dissolved oxygen (O^*), and transformed strain rate ($\dot{\varepsilon}^*$). For the transformed strain rate, temperature, and oxygen parameters, the three approaches are discussed. Transformed sulfur does not vary over the three approaches. A single approach should be utilized for all of the transformed parameters in a single load-pair F_{en} determination, although different approaches may be utilized for different load-pair F_{en}s.

First, the typical content of a fatigue calculation is presented.

4.2.1 Contents of a Typical Fatigue Evaluation

This section provides the content of a typical fatigue calculation. Whereas fatigue calculations have varied over the years, their basic content is the same. With the advent of computer technology, the calculations have basically maintained the same content, but computations have become more refined and exhaustive. For example, 30 years ago it was computationally difficult for a stress analyst to evaluate 100 different transients in a fatigue calculation. Therefore, the analyst would have grouped the transients into as few as one transient grouping and performed as few incremental fatigue calculations as possible. With today's computer technology and desire to show more margin, it is relatively easy for the modern-day analyst to evaluate all 100 incremental fatigue calculations for this same problem. Also, older technology would have likely utilized conservative shell interaction hand solutions for computing stress, whereas today finite element techniques are commonly deployed. This improvement in technology would not have changed the basic inputs to the fatigue calculation (i.e., stress), but it would have typically yielded significantly more representative input values.

The discussion here is limited to the general content of most typical fatigue calculations. Discussions of removing excess conservatisms from the input (stress) values of these calculations are not included, as it is assumed that those techniques are generally well understood by engineers performing these assessments throughout the industry.

Two typical fatigue calculations are shown in Figures 4-1 through 4-4. Figure 4-1 reflects an "old" calculation, i.e., one that is typical from a stress report from a plant designed in the 1960s. Figures 4-2 through 4-4 reflect a "new" calculation, i.e., one that is typical from a 1990s vintage stress report. A description of the content of these two calculations is provided below.

The same basic content is readily apparent in both CUF calculations shown in Figures 4-1 through 4-4. However, it is also apparent that much more detail is present in Figures 4-2 through 4-4 for the "new" calculation compared to Figure 4-1 for the "old" calculation. Therefore, with respect to applying F_{en} methodology to a CUF calculation, the guidance provided in the following sections equally applies to both vintages of calculations. The main difference is in assumptions that need to be made for the F_{en} transformed variables due to a lack of detail backing up the calculations in the stress report. Guidance for these assumptions is described in Sections 4.2.2 through 4.2.5, with appropriate reference to the calculations shown in Figures 4-1 through 4-4.

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 7

Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249
Tel (914) 788-2055

Fred Dacimo
Vice President
License Renewal

January 22, 2008

Re: Indian Point Units 2 & 3
Docket Nos. 50-247 & 50-286
NL-08-021

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Entergy Nuclear Operations Inc.
Indian Point Nuclear Generating Unit Nos. 2 & 3
Docket Nos. 50-247 and 50-286
License Renewal Application Amendment 2

REFERENCES:

1. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application" (NL-07-039)
2. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Boundary Drawings" (NL-07-040)
3. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Environmental Report References" (NL-07-041)

Dear Sir or Madam:

In the referenced letters, Entergy Nuclear Operations, Inc. (Entergy) applied for renewal of the Indian Point Energy Center operating licenses for Unit 2 and 3.

Based on discussions during license renewal audits, clarification to the LRA is provided in Attachment 1. This information clarifies the relationship between Commitment 33 regarding environmentally assisted fatigue and the Fatigue Monitoring Program described in LRA Section B.1.12. The Fatigue Monitoring Program includes the actions identified in Commitment 33 to address the evaluation of the effects of environmentally assisted fatigue in accordance with 10 CFR 54.21(c)(1)(iii). The revised Regulatory Commitment List is provided in Attachment 2.

If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 1-22-08.

Sincerely,

Patric W Conroy for
Fred R. Dacimo per telecon
Vice President
License Renewal

Attachments:

1. Fatigue Monitoring Program Clarification
2. Regulatory Commitment List, Revision 3

cc: Mr. Samuel J. Collins, Regional Administrator, NRC Region I
Mr. Kenneth Chang, NRC Branch Chief, Engineering Review Branch I
Mr. Bo M. Pham, NRC Environmental Project Manager
Mr. John Boska, NRR Senior Project Manager
Mr. Paul Eddy, New York State Department of Public Service
NRC Resident Inspector's Office
Mr. Paul D. Tonko, President, New York State Energy, Research, & Development Authority

ATTACHMENT 1 TO NL-08-021

Fatigue Monitoring Program Clarification

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3
DOCKET NOS. 50-247 AND 50-286

License Renewal Application
Amendment 2

Fatigue Monitoring Program Clarification

LRA and commitment list revisions are provided below. (underline - added, strikethrough – deleted)

LRA Table 4.1-1, List of IP2 TLAA and Resolution, line item titled "Effects of reactor water environment on fatigue life", is revised as follows.

Effects of reactor water environment on fatigue life	~~Analyses remain valid~~ ~~10 CFR 54.21(c)(1)(i)~~ ~~OR~~ Aging effect managed 10 CFR 54.21(c)(1)(iii)	4.3.3

LRA Table 4.1-2, List of IP3 TLAA and Resolution, line item titled "Effects of reactor water environment on fatigue life", is revised as follows.

Effects of reactor water environment on fatigue life	~~Analyses remain valid~~ ~~10 CFR 54.21(c)(1)(i)~~ ~~OR~~ Aging effect managed 10 CFR 54.21(c)(1)(iii)	4.3.3

LRA Section 4.3.3, paragraph 10 is revised as follows.

At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3)~~NUREG/CR-6260 for Westinghouse PWRs of the IPEC vintage~~, under the Fatigue Monitoring Program IPEC will implement one or more of the following.

(1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using ~~R~~refined ~~the~~ fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined in accordance with one of the following.

For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3)~~including NUREG/CR-6260 locations~~, with existing fatigue analysis valid for the period of extended operation, use the existing CUF ~~to determine the environmentally adjusted CUF~~.

~~More limiting IPEC~~ Additional plant-specific locations with a valid CUF may be ~~added in addition to the NUREG/CR-6260 locations~~evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.

Representative CUF values from other plants, adjusted to or enveloping the IPEC plant-specific external loads may be used if demonstrated applicable to IPEC.

An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

~~(2) Manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC).~~

(~~3~~2) Consistent with the Fatigue Monitoring Program, Corrective Actions, ~~R~~repair or replace the affected locations before exceeding a CUF of 1.0.

~~Should IPEC select the option to manage the aging effects due to environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.~~

~~Depending on the option chosen, which may vary by component, this TLAA will be projected through the period of extended operation per 10CFR54.21(c)(1)(ii) or t~~The effects of environmentally assisted fatigue will be managed per 10CFR54.21(c)(1)(iii).

LRA Section A.2.1.11, Fatigue Monitoring Program, is revised as follows.

The Fatigue Monitoring Program is an existing program that tracks the number of critical thermal and pressure transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly analyzed a specified number of fatigue transients by assuring that the actual effective number of transients does not exceed the analyzed number of transients. The program provides for update of the fatigue usage calculations to maintain a CUF of < 1.0 for the period of extended operation. For the locations identified in Section A.2.2.2.3, updated calculations will account for the effects of the reactor water environment. These calculation updates are governed by Entergy's 10 CFR 50 Appendix B Quality Assurance (QA) program and include design input verification and independent reviews ensuring that valid assumptions, transients, cycles, external loadings, analysis methods, and environmental fatigue life correction factors will be used in the fatigue analyses. The program requires corrective actions including repair or replacement of affected components before fatigue usage calculations determine the CUF exceeds 1.0. Specific corrective actions are implemented in accordance with the IPEC corrective action program. Repair or replacement of the affected component(s), if necessary, will be in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by Entergy's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements of the ASME Code Section XI.

LRA Section A.2.2.2.3, Environmental Effects on Fatigue, is revised as follows.

The effects of reactor water environment on fatigue were evaluated for license renewal. Projected cumulative usage factors (CUFs) were calculated for the limiting locations ~~identified in~~ based on NUREG/CR-6260. The identified IP2 locations are those listed in the license renewal application, Table 4.3-13. ~~For the locations with CUFs less than 1.0, the TLAA has been projected through the period of extended operation per 10 CFR 54.21(c)(1)(ii).~~ Several locations may exceed a CUF of 1.0 with consideration of environmental effects during the period of extended operation. The Fatigue Monitoring Program requires that ~~A~~at least two years prior to entering the period of extended operation, the site will implement one or more of the following:

(1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using ~~R~~refined ~~the~~fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined in accordance with one of the following.

For locations~~, including NUREG/CR-6260 locations,~~ with existing fatigue analysis valid for the period of extended operation, use the existing CUF ~~to determine the environmentally adjusted CUF~~.

~~In addition to the NUREG/CR-6260 locations, more limiting~~Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.

Representative CUF values from other plants, adjusted to or enveloping the plant-specific external loads may be used if demonstrated applicable.

An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

~~(2) Manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-~~

~~destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC).~~

(~~3~~2) Consistent with the Fatigue Monitoring Program, Corrective Actions, ~~R~~repair or replace the affected locations before exceeding a CUF of 1.0.

~~Should IPEC select the option to manage the aging effects due to environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.~~

LRA Section A.3.1.11, Fatigue Monitoring Program, is revised as follows.

The Fatigue Monitoring Program is an existing program that tracks the number of critical thermal and pressure transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly analyzed a specified number of fatigue transients by assuring that the actual effective number of transients does not exceed the analyzed number of transients. The program provides for update of the fatigue usage calculations to maintain a CUF of < 1.0 for the period of extended operation. For the locations identified in Section A.3.2.2.3, updated calculations will account for the effects of the reactor water environment. These calculation updates are governed by Entergy's 10 CFR 50 Appendix B Quality Assurance (QA) program and include design input verification and independent reviews ensuring that valid assumptions, transients, cycles, external loadings, analysis methods, and environmental fatigue life correction factors will be used in the fatigue analyses. The program requires corrective actions including repair or replacement of affected components before fatigue usage calculations determine the CUF exceeds 1.0. Specific corrective actions are implemented in accordance with the IPEC corrective action program. Repair or replacement of the affected component(s), if necessary, will be in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by Entergy's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements of the ASME Code Section XI.

LRA Section A.3.2.2.3, Environmental Effects on Fatigue, is revised as follows.

The effects of reactor water environment on fatigue were evaluated for license renewal. Projected cumulative usage factors (CUFs) were calculated for the limiting locations ~~identified in~~ based on NUREG/CR-6260. The identified IP3 locations are those listed in the license renewal application, Table 4.3-14. ~~For the locations with CUFs less than 1.0, the TLAA has been projected through the period of extended operation per 10 CFR 54.21(c)(1)(ii).~~ Several locations may exceed a CUF of 1.0 with consideration of environmental effects during the period of extended operation. The Fatigue Monitoring Program requires that ~~A~~at least two years prior to entering the period of extended operation, ~~for the locations identified in NUREG/CR-6260 for Westinghouse PWRs of this vintage,~~ the site will implement one or more of the following:

(1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculation using ~~R~~refined ~~the~~ fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined in accordance with one of the following.

For locations~~, including NUREG/CR-6260 locations,~~ with existing fatigue analysis valid for the period of extended operation, use the existing CUF~~ to determine the environmentally adjusted CUF~~.

~~In addition to the NUREG/CR-6260 locations, more limiting~~Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.

Representative CUF values from other plants, adjusted to or enveloping the plant-specific external loads may be used if demonstrated applicable.

An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

~~(2) Manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC).~~

(~~3~~2) Consistent with the Fatigue Monitoring Program, Corrective Actions, ~~R~~repair or replace the affected locations before exceeding a CUF of 1.0.

~~Should IPEC select the option to manage the aging effects due to environmental assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.~~

LRA Section B.1.12, Fatigue Monitoring, Program Description, is revised as follows.

The Fatigue Monitoring Program is an existing program that tracks the number of critical thermal and pressure transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly analyzed a specified number of fatigue transients by assuring that the actual effective number of transients does not exceed the analyzed number of transients.

The program provides for update of the fatigue usage calculations to maintain a CUF of < 1.0 for the period of extended operation. For the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), updated calculations will account for the effects of the reactor water environment. These calculation updates are governed by Entergy's 10 CFR 50 Appendix B Quality Assurance (QA) program and include design input verification and independent reviews ensuring that valid assumptions, transients, cycles, external loadings, analysis methods, and environmental fatigue life correction factors will be used in the fatigue analyses.

The analysis methods for determination of stresses and fatigue usage will be in accordance with an NRC endorsed Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III Rules for Construction of Nuclear Power Plant Components Division 1 Subsection NB, Class 1 Components, Sub articles NB-3200 or NB-3600 as applicable to the specific component. IPEC will utilize design transients from IPEC Design Specifications to bound all operational transients. The numbers of cycles used for evaluation will be based on the design number of cycles and actual IPEC cycle counts projected out to the end of the license renewal period (60 years).

Environmental effects on fatigue usage will be assessed using methodology consistent with the Generic Aging Lessons Learned Report, NUREG-1801, Rev. 1, (GALL) that states: "The sample of critical components can be evaluated by applying environmental life correction factors to the existing ASME Code fatigue analyses. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels."

The Fatigue Monitoring Program tracks actual plant transients and evaluates these against the design transients. Cycle counts show no limits are expected to be approached for the current license term. The Fatigue Monitoring Program will ensure that the numbers of transient cycles experienced by the plant remain within the analyzed numbers of cycles and hence, the component CUFs remain below the values calculated in the design basis fatigue evaluations. If ongoing monitoring indicates the potential for a condition outside that analyzed above, IPEC may perform further reanalysis of the identified configuration using established configuration management processes as described above.

The program requires corrective actions including repair or replacement of affected components before fatigue usage calculations determine the CUF exceeds 1.0. Specific corrective actions are implemented in accordance with the IPEC corrective action program. Repair or replacement of the affected component(s), if necessary, will be in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by Entergy's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements

ATTACHMENT 2 TO NL-08-021

<u>Regulatory Commitment List, Revision 3</u>

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3
DOCKET NOS. 50-247 AND 50-286

List of Regulatory Commitments

Rev. 3

The following table identifies those actions committed to by Entergy in this document.

Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
1	Enhance the Aboveground Steel Tanks Program for IP2 and IP3 to perform thickness measurements of the bottom surfaces of the condensate storage tanks, city water tank, and fire water tanks once during the first ten years of the period of extended operation. Enhance the Aboveground Steel Tanks Program for IP2 and IP3 to require trending of thickness measurements when material loss is detected.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.1 A.3.1.1 B.1.1
2	Enhance the Bolting Integrity Program for IP2 and IP3 to clarify that actual yield strength is used in selecting materials for low susceptibility to SCC and clarify the prohibition on use of lubricants containing MoS_2 for bolting. The Bolting Integrity Program manages loss of preload and loss of material for all external bolting.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.2 A.3.1.2 B.1.2 Audit Items 201, 241, 270
3	Implement the Buried Piping and Tanks Inspection Program for IP2 and IP3 as described in LRA Section B.1.6. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.M34, Buried Piping and Tanks Inspection.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.5 A.3.1.5 B.1.6 Audit Item 173

<table>
<tr><th>#</th><th>COMMITMENT</th><th>IMPLEMENTATION SCHEDULE</th><th>SOURCE</th><th>RELATED LRA SECTION / AUDIT ITEM</th></tr>
<tr><td>33</td><td>At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), <u>under the Fatigue Monitoring Program,</u> IP2 and IP3 will implement one or more of the following:

(1) <u>Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using</u> ~~R~~refine<u>d</u> ~~the~~ fatigue analyses to determine valid CUFs less than 1<u>.0</u> when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following:

1. For locations <u>in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3)</u> ~~including NUREG/CR-6260 locations~~, with existing fatigue analysis valid for the period of extended operation, use the existing CUF. ~~to determine the environmentally adjusted CUF.~~

2. ~~In addition to the NUREG/CR-6260 locations, more limiting~~ <u>Additional</u> plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.

3. Representative CUF values from other plants, adjusted to or enveloping the IPEC plant specific external loads may be used if demonstrated applicable to IPEC.

4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

~~(2) Manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC).~~

~~(3)~~ <u>(2) Consistent with the Fatigue Monitoring Program, Corrective Actions,</u> ~~R~~repair or replace the affected locations before exceeding a CUF of 1.0.

~~Should IPEC select the option to manage the aging effects due to environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.~~</td><td>IP2:
September 28, 2011

IP3:
December 12, 2013</td><td>NL-07-039

NL-07-153

NL-08-021</td><td>A.2.2.2.3
A.3.2.2.3
4.3.3
Audit item 146</td></tr>
</table>

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
34	IP2 SBO / Appendix R diesel generator will be installed and operational by April 30, 2008. This committed change to the facility meets the requirements of 10 CFR 50.59(c)(1) and, therefore, a license amendment pursuant to 10 CFR 50.90 is not required.	April 30, 2008	NL-07-078	2.1.1.3.5

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 8

SAND94-0187, *Evaluation of Conservatisms and Environmental Effects in ASME Code, Section III, Class 1 Fatigue Analysis* (Aug. 1994) is available to the public for purchase from the National Technical Information Service (NTIS) (http://www.ntis.gov/)

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 9

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG-1916, Vol. 2

Safety Evaluation Report

Related to the License Renewal of Shearon Harris Nuclear Power Plant, Unit 1

Docket No. 50-400

Carolina Power & Light Company

Manuscript Completed: August 2008
Date Published: November 2008

Prepared by
M. Heath

Office of Nuclear Reactor Regulation

concludes that the FSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

4.3.1.6 Pressurizer

4.3.1.6.1 Summary of Technical Information in the Application

LRA Section 4.3.1.6 summarizes the evaluation of the pressurizer for the period of extended operation. There are TLAAs for several pressurizer subcomponents. The use of transients from the steam generator replacement/uprating analysis is reasonable and limiting for the primary equipment with the exceptions of the pressurizer surge line and portions of the pressurizer lower head analyzed separately; therefore, the NSSS design transients are those shown in the steam generator replacement/uprating analysis, in which 40-year design CUF values also were determined.

The pressurizer fatigue analysis demonstrated that, if pressurizer subcomponents were exposed to a bounding set of postulated transient cycles, CUF values would not exceed 1.0 for all components; however, certain pressurizer lower head locations are not bounded by the original design fatigue analysis because it did not consider insurge/outsurge transients discovered subsequently.

For the pressurizer (other than the lower head and surge line nozzle), the highest 40-year design fatigue usage value is 1.00 for the "Trunnion Bolt Hole" (LRA Table 4.3-2). Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a CUF of 1.50.

The applicant used Westinghouse Owners Group (WOG) recommendations to address operational pressurizer insurge/outsurge transients by reviewing plant operating records in sufficient detail to determine pressurizer insurge/outsurge transients for past operation, updating pressurizer lower head and surge nozzle transients to reflect past and projected future operations, and evaluating the impact of the updated transients on the structural integrity of the pressurizer. The WOG also recommended operating strategies that may be useful in addressing the insurge/outsurge issue. On January 20, 1994, the applicant adopted the modified operating procedures recommended by the WOG to mitigate pressurizer insurge/outsurge transients.

The applicant used plant data from hot functional testing to January 20, 1994, to establish pre-modified operating procedure transients that represent past plant heat-up and cool-down operations and collected and processed plant data from July 19, 1999, to October 18, 2004, for post-modified operating procedures operations. The 5.26 years of data history with the pre-modified operating procedure transients was projected to predict 60-year fatigue usage based on current operating practices.

Fatigue evaluations of the pressurizer lower head and surge line nozzle used the online monitoring and Westinghouse proprietary design analysis features of the WESTEMS™ Integrated Diagnostics and Monitoring System. The fatigue evaluations follow the procedures of

ASME Code, Section III, NB-3200. Calculations of stress ranges, cycle pairing, and fatigue usage factors were by use of WESTEMS™ consistent with the ASME Code and WOG recommendations.

The fatigue evaluations at critical locations of the pressurizer lower head (including the pressurizer surge line nozzle) and of the surge line RCS hot leg nozzle were based upon pre-modified operating procedure transients with the post-modified operating procedure transients that include the effects of insurge/outsurge and surge line stratification. These transients were developed based upon plant-specific data and WOG information and guidelines. The predicted fatigue usage was determined assuming future operations following current operating procedures.

For 40 years of plant life, the pressurizer lower head has the highest fatigue usage of 0.36 at the inside surface of the lower head at the heater penetration region. Multiplying this fatigue usage by 1.5 to account for 60 years of operation yields a fatigue usage of 0.54. Evaluation of this location also accounted for the effects of reactor water environment on fatigue. The 60-year fatigue usage for this location is 1.35 as shown in LRA Table 4.3-3.

For the pressurizer, the maximum fatigue usage for 60 years of operation is 1.35. This value exceeds the design limit of 1.0 and, therefore, requires an AMP. The Reactor Coolant Pressure Boundary Fatigue Monitoring Program will maintain the design limit fatigue usage or take appropriate re-evaluation or corrective action to manage the effects of fatigue on the pressurizer for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.1.6.2 Staff Evaluation

During audit and review, the staff asked the applicant what components are in the stress-based fatigue monitoring portion of the HNP program. The applicant responded as follows:

> The HNP Fatigue Evaluation for License Renewal (WCAP-16353-P) resulted in the following locations recommended for inclusion into the program.
>
> - Pressurizer Lower Head
> - Pressurizer Surge Line
> - CVCS Piping and Heat Exchanger
>
> Based on the Westinghouse recommendations, the HNP fatigue monitoring program will be enhanced to include the above components by monitoring fatigue usage for these locations using online fatigue monitoring software.

In this letter, the applicant also indicated its stress-based fatigue monitoring locations and stress-based alarm limit of 0.9. On the basis that the 0.9 alarm limit will provide adequate time for actions, the staff concluded that the applicant's stress-based alarm limit is adequate. For all other locations managed through a cycle-based monitoring program, the applicant also provided its alarm limit. Commitment 32 states that the enhanced program will address corrective actions through the Corrective Action Program for components exceeding alarm limits, including a revised fatigue analysis or repair or replacement of the component.

LRA Amendment 2 states that the applicant used plant data from July 19, 1999, to October 18, 2004, to predict 60-year fatigue usage based on current operating practices. The staff does not agree with this prediction, which used 5.26 years of data to determine the next 40 years of operation transients; however, the applicant, by letter date January 17, 2008, committed to a stress-based fatigue monitoring program to manage those components. On this basis, the staff finds this LRA amendment acceptable. Therefore the applicant projections will not be used. The applicant will manage the effects of aging for the period of extended operation.

LRA Amendment 2 also states that the pressurizer lower head heater penetration region has the highest fatigue usage (0.36) for the 40 years of plant life. LRA Table 4.3-2 lists a design fatigue usage factor of 0.909 for this location. The staff asked the applicant to address the difference. This item was confirmatory item (CI) 4.3 and needed the applicant's docketed response to complete the staff's review.

In letter dated April 23, 2008, the applicant stated that HNP will update the piping design specification to reflect the current design basis operational transients used in the Time-Limited Aging Analyses for the reactor coolant pressure boundary (See Commitment No. 37). The applicant also amended LRA FSAR Supplement Section A.1.2.2.2.10 to indicate that the TLAA on metal fatigue of the charging nozzle, surge line, and pressurizer lower head and surge nozzle will be managed in accordance with the 10 CFR 54.21(c)(1)(iii). This is consistent with the applicant's TLAA on metal fatigue of the Class 1 piping components (as provided in LRA Section 4.3.5), which indicates that the Fatigue Monitoring Program will be used to manage the effects of aging for these components in accordance with the TLAA acceptance criterion requirement in 10 CFR 54.21(c)(1)(iii).

Based on this review, the staff finds that the applicant has appropriately addressed the staff's confirmatory item on the TLAA on metal fatigue of the reactor coolant pressure boundary. Confirmatory Item 4.3 is closed.

During the audit and review, the staff asked the applicant to explain the input of stresses to apply the stress transfer function of fatigue analysis software, WESTEMS™, to the stressed components or the stress intensity and asked for input and results of any benchmarking problems for pressure, temperature, or moment loadings.

The applicant's response is in pages 67 to 93 of Enclosure 3 of LRA Amendment 2 by letter dated August 31, 2007.

The staff reviewed the applicant's response explaining the method for the stress transfer function of fatigue analysis software WESTEMS. On the basis of its review, the staff confirmed that the applicant superimposed stress at the component stress level for each time step and for each applied loading type. The staff concluded that the method is in accordance with ASME Section III, Division 1, NB-3200 criteria.

The applicant also stated,

> The verification of fatigue analysis software thermal and mechanical stress calculations have been performed in the programs verification and validation documentation. However, each application verification of the finite element model and of the final thermal transfer function databases should be performed in order to show applicability to the problem being modeled. To do this for mechanical loads, Westinghouse verifies the finite element model results by comparing them to the expected theoretical values. For the time varying thermal results, the applicant performs thermal stress analyses using both the finite element program and WESTEMS™."

On the basis that verified fatigue analysis software stress results had the theoretical values and traditional finite element analysis, the staff finds the applicant's transfer function method for evaluating stress results acceptable.

The staff also reviewed the applicant's benchmark verification results plotted in Figures B-1 through B-11 and additional results of samples 1 and 2 all indicating that the stress results generated from fatigue analysis software and those generated from traditional finite element ANSYS analysis have negligible differences. On this basis, the staff concludes that stress evaluation by fatigue analysis software is acceptable.

The GALL Report recommends a fatigue monitoring program to manage metal fatigue according to 10 CFR 54.21(c)(1)(iii). The staff has evaluated the applicant's AMP B3.1, "Reactor Coolant Pressure Boundary Fatigue Monitoring Program," for monitoring and tracking the number of critical thermal and pressure transients for RCS components, determined that this program is acceptable to address metal fatigue of RCS components according to 10 CFR 54.21(c)(1)(iii), and documented its evaluation and acceptance in SER Section 3.0. On the basis that the applicant's action is consistent with the GALL Report recommendation, the staff finds that management of the effects of aging on intended functions will be adequate for the period of extended operation.

4.3.1.6.3 FSAR Supplement

The applicant provided an FSAR supplement summary description, as amended by letter dated April 23, 2208, of its TLAA evaluation of the pressurizer in LRA Section A.1.2.2.6. On the basis of its review of the FSAR supplement, the staff concludes that the summary description of the applicant's actions to address pressurizer is adequate.

4.3.1.6.4 Conclusion

On the basis of its review, as discussed above, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that management of the effects of aging on intended functions will be adequate for the period of extended operation. The staff also concludes that the FSAR supplement is an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d).

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 10

Progress Energy

AUG 3 1 2007

SERIAL: HNP-07-119
10 CFR 54

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

Subject: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400 / LICENSE NO. NPF-63

LICENSE RENEWAL APPLICATION, AMENDMENT 2: CHANGES RESULTING FROM RESPONSES TO SITE AUDIT QUESTIONS REGARDING TIME-LIMITED AGING ANALYSES

References: 1. Letter from Cornelius J. Gannon to the U. S. Nuclear Regulatory Commission (Serial: HNP-06-136), "Application for Renewal of Operating License," dated November 14, 2006

Ladies and Gentlemen:

On November 14, 2006, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, requested the renewal of the operating license for the Shearon Harris Nuclear Power Plant, Unit No. 1, also known as the Harris Nuclear Plant (HNP), to extend the term of its operating license an additional 20 years beyond the current expiration date.

The Nuclear Regulatory Commission (NRC) staff conducted audits of HNP License Renewal activities related to time-limited aging analyses (TLAAs) during the periods from May 21 to 25, from June 25 to 29, and from August 13 to 15, 2007. In the course of these audits, questions were identified by the auditors. Responses to these TLAA-related questions are enclosed.

Also, enclosed is the list of regulatory commitments supporting License Renewal modified to reflect the information provided in the responses to TLAA-related audit questions. Any other actions discussed should be considered intended or planned actions; they are included for informational purposes but are not considered to be regulatory commitments.

Based on the above activities, required changes to the HNP License Renewal Application (LRA) have been identified. This LRA amendment consists of three enclosures. Enclosure 1 is the revised list of License Renewal Commitments. Enclosure 2 is a table that identifies

Progress Energy Carolinas, Inc.
Harris Nuclear Plant
P.O. Box 165
New Hill, NC 27562

A126
NRR

changes to the LRA and the source of those changes. Enclosure 3 is a report of TLAA-related questions and responses from the NRC audits.

Please refer any questions regarding this submittal to Mr. Roger Stewart, Supervisor - License Renewal, at (843) 857-5375.

I declare, under penalty of perjury, that the foregoing is true and correct (Executed on AUG 31 2007).

Sincerely,

Thomas J. Natale
Manager - Support Services
Harris Nuclear Plant

TJN/mhf

Enclosures:

1. HNP License Renewal Commitments, Revision 2
2. Amendment 2 Changes to the License Renewal Application
3. Harris Nuclear Plant License Renewal Audit Question and Response Database Report

cc:

Mr. P. B. O'Bryan (NRC Senior Resident Inspector, HNP)
Ms. B. O. Hall (Section Chief, N.C. DENR)
Mr. M. L. Heath (NRC License Renewal Project Manager, HNP)
Ms. M. G. Vaaler (NRC Project Manager, HNP)
Dr. W. D. Travers (NRC Regional Administrator, Region II)

Harris Nuclear Plant License Renewal Audit Question and Response Database

Question No: LRA 4.3-17

NRC Request:
Please explain how stresses are input to apply the stress transfer function of WESTEMS™, from the 6 stress components or the stress intensity. Please provide input and results of any benchmarking problems for pressure, temperature, or moment loadings.

HNP Response:
WESTEMS™ uses the transfer function method (TFM) [1] to calculate six (6) components of stresses due to time varying mechanical and thermal loads. Time varying component stresses are calculated through wall as a function of the time varying mechanical and thermal boundary conditions. The resulting through wall stress components are processed and categorized according ASME Section III, Division 1, Subsection NB criteria. The processing first involves the calculation and categorization of the membrane and bending and peak components mechanical and thermal stresses. These calculations are performed at the component stress levels, for each time step and for each applied loading type. The resulting stresses are then added to form the total stress and primary plus secondary stress according to ASME rules. Stress peak selection for fatigue evaluation purposes is based on analysis of the total stress time history and of the primary plus secondary stress time history. Both the total stress and primary plus secondary stress are retained for future consideration in online fatigue evaluations. The discussion below will help to clarify the transfer function methodology, the transfer function database role, and provide an example of the current benchmarking process.

The transfer function method is a mathematical device that is capable of quantifying the effects experienced by a system due to an external disturbance, or excitation, with the aid of a characteristics function known as transfer function. In essence, the transfer function method is a means that correlates time-dependent behavior, in terms of input and output, of a system as seen in the thermal and dynamical problems. Examples of "disturbance" are mechanical forces, and thermal transients, etc. Examples of "effects" include stresses, strains, displacements, and temperature, etc. For typical structural applications, the "disturbance" can be surface temperature changes T(t), pressure P variation, forces (Fx, Fy, Fz), and moments (Mx, My, Mz) in a structural body (in vector notations: $\vec{F}$, and $\vec{M}$), whereas the typical "effects" mostly refers to the stresses, displacements and metal interior temperatures.

Harris Nuclear Plant License Renewal Audit Question and Response Database

In WESTEMS™, the transfer function methodology uses 2 or more unit load databases that have 4 or 6 components of stress depending on the nature of the original finite element model method that was used. If a two dimensional finite element model was used to create the transfer function database then 4 components of stress are capable (Sxx, Syy, Szz, Sxy). If a three dimensional model was used then there are 6 components of stress in the transfer function databases (Sxx, Syy, Szz, Sxy, Syz, and Szx). The total number of stress states in the transfer function databases is dependent on the complexity of the thermal and mechanical boundary conditions being simulated.

For thermal applications, the transfer function is a characteristics function of a thermal-mechanical system. The characteristics include geometry, boundary conditions, insulation conditions, material properties, and thermal zones. These characteristics are all built into the transfer function for a predefined thermal-mechanical system. Therefore, a transfer function database is fixed for a particular type of thermal-mechanical problem. However, a single set of transfer function database can be used to evaluate the system responses caused by any kind of transients. This means that transfer function database is created only once but can be used to obtain solutions for unlimited numbers of transient cases.

It is important to realize that thermal stresses in materials or any structural systems arisen from temperature transients are evolving because heat transfer is an energy transport process that will continue until thermal equilibrium is established. This means that it requires appreciable amount of time for a thermally disturbed material or structural system to come to a steady state even if the disturbance is as brief as an impulse. In short, thermal transient is a time-dependent problem. On the contrary, all mechanical loads, pressure, direct forces, and moments, encountered in the general structural applications are treated as static problems unless the loading rates are so high that the dynamic effects can not be ignored. To appropriately reflect to the types of loads being dealt with, the databases are split into two types:

- Thermal transfer Function DataBase (TFDB)
- MEchanical transfer function DataBase (MEDB).

Westinghouse has validated the thermal stress capability of the WESTEMS™ transfer function method by performing identical analyses using the Westinghouse transfer function method and an independent finite element program like ANSYS or WECAN. Examples of the predicted stress components results for benchmarking the transfer function models are shown below. The benchmarking process is generally performed for every transfer function database created. The following example was taken directly from the appendix of a recent Westinghouse Transfer function database calculation note. The verification of WESTEMS™ thermal and mechanical stress calculations have been performed in the programs verification and validation documentation. However, each application verification of the finite element models and of the final thermal transfer function databases should be performed in order to show applicability to the problem being modeled. To do this for mechanical loads, Westinghouse verifies the finite element model results by comparing them to the expected theoretical values. For the time varying thermal results Westinghouse performs thermal stress analyses using both the finite element program and WESTEMS™. The example below shows these comparisons and results. Certain information has been removed and text has been modified in order to clarify the example.

Verification of the Surge Nozzle With Reducer TFDB and MEDB

Verification of the databases being used for the WESTEMS™ analyses is a required step to ensure good analysis results. All databases are herein examined through suitable benchmarking problems.

The database files, TFDB and MEDB, generated in the unit load finite element analyses represent the thermal and mechanical characteristics of the structural component considered. By using these databases, the stresses at the specified analysis sections (ASN or cut) can be evaluated for any combination of load conditions. To correctly produce the results, each load type requires an appropriate scaling factor which is being developed in the following subsections. The scaling factor provides a means to correct the effects arisen from differences on the stress units used in ANSYS and in WESTEMS™. It also is a means which permits non-standard unit loads to be used to generate the database.

Verifying the bending moment database - M_z

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the bending moment portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

Moment M_z represents bending about the global z-axis. According to Reference 1, the z-axis is perpendicular to the x and y axes. The analysis for this bending case was performed and documented in Reference 1. The applied moment is 1000 in-kips.

Harris Nuclear Plant License Renewal Audit Question and Response Database

Consider the well known bending stress equation

$\sigma = Mr/I$

where M is the applied bending moment, r is distance from the neutral axis, and I is the moment of inertia of the cross sectional area.

Two nodes, as listed in Table B-1, are considered to benchmark/verify the results. These nodes are located at the stainless steel pipe section of the model, remote from the reinforced section of the nozzle. Therefore, the above bending equation can be applied.

At this location, the following data apply:

$R_o = 5.250$ in.

$R_i = 6.375$ in.

$I = \pi (R_o^4 - R_i^4) / 4 = 700.55\ in^4$

Comparison of the ANSYS FE and analytical results are shown in Table B-1. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied moment is in-kips whereas 1000 in-kips of bending was used in the database creation, a second scaling factor, f_2=0.001, is required. Combining the two and the scaling factor for the bending load to be used for WESTEMS™ analyses is found to be $f_b = f_1 * f_2 = 10^{-6}$.

Table B-1: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	Error (%)
Inside node	275	7494.11	7533.20	-0.52
Outside node	187	9099.99	9067.20	0.36

Verifying the torsion database - M_y

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the torsion moment portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

The moment M_y represents the moment about the global y-axis. The y-axis is in coincidence with the centerline of the nozzle. Moment M_y therefore represents twist of the nozzle. The analysis for this torsion case was performed and documented. The applied moment is 1000 in-kips.

Consider the well known torsion shearing stress equation

$$\tau = Mr/J$$

where M is the applied torque, r is distance from the neutral axis, and J is the polar moment of inertia in torsion of the cross sectional area.

Two nodes, as listed in Table B-2, are considered to benchmark/verify the results. These nodes are located at the stainless steel pipe section of the model, remote from the reinforced section of the nozzle. Therefore, the above torsion equation can be applied.

At this location, the following data apply:

R_o = 6.375 in.

R_i = 5.25 in.

$J = \pi (R_o^4 - R_i^4) / 2 = 1401.1 \text{ in}^4$

Comparison of the ANSYS FE and analytical results are shown in Table B-2. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied torque is in-kips whereas 1000 in-kips of torque was used in the database creation, a second scaling factor, f_2=0.001, is required. Combining the two and the scaling factor for the torsion load to be used for WESTEMS™ analyses is found to be $f_t = f_1 * f_2 = 10^{-6}$.

Table B-2: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	Error (%)
Inside node	275	3747.05	3747.10	0.00
Outside node	187	4550.00	4550.00	0.00

Verifying the pressure database

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the pressure portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

The analysis for the pressure loading case was performed and documented. The applied pressure is 1000 psi.

Consider the well known hoop stress equation for a pressurized pipe

$$\sigma_\theta = \frac{p\,R_i^2}{R_o^2 - R_i^2}\left(1 + \frac{R_o^2}{r^2}\right)$$

where p is the internal pressure, R_o is the outside radius, R_i is the inside radius, and r is the radius at any point.

Two nodes, as listed in Table B-3, are considered to benchmark/verify the results. These nodes are located at the stainless steel pipe section of the model, remote from the reinforced section of the nozzle. Therefore, the above hoop stress equation can be applied.

Harris Nuclear Plant License Renewal Audit Question and Response Database

At this location, the following data apply:

R_o = 6.375 in.

R_i = 5.25 in.

Comparison of the ANSYS FE and analytical results are shown in Table B-3. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied pressure is psi whereas 1000 psi of pressure was used in the database creation, a second factor, f_2=0.001, is required. Combining the two factors and the scaling factor for the pressure load to be used for WESTEMS™ analyses is found to be $f_p = f_1 * f_2 = 10^{-6}$.

Table B-3: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	error (%)
Inside node	275	5215.05	5230.30	-0.29
Outside node	187	4215.05	4208.60	0.15

Verifying the axial database

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the axial force portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

The analysis for the axial force loading case was performed and documented. The applied axial force is 1000 lb.

Consider the well known axial stress equation for a pipe

Harris Nuclear Plant License Renewal Audit Question and Response Database

$$\sigma = \frac{F}{A}$$

where F is the applied force, and A is the cross sectional area.

Two nodes, as listed in Table B-4, are considered to benchmark/verify the results. These nodes are located at the stainless steel pipe section of the model, remote from the reinforced section of the nozzle. Therefore, the above axial stress equation can be applied.

At this location, the following data apply:

R_o = 6.375 in.

R_i = 5.25 in.

$A = \pi (R_o^2 - R_i^2) = 41.09\ in^2$

Comparison of the ANSYS FE and analytical results are shown in Table B-4. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied force is 1 kip whereas 1 kip of force was used in the database creation, a second factor, f_2=1.0, is required. Combining the two factors and the scaling factor for the axial load to be used for WESTEMS™ analyses is found to be $f_a = f_1 * f_2 = 10^{-3}$.

Table B-4: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	Error (%)
Inside node	275	24.34	24.38	0.18
Outside node	187	24.34	24.30	0.16

Verifying the thermal stress database

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the transfer function thermal stress database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

To benchmark and verify this portion of the database and determine the appropriate scaling factor for the thermal loads, an arbitrary transient was used. The transient used for this benchmarking problem is defined in the data shown in Table B-6 and Figure B-1.

Table B-6: Temperature and Film Coefficient used for the Benchmarking.

Time (s)	tzone1 (°F)	tzone2 (°F)	tzone3 (°F)	hzone1 (Btu/s-in^2-°F)	hzone2 (Btu/s-in^2-°F)	hzone3 (Btu/s-in^2-°F)
0.001	550	550	550	0.007716	0.007716	0.007716
10	550	550	550	0.007716	0.007716	0.007716
12	250	250	250	0.007716	0.007716	0.007716
13	250	250	250	0.007716	0.007716	0.007716
14	250	250	250	0.007716	0.007716	0.007716
15	250	250	250	0.007716	0.007716	0.007716
16	250	250	250	0.007716	0.007716	0.007716
18	250	250	250	0.007716	0.007716	0.007716
20	250	250	250	0.007716	0.007716	0.007716
30	250	250	250	0.007716	0.007716	0.007716
40	250	250	250	0.007716	0.007716	0.007716
55	250	250	250	0.007716	0.007716	0.007716
70	250	250	250	0.007716	0.007716	0.007716
90	250	250	250	0.007716	0.007716	0.007716
110	250	250	250	0.007716	0.007716	0.007716
135	250	250	250	0.007716	0.007716	0.007716
160	250	250	250	0.007716	0.007716	0.007716
185	250	250	250	0.007716	0.007716	0.007716
210	250	250	250	0.007716	0.007716	0.007716

Harris Nuclear Plant License Renewal Audit Question and Response Database

Time (s)	tzone1 (°F)	tzone2 (°F)	tzone3 (°F)	hzone1 (Btu/s-in^2-°F)	hzone2 (Btu/s-in^2-°F)	hzone3 (Btu/s-in^2-°F)
212	550	550	550	0.007716	0.007716	0.007716
213	550	550	550	0.007716	0.007716	0.007716
214	550	550	550	0.007716	0.007716	0.007716
215	550	550	550	0.007716	0.007716	0.007716
216	550	550	550	0.007716	0.007716	0.007716
217	550	550	550	0.007716	0.007716	0.007716
219	550	550	550	0.007716	0.007716	0.007716
221	550	550	550	0.007716	0.007716	0.007716
225	550	550	550	0.007716	0.007716	0.007716
230	550	550	550	0.007716	0.007716	0.007716
235	550	550	550	0.007716	0.007716	0.007716
250	550	550	550	0.007716	0.007716	0.007716
265	550	550	550	0.007716	0.007716	0.007716
280	550	550	550	0.007716	0.007716	0.007716
300	550	550	550	0.007716	0.007716	0.007716
320	550	550	550	0.007716	0.007716	0.007716
345	550	550	550	0.007716	0.007716	0.007716
370	550	550	550	0.007716	0.007716	0.007716
395	550	550	550	0.007716	0.007716	0.007716
410	550	550	550	0.007716	0.007716	0.007716
470	548	548	548	0.007716	0.007716	0.007716
530	546	546	546	0.007716	0.007716	0.007716
590	544	544	544	0.007716	0.007716	0.007716
650	542	542	542	0.007716	0.007716	0.007716
710	540	540	540	0.007716	0.007716	0.007716
770	538	538	538	0.007716	0.007716	0.007716
830	536	536	536	0.007716	0.007716	0.007716
890	534	534	534	0.007716	0.007716	0.007716
950	532	532	532	0.007716	0.007716	0.007716
1010	530	530	530	0.007716	0.007716	0.007716
1070	528	528	528	0.007716	0.007716	0.007716

Time (s)	tzone1 (°F)	tzone2 (°F)	tzone3 (°F)	hzone1 (Btu/s-in^2-°F)	hzone2 (Btu/s-in^2-°F)	hzone3 (Btu/s-in^2-°F)
1130	526	526	526	0.007716	0.007716	0.007716
1190	524	524	524	0.007716	0.007716	0.007716
1250	522	522	522	0.007716	0.007716	0.007716
1310	520	520	520	0.007716	0.007716	0.007716
1370	518	518	518	0.007716	0.007716	0.007716
1430	516	516	516	0.007716	0.007716	0.007716
1490	514	514	514	0.007716	0.007716	0.007716
1550	512	512	512	0.007716	0.007716	0.007716
1610	510	510	510	0.007716	0.007716	0.007716
1670	508	508	508	0.007716	0.007716	0.007716
1730	506	506	506	0.007716	0.007716	0.007716
1790	504	504	504	0.007716	0.007716	0.007716
1850	502	502	502	0.007716	0.007716	0.007716
1910	500	500	500	0.007716	0.007716	0.007716
1970	498	498	498	0.007716	0.007716	0.007716
2030	496	496	496	0.007716	0.007716	0.007716
2090	494	494	494	0.007716	0.007716	0.007716
2150	492	492	492	0.007716	0.007716	0.007716
2210	490	490	490	0.007716	0.007716	0.007716
2212	490	490	490	0.007716	0.007716	0.007716
2213	490	490	490	0.007716	0.007716	0.007716
2214	490	490	490	0.007716	0.007716	0.007716
2215	490	490	490	0.007716	0.007716	0.007716
2216	490	490	490	0.007716	0.007716	0.007716
2217	490	490	490	0.007716	0.007716	0.007716
2219	490	490	490	0.007716	0.007716	0.007716
2221	490	490	490	0.007716	0.007716	0.007716
2225	490	490	490	0.007716	0.007716	0.007716
2230	490	490	490	0.007716	0.007716	0.007716
2235	490	490	490	0.007716	0.007716	0.007716
2250	*490*	*490*	*490*	*0.007716*	*0.007716*	*0.007716*

Harris Nuclear Plant License Renewal Audit Question and Response Database

Time (s)	tzone1 (°F)	tzone2 (°F)	tzone3 (°F)	hzone1 (Btu/s-in^2-°F)	hzone2 (Btu/s-in^2-°F)	hzone3 (Btu/s-in^2-°F)
2265	490	490	490	0.007716	0.007716	0.007716
2280	490	490	490	0.007716	0.007716	0.007716
2300	490	490	490	0.007716	0.007716	0.007716
2320	490	490	490	0.007716	0.007716	0.007716
2345	490	490	490	0.007716	0.007716	0.007716
2370	490	490	490	0.007716	0.007716	0.007716
2395	490	490	490	0.007716	0.007716	0.007716
2410	490	490	490	0.007716	0.007716	0.007716

In this benchmarking transient, Zone 1, Zone 2, and Zone 3 undergo a severe thermal shock. The transient considered is hypothetical but is intentionally made severe on the temperature rate so as to allow a vigorous examination of the integrity of the transfer function database. The transient is shown in the Figure B-1. This transient was analyzed by both WESTEMS™ and ANSYS. Note that the ANSYS results represent full finite element analyses whereas the WESTEMS™ results are produced by the transfer function method, which utilizes the transfer function databases produced by ANSYS.

The metal surface temperatures for all three zones were calculated using the 1D Simplified Stress Model (SSM) in WESTEMS. The input data for this part of the calculations are shown in Table B-7. The metal surface temperature solutions from the SSM evaluations are represented by tagnames tzone1m, tzone2m, and tzone3m, which are saved in the WESTEMS benchmark history file.

Table B-7: Simplified Stress Models for Metal Surface Temperature Calculations.

Component ID	Name	OutPut Tag	T ambi	Material_A	Temp Tag_A	HFilm Tag_A	Wall Thick_A	Inside Diameter_A	Num Nodes_A
200	PZR Surge Nozzle with Reducer Zone 1	tzone1m	70	3	tzone1	hzone1	1.1	11.3	0
201	PZR Surge	tzone2m	70	60	tzone2	hzone2	1.56	11.88	0

Harris Nuclear Plant License Renewal Audit Question and Response Database

Component ID	Name	OutPut Tag	T ambi	Material_A	Temp Tag_A	HFilm Tag_A	Wall Thick_A	Inside Diameter_A	Num Nodes_A
	Nozzle with Reducer Zone 2								
202	PZR Surge Nozzle with Reducer Zone 3	tzone3m	70	2	tzone3	hzone3	3.0	12.0	0

Note:

1) Material 3 is SA 182 F316 SS 1989
2) Material 60 is SB166 Alloy 600 (Rod) 1998
3) Material 2 is SA 216 Gr WCC 1989

The results, as shown in Figures B-2 through B-11 (units: stress ksi, time seconds), are then graphically compared on both the shapes and the magnitudes. It can be seen from these figures that the WESTEMS™ results compare very well with those calculated by ANSYS, both in magnitudes and curve shapes.

The shapes of the curves of the stresses from the WESTEMS™ analysis are visually compared with those from the ANSYS full finite element analysis. In general, good comparisons are observed for all cases.

Overall, very good benchmarking results have been achieved which assures good results can be produced through the TFDB created in Reference 1. In order to maintain a conservative answer a correction factor of 4% is applied, that is, f_1=1.04. Since the stress unit in the ANSYS FEA results is psi whereas the stress unit to be used for WESTEMS^{TM} calculations is ksi, a factor, f_2=0.001, is required. Combining the two factors for the thermal load to be used for WESTEMS™ analyses is $f_T = f_1 * f_2$ = 0.00104, which is to be registered to the "TFDB_Factor" box in the WESTEMS™ ASN Analysis Models.

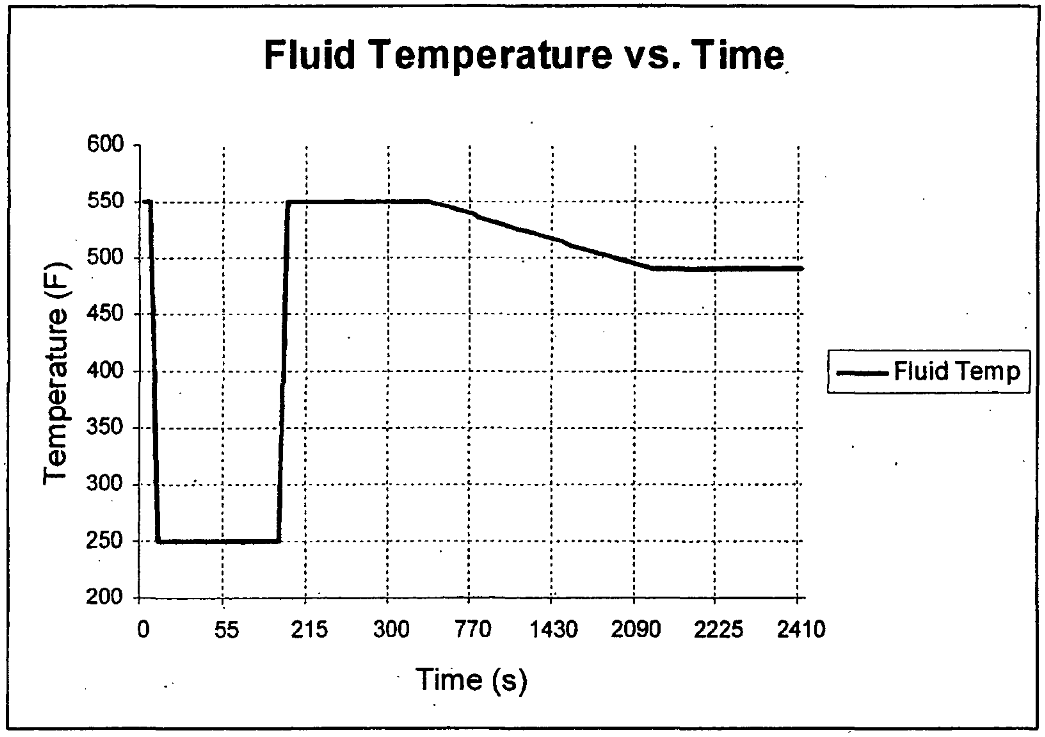


Figure B-1: Benchmark Transient.

Harris Nuclear Plant License Renewal Audit Question and Response Database

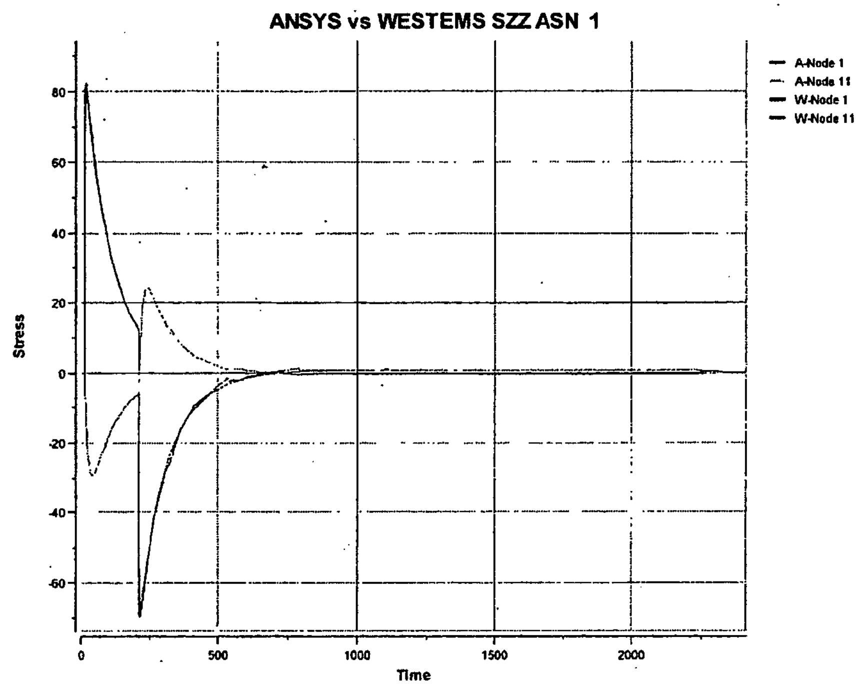


Figure B-2: ASN 1 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

Harris Nuclear Plant License Renewal Audit Question and Response Database

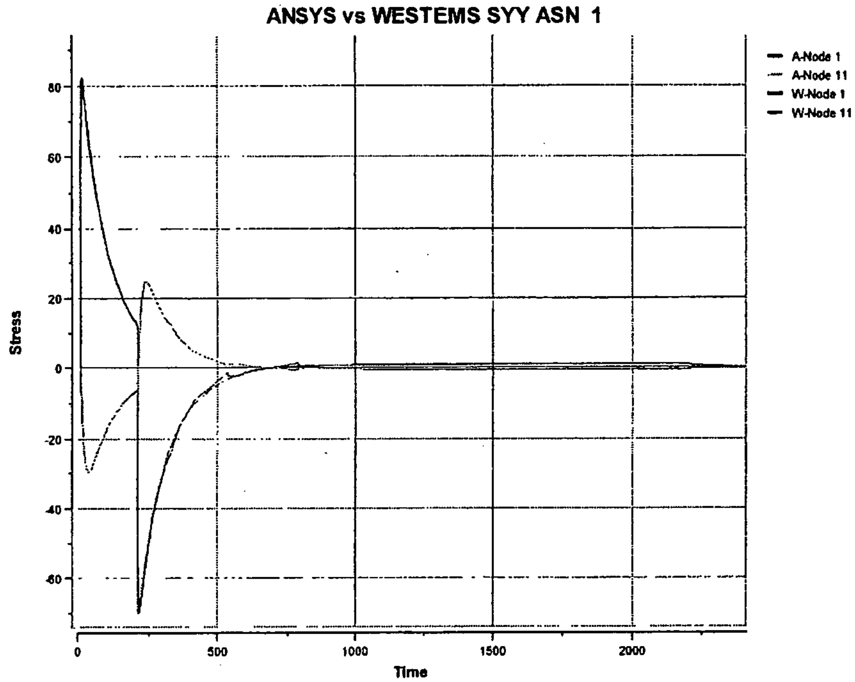


Figure B-3: ASN 1 Axial Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

Harris Nuclear Plant License Renewal Audit Question and Response Database

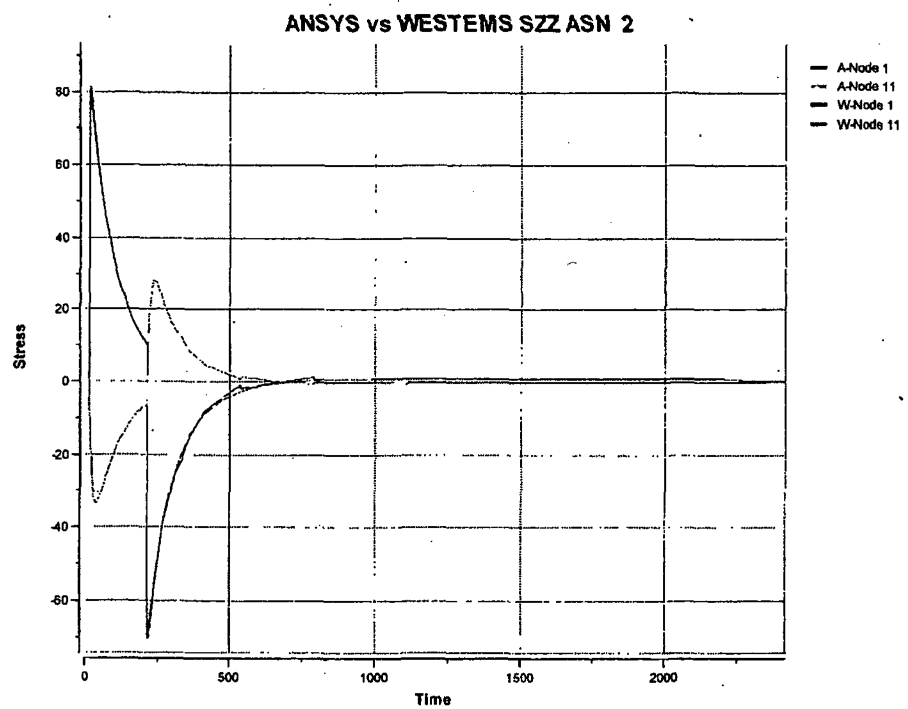


Figure B-4: ASN 2 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

Harris Nuclear Plant License Renewal Audit Question and Response Database

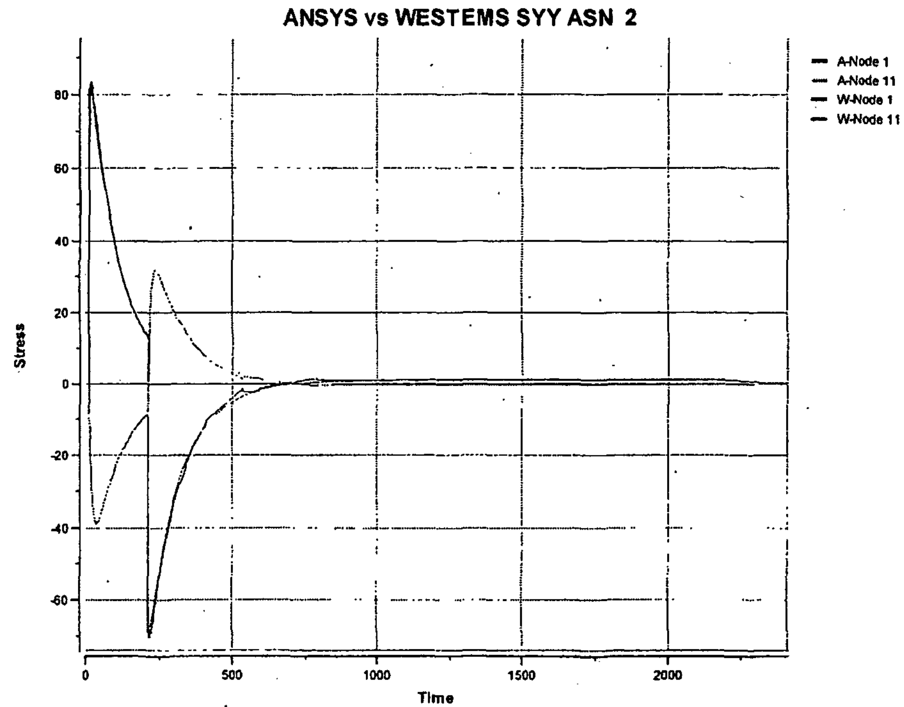


Figure B-5: ASN 2 Axial Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

Harris Nuclear Plant License Renewal Audit Question and Response Database

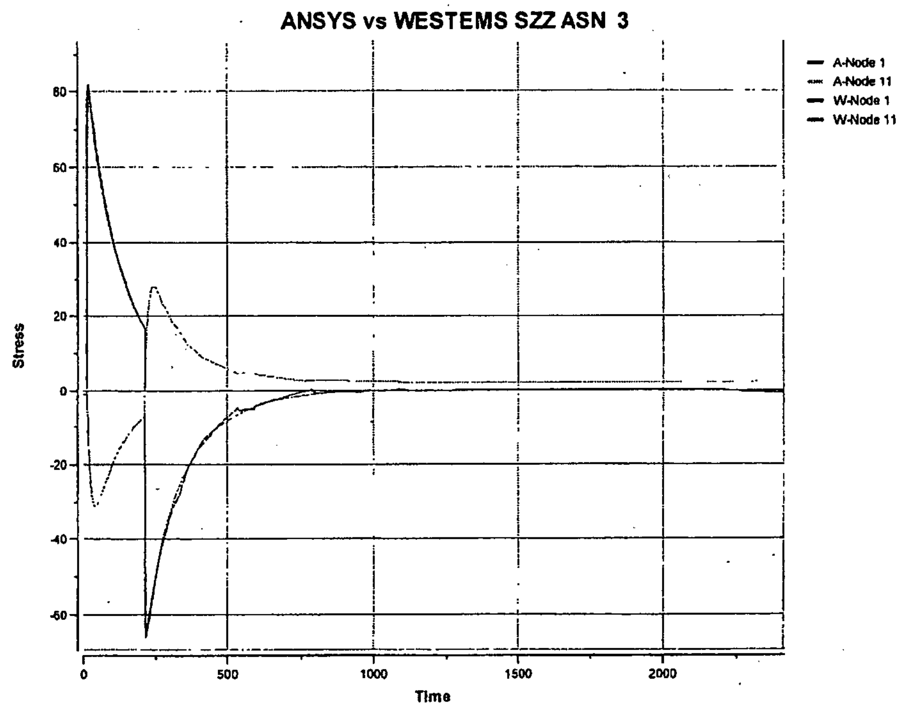


Figure B-6: ASN 3 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

Harris Nuclear Plant License Renewal Audit Question and Response Database

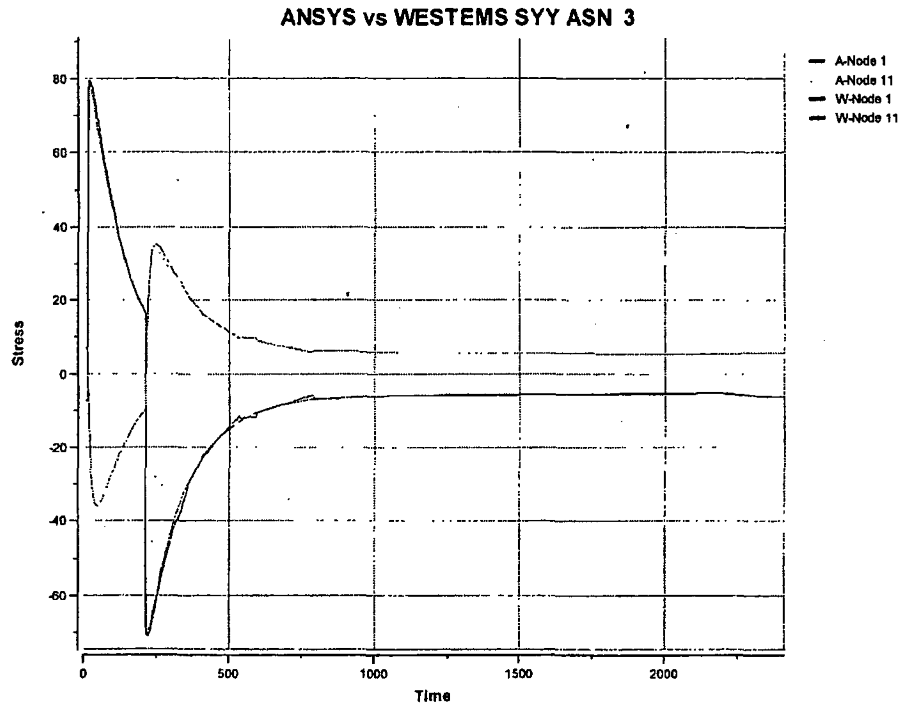


Figure B-7: ASN 3 Axial Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

Harris Nuclear Plant License Renewal Audit Question and Response Database

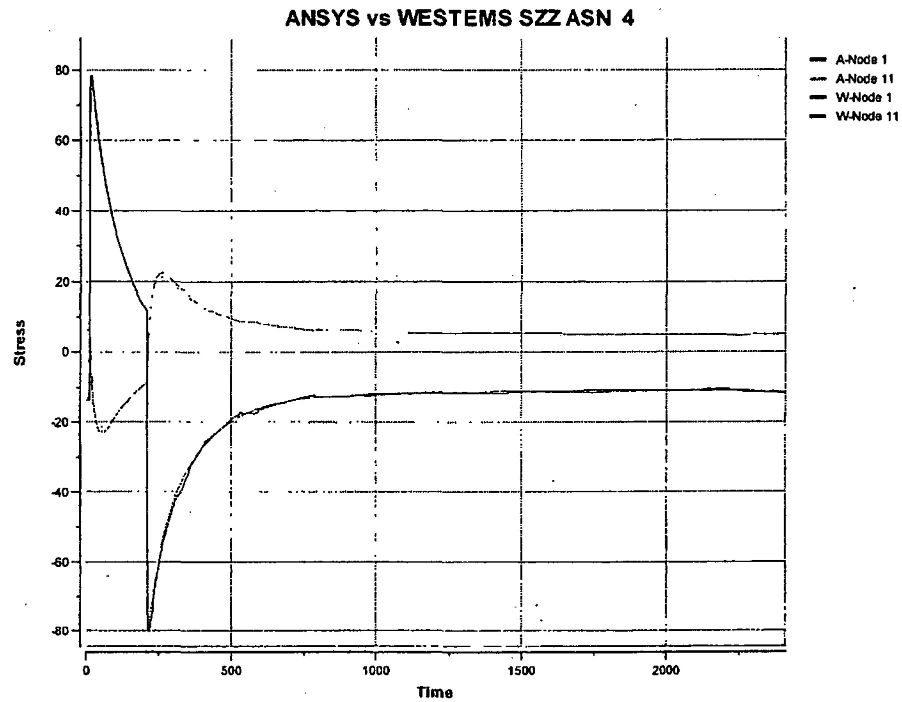


Figure B-8: ASN 4 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

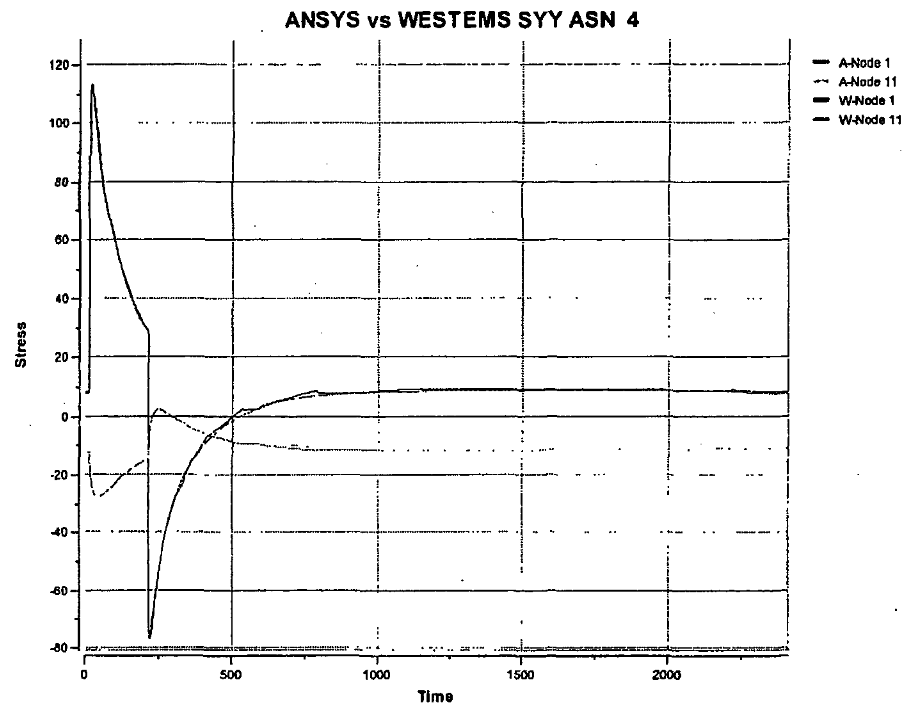


Figure B-9: ASN 4 Axial Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

Harris Nuclear Plant License Renewal Audit Question and Response Database

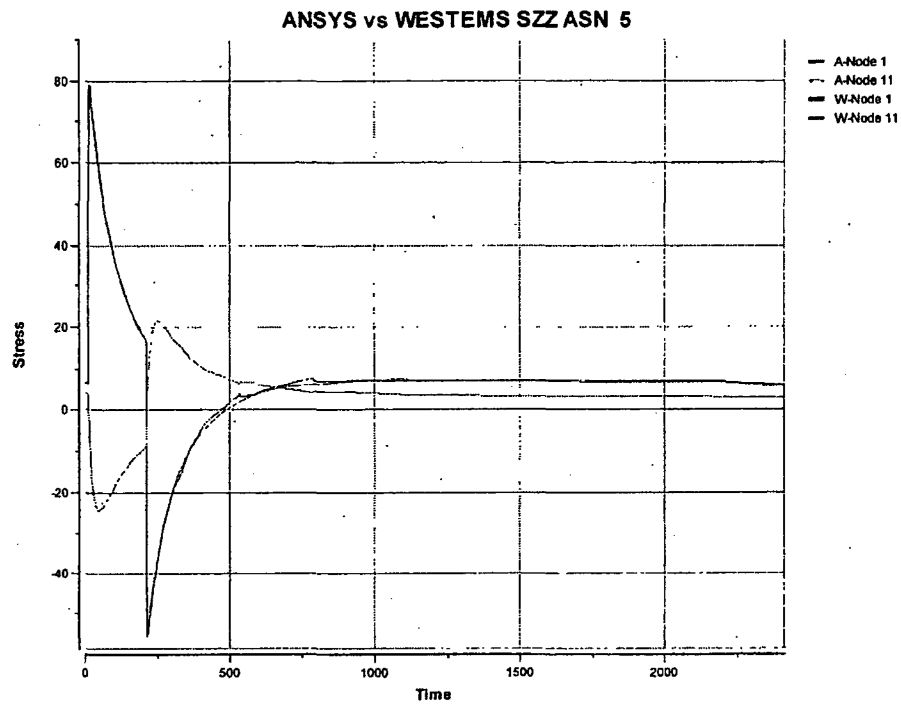


Figure B-10: ASN 5 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

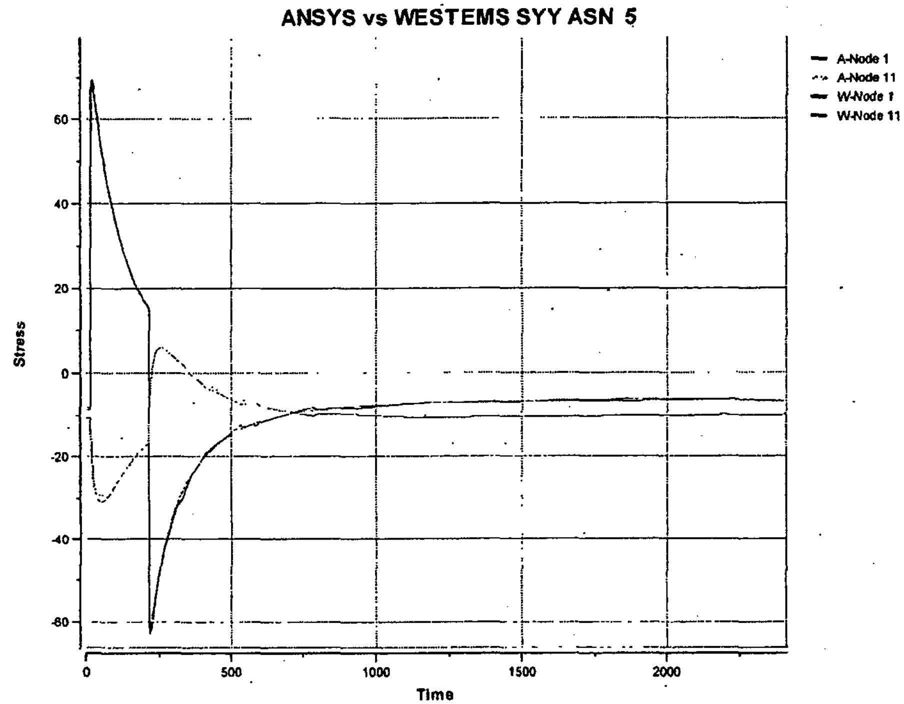
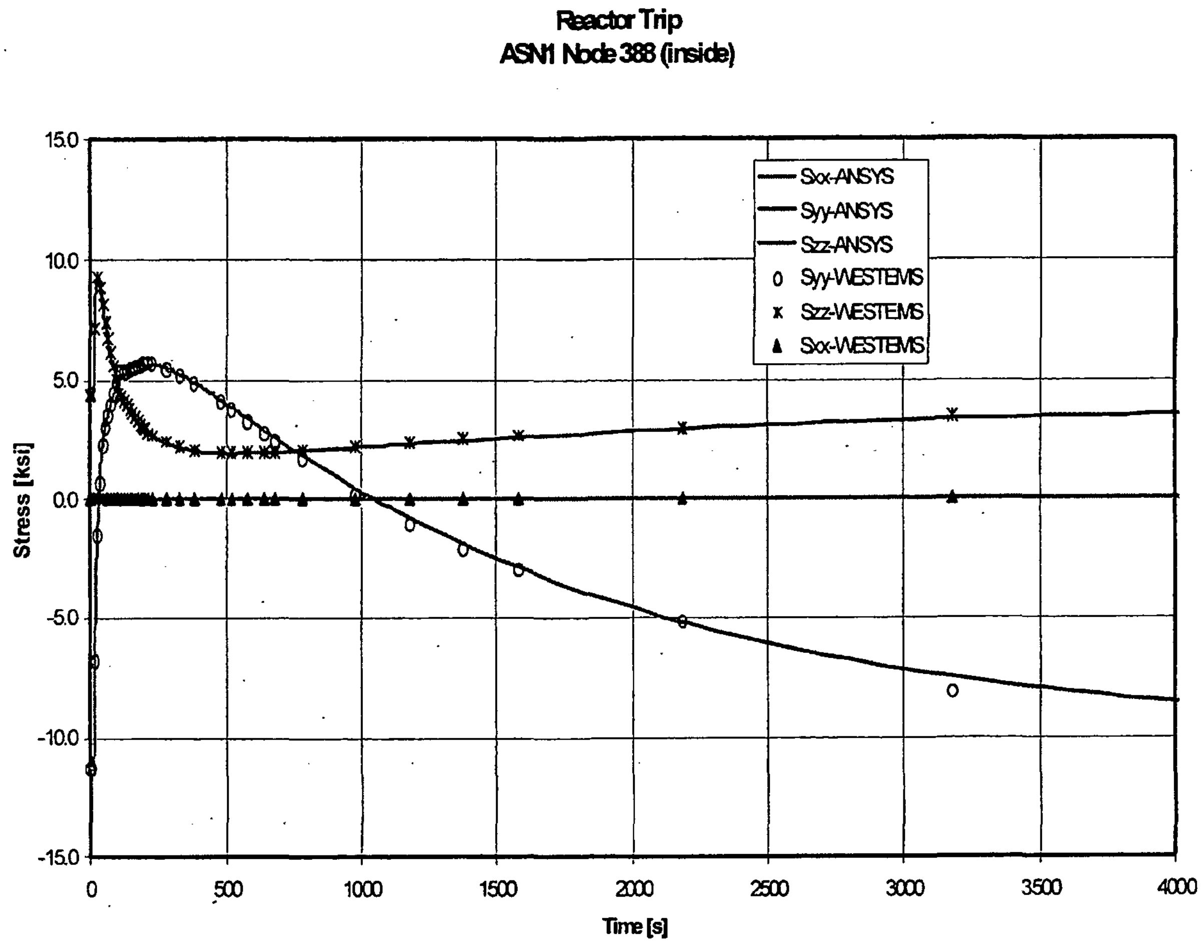


Figure B-11: ASN 5 Axial Stress Comparison (ANSYS vs. WESTEMS) for Benchmark Transient Loading.

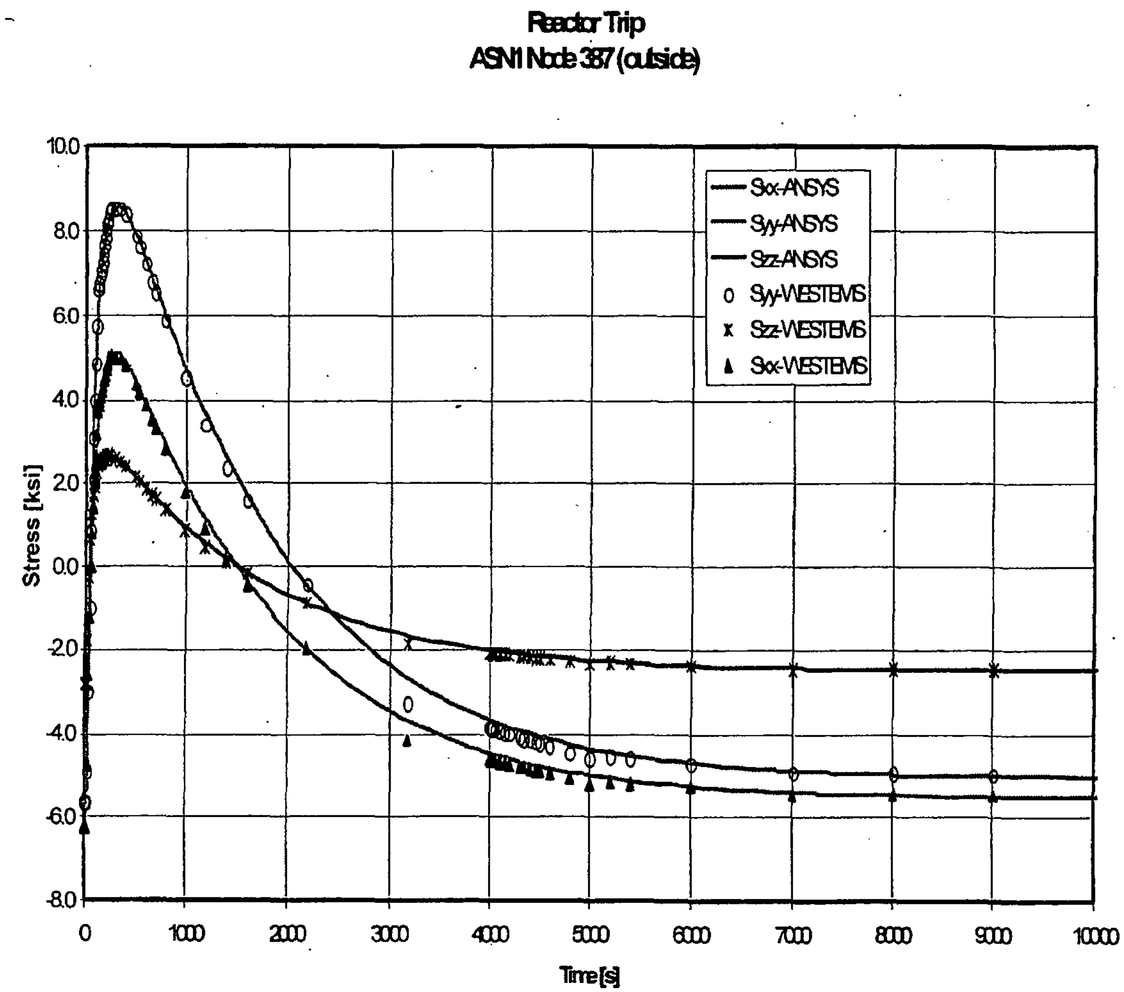
The results shown below, obtained by a WESTEMS user in a Westinghouse European site, serve additional verification of the transfer function methodology..

Harris Nuclear Plant License Renewal Audit Question and Response Database



Additional Thermal Stress Benchmark Results, Sample 1.

Harris Nuclear Plant License Renewal Audit Question and Response Database



Additional Thermal Stress Benchmark Results, Sample 2.

References:
1. "Transfer Function Method for thermal Stress and Fatigue Analysis: Technical Basis", WCAP-12315, Westinghouse Proprietary Class 2, C. Y. Yang, May 1990.

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 11

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

July 1, 2008

MEMORANDUM TO: ACRS MEMBERS

FROM: Peter Wen, Sr. Staff Engineer **/RA/**
Reactor Safety Branch - A
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS PLANT LICENSE RENEWAL SUBCOMMITTEE MEETING REGARDING SHEARON HARRIS NUCLEAR POWER PLANT ON MAY 7, 2008- ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued on June 27, 2008, have been certified as the official record of the proceedings for that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc via e-mail: ACRS Staff Engineers
S. Duraiswamy
J. Flack

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

June 27, 2008

MEMORANDUM TO: Peter Wen, Senior Staff Engineer
Reactor Safety Branch – A
ACRS

FROM: John Stetkar, Chairman,
Plant License Renewal Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE SUBCOMMITTEE ON PLANT LICENSE RENEWAL REGARDING SHEARON HARRIS NUCLEAR POWER PLANT ON MAY 7, 2008, IN ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on May 7, 2008, are an accurate record of the proceedings for that meeting.

__/RA/____________________**6/27/2008**____
J. Stetkar, Date
Plant License Renewal Subcommittee Chairman

Certified By: J. Stetkar
Certified on June 27, 2008

Issued on: July 1, 2008

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF THE MEETING OF THE SUBCOMMITTEE ON PLANT LICENSE RENEWAL REGARDING SHEARON HARRIS NUCLEAR POWER PLANT
MAY 7, 2008,
ROCKVILLE, MARYLAND

INTRODUCTION

On May 7, 2008, the ACRS Subcommittee on Plant License Renewal held a meeting regarding Shearon Harris Nuclear Power Plant (HNP) in Room T-2B3, 11545 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to discuss the HNP application for license renewal and NRC staff review of it. In addition to the NRC staff, representatives from Carolina Power & Light (CP&L) (the HNP operator and the licensee) made presentations to the Committee. The meeting was convened at 10:30 a.m. and adjourned at 3:08 p.m.

ATTENDEES

ACRS Members

John Stetkar, Subcommittee Chairman	Otto Maynard, Member
William Shack, Member	Mario Bonaca, Member
Said Abdel-Khalik, Member	John Sieber, Member
John. Barton (Consultant)	Christopher Brown (DFO)
Peter Wen, Cognizant Staff Engineer	

Principal NRC Speakers

S. Lee, NRR	L. Lund, NRR	M. Heath, NRR	K. Chang, NRR
C. Julian, Region II			

HNP Presenters

C. Burton	J. Caves	R. Stewart	C. Mallner
D. Corlett			

Other Attendees

NRC Staff		HNP	OTHER
R. Hsu	B. Rogers	R. Reynolds	A. Saunders
S. Sakai	L. Lake	M. Heath	W. Lunceford
R. Matthew	S. Jones	M. Fletcher	J. Hilbish
Y-K Chung	B. Parks		P. Ghosal
G. Cheruvenki	Z. Xi	**OTHER**	L. Bohn
K. Green	F. Saba	S. Kim	J. Tweddell
D. Nguyen	K. Howard	M. Fallin	C. Myer
J. Fair	M. Sayoc	C. Custer	
R. Gullucci	Q. Gan	K. Putnan	

stainless tank, sits on a concrete platform, and there is an enclosure around the tank. The one-time inspection program will be implemented to perform the aging management-related inspection. Mr. Dave Corlett added that HNP operators perform the normal rounds of looking into the enclosure area at least once per day.

AMP Exceptions

- Member Bonaca asked the applicant to characterize the nature of exceptions that the HNP are taking in its AMPs. Mr. Roger Stewart of HNP staff replied that the majority of HNP's exceptions to the GALL Report were due to either ASME Code edition or revision of EPRI guidelines or in one case, due to the revision of NEI 97-06. Mr. Chris Mallner of HNP staff added that in a few cases, some exceptions were taken because of the GALL Report's inadequate, prescriptive description. He gave an example of the Brinell hardness testing, which was specifically recommended in the GALL Report XI, M33, Selective Leaching Program. He pointed out that almost all the applicants took this exception because the Brinell hardness testing could be problematic. Mr. Roger Stewart of HNP staff briefly discussed the other exceptions contained in the Fuel Oil Chemistry Program, the One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program, and the Electric Cable Connections (E-6) Program.

Metal Fatigue TLAA

- Member Maynard asked the applicant to discuss metal fatigue issues that were identified in recent staff's review of several other LRAs. Mr. Chris Mallner of HNP staff replied that those technical issues are centered around the "1-D stress" methodology, which was adopted by one vendor, used in its on-line fatigue monitoring software, and was used in some plants' LRAs. The staff's concern is that the simplified "1-D stress" methodology may not provide conservative results consistently. Mr. Mallner stated that the HNP uses Westinghouse's "WESTEMS" for the fatigue analysis software, which is different from the "1-D stress" methodology. This WESTEMS software uses six stress tensors to calculate the stress intensity for the fatigue evaluation, which is consistent with the methodology described in the ASME Code. Therefore, the metal fatigue issue discussed in the previous reviews of other plants' LRA does not apply to the HNP license renewal.

- Mr. Robert Hsu of NRC staff described the benchmark of the software (WESTEMS) used by the HNP. He presented a slide (Slide #30) which showed excellent agreement of calculated stresses at one node between the results from WESTEMS and ANSYS, a well-known stress analysis computer software. Member Abdel-Khalik asked why showing agreement at one node location was a representative of all other locations. Mr. Chris Mallner of HNP staff replied that HNP generated not just one, but about 18 different plots to benchmark the WESTEMS results. Member Maynard asked whether the other plots also showed good agreement. Dr. Ken Chang of the staff replied that all plots showed good agreement of WESTEMS and ANSYS results, with the calculated component stress intensity comparison within plus/minus half a percent.

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 12

FENOC

FirstEnergy Nuclear Operating Company

Mark A. Manoleras
Director Site Engineering

724-682-5101
Fax: 724-682-1840

July 11, 2008
L-08-209

10 CFR 54

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Beaver Valley Power Station, Unit Nos. 1 and 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Reply to Request for Additional Information for the Review of the Beaver Valley Power Station, Units 1 and 2, License Renewal Application (TAC Nos. MD6593 and MD6594), and License Renewal Application Amendment No. 15

Reference 1 provided the FirstEnergy Nuclear Operating Company (FENOC) License Renewal Application (LRA) for the Beaver Valley Power Station (BVPS). Reference 2 requested additional information from FENOC regarding the BVPS license renewal integrated plant assessment in Sections B.2.27, 4.3, and 4.7.4 of the BVPS LRA.

Attachment 1 provides the FENOC reply to the U.S. Nuclear Regulatory Commission request for additional information. Attachment 2 provides the Regulatory Commitment List. Enclosure A provides Amendment No. 15 to the BVPS License Renewal Application. Enclosure B provides a copy of Westinghouse letter FENOC-08-109, "FirstEnergy Nuclear Operating Company, Beaver Valley Unit 1 and 2, Responses to NRC RAIs Regarding Pressurizer Surge Line Environmental Fatigue," Revision 1, dated June 25, 2008.

If there are any questions or if additional information is required, please contact Mr. Clifford I. Custer, Fleet License Renewal Project Manager, at 724-682-7139.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 11, 2008.

Sincerely,

Mark A. Manoleras

A108
NRR

References:
1. FENOC Letter L-07-113, "License Renewal Application," August 27, 2007.
2. NRC Letter, "Request for Additional Information for the Review of the Beaver Valley Power Station, Units 1 and 2, License Renewal Application (TAC Nos. MD6593 and MD6594)," May 28, 2008.

Attachments:
1. Reply to Request for Additional Information Regarding Beaver Valley Power Station, Units 1 and 2, License Renewal Application, Sections B.2.27, 4.3, and 4.7.4
2. Regulatory Commitment List

Enclosures:
A. Amendment No. 15 to the BVPS License Renewal Application
B. Westinghouse letter FENOC-08-109, "FirstEnergy Nuclear Operating Company, *Beaver Valley Unit 1 and 2*, Responses to NRC RAIs Regarding Pressurizer Surge Line Environmental Fatigue," Revision 1, dated June 25, 2008.

cc: Mr. K. L. Howard, NRC DLR Project Manager
Mr. S. J. Collins, NRC Region I Administrator

cc: w/o Attachments or Enclosures
Mr. B. E. Holian, NRC DLR Director
Mr. D. L. Werkheiser, NRC Senior Resident Inspector
Ms. N. S. Morgan, NRC DORL Project Manager
Mr. D. J. Allard, PA BRP/DEP Director
Mr. L. E. Ryan, PA BRP/DEP

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

Mr. Cliff Custer
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
P. O. Box 4
Shippingport, PA 15077

Direct tel: 412-374-5216
Direct fax: 412-374-2252
e-mail: blanchk@westinghouse.com

W Sales Order: 49098 LI 50
FENOC P.O. No.: 55106227

Our ref: FENOC-08-109, Revision 1

June 25, 2008

Note: Revision 1 is being issued to change the document to Westinghouse Non-Proprietary Class 3

FirstEnergy Nuclear Operating Company
Beaver Valley Unit 1 and 2
Responses to NRC RAIs Regarding Pressurizer Surge Line Environmental Fatigue

Dear Mr. Custer:

Attached are the Westinghouse inputs to the BVPS responses for the following NRC RAIs concerning pressurizer surge line environmental fatigue evaluations: RAI B.2.27-3, RAI 4.3-3 (b) and (c).

Should you have any questions, please feel free to contact Mr. Charlie Meyer at (724) 722-6017, or me at (412) 374-5216.

Regards,
WESTINGHOUSE ELECTRIC COMPANY

K. Blanchard
Customer Project Manager

with attachment

cc: BVRC Central File, SEB-1
Larry Hinkle – (FENOC)
Steve Buffington - (FENOC)

bcc: K. Blanchard – Energy Center
N. B. Closky – Energy Center
R. R. Jewell – Energy Center
C. Meyer – Waltz Mill
M. A. Gray – Waltz Mill

Responses to NRC Requests for Additional Information Regarding Pressurizer Surge Line Environmental Fatigue

RAI-B.2.27-3:

During the on-site audit, the applicant stated "the surge line to hot leg nozzle, for BVPS units 1 and 2, is included in a stress and fatigue model to be used in an on-line monitoring system (WESTEMS)" ...

b. Please provide the benchmarking results for the WESTEMS software using relevant transient data, and proper 3-D model. Please justify the use of WESTEMS™ to update the CUF calculation by using the monitored or projected transient data (cycles) and discuss the conservatisms in the calculation on a plant specific basis.

Westinghouse Input to Response:

The following provides the benchmarking results for WESTEMS™.

WESTEMS™ uses the transfer function method (TFM) [reference 1] to calculate six (6) components of stresses due to time varying mechanical and thermal loads. Time varying component stresses are calculated through wall as a function of the time varying mechanical and thermal boundary conditions. The resulting through wall stress components are processed and categorized according ASME Section III, Division 1, Subsection NB criteria. The processing first involves the calculation and categorization of the membrane, bending, and peak categories of mechanical and thermal stresses. These calculations are performed at the stress component levels, for each time step and for each applied loading type. The resulting stresses are then added to form the total stress and primary plus secondary stress according to ASME rules. Stress peak selection for fatigue evaluation purposes is based on analysis of the total stress time history and of the primary plus secondary stress time history. Both total stress and primary plus secondary stress are retained for future consideration in online fatigue evaluations. The discussion below will help to clarify the transfer function methodology, the transfer function database role, and provide an example of the current benchmarking process.

The transfer function method is a mathematical device that is capable of quantifying the effects experienced by a system due to an external disturbance, or excitation, with the aid of a characteristics function known as transfer function. In essence, the transfer function method is a means that correlates time-dependent behavior, in terms of input and output, of a system as seen in the thermal and dynamic problems. Examples of "disturbance" are mechanical forces, thermal transients, etc. Examples of "effects" include stresses, strains, displacements, temperature, etc. For typical structural applications, the "disturbance" can be surface temperature changes T(t), pressure P variation, forces (Fx, Fy, Fz), and moments (Mx, My, Mz) in a structural body (in vector notations: $\vec{F}$, and $\vec{M}$), whereas typical "effects" refer mostly to the stresses, displacements and metal interior temperatures.

In WESTEMS™, the transfer function methodology uses 2 or more unit load databases that have 4 or 6 components of stress depending on the nature of the original finite element model method that was used. If a two dimensional finite element model was used to create the transfer function database, then 4 components of stress are applicable (Sxx, Syy, Szz, Sxy). If a three dimensional model was used, then there are 6 components of stress in the transfer function databases (Sxx, Syy, Szz, Sxy, Syz, and Szx). The total number of stress states in the transfer function databases is dependent on the complexity of the thermal and mechanical boundary conditions being simulated.

For thermal applications, the transfer function is a characteristics function of a thermal-mechanical system. The characteristics include geometry, boundary conditions, insulation conditions, material properties, and thermal zones. These characteristics are all built into the transfer function for a predefined thermal-mechanical system. Therefore, a transfer function database is fixed for a particular type of thermal-mechanical problem. However, a single set of transfer function databases can be used to evaluate the system responses caused by any kind of transients. This means that a transfer function database is created only once but can be used to obtain solutions for unlimited numbers of transient cases.

It is important to realize that thermal stresses in materials or any structural systems arising from temperature transients are evolving because heat transfer is an energy transport process that will continue until thermal equilibrium is established. This means that it requires an appreciable amount of time for a thermally disturbed material or structural system to come to a steady state, even if the disturbance is as brief as an impulse. In short, a thermal transient is a time-dependent problem. On the contrary, all mechanical loads, pressure, direct forces, and moments, encountered in the general structural applications are treated as static problems, unless the loading rates are so high that the dynamic effects cannot be ignored. To appropriately reflect the types of loads being dealt with, the databases are split into two types:

- Thermal transfer Function DataBase (TFDB)
- MEchanical transfer function DataBase (MEDB).

Westinghouse has validated the thermal stress capability of the WESTEMS™ transfer function method by performing identical analyses using the Westinghouse transfer function method and an independent finite element program like ANSYS or WECAN. Examples of the predicted stress component results for benchmarking the transfer function models are shown below. The benchmarking process is generally performed for every transfer function database created. The following example was taken directly from the appendix of a Westinghouse Transfer function database calculation note. The verification of WESTEMS™ thermal and mechanical stress calculations have been performed in the program's verification and validation documentation. However, each application verification of the finite element models and of the final thermal transfer function databases should be performed to show applicability to the problem being modeled. To do this for mechanical loads, Westinghouse verifies the finite element model results by comparing them to the expected theoretical values. For the time varying thermal results, Westinghouse performs thermal stress analyses using both the finite element program and WESTEMS™. The example below shows these comparisons and results. Certain information has been removed and text has been modified in order to clarify the example, which is taken from reference 2.

Verification of the Surge Line Hot Leg Nozzle TFDB and MEDB Databases

Verification of the databases being used for the WESTEMS™ analyses is a required step to ensure good analysis results. All databases are herein examined through suitable benchmarking problems.

The database files, TFDB and MEDB, generated in the unit load finite element analyses, represent the thermal and mechanical characteristics of the structural component considered. By using these databases, the stresses at the specified analysis sections (ASN or cut) can be evaluated for any combination of load conditions. To correctly produce the results, each load type requires an appropriate scaling factor, which is being developed in the following subsections. The scaling factor provides a means to correct the effects arising from differences in the stress units used in ANSYS and in WESTEMS™. It also is a means that permits non-standard unit loads to be used to generate the database.

Verifying the Bending Moment Database – M_x

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the bending moment about the x-axis portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

Moment M_x represents bending about the global x-axis. The analysis for this bending case was performed using an ANSYS model and documented. The applied moment is 1000 in-kips.

Consider the well known bending stress equation

$$\sigma = Mr/I$$

where M is the applied bending moment, r is distance from the neutral axis, and I is the moment of inertia of the cross sectional area.

Two nodes, as listed in Table B-1, are considered to benchmark/verify the results. These nodes are both on the surge line pipe section of the model (one on the inside and one on the outside diameter) and are remote from the reinforced section of the nozzle. Therefore, the above bending equation can be applied.

At this location, the following data apply:

Ro = 7.0 in.

Ri = 5.594 in.

$I = \pi (Ro^4 - Ri^4) / 4 = 1116.6 \text{ in}^4$

Comparisons of the ANSYS FE and analytical results are shown in Table B-1. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi, whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied moment is in-kips, whereas 1000 in-kips of bending was used in the database creation, a second scaling factor, f_2=0.001, is required. Combining the two, the scaling factor for the bending load to be used for WESTEMS™ analyses is found to be $fb = f_1 * f_2 = 10^{-6}$.

Table B-1: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	error (%)
Inside node	28106	5010	-5010	0.0
Outside node	26787	6269	-6269	0.0

Note: The sign of stress produced by ANSYS was negative since "CUT4" is in compression due to the direction of moment loading in ANSYS.

Verifying the Torsion Database – My

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the torsion moment portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

The moment M_y represents the moment (or twist) about the global y-axis. The analysis for this torsion case was performed in ANSYS and documented. The applied moment is 1000 in-kips.

Consider the well known torsion shearing stress equation

$$\tau = Mr/J$$

where M is the applied torque, r is distance from the neutral axis, and J is the polar moment of inertia in torsion of the cross sectional area.

Two nodes, as listed in Table B-2, are considered to benchmark/verify the results. These nodes are both on the surge line pipe section of the model (one on the inside and one on the outside diameter) and are remote from the reinforced section of the nozzle. Therefore, the above equation can be applied.

At this location, the following data apply:

$R_o = 7.0$ in.

$R_i = 5.594$ in.

$J = \pi (D_o^4 - D_i^4) / 32 = 2233.3\ in^4$

Comparisons of the ANSYS FE and analytical results are shown in Table B-2. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi, whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied torque is in-kips, whereas 1000 in-kips of torque was used in the database creation, a second scaling factor, f_2=0.001, is required. Combining the two, the scaling factor for the torsion load to be used for WESTEMS™ analyses is found to be $f_t = f_1 * f_2 = 10^{-6}$.

Table B-2: Comparison of ANSYS and Analytical Results

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	error (%)
Inside node	28106	2504.82	2531.80	-1.08
Outside node	26787	3134.39	3131.90	0.08

Verifying the Bending Moment Database – Mz

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the bending moment about the z axis portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

Moment M_z represents bending about the global z-axis. The analysis for this bending case was performed in ANSYS and documented. The applied moment is 1000 in-kips.

Consider the well known bending stress equation

$$\sigma = Mr/I$$

where M is the applied bending moment, r is distance from the neutral axis, and I is the moment of inertia of the cross sectional area.

Two nodes, as listed in Table B-3, are considered to benchmark/verify the results. These nodes are both on the surge line pipe section of the model (one on the inside and one on the outside diameter) and are remote from the reinforced section of the nozzle. Therefore, the above bending equation can be applied.

At this location, the following data apply:

$R_o = 7.0$ in.

$R_i = 5.594$ in.

$I = \pi (R_o^4 - R_i^4) / 4 = 1116.6 \text{ in}^4$

Comparisons of the ANSYS FE and analytical results are shown in Table B-3. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi, whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied moment is in-kips, whereas 1000 in-kips of bending was used in the database creation, a second scaling factor, f_2=0.001, is required. Combining the two, the scaling factor for the bending load to be used for WESTEMS™ analyses is found to be $f_b = f_1 * f_2 = 10^{-6}$.

Table B-3: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	error (%)
Inside node	28093	5010	-5010	0.0
Outside node	26754	6269	-6268	0.0

Verifying the Pressure Database

A benchmarking problem is considered here, which serves two purposes: (1) to determine the scaling factor corresponding to the pressure portion of the database, and (2) to verify the database created. This process ensures the correctness of the results produced by WESTEMS™.

The analysis for the pressure loading case was performed in ANSYS and documented. The applied pressure is 1000 psi.

Consider the well known hoop stress equation for a pressurized pipe:

$$\sigma_\theta = \frac{p R_i^2}{R_o^2 - R_i^2}\left(1 + \frac{R_o^2}{r^2}\right)$$

where p is the internal pressure, R_o is the outside radius, R_i is the inside radius, and r is the radius at any point.

Two nodes, as listed in Table B-4, are considered to benchmark/verify the results. These nodes are both on the surge line pipe section of the model (one on the inside and one on the outside diameter) and are remote from the reinforced section of the nozzle. Therefore, the above equation can be applied. At this location, the following data apply:

R_o = 7.0 in.

R_i = 5.594 in.

Comparisons of the ANSYS FE and analytical results are shown in Table B-4. The results are in good agreement. The scaling factor, which depends on the benchmarking results, the stress units used in FE and WESTEMS™, and the unit of the input load for the WESTEMS™ analysis, can now be determined. Since the stress unit in the ANSYS FE results is psi, whereas the stress unit to be used for WESTEMS™ calculations is ksi, a required scaling factor is f_1=0.001. Since the unit of the applied pressure is psi, whereas 1000 psi of pressure was used in the database creation, a second factor, f_2=0.001, is required. Combining the two factors, the scaling factor for the pressure load to be used for WESTEMS™ analyses is found to be $f_p = f_1 * f_2 = 10^{-6}$.

Table B-4: Comparison of ANSYS and Analytical Results.

Hand Calculation Comparison				
		Analytical	ANSYS	
Location	Node Number	Stress (psi)	Stress (psi)	error (%)
Inside node	28093	4534.48	4756.50	-4.9
Outside node	26754	3534.48	3427.00	3.0

Verifying the Thermal Stress Database

Two benchmarking problems are considered here, which serve two purposes: (1) to determine the scaling factor corresponding to the transfer function thermal stress database, and (2) to verify the database created.

To benchmark and verify this portion of the database and determine the appropriate scaling factor for the thermal loads, an arbitrary shock transient and a stratification transient were used. The transients used for this benchmarking problem are defined in the data shown in Table B-5 and Table B-6.

Table B-5: The Transient Used for the Thermal Shock Benchmarking Case

Time (seconds)	Temperature -Zone1 through Zone10 (°F)	Heat Transfer Film Coefficient - hzone1 through hzone10 (Btu/hr-ft^2-°F)
0	100	8000
1	100	8000
10	100	8000
20	100	8000
21	500	8000
22	500	8000
23	500	8000
24	500	8000
25	500	8000
26	500	8000
27	500	8000
28	500	8000
30	500	8000
31	500	8000
40	500	8000
50	500	8000
60	500	8000
75	500	8000
85	500	8000
95	500	8000
110	500	8000
125	500	8000
160	500	8000

Time (seconds)	Temperature -Zone1 through Zone10 (°F)	Heat Transfer Film Coefficient - hzone1 through hzone10 (Btu/hr-ft^2-°F)
210	500	8000
410	500	8000
710	500	8000
1000	500	8000
2000	500	8000
4000	500	8000

Table B-6: The Transient Used for the Thermal Stratification Benchmarking Case

Time (seconds)	"Nozzle Top" Temperature - Zones 1, 3, 5, 7, 8, and 9 (°F)	"Nozzle Bottom" Temperature - Zones 2, 4, 6, and 10 (°F)	Heat Transfer Film Coefficient - hzone1 through hzone 10 (Btu/hr-ft^2-°F)
0.001	110	110	275
19	110	110	275
21	430	110	275
22	430	110	275
23	430	110	275
24	430	110	275
25	430	110	275
26	430	110	275
27	430	110	275
28	430	110	275
29	430	110	275
30	430	110	275
31	430	110	275
32	430	110	275
35	430	110	275
40	430	110	275
45	430	110	275
50	430	110	275
55	430	110	275
60	430	110	275
65	430	110	275
70	430	110	275
75	430	110	275
80	430	110	275
85	430	110	275
90	430	110	275
95	430	110	275
100	430	110	275

Time (seconds)	"Nozzle Top" Temperature - Zones 1, 3, 5, 7, 8, and 9 (°F)	"Nozzle Bottom" Temperature - Zones 2, 4, 6, and 10 (°F)	Heat Transfer Film Coefficient - hzone1 through hzone 10 (Btu/hr-ft^2-°F)
105	430	110	275
110	430	110	275
115	430	110	275
120	430	110	275
140	430	110	275
150	430	110	275
160	430	110	275
210	430	110	275
410	430	110	275
610	430	110	275
810	430	110	275
1210	430	110	275
1610	430	110	275
2500	430	110	275
5000	430	110	275

In the shock benchmarking transient, all zones undergo a severe thermal shock. In the stratification benchmarking transient, the surge line piping undergoes stratification with a temperature change of 320 °F. The transients are intentionally made severe on the temperature rate so as to allow a vigorous examination of the integrity of the transfer function database. This set of transients was analyzed by both WESTEMS™ and ANSYS. Note that the ANSYS results represent full finite element analyses whereas the WESTEMS™ results are produced by the transfer function method, which utilizes the transfer function databases produced by ANSYS.

The results, as shown in Figures B-1 through B-4 for the controlling location of the nozzle (units: ksi for stress, seconds for time), are then graphically compared on both the shapes and the magnitudes. It can be seen from these figures that the WESTEMS™ results compare very well with those calculated by ANSYS, both in magnitudes and curve shapes for both the shock transient loading and the stratification transient loading.

The shapes of the curves of the stresses from the WESTEMS™ analysis are visually compared with those from the ANSYS full finite element analysis. In general, good comparisons are observed for all cases. The stratification case shows slight differences in stress magnitude in the steady state stratification condition, which is expected. This is caused by the inside surface film coefficients changing values between zones, which is accounted for in ANSYS by two-dimensional heat transfer, but is not fully accounted for in the WESTEMS™ benchmark run. The results from WESTEMS™ predict slightly higher stresses at the stratified steady state condition, which therefore leads to conservative answers and is not considered a concern.

Overall, very good benchmarking results have been achieved, which assures good results can be produced through the TFDB created. Since WESTEMS™ results are either close in magnitude or slightly higher than the ANSYS benchmark results, the factor, f_l=1.00, is applied. Since the stress unit in the ANSYS FEA results is psi, whereas the stress unit to be used for WESTEMS™ calculations is ksi, a factor, f_2=0.001, is required. Combining the two factors, the scaling factor for the thermal load to be used for WESTEMS™ analyses is $f_T = f_l * f_2 = 0.001$, which is to be registered to the "TFDB_Factor" box in the WESTEMS™ ASN Analysis Models.

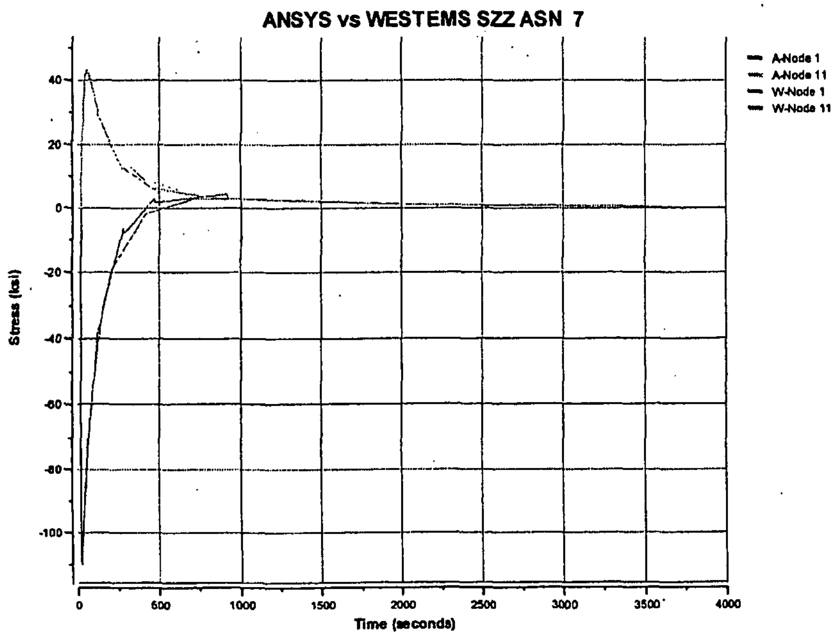


Figure B-1: ASN 7 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Shock Transient Loading

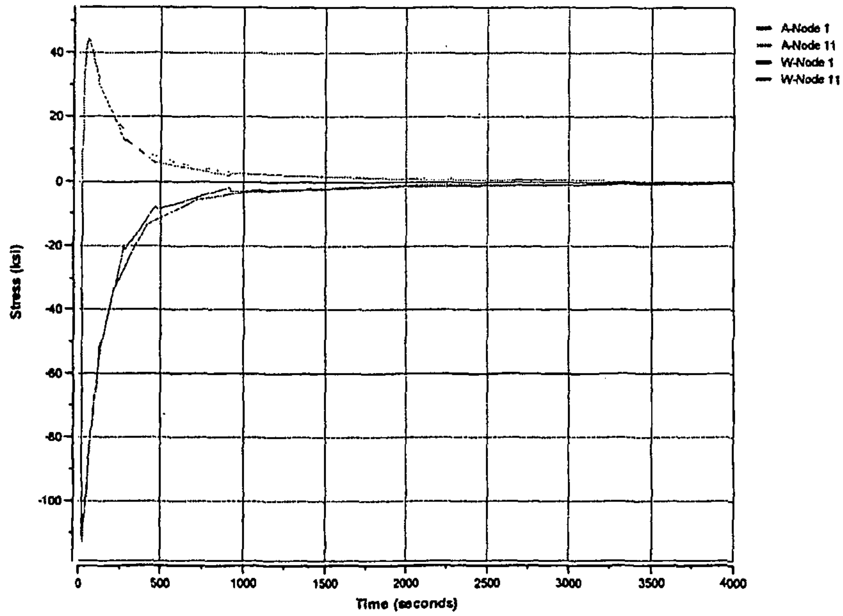


Figure B-2: ASN 7 Axial Stress Comparison (ANSYS vs. WESTEMS) for Shock Transient Loading

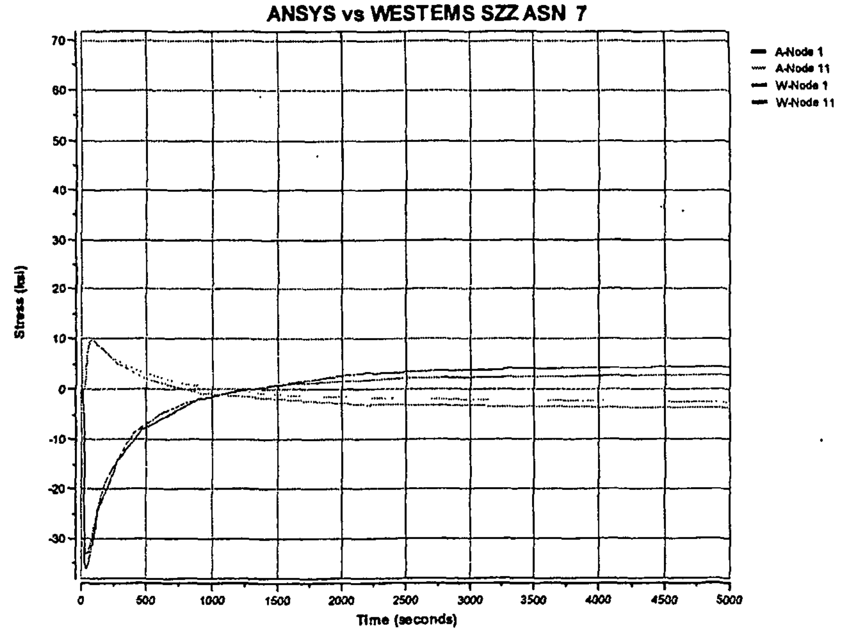


Figure B-3: ASN 7 Hoop Stress Comparison (ANSYS vs. WESTEMS) for Stratification Transient Loading

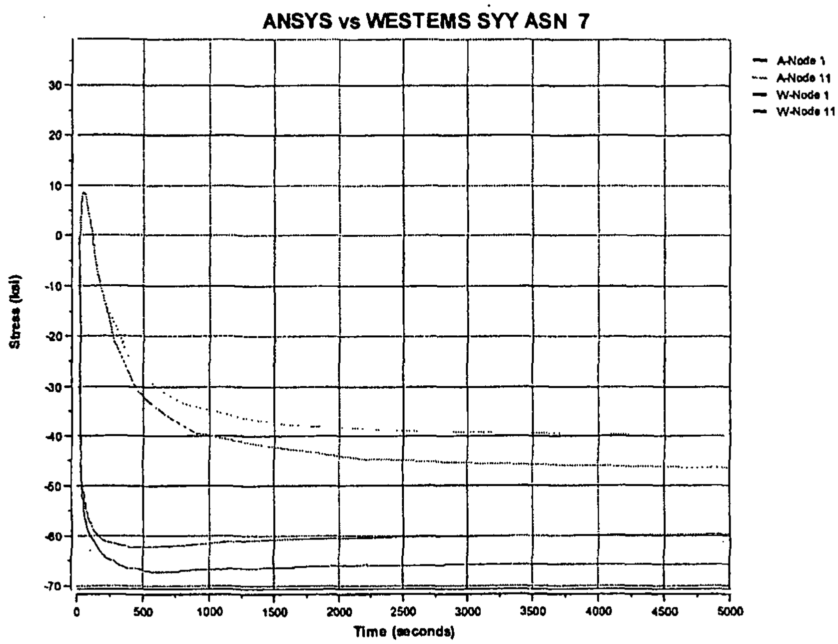
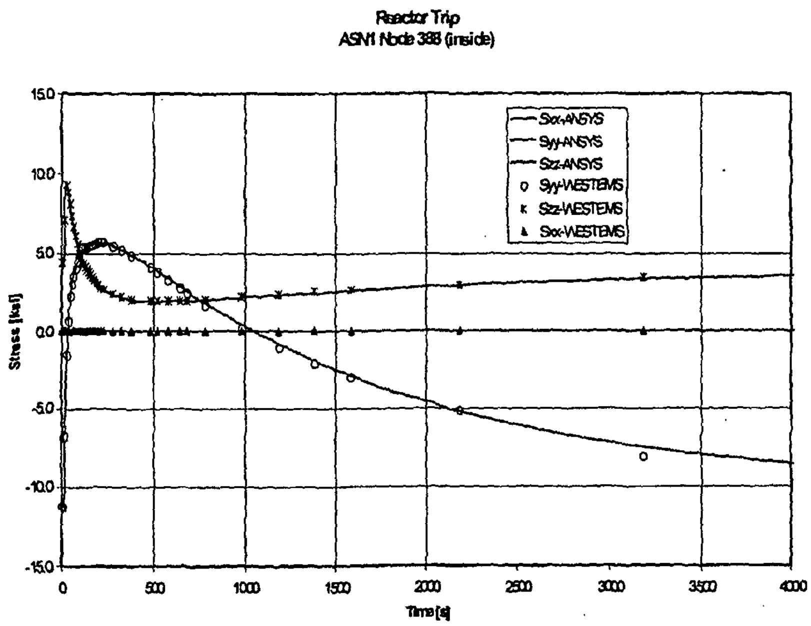
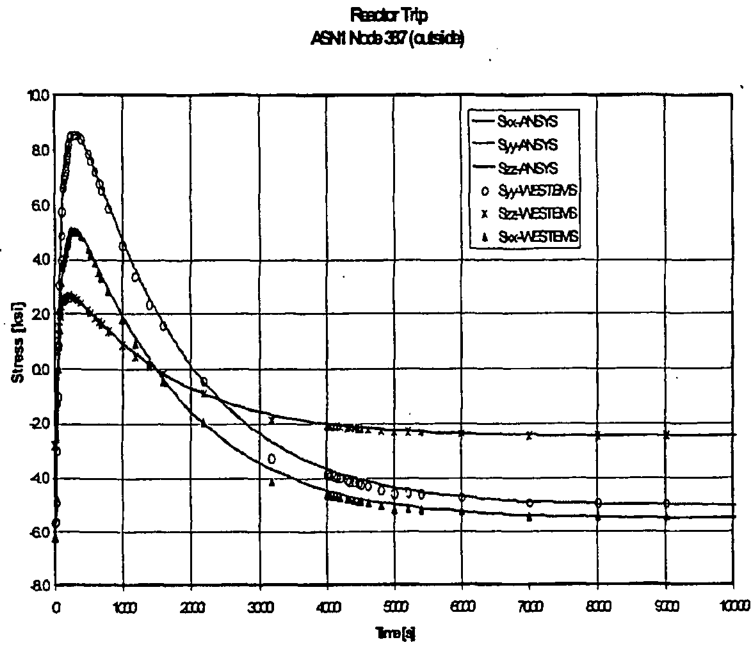


Figure B-4: ASN 7 Axial Stress Comparison (ANSYS vs. WESTEMS) for Stratification Transient Loading

The results shown below, obtained by a WESTEMS™ user in a Westinghouse European site, serve as *additional* verification of the transfer function methodology.



Additional Thermal Stress Benchmark Results, Sample 1.



Additional Thermal Stress Benchmark Results, Sample 2.

References:

1. "Transfer Function Method for thermal Stress and Fatigue Analysis: Technical Basis", WCAP-12315, Westinghouse Proprietary Class 2, C. Y. Yang, May 1990.
2. Westinghouse Calculation No. CN-PAFM-07-60, Rev. 0, "Beaver Valley Unit 2: Transfer Function Database Development for a 14-inch Hot Leg Surge Nozzle." S. F. Hankinson.

RAI-4.3-3 (b & c)

LRA Section 4.3.3.3 discusses the effects of primary coolant environment on fatigue life. During the audit, the applicant indicated that it will refine the analysis for NUREG/CR-6260 components in the near future. To assist the staff it its review:

b. Please explain how the calculations for the fatigue life correction factor (Fen), used to express the effects of the reactor coolant environment, will be performed. Specifically, how the transient pairs will be considered in the calculations.

c. Please describe the criteria and methodology that will be performed for the additional analyses in calculating the CUF, including environmental effects, for the components where the CUF exceeds the design code allowable value of 1.0.

Westinghouse Input to Response for Part b

For the surge line hot leg nozzle, reactor water environmental effects were evaluated by calculating F_{en} factors on fatigue usage using the general methodology in NUREG/CR-5704 for stainless steel.

According to this method, fatigue usage is calculated with environmental fatigue correction factors on each load pair incremental usage as:

$$U_{en} = U_1 * F_{en,1} + \ldots + U_i * F_{en,i} + \ldots + U_n * F_{en,n}$$

where $i = 1, 2, \ldots n$
U_i = incremental fatigue usage contribution, calculated according to NB-3222.4
$F_{en,i}$ = environmental fatigue penalty factor

For stainless steels, F_{en} is calculated as follows:

$$F_{en} = \exp[0.935 - T^*O^*\varepsilon'^*]$$

where T^* = transformed temperature

O^* = transformed oxygen content

ε'^* = transformed strain rate

The terms are explained below in detail.

Thresholds are defined where the following parameters for the pair are within the following ranges for stainless steel (per NUREG/CR-5704):

$T <= 200°C$

$\varepsilon' > 0.4\%/sec$

When one of these is satisfied, the negative term in the F_{en} equation above is zero, and the minimum value of F_{en} is calculated as:

$$F_{en} = \exp(0.935) = 2.547$$

A strain amplitude threshold is also discussed in NUREG/CR-5704 and clarified in NUREG/CR-6717 (ε_{amp} <= 0.10%), where the environmental effect is insignificant (F_{en} = 1.0) for the pair. This was not applied in the evaluation since pairs in this range did not have a significant effect on fatigue.

T* = transformed temperature

$T^* = 0 \qquad (T < 200°C)$

$T^* = 1.0 \qquad (T \geq 200°C)$

Where T = metal surface temperature of the component being considered

O* = transformed oxygen content

$O^* = 0.260 \qquad (DO < 0.05 \text{ ppm})$

$O^* = 0.172 \qquad (DO \geq 0.05 \text{ ppm})$

Where DO = dissolved oxygen (DO) content (ppm).

For PWRs, it is easily assumed that: DO < 0.05 ppm.

Therefore, O* = 0.260 for all cases for stainless steels.

ε'* = transformed strain rate, for stainless steels is:

$\varepsilon'^* = 0 \qquad (\varepsilon' > 0.4\%/\text{sec})$

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

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

This may be determined using various methods depending on the degree of conservatism retained for qualification.

A detailed integrated method was used to incorporate strain rate, called the modified rate approach, where the Fen is integrated over the strain range for the tensile strain producing cycle of the transient pair. The modified rate approach is represented below:

$$Fen = \frac{\sum Fen_i \Delta\varepsilon_i}{\sum \Delta\varepsilon_i}$$

where:

Fen_i = Fen computed for time interval i, based on $\varepsilon_i' = 100\Delta\varepsilon_i/\Delta t_i$ and transformed parameters (T*), (ε*), and (O*) computed for the interval

$\Delta\varepsilon_i$ = change in strain for time interval i, $(\sigma_i - \sigma_{i-1})/E$

σ_i = stress intensity for time i

σ_{i-1} = stress intensity for time i-1

Δt_i = change in time for time interval i, $\Delta t = t_i - t_{i-1}$

E = Young's Modulus

For load pairs that include dynamic OBE loading, a minimum Fen = 2.55 was used for the dynamic portion of the strain included in the pair. This was considered to be conservative, since dynamic load cycling occurs at a frequency that is too high for environmental effects to be significant, and an Fen = 1.0 could be justified. Based on this, when OBE occurred in a pair with a thermal transient, it was conservative to use the Fen determined based on the thermal transient only.

The stress cycle pairs obtained from the fatigue analysis of the safe end to pipe weld of the surge line hot leg nozzle were used to calculate F_{en} factors. The most dominant stress cycle pairs in these evaluations came from the heatup and cooldown transients. F_{en} for all stress cycle pairs was calculated using the F_{en} modified rate approach discussed above. This approach, integrating the F_{en} over the positive strain rate portions of the pair's history, resulted in F_{en} values for each stress cycle pair. After calculation of the appropriate F_{en} values for the respective stress cycle pairs, the final cumulative usage factor for the surge line hot leg nozzle with environmental effects was calculated by summing the corrected usage for each pair.

Westinghouse Input to Response for Part c (for surge line hot leg nozzles)

The surge line hot leg nozzle fatigue analyses were performed according to the detailed methods of ASME Code Section III, NB-3200, as permitted by the NB-3600 piping design section. The method used to evaluate the effects of reactor water environment on the ASME fatigue usage is discussed in the response to RAI 4.3-3.b. The NB-3200 evaluation was performed using program WESTEMS™.

Inputs to the fatigue evaluation were provided or confirmed by Beaver Valley engineering in a Design Information Transmittal (DIT), "DIT-WEST-ENV-02". The information provided included the design mechanical loads for Units 1 and 2. It also confirmed the applicability of thermal loads related to stratified and non-stratified conditions. The inputs were consistent with those used in the evaluations of surge line stratification in WCAP-12727, Supplement 1 for Unit 1 and WCAP-12093, Supplement 5 for Unit 2. *The DIT also provided primary stresses calculated separately from the fatigue evaluations* performed by Westinghouse.

Transients used in the fatigue evaluations were developed based on the design transients used in the original evaluations of surge line stratification, WCAP-12727 and WCAP-12093, and updated information on stratification loading developed from plant operating data. Transient input information was supplied and/or confirmed in the Beaver Valley DIT.

The fatigue evaluation followed the procedures given in the ASME Code Section III, NB-3200. Transient loadings representing the transients defined for the surge line hot leg nozzle were input to WESTEMS™ using binary "history files." The history files contain all the local parameter tagnames needed to calculate stress at the controlling locations, using the WESTEMS™ Derived Value functions and transfer functions. The methodology used to develop and benchmark transfer functions was discussed in the response to RAI B.2.27-3.

The stress ranges, cycle pairing and fatigue usage factors were calculated using WESTEMS™, consistent with the ASME Code as outlined by the steps below.

1. The stress histories were calculated for stress cuts (ASNs) in structural components subjected to thermal, pressure, and piping loads from the defined transients using the unit load transfer function databases. WESTEMS™ model information was used to calculate stress and related inputs for the fatigue evaluation based on ASME Section III, NB-3200, methodology. Stress component histories and stress component ranges were determined and used in the fatigue evaluations.

2. The stress peak and valley times were determined for each transient stress history, and associated stress component values at each selected time were input to the fatigue usage calculation.

3. The Primary plus Secondary Stress Intensity Ranges were calculated. Since the ASME Code fatigue curves are based on elastic stress results, adjustments to the alternating stress intensity range were required if this stress range exceeded the elastic range. In the ASME Code evaluation, the linearized primary plus secondary stress ranges are compared to the $3S_m$ allowable to determine if the elastic range is exceeded. If the 3Sm allowable is exceeded, then a Simplified Elastic-plastic analysis per NB-3228.5 is performed to obtain the appropriate adjustment factor (K_e).

4. Appropriate correction factors were calculated for each possible load set range pair formed from the stress components at each peak and valley time.

    a. Primary plus Secondary Stress Intensity Range (S_n) was compared to the 3Sm allowable. If $S_n > 3S_m$, the elastic-plastic penalty factor, K_e, for that pair was applied, in addition to evaluation of other requirements of NB-3228.5.

b. If $S_n \leq 3S_m$, then Poisson's ratio correction factor for Local Thermal stress was calculated according to NB-3227.6.

c. The elastic modulus correction factor, $E_{curve}/E_{analysis}$, was calculated according to NB-3222.4(e)(4).

5. For each load set pair, correction factors were applied, and final adjusted alternating stress, S_a, was determined. Cumulative fatigue usage was calculated using the method of NB-3222.4(e)(5) and the appropriate material fatigue curve.

The surge line hot leg nozzle environmental fatigue evaluations are documented in WCAP-16830-P for Unit 1 and WCAP-16867-P for Unit 2

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 13

American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III , Rules for Construction of Nuclear Facility Components, Division 1, Subarticle NB-3200 (Design by Analysis) (1998 Edition) available for purchase from the ASME (**http://www.asme.org/**)

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 14

NUREG/CR-5704
ANL-98/31

Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels

Argonne National Laboratory

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

NUREG/CR-5704
ANL-98/31

Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels

Manuscript Completed: November 1998
Date Published: April 1999

Prepared by
O. K. Chopra

Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

M. McNeil, NRC Project Manager

Prepared for
Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
NRC Job Code W6610

EFFECTS OF LWR COOLANT ENVIRONMENTS ON FATIGUE DESIGN CURVES OF AUSTENITIC STAINLESS STEELS

by

O. K. Chopra

Abstract

The ASME Boiler and Pressure Vessel Code provides rules for the construction of nuclear power plant components. Figures I-9.1 through I-9.6 of Appendix I to Section III of the Code specify fatigue design curves for structural materials. While effects of reactor coolant environments are not explicitly addressed by the design curves, test data indicate that the Code fatigue curves may not always be adequate in coolant environments. This report summarizes work performed by Argonne National Laboratory on fatigue of austenitic stainless steels in light water reactor (LWR) environments. The existing fatigue S-N data have been evaluated to establish the effects of various material and loading variables such as steel type, dissolved oxygen level, strain range, strain rate, and temperature on the fatigue lives of these steels. Statistical models are presented for estimating the fatigue S-N curves as a function of material, loading, and environmental variables. Design fatigue curves have been developed for austenitic stainless steel components in LWR environments. The extent of conservatism in the design fatigue curves and an alternative method for incorporating the effects of LWR coolant environments into the ASME Code fatigue evaluations are discussed.

hydrogen-induced cracking. Fatigue striations should not be observed if enhancement of crack growth is caused by the slip oxidation/dissolution process.

5 Statistical Model

The fatigue S-N curves are generally expressed in terms of the Langer equation,[6] which may be used to represent either strain amplitude in terms of life or life in terms of strain amplitude. The parameters of the equation are commonly established through least-squares curve-fitting of the data to minimize the sum of the square of the residual errors for either fatigue life or strain amplitude. A predictive model based on least-squares fit on life is biased for low strain amplitude. The model leads to probability curves that converge to a single value of strain, and fails to address the fact that at low strain values, most of the error in life is due to uncertainty associated with either measurement of strain or variation in fatigue limit caused by material variability. On the other hand, a least-squares fit on strain does not work well for higher strain amplitudes. Statistical models have been developed at ANL[33,34] by combining the two approaches and minimizing the sum of the squared Cartesian distances from the data point to the predicted curve; the models were later updated with a larger fatigue S-N data base.[31] The functional forms and transformation for the different variables were based on experimental observations and data trends.

In air, the model assumes that fatigue life is independent of temperature and that strain rate effects occur at temperatures >250°C. It is also assumed that the effect of strain rate on life depends on temperature. One data set, obtained on Type 316 SS in room-temperature air, was excluded from the analysis. The tests in this data set were conducted in load-control mode at stress levels in the range of 190-230 MPa. The strain amplitudes were calculated only as elastic strains, i.e., strain amplitudes of 0.1-0.12% (the data are shown as circles in Fig. 5, with fatigue lives of 4×10^5 to 3×10^7). Based on cyclic stress vs. strain correlations for Type 316 SS (Eqs. 4a-4f), actual strain amplitudes for these tests should be 0.23-0.32%. In air, the fatigue life N of Types 304 and 316 SS is expressed as

$$\ln(N) = 6.703 - 2.030 \ln(\varepsilon_a - 0.126) + T^* \dot{\varepsilon}^* \qquad (5a)$$

and that of Type 316NG, as

$$\ln(N) = 7.422 - 1.671 \ln(\varepsilon_a - 0.126) + T^* \dot{\varepsilon}^*, \qquad (5b)$$

where ε_a is the strain amplitude (%) and T^* and $\dot{\varepsilon}^*$ are transformed temperature and strain rate, respectively, defined as follows:

$$\begin{aligned} T^* &= 0 && (T < 250°C) \\ T^* &= [(T - 250)/525]^{0.84} && (250 \le T < 400°C) \end{aligned} \qquad (6a)$$

$$\begin{aligned} \dot{\varepsilon}^* &= 0 && (\dot{\varepsilon} > 0.4\%/s) \\ \dot{\varepsilon}^* &= \ln(\dot{\varepsilon}/0.4) && (0.0004 \le \dot{\varepsilon} \le 0.4\%/s) \\ \dot{\varepsilon}^* &= \ln(0.0004/0.4) && (\dot{\varepsilon} < 0.0004\%/s). \end{aligned} \qquad (6b)$$

In LWR environments, the fatigue lives of austenitic SSs depends on strain rate, DO level, and temperature; the decrease in life is greater at low-DO levels and high temperatures. However, existing data are inadequate to establish the functional form for the dependence of fatigue life on DO level or temperature. Separate correlations have been developed for low- and high-DO levels (< or ≥ 0.05 ppm), and low and high temperatures (< or ≥ 200°C). Also, a threshold strain rate of

0.4%/s and saturation rate of 0.0004%/s is assumed in the model. Furthermore, for convenience in incorporating environmental effects into fatigue evaluations, the slope of the S-N curve in LWR environments was assumed to be the same as that in air although the best-fit of the experimental data in water yielded a slope for the S-N curve that differed from the slope of the curve that was obtained in air. In LWR environments, the fatigue life N of Types 304 and 316 SS is expressed as

$$\ln(N) = 5.768 - 2.030 \ln(\varepsilon_a - 0.126) + T^* \dot{\varepsilon}^* O^* \tag{7a}$$

and that of Type 316NG, as

$$\ln(N) = 6.913 - 1.671 \ln(\varepsilon_a - 0.126) + T^* \dot{\varepsilon}^* O^*, \tag{7b}$$

where the constants for transformed temperature, strain rate, and DO are defined as follows:

$$\begin{aligned} T^* &= 0 && (T < 200°C) \\ T^* &= 1 && (T \geq 200°C) \end{aligned} \tag{8a}$$

$$\begin{aligned} \dot{\varepsilon}^* &= 0 && (\dot{\varepsilon} > 0.4\%/s) \\ \dot{\varepsilon}^* &= \ln(\dot{\varepsilon}/0.4) && (0.0004 \leq \dot{\varepsilon} \leq 0.4\%/s) \\ \dot{\varepsilon}^* &= \ln(0.0004/0.4) && (\dot{\varepsilon} < 0.0004\%/s) \end{aligned} \tag{8b}$$

$$\begin{aligned} O^* &= 0.260 && (DO < 0.05\ ppm) \\ O^* &= 0.172 && (DO \geq 0.05\ ppm). \end{aligned} \tag{8c}$$

The model is recommended for predicted fatigue lives $\leq 10^6$ cycles. Recent test results indicate that for high-DO environments, conductivity of water is important for environmental effects on fatigue life of austenitic SSs. Therefore, the above correlations may be conservative for high-DO, i.e., ≥0.05 ppm DO, environments. The experimental values of fatigue life in air and water and those predicted from Eqs. 5-8 are plotted in Fig. 24. The estimated fatigue S-N curves for Types 304, 316, and 316NG SSs in air and LWR environments are shown in Figs. 5 and 25, respectively. The predicted fatigue lives show good agreement with the experimental data. Note that the ASME mean curve is not consistent with the existing fatigue S-N data (Fig. 5). Also, although the best-fit of the S-N data in LWR environments (Fig. 25) yields a steeper slope, the slope of the S-N curve in water was assumed to be the same as in air.

Upon completion of the modeling phase, the residual errors (i.e., the Cartesian distance from the prediction curve) should not show significant patterns, such as heteroskedasticity (changing variance), or a nonzero slope. The residual errors for each variable, grouped by steel type and environment (air or water), are plotted in Figs. 26-30. Most data subsets and plots

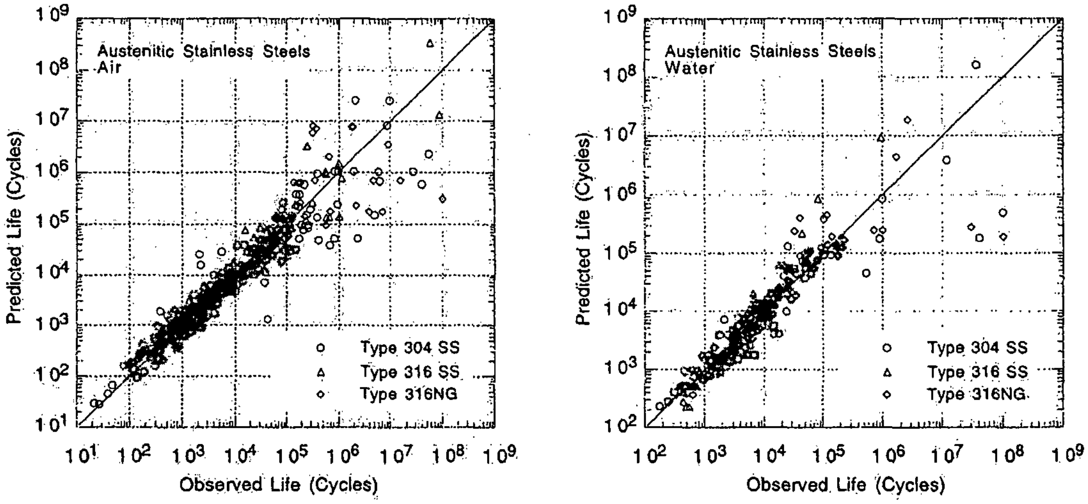


Figure 24. Experimental and predicted values of fatigue lives of austenitic SSs in air and water environments

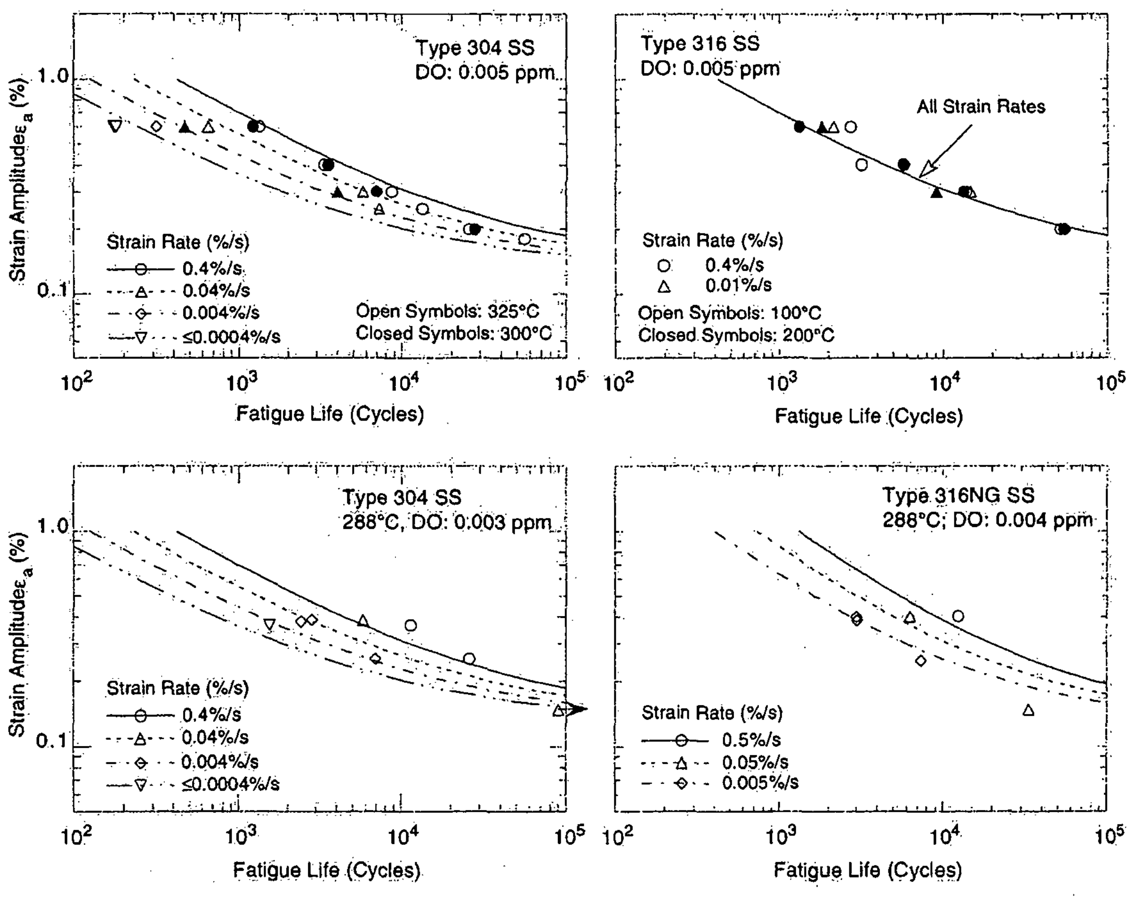


Figure 25. Experimental fatigue lives and those estimated from statistical models for austenitic SSs in water environments

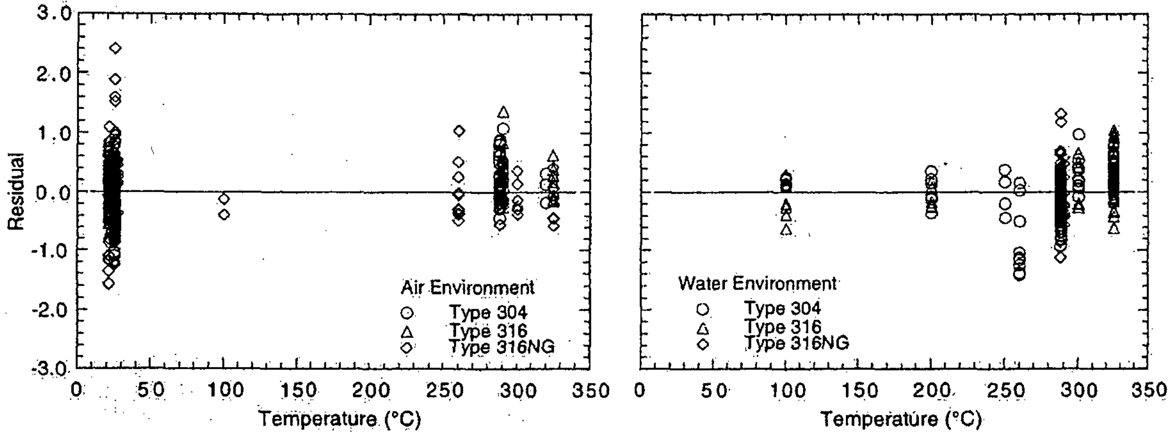


Figure 26. Residual error for austenitic SSs as a function of test temperature

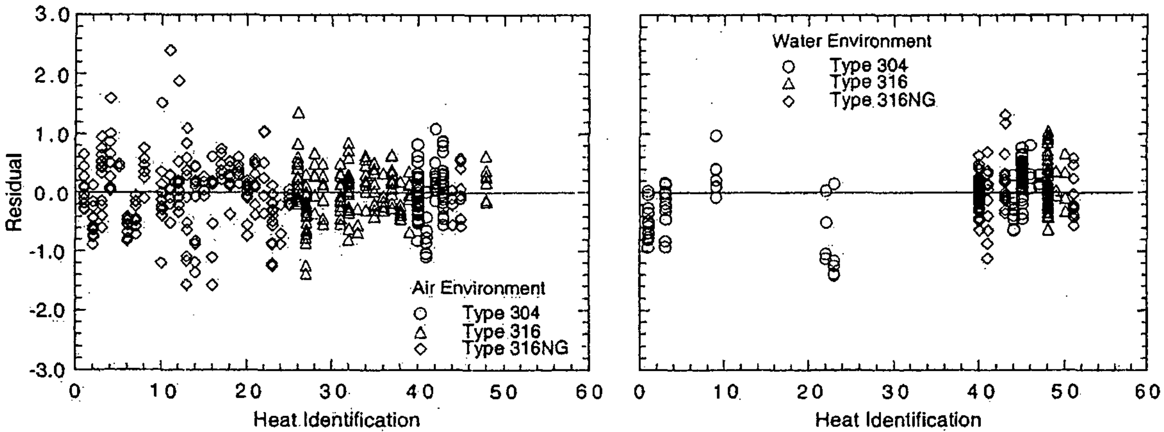


Figure 27. Residual error for austenitic SSs as a function of material heat

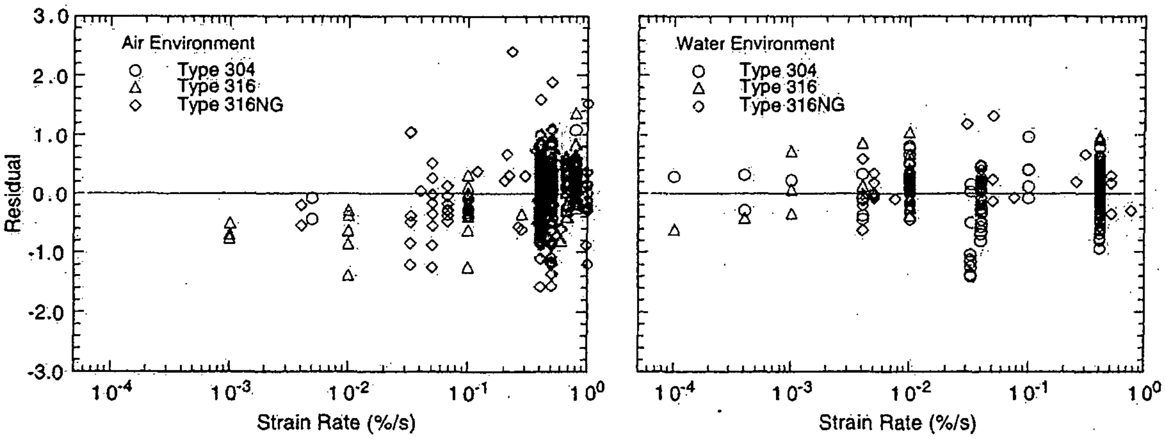
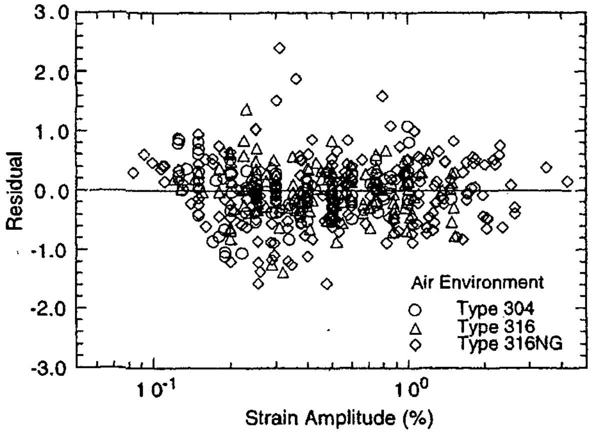


Figure 28. Residual error for austenitic SSs as a function of loading strain rate



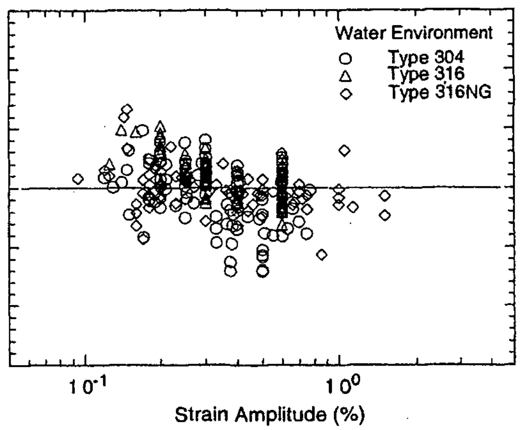


Figure 29. Residual error for austenitic SSs as a function of applied strain amplitude

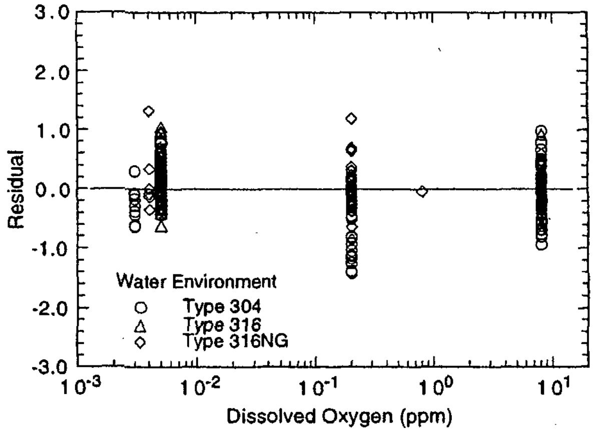


Figure 30.
Residual error for austenitic SSs as a function of dissolved oxygen in water

do not show patterns. In general, high variance tends to be associated with longer lives and lower strain amplitudes. Furthermore, biases seem to be traceable to heat-to-heat variation.

6 Design Fatigue Curves

The design fatigue curves in the current ASME Section III Code were based on experimental data on small polished test specimens. The curves were obtained by adjusting the best-fit curve for the effect of mean stress and then lowering the adjusted curve by a factor of 2 on stress or 20 on life, whichever was more conservative, at each point of the curve. The best-fit curve to the experimental data,[51] expressed in terms of strain amplitude ε_a (%) and fatigue cycles N, for austenitic SSs is given by

$$\ln[N] = 6.954 - 2.0 \ln(\varepsilon_a - 0.167). \quad (9)$$

The mean curve, expressed in terms of stress amplitude S_a (MPa), which is the product of ε_a and elastic modulus E, is given by

$$S_a = 58020/(N)^{1/2} + 299.92. \quad (10)$$

The strain-vs.-life data were converted to stress-vs.-life curves by using the room-temperature value of 195.1 GPa (28300 ksi) for the elastic modulus. The best-fit curves were adjusted for the effect of mean stress by using the modified Goodman relationship[46]

$$S'_a = S_a\left(\frac{\sigma_u - \sigma_y}{\sigma_u - S_a}\right) \qquad \text{for } S_a < \sigma_y, \tag{10a}$$

and
$$S'_a = S_a \qquad \text{for } S_a > \sigma_y, \tag{10b}$$

where S'_a is the adjusted value of stress amplitude, and σ_y and σ_u are yield and ultimate strengths of the material, respectively. The Goodman relationship assumes the maximum possible mean stress and typically gives a conservative adjustment for mean stress, at least when environmental effects are not significant. The design fatigue curves were then obtained by lowering the adjusted best-fit curve by a factor of 2 on stress or 20 on cycles, whichever was more conservative, to account for differences and uncertainties in fatigue life associated with material and loading conditions.

The same procedure has been used to develop design fatigue curves for LWR environments. However, because of the differences between the ASME mean curve and the best-fit curve to existing fatigue data (Fig. 5), the margin on strain for the current ASME Code design fatigue curve is closer to 1.5 than 2. Therefore, to be consistent with the current Code design curve, a factor of 1.5 rather than 2 was used in developing the design fatigue curves from the updated statistical models in air and LWR environments.

The design fatigue curves based on the statistical model for Types 304 and 316 SS in air and low- and high-DO water are shown in Figs. 31–33. A similar set of curves can be obtained for Type 316NG SS. Because the fatigue life of Type 316NG is superior to that of Types 304 or 316 SS, Figs. 31–33 may be used conservatively for Type 316NG SS. Also, as mentioned earlier, recent test results indicate that the conductivity of water is important for environmental effects on fatigue life of austenitic SSs in high-DO environments. Therefore, the design fatigue curves for Type 304 and 316 SS in water with ≥0.05 ppm DO (Fig. 33) may be conservative.

Although, in air at low stress levels, the differences between the current ASME Code design curve and the design curve obtained from the updated statistical model at temperatures <250°C have been reduced or eliminated by reducing the margin on stress from 2 to 1.5, significant differences still exist between the two curves. For example, at stress amplitudes >300 MPa, estimates of life from the updated design curve are a factor of ≈2 lower than those from the ASME Code curve. Therefore, the actual margins on stress and life for the current ASME Code design fatigue curve are 1.5 and 10, respectively, instead of 2 and 20.

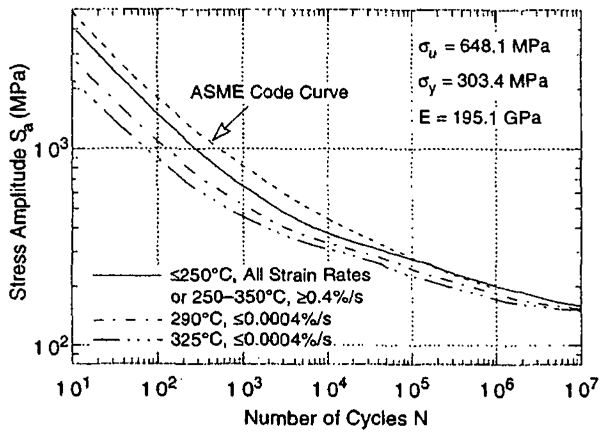


Figure 31.
ASME and statistical-model design fatigue curves for Types 304 and 316 SS in air

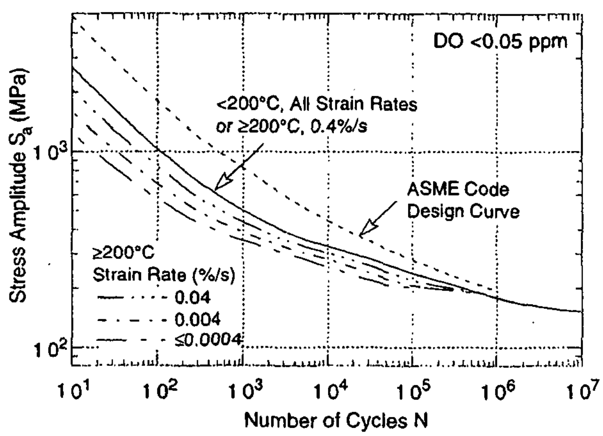


Figure 32.
ASME and statistical-model design fatigue curves for Types 304 and 316 SS in water with <0.05 ppm DO

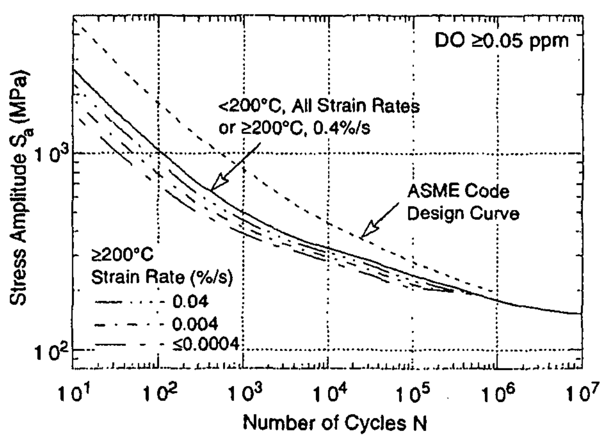


Figure 33.
ASME and statistical-model design fatigue curves for Types 304 and 316 SS in water with ≥0.05 ppm DO

As discussed above, the existing fatigue data indicate a threshold strain range of ≈0.32%, below which environmental effects on the fatigue life of austenitic SSs either do not occur or are insignificant. This value must be adjusted for the effects of mean stress and uncertainties due to material and loading variability. Threshold strain amplitudes are decreased by ≈10% to account for mean stress effects and by a factor of 1.5 to account for uncertainties in fatigue life associated with material and loading variability. Thus, a threshold strain amplitude of 0.097% (stress amplitude of 189 MPa) was selected, below which environmental effects on life are modest and are represented by the design curve for temperatures <200°C (shown by the solid line in Figs. 31 and 32).

These curves can be used to perform ASME Code fatigue evaluations of components that are in service in LWR environments. For each set of load pairs, a partial usage factor is obtained from the appropriate design fatigue curve. Information about the service conditions, such as temperature, strain rate, and DO level, are required for the evaluations. The procedure for obtaining these parameters depends on whether the elapsed-time-vs.-temperature information for the transient is available. The maximum values of temperature and DO level and the slowest strain rate during the transient may be used for a conservative estimate of life. Note that the design curves in LWR environments not only account for environmental effects on life but also include the difference between the current Code design curve and the updated design curve in air, i.e., the difference between the solid and dashed curves in Fig. 31.

7 Fatigue Life Correction Factor

The effects of reactor coolant environments on fatigue life have also been expressed in terms of a fatigue life correction factor F_{en}, which is the ratio of the life in air at room temperature to that in water at the service temperature.[11,52,53] To incorporate environmental effects into the ASME Code fatigue evaluation, a fatigue usage for a specific load pair, based on the current Code fatigue design curve, is multiplied by the correction factor. A fatigue life correction factor F_{en} can also be obtained from the statistical model, where

$$\ln(F_{en}) = \ln(N_{air}) - \ln(N_{water}). \tag{12}$$

From Eqs. 5a and 7a, the fatigue life correction factor relative to room-temperature air for Types 304 and 316 SSs is given by

$$F_{en} = \exp(0.935 - T^* \dot{\varepsilon}^* O^*), \tag{13}$$

where the threshold and saturation values for T^*, $\dot{\varepsilon}^*$, and O^* are defined in Eqs. 8a–8c. At temperatures ≥200°C and strain rates ≤0.0004%/s, Eq. 13 yields an F_{en} of ≈15 in low-DO PWR water (<0.05 ppm DO) and ≈8 in high-DO water (≥0.05 ppm DO). At temperatures <200°C, F_{en} is ≈2.5 in both low- and high-DO water at all strain rates.

8 Conservatism in Design Fatigue Curves

The overall conservatism in ASME Code fatigue evaluations has also been demonstrated in fatigue tests on piping welds and components.[54] In air, the margins on the number of cycles to failure for austenitic SS elbows and tees were 40–310 and 104–510, respectively. The margins for girth butt welds were significantly lower at 6–77. In these tests, fatigue life was expressed as the number of cycles for the crack to penetrate through the wall, which ranged in thickness from 6 to 18 mm (0.237 to 0.719 in). The fatigue design curves represent the number of cycles that are necessary to form a 3-mm-deep crack. Consequently, depending on wall thickness, the actual margins to failure may be lower by a factor of >2.

Deardorff and Smith[55] have discussed the types and extent of conservatisms present in the ASME Section III fatigue evaluations and the effects of LWR environments on fatigue margins. The sources of conservatism include design transients considerably more severe than those experienced in service, grouping of transients, and simplified elastic-plastic analysis. Environmental effects on two components, the BWR feedwater nozzle/safe end and PWR steam generator feedwater nozzle/safe end, both constructed from LAS and known to be affected by severe thermal transients, were also investigated in the study. When environmental effects on fatigue life were not considered, Deardorff and Smith[55] estimated that the ratio of the CUFs for the PWR and BWR nozzles (both constructed from LAS), computed with the mean experimental curve for test specimen data, to CUFs computed with the Code fatigue design curve were ≈60 and 90, respectively. To maintain the factor of 20 on life that was used in the present Code fatigue design curves to account for the uncertainties due to material and loading variability, the margins for the PWR and BWR nozzles are reduced to 3 and 4.5, respectively. These results suggest that, for carbon and low-alloy steels, the Code Design procedures provide some margin in life that can be used to account for environmental effects on life. However, as noted previously in Section 6, the Code fatigue design curve for austenitic SSs is not consistent with the existing fatigue S–N data; the actual margins on stress and life are 1.5 and 10, respectively, instead of 2 and 20. Consequently, the Code fatigue design curve for austenitic SSs provides little or no margin in life to account for environmental effects.

Data available in the literature have been reviewed to evaluate the effects of various material, loading, and environmental variables on the fatigue life of structural materials in air and LWR environments.[33] The subfactors that may be used to account for the effects of these variables on fatigue life are summarized in Table 5. The factors on strain primarily account for variation in the fatigue limit of a material caused by material variability, component size and surface finish, and loading history. Because the reduction in fatigue life is associated with the growth of short cracks (<100 μm), the effects of these variables on threshold strain are typically not cumulative but rather are controlled by the variable that has the largest effect. The values in Table 5 suggest that a factor of at least 1.5 on strain and 10 on cycles is needed to account for the differences and uncertainties of relating the fatigue lives of laboratory test specimens to those of large components. Beacuse SSs develop a corrosion scale in LWR environments, the effect of surface finish may not be significant; the subfactor on life to account for surface finish effects may be as low as 1.5 or may be eliminated completely. Therefore, a factor of 1.5 or 2 on life may be able to account for the effects of environment on the fatigue lives of austenitic SSs.

Table 5. Subfactors that may be used to account for effects of various variables on fatigue life

Variable	Factor on Life	Factor on Strain
Material variability and experimental scatter	2.5	1.4–1.7
Size	1.4	1.25
Surface finish	2.0–3.0	1.3
Loading history	1.5–2.5	1.5
Total adjustment	10.0–26.0	1.5–1.7

9 Fatigue Evaluations in LWR Environments

Section III of the ASME Boiler and Pressure Vessel Code contains rules for the construction of nuclear power plant Class 1 components.[5] It provides requirements for designs that will withstand cyclic loadings on a structural component that occur because of changes in mechanical and thermal loadings as the system goes from one load set (pressure, temperature, moment, and force) to any other load set. ASME Section III, NB-3600 (piping design) methodology is used exclusively for piping and sometimes for branch nozzles. ASME Section III, NB-3200 (design by analysis) methodology is generally used for vessels and frequently for nozzles. In both cases, the various sets of load states at the most highly stressed locations in the component are defined first. The load states are defined in terms of the three principal stresses in NB-3200 methodology, and in terms of

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 15

NUREG/CR-6583
ANL-97/18

Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels

Argonne National Laboratory

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

NUREG/CR-6583
ANL-97/18

Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels

Prepared by
O. K. Chopra, W. J. Shack

Argonne National Laboratory

**Prepared for
U.S. Nuclear Regulatory Commission**

EFFECTS OF LWR COOLANT ENVIRONMENTS ON FATIGUE DESIGN CURVES OF CARBON AND LOW-ALLOY STEELS

by

O. K. Chopra and W. J. Shack

Abstract

The ASME Boiler and Pressure Vessel Code provides rules for the construction of nuclear power plant components. Figures I-9.1 through I-9.6 of Appendix I to Section III of the Code specify fatigue design curves for structural materials. While effects of reactor coolant environments are not explicitly addressed by the design curves, test data indicate that the Code fatigue curves may not always be adequate in coolant environments. This report summarizes work performed by Argonne National Laboratory on fatigue of carbon and low-alloy steels in light water reactor (LWR) environments. The existing fatigue S-N data have been evaluated to establish the effects of various material and loading variables such as steel type, dissolved oxygen level, strain range, strain rate, temperature, orientation, and sulfur content on the fatigue life of these steels. Statistical models have been developed for estimating the fatigue S-N curves as a function of material, loading, and environmental variables. The results have been used to estimate the probability of fatigue cracking of reactor components. The different methods for incorporating the effects of LWR coolant environments on the ASME Code fatigue design curves are presented.

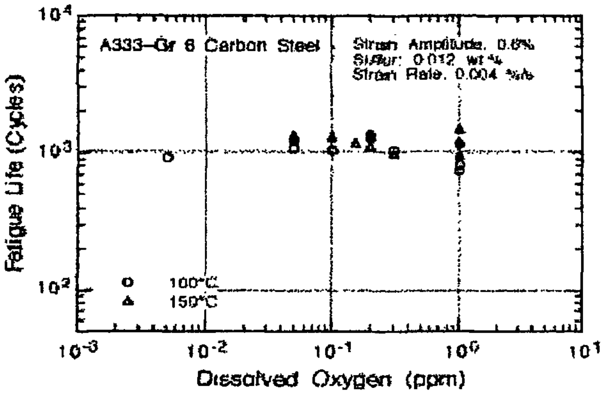


Figure 68.
Fatigue life of A333–Gr 6 carbon steel as a function of dissolved oxygen in water at 100 and 150°C

5 Statistical Model

5.1 Modeling Choices

In attempting to develop a statistical model from incomplete data and where physical processes are only partially understood, care must be taken to avoid overfitting the data. Different functional forms of the predictive equations (e.g., different procedures for transforming the measured variables into data used for fitting equations) were tried for several aspects of the model. Fatigue S–N data are generally expressed by Eq. 1.1, which may be rearranged to express fatigue life N in terms of strain amplitude ε_a as

$$\ln(N) = [\ln B - \ln(\varepsilon_a - A)]/b. \tag{5.1}$$

Additional terms may be added to the model that would improve agreement with the current data set. However, such changes may not hold true in other data sets, and the model would typically be less robust, i.e., it would not predict new data well. In general, complexity in a statistical model is undesirable unless it is consistent with accepted physical processes. Although there are statistical tools that can help manage the tradeoff between robustness and detail in the model, engineering judgment is required. Model features that would be counter to known effects are excluded. Features that are consistent with previous studies use such results as guidance, e.g., defining the threshold or saturation values for an effect, but where there are differences from previous findings, the reasons for the differences are evaluated and an appropriate set of assumptions is incorporated into the model.

5.2 Least–Squares Modeling within a Fixed Structure

The parameters of the model are commonly established through least-squares curve-fitting of the data to either Eq. 1.1 or 5.1. An optimization program sets the parameters so as to minimize the sum of the square of the residual errors, which are the differences between the predicted and actual values of ε_a or ln(N). A predictive model based on least-squares fit on ln(N) is biased for low ε_a; in particular, runoff data cannot be included. The model also leads to probability curves that converge to a single value of threshold strain.

However, the model fails to address the fact that at low ε_a, most of the error in life is due to uncertainty associated with either measurement of stress or strain or variation in threshold strain caused by material variability. On the other hand, a least-squares fit on ε_a does not work well for higher strain amplitudes. The two kinds of models are merely transformations of each other, although the precise values of the coefficients differ.

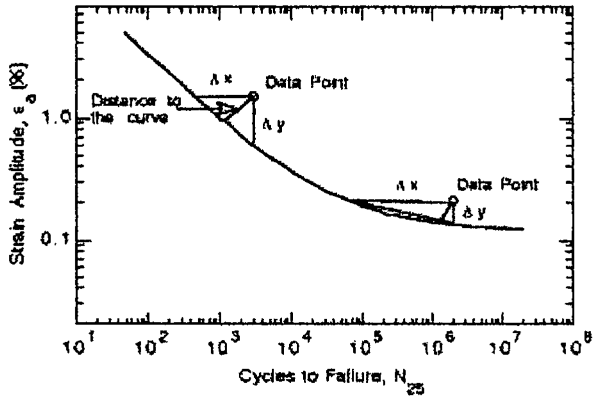


Figure 69.
Schematic of least-squares curve-fitting of data by minimizing sum of squared Cartesian distances from data points to predicted curve

The statistical models[27,28] were developed by combining the two approaches and minimizing the sum of squared Cartesian distances from the data points to the predicted curve (Fig. 69). For low ε_a, this is very close to optimizing the sum of squared errors in predicted ε_a; at high ε_a, it is very close to optimizing the sum of squared errors in predicted life; and at medium ε_a, this model combines both factors. However, because the model includes many nonlinear transformations of variables and because different variables affect different parts of the data, the actual functional form and transformations are partly responsible for minimizing the squares of the errors. The functional forms and transformation are chosen a priori, and no direct computational means exist for establishing them.

To perform the optimization, it was necessary to normalize the x and y axes by assigning relative weights to be used in combining the error in life and strain amplitude because x and y axes are not in comparable units. In this analysis, errors in strain amplitude (%) are weighted 20 times as heavily as errors in ln(N). A value of 20 was selected for two related reasons. First, this factor leads to approximately equal weighting of low- and high-strain-amplitude data in the least-squared error computation of model coefficients. Second, when applied to the model to generate probability curves, it yielded a standard deviation on strain amplitude comparable to that obtained from the best-fit of the high cycle fatigue data to Eq. 1.1. Because there is necessarily judgment applied in the selection of this value, a sensitivity analysis was performed, and it showed that the coefficients of the model do not change much for weight factors between 10 and 25. Distance from the curve was estimated as

$$D = \left\{(x - \hat{x})^2 + \left[k(y - \hat{y})\right]^2\right\}^{1/2}, \qquad (5.2)$$

where $\hat{x}$ and $\hat{y}$ represent predicted values, and k = 20.

5.3 The Model

Based on the existing fatigue S-N data base, statistical models have been developed for estimating the effects of material and loading conditions on the fatigue lives of CSs and LASs.[27,28] The dependence of fatigue life on DO level has been modified because it was determined that in the range of 0.05-0.5 ppm, the effect of DO was more logarithmic than linear.[45,93] In this report, the models have been further optimized with a larger fatigue S-N data base. Because of the conflicting possibilities that with decreasing strain rate, fatigue life may either be unaffected, decrease for some heats, or increase for others, effects of strain rate in air were not explicitly considered in the model. The effects of orientation, i.e., size and distribution of sulfide inclusions, on fatigue life were also excluded because the existing data base does not include information on sulfide distribution and morphology. In air, the fatigue data for CSs are best represented by

$$\ln(N_{25}) = 6.595 - 1.975 \ln(\varepsilon_a - 0.113) - 0.00124\,T \quad (5.3a)$$

and for LASs by

$$\ln(N_{25}) = 6.658 - 1.808 \ln(\varepsilon_a - 0.151) - 0.00124\,T. \quad (5.3b)$$

In LWR environments, the fatigue data for CSs are best represented by

$$\ln(N_{25}) = 6.010 - 1.975 \ln(\varepsilon_a - 0.113) + 0.101\,S^*\,T^*\,O^*\,\dot{\varepsilon}^* \quad (5.4a)$$

and for LASs by

$$\ln(N_{25}) = 5.729 - 1.808 \ln(\varepsilon_a - 0.151) + 0.101\,S^*\,T^*\,O^*\,\dot{\varepsilon}^*, \quad (5.4b)$$

where S^*, T^*, O^*, and $\dot{\varepsilon}^*$ = transformed sulfur content, temperature, DO, and strain rate, respectively, defined as follows:

$$\begin{aligned} S^* &= S && (0 < S \leq 0.015 \text{ wt.\%}) \\ S^* &= 0.015 && (S > 0.015 \text{ wt.\%}) \end{aligned} \quad (5.5a)$$

$$\begin{aligned} T^* &= 0 && (T < 150°C) \\ T^* &= T - 150 && (T = 150\text{-}350°C) \end{aligned} \quad (5.5b)$$

$$\begin{aligned} O^* &= 0 && (DO < 0.05 \text{ ppm}) \\ O^* &= \ln(DO/0.04) && (0.05 \text{ ppm} \leq DO \leq 0.5 \text{ ppm}) \\ O^* &= \ln(12.5) && (DO > 0.5 \text{ ppm}) \end{aligned} \quad (5.5c)$$

$$\begin{aligned} \dot{\varepsilon}^* &= 0 && (\dot{\varepsilon} > 1\ \%/s) \\ \dot{\varepsilon}^* &= \ln(\dot{\varepsilon}) && (0.001 \leq \dot{\varepsilon} \leq 1\ \%/s) \\ \dot{\varepsilon}^* &= \ln(0.001) && (\dot{\varepsilon} < 0.001\ \%/s) \end{aligned} \quad (5.5d)$$

The functional form and bounding values of the transformed parameters S^*, T^*, O^*, and $\dot{\varepsilon}^*$ were based upon experimental observations and data trends discussed in Section 4.2. Significant features of the model for estimating fatigue life in LWR environments are as follows:

(a) The model assumes that environmental effects on fatigue life occur primarily during the tensile-loading cycle; minor effects during the compressive loading cycle have been excluded. Consequently, the loading and environmental conditions, e.g., temperature, strain rate, and DO, during the tensile-loading cycle are used for estimating fatigue lives.

(b) When any one of the threshold condition is not satisfied, e.g., <0.05 ppm DO in water, the effect of strain rate is not considered in the model, although limited data indicate that heats of steel that are sensitive to strain rate in air also show a decrease in life in water with decreasing strain rate.

(c) The model assumes a linear dependence of S^* on S content in steel and saturation at 0.015 wt.% S.

The model is recommended for predicted fatigue lives of $\leq 10^6$ cycles. For fatigue lives of 10^6 to 10^8 cycles, the results should be used with caution because, in this range, the model is based on very limited data obtained from relatively few heats of material.

The estimated and experimental S-N curves for CS and LAS in air at room temperature and 288°C are shown in Fig. 70. The mean curves used in developing the ASME Code design curve and the average curves of Higuchi and Iida[7] are also included in the figure. The results indicate that the ASME mean curve for carbon steels is not consistent with the experimental data; at strain amplitudes <0.2%, the mean curve predicts significantly lower fatigue lives than those observed experimentally. The estimated curve for low-alloy steels is comparable with the ASME mean curve. For both steels, Eq. 5.3 shows good agreement with the average curves of Higuchi and Iida.

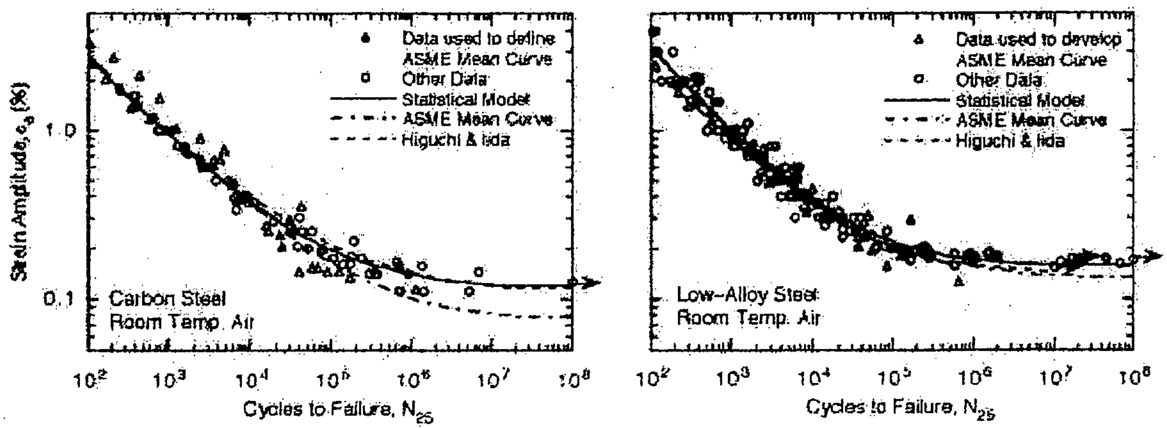


Figure 70. Fatigue S-N behavior for carbon and low-alloy steels estimated from model and determined experimentally in air at room temperature

5.4 Distribution of Fatigue Life

For a given steel type, the average distance of data points from the mean curve does not vary much for different environmental conditions. To develop a distribution on life, we start with the assumption that there are three sources of prediction error: (a) measurement errors for the applied strain amplitude, (b) variations in the threshold strain amplitude due to material variability, and (c) errors due to uncertainty in test and material conditions or other unexplained variation. Because measurement errors are small at high strain amplitudes, the standard deviation of distance from the mean curve at high strain amplitudes is a good measure of the scatter in fatigue life due to unexplained variations. At low amplitudes where the S-N curve is almost horizontal, the errors (as measured by the distance from the mean curve) are dominated by the variation in strain amplitude. The standard deviation of the error in strain amplitude was taken to be equal to the standard deviation in the predicted fatigue life divided by a factor of 20 consistent with the weighting factor used for optimization. The standard deviation on life was 0.52 for CSs and LASs. These results can be combined with Eq. 5.3 to estimate the distribution in life for smooth test specimens. In air, the xth percentile of the distribution on life $N_{25}[x]$ for CSs is

$$\ln(N_{25}) = 6.595 + 0.52\ F^{-1}[x] - 1.975\ \ln(\varepsilon_a - 0.113 + 0.026\ F^{-1}[1-x]) - 0.00124\ T \qquad (5.6a)$$

and for LASs it is

$$\ln(N_{25}) = 6.658 + 0.52\ F^{-1}[x] - 1.808\ \ln(\varepsilon_a - 0.151 + 0.026\ F^{-1}[1-x]) - 0.00124\ T\,. \qquad (5.6b)$$

In LWR environments, the xth percentile of the distribution on life $N_{25}[x]$ for CSs is

$$\ln(N_{25}) = 6.010 + 0.52\ F^{-1}[x] - 1.975\ \ln(\varepsilon_a - 0.113 + 0.026\ F^{-1}[1-x]) + 0.101\ S^*\ T^*\ O^*\ \dot{\varepsilon}^* \qquad (5.7a)$$

and for LASs it is

$$\ln(N_{25}) = 5.729 + 0.52\ F^{-1}[x] - 1.808\ \ln(\varepsilon_a - 0.151 + 0.026\ F^{-1}[1-x]) + 0.101\ S^*\ T^*\ O^*\ \dot{\varepsilon}^*. \qquad (5.7b)$$

The parameters S^*, T^*, O^*, and $\dot{\varepsilon}^*$ are defined in Eqs. 5.5, and $F^{-1}[\cdot]$ denotes the inverse of the standard normal cumulative distribution function. The coefficients of distribution functions $F^{-1}[x]$ and $F^{-1}[1-x]$ represent the standard deviation on life and strain amplitude, respectively. For convenience, values of the inverse of standard normal cumulative distribution function in Eqs. 5.6 and 5.7 are given in Table 3. The standard deviation of 0.026 on strain amplitude obtained from the analysis may be an overly conservative value. A more realistic value for the standard deviation on strain could be obtained by analysis of the fatigue limits of different heats of material. The existing data are inadequate for such an analysis because (a) not enough heats of materials are included in the data base, and (b) there are very few high-cycle fatigue data for accurate estimations of the fatigue limit for specific heats.

The estimated probability curves for the fatigue life of CSs and LASs in an air and LWR environments in Figs. 71–73 show good agreement with experimental data; nearly all of the data are bounded by the 5% probability curve. Relative to the 50% probability curve, the 5% probability curve is a factor of ≈2.5 lower in life at strain amplitudes >0.3% and a factor of 1.4–1.7 lower in strain at <0.2% strain amplitudes. Similarly, the 1% probability curve is a factor of ≈3.7 lower in life and a factor of 1.7–2.2 lower in strain.

Table 3. Inverse of standard cumulative distribution function

Probability	$F^{-1}[x]$	$F^{-1}[1-x]$	Probability	$F^{-1}[x]$	$F^{-1}[1-x]$
0.01	-3.7195	3.7195	3.00	-1.8808	1.8808
0.02	-3.5402	3.5402	5.00	-1.6449	1.6449
0.03	-3.4319	3.4319	7.00	-1.4758	1.4758
0.05	-3.2905	3.2905	10.00	-1.2816	1.2816
0.07	-3.1947	3.1947	20.00	-0.8416	0.8416
0.10	-3.0902	3.0902	30.00	-0.5244	0.5244
0.20	-2.8782	2.8782	50.00	0.0000	0.0000
0.30	-2.7478	2.7478	65.00	0.3853	-0.3853
0.50	-2.5758	2.5758	80.00	0.8416	-0.8416
0.70	-2.4573	2.4573	90.00	1.2816	-1.2816
1.00	-2.3263	2.3263	95.00	1.6449	-1.6449
2.00	-2.0537	2.0537	98.00	2.0537	-2.0537

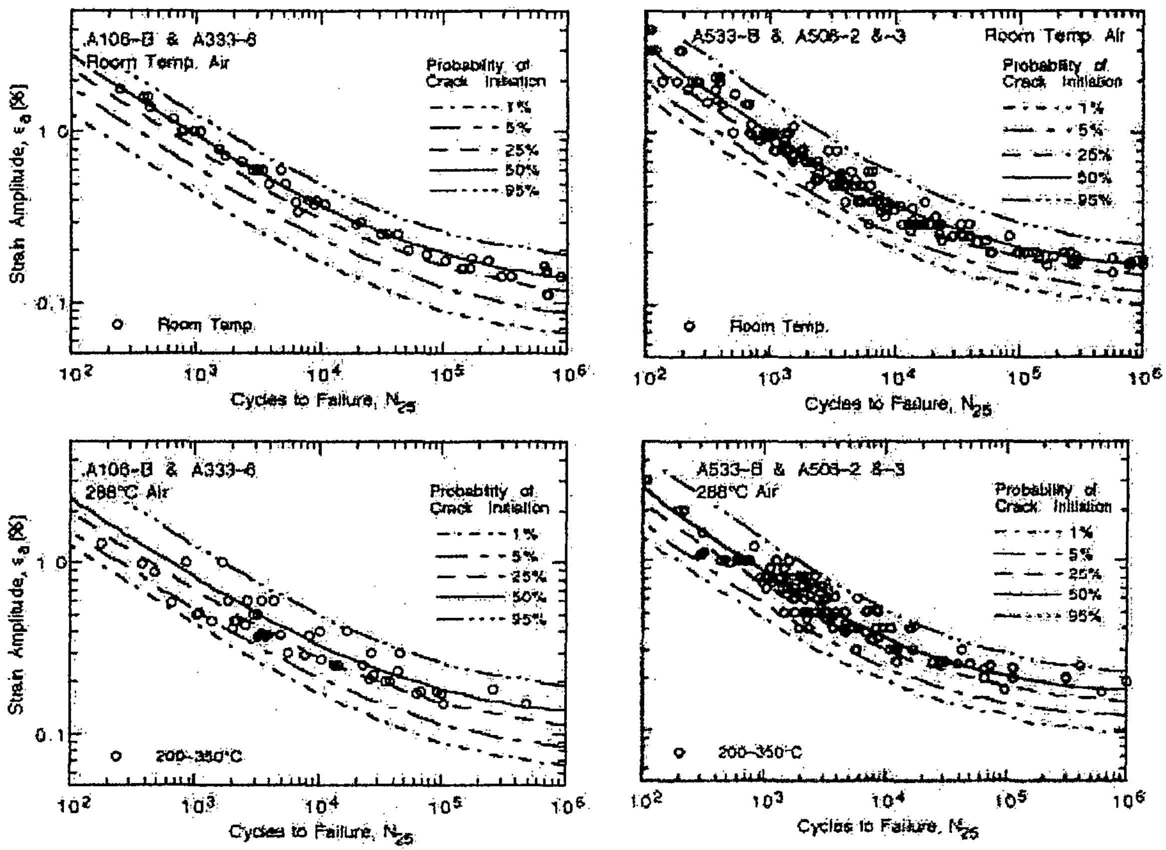


Figure 71. Experimental data and probability of fatigue cracking in carbon and low-alloy steel test specimens in air

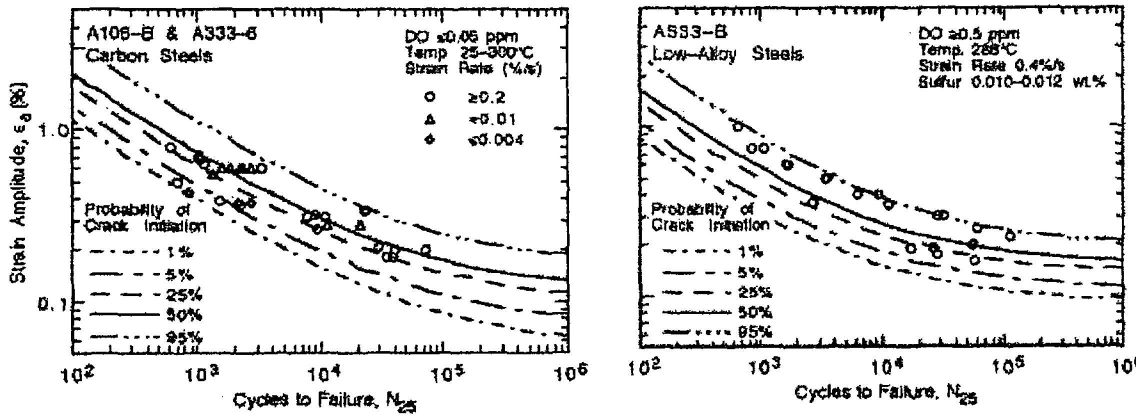


Figure 72. Experimental data and probability of fatigue cracking in carbon and low-alloy steel test specimens in simulated PWR environments

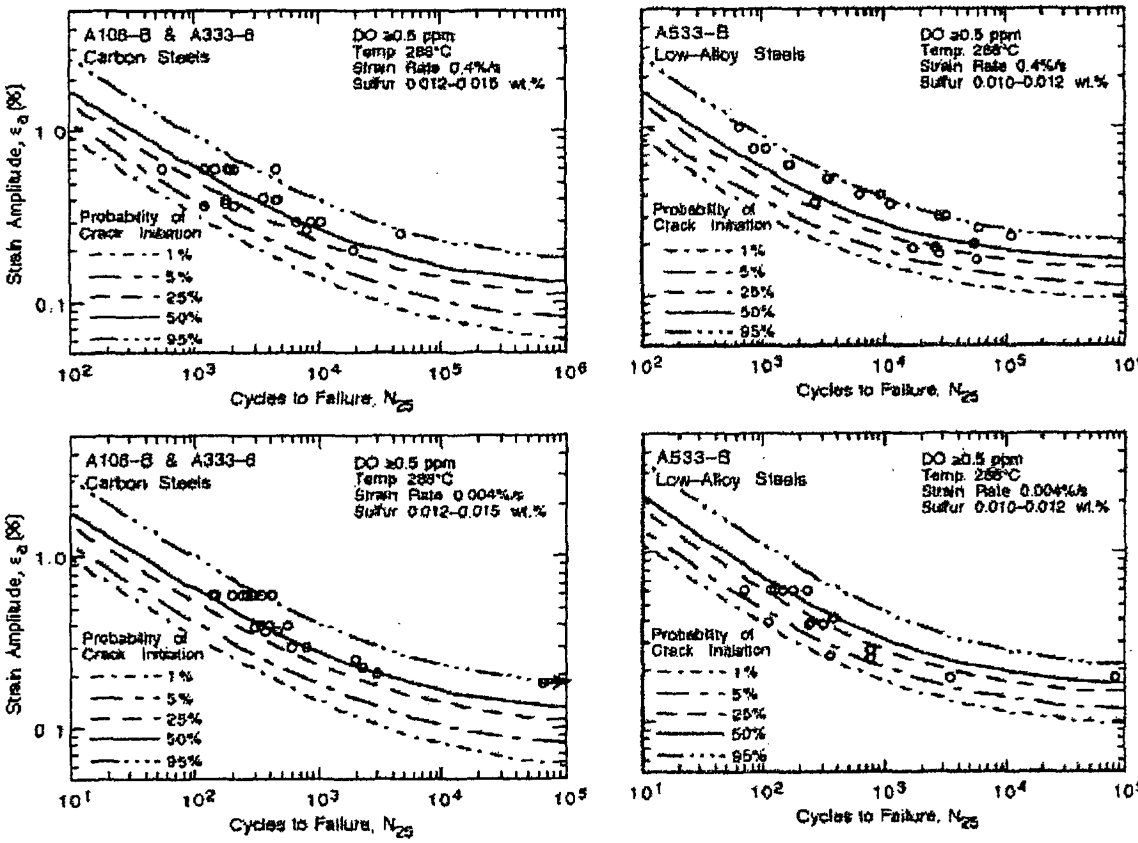


Figure 73. Experimental data and probability of fatigue cracking in carbon and low-alloy steel test specimens in high-dissolved-oxygen water

As with other aspects of this model, the estimates of the probability of cracking should not be extrapolated much beyond the data. The probabilities assume a normal distribution, which is consistent with the data for most of the range. The existing data are not sufficient to determine precise distributions because more data are required to estimate distributions than to estimate the mean curve. However, the assumption of normality is reasonable (and conservative) down to 0.1-1% probability of cracking and it is empirically verified by the number of data points that fall below the respective curves. The probability is not expected to deviate significantly from the normal curve for another order of magnitude (one more standard deviation) even if the probability distribution is not the same. Because estimates of extremely low or high probabilities are sensitive to the choice of distribution, the probability distribution curves should not be extrapolated beyond 0.02% probability.

6 Fatigue Life Correction Factor

An alternative approach for incorporating the effects of reactor coolant environments on fatigue S-N curves has been proposed by the Environmental Fatigue Data (EFD) Committee of the Thermal and Nuclear Power Engineering Society (TENPES) of Japan.* A fatigue life correction factor F_{en} is defined as the ratio of the life in air at room temperature to that in water at the service temperature. The fatigue usage for a specific load pair based on the current Code fatigue design curve is multiplied by the correction factor to account for the environmental effects. Note that the fatigue life correction factor does not account for any differences that might exist between the current ASME mean air curves and the present mean air curves developed from a larger data base. The specific expression for F_{en}, proposed initially by Higuchi and Iida,[7] assumes that life in the environment N_{water} is related to life in air N_{air} at room temperature through a power-law dependence on the strain rate

$$F_{en} = \frac{N_{air}}{N_{water}} = (\dot{\varepsilon})^{-P}, \tag{6.1a}$$

or $$\ln(F_{en}) = \ln(N_{air}) - \ln(N_{water}) = -P\ln(\dot{\varepsilon}). \tag{6.1b}$$

In air at room temperature, the fatigue life N_{air} of CSs is expressed as

$$\ln(N_{air}) = 6.653 - 2.119\ln(\varepsilon_a - 0.108) \tag{6.2a}$$

and for LASs by

$$\ln(N_{air}) = 6.578 - 1.761\ln(\varepsilon_a - 0.140), \tag{6.2b}$$

where ε_a is the applied strain amplitude (%). Only the tensile loading cycle is considered to be important for environmental effects on fatigue life. The exponent P is a product of a environmental factor R_p, which depends on temperature T (°C) and DO level (ppm), and a material factor P_c, which depends on the ultimate tensile strength σ_u (MPa) and sulfur content S (wt/.%) of the steel. Thus

$$P = R_p\,P_c, \tag{6.3a}$$

* Presented at the Pressure Vessel Research Council Meeting, April 1996, Orlando, FL.

$$P_c = 0.864 - 0.00092\,\sigma_u + 14.6\,S, \tag{6.3b}$$

$$R_p = \frac{R_{pT} - 0.2}{2.64}\ln(DO) + 1.75\,R_{pT} - 0.035, \qquad 0.2 \le R_p \le R_{pT} \tag{6.3c}$$

and $R_{pT} = 0.198\,\exp(0.00557T)$. (6.3d)

The fatigue lives of carbon and low-alloy steels measured experimentally and those estimated from the statistical and EFD models are shown in Figs. 74–78. Although the EFD correlations for exponent P were based entirely on data for carbon steels, Eqs. 6.3a–6.3d were also used for estimating the fatigue lives of LASs. Also, σ_u in Eq. 6.3b was assumed to be 520 and 650 MPa, respectively, for CSs and LASs. The significant differences between the two models are as follows:

(a) The EFD correlations have been developed from data for CSs alone.

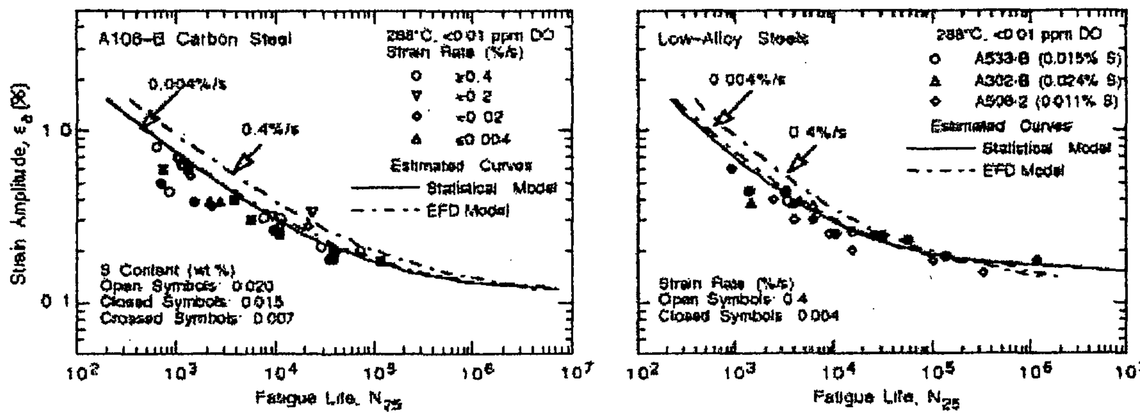


Figure 74. Experimental fatigue lives and those estimated from statistical and EFD models for carbon and low-alloy steels in simulated PWR water

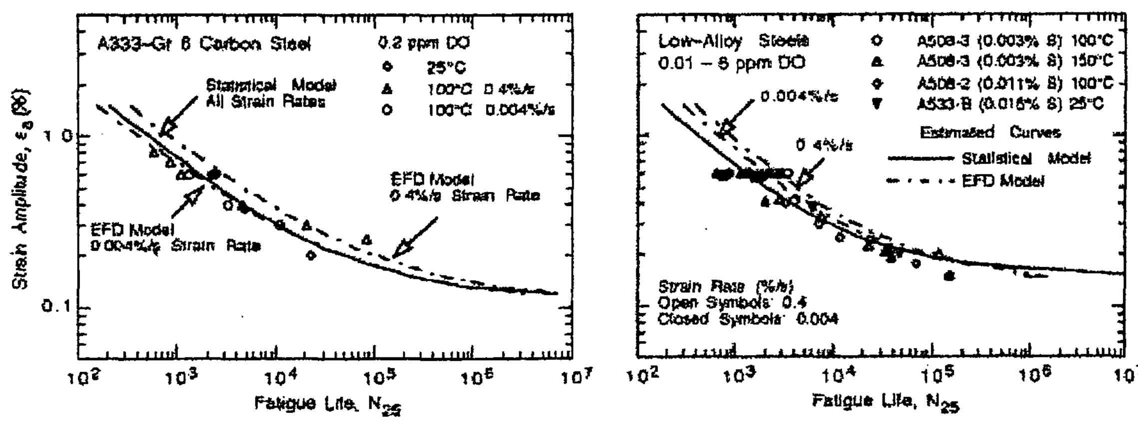


Figure 75. Experimental fatigue lives and those estimated from statistical and EFD models for carbon and low-alloy steels in water at temperatures below 150°C

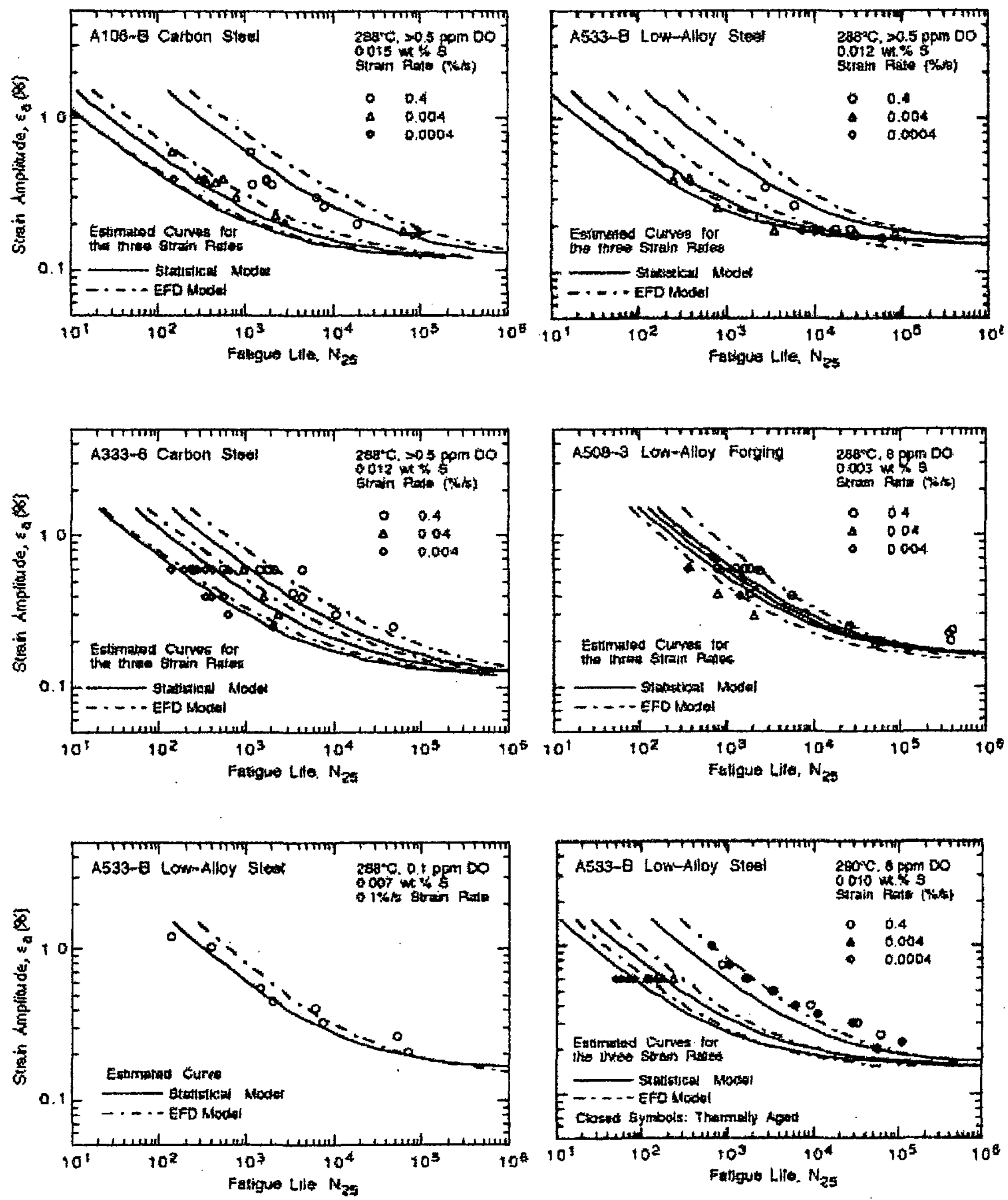


Figure 76. Experimental fatigue lives and those estimated from statistical and EFD models for carbon and low-alloy steels in high-dissolved-oxygen water

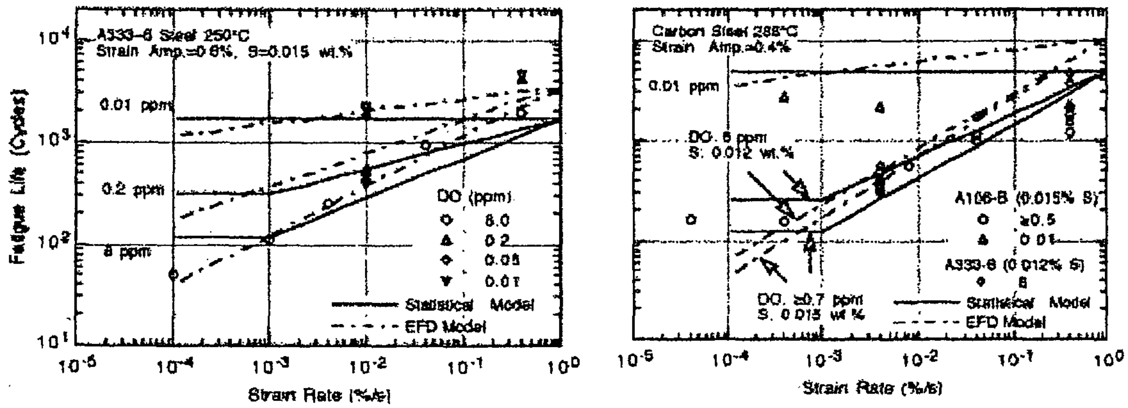


Figure 77. Dependence on strain rate of fatigue life of carbon steels observed experimentally and that estimated from statistical and EFD models

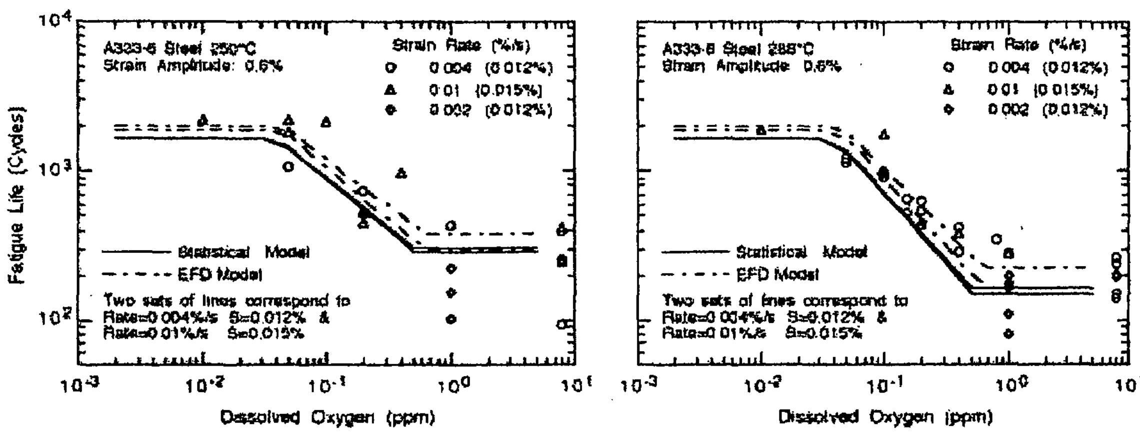


Figure 78. Dependence on dissolved oxygen of fatigue life of carbon steels observed experimentally and that estimated from statistical and EFD models

(b) The statistical model assumes that the effects of strain rate on fatigue life saturate below 0.001%/s, Fig. 77. Such a saturation is not considered in the EFD model.

(c) A threshold temperature of 150°C below which environmental effects on fatigue life are modest is incorporated in the statistical model but not in the EFD model.

(d) The EFD model includes the effect of tensile strength on fatigue life of CSs in LWR environments.

Another estimate of the fatigue life correction factor F_{en} can also be obtained from the statistical model. Since

$$\ln(F_{en}) = \ln(N_{air}) - \ln(N_{water}), \tag{6.4}$$

from Eqs. 5.3a and 5.4a, the fatigue life correction factor for CSs is given by

$$\ln(F_{en}) = 0.585 - 0.00124T - 0.101S^*T^*O^*\dot{\varepsilon}^* \qquad (6.5a)$$

and from Eqs. 5.3b and 5.4b, the fatigue life correction factor for LASs is given by

$$\ln(F_{en}) = 0.929 - 0.00124T - 0.101S^*T^*O^*\dot{\varepsilon}^*, \qquad (6.5b)$$

where the threshold and saturation values for S^*, T^*, O^*, and $\dot{\varepsilon}^*$ are defined in Eqs. 5.5. A value of 25°C is used for T in Eqs. 6.5a and 6.5b if the fatigue life correction factor is defined relative to RT air. Otherwise, both T and T^* represent the service temperature. A fatigue life correction factor F_{en} based on the statistical model has been proposed as part of a nonmandatory Appendix to ASME Section IX fatigue evaluations.[94,95]

7 Fatigue S–N Curves for Components

The current ASME Section III Code design fatigue curves were based on experimental data on small polished test specimens. The best-fit or mean curve to the experimental data used to develop the Code design curve, expressed in terms of stress amplitude S_a (MPa) and fatigue cycles N, for carbon steels is given by

$$S_a = 59{,}736/\sqrt{N} + 149.24 \qquad (7.1a)$$

and for low-alloy steels by

$$S_a = 49{,}222/\sqrt{N} + 265.45. \qquad (7.1b)$$

The stress amplitude S_a is the product of strain amplitude ε_a and elastic modulus E; the room temperature value of 206.8 GPa (30,000 ksi) for the elastic modulus for carbon and low-alloy steels was used in converting the experimental strain-versus-life data to stress-versus-life curves. To obtain design fatigue curves the best-fit curves (Eqs. 7.1a and 7.1b) were first adjusted for the effect of mean stress based on the modified Goodman relation

$$S'_a = S_a\left(\frac{\sigma_u - \sigma_y}{\sigma_u - S_a}\right) \qquad \text{for } S_a < \sigma_y, \qquad (7.2a)$$

and

$$S'_a = S_a \qquad \text{for } S_a > \sigma_y, \qquad (7.2b)$$

where S'_a is the adjusted value of stress amplitude, and σ_y and σ_u are yield and ultimate strengths of the material, respectively. The Goodman relation assumes the maximum possible mean stress and typically gives a conservative adjustment for mean stress at least when environmental effects are not significant. The design fatigue curves were then obtained by lowering the adjusted best-fit curve by a factor of 2 on stress or 20 on cycles, whichever was more conservative, at each point on the curve. The factor of 20 on cycles was intended to account for the uncertainties in fatigue life associated with material and loading conditions, and the factor of 2 on strain was intended to account for uncertainties in threshold strain caused by material variability. This procedure is illustrated for CSs and LASs in Fig. 79.

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 16

R&D Status Report
NUCLEAR POWER DIVISION

John J. Taylor, Director

BWR WATER CHEMISTRY

Many of the stress corrosion problems in boiling water reactors (BWRs) result from the presence of a very small amount of dissolved oxygen in the reactor water. Radiolysis in the reactor core continually decomposes a small amount of the very pure water used in BWRs into free oxygen and hydrogen. Most of the gas is stripped from the water by the steam, leaving only trace amounts of oxygen and hydrogen dissolved in the reactor water. Although the amount of dissolved oxygen is only about 200 ppb, it is sufficient to facilitate stress corrosion cracking. Hydrogen water chemistry can reduce dissolved oxygen to a level that will no longer facilitate stress corrosion.

Pipe cracking in BWRs first came to the attention of U.S. electric utilities in 1974. This problem has resulted in costly repairs and lost operating time. The potential seriousness of the problem was recently emphasized by the discovery of cracks in large-diameter (26-in; 660-mm) recirculation piping at a domestic BWR. These cracks necessitated replacement of the complete recirculation piping system and will cost 12 to 18 months of operating time.

Earlier EPRI reports (*EPRI Journal*, September 1981, p. 6; November 1981, p. 18) have helped familiarize the industry with the various factors involved in pipe cracking. In most cases, cracks have resulted from intergranular stress corrosion cracking (IGSCC). This status report describes how changing reactor water chemistry can help prevent IGSCC.

Three conditions must be present simultaneously for IGSCC to occur: stress, a sensitized microstructure, and an environment (water chemistry and temperature) that will facilitate cracking. Theoretically, no pipe will ever crack if any one factor is completely eliminated. Eight pipe-cracking remedies have been developed: three that affect stress, three that affect sensitization, and two that affect environment (Table 1). By their very nature, all the stress and sensitization remedies are limited to the specific component to which they are applied. For example, induction heating stress improvement affects cracking in the pipe weld to which it is applied; it does not affect any other weld. Only the water chemistry remedies have the potential of protecting the whole system.

The water in a BWR is similar in purity to laboratory distilled water. It is converted into steam by reactor core heat, condensed into liquid again after passing through the turbine, and reconverted into steam on reentering the core. This process is repeated continuously.

During reactor operation, radiolysis in the reactor core continually decomposes a small amount of water to form free oxygen and hydrogen. Most of the oxygen and hydrogen is stripped from the water by the steam and is subsequently removed from the water circuit by special equipment in the condenser. However, about 200 ppb oxygen and 12 ppb hydrogen remain dissolved in the water in the core when the reactor is at the steady-state full-power operating temperature (288°C; 550°F). During reactor startups and shutdowns oxygen concentration varies with temperature (Figure 1). The important question of which temperature-oxygen combinations facilitate IGSCC has been answered in part under EPRI research (RP1332 and RPT115). The shaded IGSCC danger zone in the figure represents those combinations.

Reducing oxygen levels during reactor startups and shutdowns by deaeration has been highly publicized in the BWR industry. Although helpful during transients, this remedy does little, if anything, to reduce pipe cracking during steady-state conditions (RP1332-2, RPT112-1, RPT115-3, RPT115-4). Deaeration does not affect oxygen levels during steady-state operating conditions, which definitely facilitate IGSCC. The amount of time spent at steady state is about 140 times greater than the amount of time spent in startups. Therefore, to reduce IGSCC further, it is necessary to change water chemistry during steady-state conditions.

Table 1
CAUSES AND REMEDIES FOR BWR PIPE CRACKING

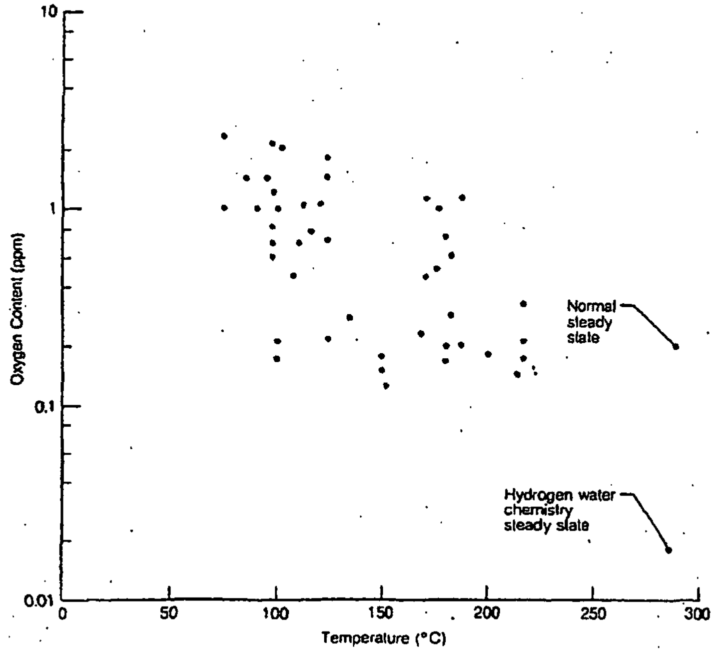
Cause	Remedy
Stress	Induction heating stress improvement
	Heat sink welding
	Last-pass heat sink welding
Sensitization	Solution heat treatment
	Corrosion-resistant cladding
	Alternative materials
Environment	Hydrogen water chemistry
	Impurity control

Hydrogen water chemistry

In hydrogen water chemistry, small amounts of hydrogen gas are added to the reactor feedwater. In the reactor core the added hydrogen recombines with oxygen and other radiolysis products to suppress the net amount of oxygen produced at the steady-state temperature (Figure 1).

Although hydrogen water chemistry experiments were conducted over 20 years ago in several early Norwegian and U.S. test reactors, the concept was not further developed until 1979, when the Swedish utilities and ASEA–Atom conducted a short eight-hour test of hydrogen water chemistry at Oskarshamn-2 and demonstrated that hydrogen water chemistry was economically feasible. In 1981 the Swedes conducted a second test at Oskarshamn-2 for four days and obtained detailed water chemistry measurements. These tests showed that hydrogen water chemistry lowered the oxygen concentration to levels that would no longer be expected to facilitate stress corrosion. However, no actual in-reactor corrosion tests were performed. In June 1982 DOE funded

Figure 1 The shaded area represents the temperature-oxygen combinations that facilitate IGSCC in high-purity water. The data points are examples of temperature-oxygen combinations that have been measured in operating BWRs during startup, shutdown, normal steady state, and hydrogen water chemistry steady state.



a 30-day hydrogen water chemistry experiment at Commonwealth Edison Co.'s Dresden-2 plant. During this experiment, EPRI sponsored in-reactor stress corrosion tests that helped confirm hydrogen water chemistry as a powerful antidote for stress corrosion problems (RP1930-2). A $1 million EPRI laboratory research project on hydrogen water chemistry, which has been in progress for two years, supports this conclusion (RP1930-1).

The combined results of the in-reactor and laboratory IGSCC tests show that the oxygen level must be suppressed to 20 ppb to eliminate IGSCC completely. For example, during the Dresden-2 test, a severely sensitized sample of stainless steel was tested under extreme stress and strain, and absolutely no IGSCC was detected. In laboratory tests on full-scale pipes the growth rates of preexisting cracks have been slowed by a factor of 10 as a result of hydrogen water chemistry. If no cracks are present before hydrogen treatment of water, no new cracks are expected to start.

To achieve an oxygen level of 20 ppb during the Dresden-2 test, it was necessary to add 1.5 ppm hydrogen to the feedwater and to use pure oxygen in the off-gas system instead of air. The total cost of both hydrogen and oxygen was less than $1000/day. If a BWR had a 70% capacity factor and a remaining lifetime of 20 years, the total would be about $5 million. Equipment installation would cost an additional $1 million. In contrast, replacement of a complete recirculation piping system is estimated to cost on the order of $500 million, including the cost of replacement power.

Although the stress corrosion benefits from hydrogen water chemistry are expected to be very high, at least one negative side effect exists. The amount of the radioactive isotope nitrogen-16 (N-16) in the steam will increase. The N-16 is formed in the reactor core by the nuclear reaction: oxygen-16 + neutron ⟶ nitrogen-16 + proton. Under normal water chemistry conditions the N-16 reacts with dissolved oxygen to form nitrate (NO_3^-), which is soluble in the reactor water.

Under hydrogen water chemistry conditions there is not enough dissolved oxygen to react with the N-16 to form NO_3^-; the N-16 combines with the hydrogen to form ammonia, NH_3. Ammonia is a volatile gas and is therefore removed from the water by the steam. The N-16 is a very unstable isotope and decays with a half-life of 7.11 s, giving off high-energy gamma rays. Because more N-16 ends up in the steam when hydrogen water chemistry is used, the steam lines and steam turbine will emit more gamma radiation than when normal BWR water chemistry is used. At Dresden-2, the amount of N-16 gamma radiation increased by a factor of 5 during the hydrogen water chemistry test. The turbine is heavily shielded and therefore the increase in N-16 did not significantly increase the radiation dose rate to plant personnel. In general, the N-16 side effect was manageable during the tests at Dresden-2. When maintenance crews had to enter an area where N-16 radiation was high, the hydrogen injection was stopped, and N-16 radiation levels quickly returned to normal. After the maintenance crew left the area, the hydrogen injection was resumed.

The major uncertainties about hydrogen water chemistry revolve around the possibility of long-term negative side effects. The two most important concerns are the hydrogen embrittlement of the nuclear fuel cladding and the redistribution of corrosion products (radiation buildup) within the plant. Although the best technical judgment available indicates that the possibility of either of these effects becoming unmanageable is extremely remote, there is no data base on which to build firm conclusions. At least one fuel cycle with hydrogen water chemistry will be required before a recommendation can be made to the utilities. EPRI is developing a long-term in-reactor test program to address these major uncertainties.

Control of Impurities

Although reactor water contains impurities in small amounts (at the ppm or ppb levels), BWRs generally operate with high-purity water. For example, NRC guidelines specify that reactor water chloride (Cl) concentration be kept below 0.2 ppm and the conductivity below 1 $\mu S/cm$ during plant operation. A solution containing 1 ppm of sodium chloride (NaCl) would have a conductivity of about 2 $\mu S/cm$ and a Cl concentration of 0.6 ppm. Therefore, 1 ppm of NaCl would exceed the NRC specifications. The results of EPRI research projects have shown that maintaining water purity may be just as important as controlling oxygen levels (RP1563-2, RPT115-3, RPT115-6). Impurities increase the size of the IGSCC danger zone.

In accelerated laboratory IGSCC tests as little as 1 ppm of certain impurities eradicated hydrogen water chemistry benefits. To benefit from hydrogen water chemistry, utilities will have to control both oxygen levels and conductivity. Reactor water with only 20 ppb oxygen and a conductivity in the vicinity of 0.2 μS/cm may eliminate any possibility of IGSCC. EPRI has recently stepped up its research to understand the role of impurities in an effort to produce cost-effective water chemistry guidelines. *Project Manager: Michael Fox*

VALVE RESEARCH

The primary goal of valve research in EPRI's Nuclear Power Division is to reduce the amount of plant unavailability attributable to valves in LWR power plants. These R&D activities seek to improve valve maintenance practices and valve performance and reliability and thus reduce the cost of producing electricity. EPRI's initial effort in this area was an assessment of industry valve problems conducted in the mid 1970s (NP-241). It was found that nuclear plant unavailability attributed to valves, valve actuators, and associated control circuits represented approximately three forced outages per plant per year, with an average outage duration of about two days. The value of such unavailability is significant. A study reported in the June 1982 EPRI Journal *(p. 18) indicates that a 1% availability improvement in base-load coal and nuclear generating units combined would represent savings of $2.2 billion nationwide over the seven-year study period.*

In the initial assessment of industry valve problems, which was conducted by MPR Associates, Inc., the concept of key valves evolved. These are valves whose malfunction can result in a forced plant outage, a power reduction, or an extension of a planned outage. It is basically to these valves that the EPRI research effort is directed.

The study concluded that only a small percentage (5–10%) of the total valve population in a nuclear power plant is applied in such a way that failure would result in a forced outage. It should be noted that these key valves are not necessarily safety-related valves. No major differences were found between PWRs and BWRs regarding the causes (seat leakage, stem leakage, actuator malfunction) of valve-related shutdowns.

The study also concluded that forced outages attributable to valves are underreported because of an umbrella or shadowing effect—situations where a valve requires maintenance or repair work during an outage attributed to another system or component. Thus, although the valve could be considered a contributing cause of the outage, this is not reflected in the reported data.

Nuclear plant data collection and evaluation systems originally had many shortcomings. As a result of improvements in these systems, data quantity and usefulness have been increased. Other existing sources of information remain to be assimilated, however, to achieve a comprehensive view of the problem. EPRI's limiting-factors analysis studies, the findings of which are published in four reports (NP-1136 through NP-1139), provide further insight into the causes and the magnitude of nuclear plant availability losses attributable to valves.

On the basis of the efforts described above, two areas were selected for initial EPRI R&D attention: the seat leakage performance of main steam isolation valves (MSIVs) in BWRs and valve stem packing improvements for both PWR and BWR application.

Figure 2 presents a cutaway view of a representative MSIV with the valve bonnet and the actuator removed. Two identical MSIVs are installed in series in each BWR steam line. Technical specifications for BWR plants establish maximum allowable seat leakage rates for MSIVs and require the periodic testing of each valve to verify that this requirement is met.

Work was initiated in early 1979 with Atwood and Morrill Co., Inc., a manufacturer of MSIVs, and General Electric Co., the nuclear steam supply system contractor for BWR plants, to develop a comprehensive test program on MSIV seat leakage performance (RP1243-1, RP1389-1). The goals were first to identify the factors that affect the valves' capability to meet the seat leakage criteria imposed by the local leak rate test (LLRT) and then to identify and verify the effectiveness of corrective actions for improving valve leakage performance.

The program evaluated the effects of such factors as local residual stresses from valve installation welding; forces and moments applied by the connecting pipe; mechanical cycling; thermal cycling; excessive wear and corrosion of critical valve surfaces; and poorly controlled maintenance practices. Of the factors investigated, corrosion of the valve seating surface (or changes in the friction coefficient) and inadequate maintenance practices were found to be the most significant contributors to the seat leakage problem. Program results are reported in NP-2381 and NP-2454.

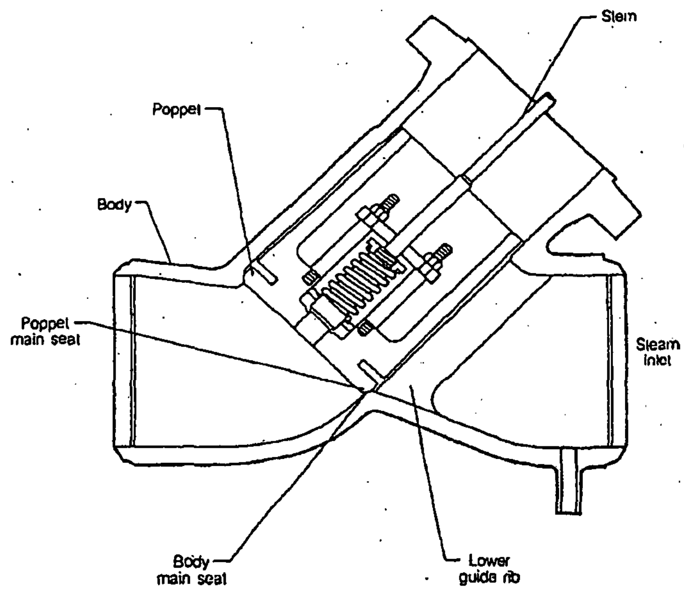


Figure 2 BWR main steam isolation valve. EPRI has sponsored a test program to determine the factors that affect valve seat leakage performance and to evaluate ways to improve this performance.

PROPOSED AMENDED CONTENTION NYS-26/26B & RIVERKEEPER TC-1/1B:

ATTACHMENT 17

Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249
Tel (914) 788-2055

Fred R. Dacimo
Vice President
License Renewal

May 16, 2008

Re: Indian Point Units 2 & 3
Docket Nos. 50-247 & 50-286

NL-08-084

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: **Reply to Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex**

Reference: NRC letter dated April 18, 2008; "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Nos. 2 and 3, License Renewal Application – Time-Limited Aging Analyses and Boraflex"

Dear Sir or Madam:

Entergy Nuclear Operations, Inc is providing, in Attachment I, the additional information requested in the referenced letter pertaining to NRC review of the License Renewal Application for Indian Point 2 and Indian Point 3. The additional information provided in this transmittal addresses staff questions for Time-Limited Aging Analyses and Boraflex.

There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. R. Walpole, Manager, Licensing at (914) 734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 5-16-08.

Sincerely,

Patrick W. Conroy for
Fred R. Dacimo
Vice President
License Renewal

A128
NRR

Attachment:

1. Reply to NRC Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex

cc: Mr. Bo M. Pham, NRC Environmental Project Manager
Ms. Kimberly Green, NRC Safety Project Manager
Mr. John P. Boska, NRC NRR Senior Project Manager
Mr. Samuel J. Collins, Regional Administrator, NRC Region I
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
IPEC NRC Senior Resident Inspectors Office
Mr. Paul D. Tonko, President, NYSERDA
Mr. Paul Eddy, New York State Dept. of Public Service

ATTACHMENT I TO NL-08-084

REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING

LICENSE RENEWAL APPLICATION

Time-Limited Aging Analyses and Boraflex

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 and 3
DOCKETS 50-247 and 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION (LRA)
REQUESTS FOR ADDITIONAL INFORMATION (RAI)
REGARDING TIME-LIMITED AGING ANALYSES AND BORAFLEX

Time-Limited Aging Analyses

RAI 4.3.1.8-1

License renewal application (LRA) Section 4.3.1 states "[c]urrent design basis fatigue evaluations calculate cumulative usage factors (CUFs) for components or sub-components based on design transient cycles." For CUF values listed in LRA Tables 4.3-13 and 4.3-14, please describe the details of how various environmental effects are factored into the calculation of the CUF using F_{en} values.

Response to RAI 4.3.1.8-1

For CUF values listed in LRA Tables 4.3-13 and 4.3-14, the F_{en} values were determined as described below.

NUREG-1801 calls for using formulas provided in NUREG/CR-5704 for austenitic stainless steel and NUREG/CR-6583 for carbon steel and low-alloy steel to calculate environmentally assisted fatigue correction factors (F_{en}). For IPEC, none of the locations identified in Tables 4.3-13 and 4.3-14 (NUREG/CR-6260 locations) are made of carbon steel, so calculation of F_{en} for carbon steel was not required.

The environmentally assisted fatigue correction factor (F_{en}) for **low alloy steel** was calculated as follows.

F_{en} =	$\exp(0.929-0.00124T-0.101\, S^{*}T^{*}O^{*}\dot{\varepsilon}^{*})$		based on NUREG/CR-6583, Eq. 6.5b
T =	25°C	Reference temperature for original fatigue curves	
S^{*} =	S	(0 < S (Sulfur) ≤ 0.015 wt.%)	
S^{*} =	0.015	(S ≥ 0.015 wt.%)	NUREG/CR-6583, Eq. 5.5a
T^{*} =	0	(T (Temperature) < 150°C)	
T^{*} =	T-150	(T = 150-350°C)	NUREG/CR-6583, Eq. 5.5b
O^{*} =	0	(DO (Dissolved Oxygen) < 0.05ppm)	
O^{*} =	ln(DO/0.04)	(0.05 ppm ≤ DO ≤ 0.5 ppm)	
O^{*} =	ln(12.5) = 2.53	(DO > 0.5 ppm)	NUREG/CR-6583, Eq. 5.5c
$\dot{\varepsilon}^{*}$ =	0	($\dot{\varepsilon}$ (strain rate) > 1%/s)	
$\dot{\varepsilon}^{*}$ =	$\ln(\dot{\varepsilon})$	(0.001 ≤ $\dot{\varepsilon}$ ≤ 1%/s)	
$\dot{\varepsilon}^{*}$ =	ln(0.001)	($\dot{\varepsilon}$ < 0.001%/s)	NUREG/CR-6583, Eq. 5.5d

There are four low alloy steel subcomponents for each unit in Tables 4.3-13 and 4.3-14 for the NUREG-6260 locations at IPEC. The F_{en} was calculated for each location on each unit as shown below.

<u>IP2</u>

$T_{(reference\ temperature\ °C)}$ = 25 Reference temperature for original fatigue curves

$O^{*}_{(RCS)}$ = 0.0 RCS dissolved oxygen is ≤ 50 ppb

Since $O^{*}_{(RCS)}$ equals 0.0, S*, T*, and ε^{*} terms are eliminated.

F_{en} = exp(0.929-(0.00124)(T))

$F_{en\ (bottom\ head\ to\ shell)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (inlet\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (outlet\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (surge\ line\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45

<u>IP3</u>

$T_{(reference\ temperature\ °C)}$ = 25 Reference temperature for original fatigue curves

Since $O^{*}_{(RCS)}$ equals 0.0, S*, T*, and ε^{*} terms are eliminated.

F_{en} = exp(0.929-(0.00124)(T))
$F_{en\ (bottom\ head\ to\ shell)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (inlet\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (outlet\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (surge\ line\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45

The environmentally assisted fatigue correction factor (F_{en}) for **austenitic stainless steel** was calculated as follows.

F_{en} = exp(0.935-T′O′ε′)		NUREG/CR-5704, Eq. 13
T′ = 0	(T < 200ºC)	
T′ = 1	(T ≥ 200ºC)	NUREG/CR-5704, Eq. 8a
O′ = 0.260	(DO < 0.05 ppm)	
O′ = 0.172	(DO ≥ 0.05 ppm)	NUREG/CR-5704, Eq. 8b
$\dot{\varepsilon}'$ = 0	($\dot{\varepsilon}$ > 0.4%/s)	
$\dot{\varepsilon}'$ = ln($\dot{\varepsilon}$ /0.4)	(0.0004 ≤ $\dot{\varepsilon}$ ≤ 0.4%/s)	
$\dot{\varepsilon}'$ = ln(0.0004/0.4)	($\dot{\varepsilon}$ < 0.0004%/s)	NUREG/CR-5704, Eq. 8c

There are four stainless steel subcomponents for each unit in Tables 4.3-13 and 4.3-14 for the NUREG-6260 locations at IPEC. The F_{en} will be calculated for each location on each unit as shown below.

<u>IP2</u>

$T'_{(Surge\ line)}$ = 1.0 (T ≥ 200ºC)
$T'_{(Charging\ nozzle)}$ = 1.0 (T ≥ 200ºC)
$T'_{(SI\ nozzle)}$ = 1.0 (T ≥ 200ºC)
$T'_{(RHR\ piping)}$ = 1.0 (T ≥ 200ºC)

$O'_{(All)}$	= 0.260	RCS dissolved oxygen is ≤ 50 ppb
$\dot{\varepsilon}'_{(all)}$	= -6.91	Assume bounding strain rate
$F_{en\ (Surge\ line)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (Charging\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (SI\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (RHR\ piping)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	

<u>IP3</u>

$T'_{(Surge\ line)}$	= 1.0	(T ≥ 200ºC)
$T'_{(Charging\ nozzle)}$	= 1.0	(T ≥ 200ºC)
$T'_{(SI\ nozzle)}$	= 1.0	(T ≥ 200ºC)
$T'_{(RHR\ piping)}$	= 1.0	(T ≥ 200ºC)
$O'_{(All)}$	= 0.260	RCS dissolved oxygen is ≤ 50 ppb
$\dot{\varepsilon}'_{(all)}$	= -6.91	Assume bounding strain rate
$F_{en\ (Surge\ line)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (Charging\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (SI\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (RHR\ piping)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	

<u>RAI 4.3.1.8-2</u>

Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (SRP-LR) Section 4.3.2.1.1.3 provides the basis for the staff acceptance of an aging management program to address environmental fatigue. It states, "[t]he staff has evaluated a program for monitoring and tracking the number of critical thermal and pressure transients for the selected reactor coolant system components. The staff has determined that this program is an acceptable aging management program to address metal fatigue of the reactor coolant system components according to 10 CFR 54.21(c)(1)(iii)." The staff is unable to determine if the Fatigue Monitoring Program of Indian Point 2 and Indian Point 3 contain sufficient details to satisfy this criterion. Please provide adequate details of the Fatigue Monitoring Program such that the staff can make a determination based on the criterion set forth in SRP-LR Section 4.3.2.1.1.3. Also, please explain in detail the corrective actions and the frequency that such actions will be taken so the acceptance criteria will not be exceeded in the period of extended operation.

<u>Response to RAI 4.3.1.8-2</u>

The IPEC Fatigue Monitoring Program was compared to the program described in NUREG-1801 (GALL), Section X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary. The program description in the GALL report is directly applicable to the IPEC units. As indicated in LRA Section B.1.12, during the period of extended operation the IPEC program will be consistent with the GALL program with one exception. The exception to GALL is that rather than performing periodic updates of CUF calculations, IPEC periodically assesses the number of transient cycles compared to calculation assumptions and updates the CUF calculations, if necessary. Based on this comparison including evaluation of the exception, the approvals set forth in the GALL report apply to the IPEC Fatigue Monitoring Program.

The program description was modified per letter NL-08-21, Indian Point Nuclear Generating Units Nos. 2 & 3 License Renewal Application Amendment 2, dated January 22, 2008. This letter commits to complete CUF calculations for all areas identified in NUREG-6260 (LRA Table 4.3-13 for IP2 and LRA Table 4.3-14 for IP3), incorporating the effect of the reactor coolant environment, for IP2 and IP3. Once these CUF calculations are complete (at least 2 years prior to the period of extended operation), IPEC will ensure that the cycles analyzed in the new or updated calculations are included in the Fatigue Monitoring Program. IPEC will continue to manage the effects of fatigue throughout the period of extended operation by monitoring cycles incurred and assuring they do not exceed the analyzed numbers of cycles.

As required by IPEC technical specifications, the Fatigue Monitoring Program tracks actual plant transients and evaluates these against the design transients. The plant transient counts are updated at least once each operating cycle. This frequency is acceptable since the evaluation during each update determines if the number of design transients could be exceeded prior to the next update. The Fatigue Monitoring Program ensures that the numbers of transient cycles experienced by the plant remain within the analyzed numbers of cycles and hence, the component CUF calculations remain valid.

The program requires corrective actions before exceeding the analyzed number of transient cycles. The corrective actions are implemented in accordance with the IPEC corrective action program. IPEC may perform further reanalysis if cycle counts approach analyzed numbers. These calculation updates will be governed by Entergy's 10 CFR 50 Appendix B Quality Assurance (QA) program and include design input verification and independent reviews ensuring that valid assumptions, transients, cycles, external loadings, analysis methods, and environmental fatigue life correction factors will be used in the fatigue analyses. Repair or replacement of the affected component(s), if necessary, will be done prior to exceeding the allowable CUF in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by Entergy's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements of the ASME Code Section XI.

RAI 3.0.3.2.3-1

Indian Point 2 Updated Final Safety Analysis Report, Revision 20, dated 2006, Section 14.2.1 on page 55 of 218, states in part that:

> "Northeast Technology Corporation report NET-173-01 and NET-173-02 are based on conservative projections of amount of boraflex absorber panel degradation assumed in each sub-region. These projections are valid through the end of the year 2006."

Please confirm that the Boraflex neutron absorber panels in the Indian Point Unit 2 spent fuel pool have been re-evaluated for service through the end of the current licensing period. Also, please discuss the plans for updating the Boraflex analysis during the period of extended operation.

Response to RAI 3.0.3.2.3-1

Boron-10 areal density gage for evaluating racks (BADGER) testing was performed in February 2000, July 2003, and again in July 2006. Using the latest test data and RACKLIFE code projections, the Boraflex neutron absorber panels in the Indian Point Unit 2 spent fuel pool will meet the Technical Specification requirements through the end of the current licensing period. The next BADGER test will be performed prior to the end of calendar year 2009. As required by the Boraflex Monitoring Program (LRA Section B.1.3), periodic BADGER testing and RACKLIFE code projections will continue through the period of extended operation to confirm acceptable Boraflex condition.

The referenced section of the Indian Point 2 Updated Final Safety Analysis Report will be updated in the next revision.