Domestic Utilities
American Electric Power
Carolina Power & Light
Commonwealth Edison
Consolidated Edison
Duquesne Light
Duke Power
Georgia Power
Florida Power & Light
Houston Lighting & Power
New York Power Authority
Northeast Utilities
Northern States Power
Pacific Gas & Electric
Public Service Electric & Gas
Rochester Gas & Electric
South Carolina Electric & Gas
Southern Nuclear
Tennessee Valley Authority
TU Electric
Union Electric
Virginia Power
Wisconsin Electric Power
Wisconsin Public Service
Wolf Creek Nuclear

International Utilities
Electrabel
Kansai Electric Power
Korea Electric Power
Nuclear Electric plc
Nuklearna Elektrana
Spanish Utilities
Taiwan Power
Vattenfall

OG-97-060

NRC Project Number 686
WCAP-14575

June 13, 1997

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

Subject: Westinghouse Owners Group
Response to NRC Request for Additional Information on WOG Generic Technical Report WCAP 14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components"

Reference: NRC letter dated April 18, 1997 from P.T. Kuo to R.A. Newton, Westinghouse Owners Group

Attached are the Westinghouse Owners Group responses to the NRC's Request for Additional Information on WCAP-14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components".

Please distribute these responses to the appropriate people in your organization for their review. These responses will provide the basis for our discussion at an NRC / WOG meeting scheduled with the License Renewal Project Directorate office for July 10, 1997 from 1-3 PM.

If you have questions on specific technical issues from these RAIs for this meeting, please contact Frank Klanica, Westinghouse, at (412) 374-6392, Charlie Meyer, Westinghouse, at (412) 374-5027, or myself at Wisconsin Electric Power Company, (414) 221-2002.

Very truly yours,

Roger A. Newton

Roger A. Newton, Chairman
LCM/LR Working Group
Westinghouse Owners Group

D043 1/1

cc: R.J. Prato, USNRC, (1L, 1A)
Pao-Tsin, USNRC, (1L, 1A)
LCM/LR Working Group (1L, 1A)
Steering Committee (1L, 1A)
C.E. Meyer, ECE 4-22 (1L, 1A)
F. Klanica, ECE 573D(1L, 1A)
A.P. Drake, W, (1L, 1A)

Proj-686

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EDRE-EMT-126

From: EQUIPMENT DESIGN & REGULATORY ENGINEERING
WIN: 284-6392
Date: June 12, 1997
Subject: License Renewal Evaluation: Aging Management for Class 1 Piping and Associated Components

To: C. E. Meyer / EC-East 4-22

cc: C. H. Boyd / EC-East 573D
M. A. Gray / EC-East 3-04
D. C. Bhowmick / EC-East 3-04
R. L. Brice-Nash / EC-East 3-04
C. C. Kim / EC-East 509D

References:
1) US NRC letter dated April 18, 1997, Project No. 686, "Request For Additional Information Regarding the Westinghouse Owners Group Topical Report WCAP-14575 (TAC No. M96439 and M92414)"

Attached are the response to the NRC RAI's on Class 1 Piping (ref.). Note that PVP-Vol. 306 (ref. 43 in WCAP-14575) is attached to the responses. The Word file, which is the electronic copy of the responses without PVP-Vol. 306, will also be sent to you.

Frank Klanica
F. Klanica
Engineering and Materials Technology

RAI and Report Section Cross Reference Table

RAI NUMBER	DESCRIPTION	REPORT SECTION
1	(a) Describe how the AMP for fatigue addresses thermowell high cycle fatigue.	Tables 3-2 through 3-17
	(b) Explain what is intended by last sentence in 3.3.2 and how the conclusion was used to develop an AMP for fatigue.	3.3.2
2	(a) Discuss how step 2 (ISI), in the AMP for fatigue, assures that the licensing basis criteria has been met.	4.2.1
	(b) Discuss how step 3 (flaw tolerance & LBB), in the AMP for fatigue, demonstrates that the licensing basis criteria has been met.	4.2.1
	(c) Discuss how environmental effects are addressed in the AMP for fatigue.	4.2.1
	(d) Describe test data used to establish PVRC criteria for water flow velocity in the AMP for fatigue.	Table 4-8
3	Identify other aging management programs to be included.	No Change
4	Describe how aging will be managed for components in inaccessible areas.	No Change
5	Describe how the owner's group reviewed applicable generic communications and associated licensee commitments.	3.1
6	Are current activities to manage boric acid corrosion consistent with the programs developed and implemented in response to Generic Letter 88-05?	No Change
7	Discuss why the program with the set of attributes identified would be an effective aging management program.	No Change
8	Will continuing commitments be addressed in plant specific applications for license renewal (rather than generically)?	No Change
9	Provide a stress corrosion cracking aging management program for components listed in NUREG-1557 (pages B-66 and B-67).	No Change
10	Provide an assessment for the cracking of thermal sleeves.	No Change

REQUEST FOR ADDITIONAL INFORMATION
WESTINGHOUSE OWNERS GROUP TOPICAL REPORT WCAP-14575
"License Renewal Evaluation: Aging Management for Class 1 Piping and Associated Pressure Boundary Components"

Request for Additional Information

1. In Section 3 of the report, aging effects that require management during the extended period of operation are identified.

1a. Section 3.3.1 of the report indicates that Class 1 thermowells are the only pressure components that are subjected to a dynamic load associated with flow-induced vibration and potentially susceptible to high-cycle fatigue damage. Describe how the aging management program for fatigue addresses this issue.

Response:

Based on the analyses of the Class 1 thermowells, the degradation sustained from the effects of low and high cycle fatigue was determined to be insignificant to all Class 1 thermowells covered by the evaluation. The hot and cold leg fast response RTD thermowells were considered to be representative for all Class 1 thermowells based on the thin walled tapered tip of the fast respo ne thermowell and the RCS flow loads. Since the thermowells are out of the frequency range c seismic excitations, the only significant fatigue loads are induced by the turbulent lift loads wh ch could potentially cause high cycle fatigue. Significant margin was calculated for high cycle fatigue based on an allowable for 10^{11} cycles. Extrapolating the design life from 40 years to 60 years has an insignificant effect on the allowable. Thus, the high cycle fatigue evaluations that were performed for the current licensing basis are valid for the license renewal term. The low cycle fatigue investigation showed that the exempt rules specified in NB-3222.4(d) were satisfied. Since the peak stress intensities are not a function of cycles, the fatigue waiver evaluations that were performed for the current licensing basis are valid for the license renewal term. Therefore, the Class 1 thermowells are not considered to be fatigue-sensitive items for license renewal. The report will be revised to include the Class 1 thermowells in Tables 3-2 through 3-17, as applicable. A rating of N, which are components that were considered to not be an issue, will be given for all of the potential aging effects for the Class 1 thermowells.

Request for Additional Information

1b. Section 3.3.2 contains a description of a fatigue assessment of B31.1 piping design. The final sentence concludes, "Based on the successful operating history of fossil plants (using the B31.1 approach) and the high cost of evaluating these stresses with a detailed fatigue analysis, this was considered to be an acceptable approach for nuclear plants." Explain what is intended by this sentence. Also, explain how this conclusion was used to develop the aging management program for fatigue.

Response:

This section discusses the fatigue methodology for B31.1 piping design and the results of the EPRI report (TR-102901) that compares the B31.1 criteria to the ASME code, Section III criteria for Class 1 piping. The last sentence is intended to explain why some operating nuclear

power plants, that have piping designed to the B31.1 code criteria. did not need to have the piping reanalyzed when the new ASME code, Section III criteria was introduced. "The successful operating history of fossil plants designed using the B31.1 approach, indicates that the structural adequacy of a piping design can be maintained using the B31.1 code criteria. In addition, since it is very costly to reanalyze the B31.1 piping design to ASME code, Report No. TR-102901 is judged adequate to show that no further evaluations are needed to ensure safe operation."

The two sentences, shown above, will replace the sentence in question.

The aging management program for fatigue is based on the proposed industry position on fatigue evaluations for license renewal and the B31.1 code requirements. Section 4.2.1.2, "Process to Qualify Fatigue for B31.1 Piping Components" applies to plants that have the B31.1 code included in the current licensing basis.

Request for Additional Information

2. In Section 4.2.1 of the report, the aging management program for fatigue is described.

2a. Step 2 of the program appears to allow the use of ASME Code, Section XI, inspection techniques to demonstrate the acceptability of a component as an alternative to meeting the licensing basis criteria in Step 1. The staff has not endorsed this position. Discuss how the use of this alternative provides assurance that the licensing basis criteria has been met at a facility.

Response

The report will be modified to incorporate the revised proposed industry position on fatigue.

The last paragraph of step 2 of the "Position", in Section 4.2 (Revision 0, page 152), will be clarified to explain how ISI requirements manage cracking caused by fatigue and why a program based on these ISI requirements will continue to be effective during an extended period of operation. This paragraph will be modified as follows:

"Since the examination methods and related evaluations described above will allow the detection, evaluation, and/or repair of minor cracks, caused by fatigue, this management option will maintain the intended function of the Class 1 piping and pressure boundary components. Specifically, the flaw acceptance standards in subsection IWB-3500, which are the current industry accepted criteria, are stringent enough that indications identified by the evaluations do not represent a loss of the reactor coolant pressure boundary intended function of the Class 1 piping and pressure boundary components under design-basis loads. It is noted that other plant programmatic requirements (Technical Specifications - RCS Operational Leakage Limits) require a plant shutdown to repair the degradation before an intended function would be lost. The criteria of IWB-3500 would allow further evaluation and/or repair of indications prior to the loss of the intended function of the Class 1 piping and pressure boundary components. These inspections are required periodically and are not tied to a specific design life. Because the transient loading frequencies are not anticipated to significantly increase during the license renewal period, these inspection periods will remain effective throughout the license renewal period, as long as CLB cyclic commitments are met (as confirmed in step 1)."

It should be noted that the proposed industry position defines the CLB as a combination of the fatigue design basis, the fatigue operating basis, and the regulatory oversight process. This definition includes any requirements for assuring that the plant operates within commitments on cyclic duty, and items such as resolution of generic fatigue issues or regulatory information notices and bulletins.

Request for Additional Information

2b. Step 3 of the program appears to allow the use of flaw tolerance or leak-before-break analysis to demonstrate the acceptability of a component as an alternative to meeting the licensing basis criteria in Step 1. The staff has not endorsed these positions. Discuss how the use of these alternatives will demonstrate that the licensing basis criteria has been met at a facility.

Response:

The report will be modified to incorporate the revised proposed industry position on fatigue.

The second and fourth paragraphs of step 3 of the "Position", in Section 4.2 (Revision 0, page 153), indicate that the structural integrity of the Class 1 piping and pressure boundary components is maintained by using either the flaw tolerance or leak-before-break analyses.

In section 4.2.1.3, "Evaluation of Actual/Postulated Flaw", the second paragraph on page 160, (shown below), explains how the structural integrity of the Class 1 piping is maintained by using the flaw tolerance approach.

> "Fracture mechanics techniques can be employed for justification of continued operation in two ways. First, inspection results may identify flaws that can be shown by analysis either not to grow as the consequence of continued service or to grow at such a low rate that current inservice inspection schedules (or more frequent inservice inspections) will be able to ensure the integrity of Class 1 piping and pressure boundary components. Second, flaws can be postulated in Class 1 piping and pressure boundary components

and shown either not to grow or to grow slowly. Through such a "flaw tolerance" approach, appropriate inservice inspection schedules can be formulated and compared to current requirements. This comparison may show that the current schedule is adequate, may need to be accelerated, or can be less frequent. Also, an appropriate schedule can be determined for Class 1 piping and pressure boundary components not included in the current inspection program."

The Leak-Before-Break (LBB) criteria is more stringent than the flaw tolerance criteria. In section 4.2.2.1, "Aging management program for Thermal Aging - LBB Analyses (AMP-3.6)", the second and third paragraphs on page 160, (excerpts shown below), explain how the structural integrity of the Class 1 piping is maintained by using the LBB criteria.

> "The LBB criteria includes elastic-plastic fracture mechanics analyses and leak rate calculation methodologies. These evaluations follow the recommendations and criteria proposed in NUREG 1061, Volume 3 and the methodology described in the Standard Review Plan 3.6.3. The procedures include: (1) the postulation of an assumed flaw at the governing location, which will be the location with the combination of highest stress and limiting fracture toughness, with a demonstration of flaw stability under applied loads and any subsequent flaw growth; (2) postulation of an assumed through-wall flaw at the governing location with a demonstration that any leakage is assured of detection, with appropriate margin, using the installed leak detection system at the plant, when the piping is subjected to normal operating loads; and (3) demonstration of adequate margin between the leakage flaw size and the critical flaw size under faulted condition loading."

Satisfying this criteria will demonstrate that the structural integrity of the Class 1 piping and pressure boundary components will be maintained for a through wall crack in the piping.

Request for Additional Information

2c. The discussion following Step 3 of the program describes issues regarding environmental effects on fatigue. The location of this discussion is confusing because Step 3 appears to be an alternative to Step 1. SECY 95-245 provided the staff recommendation regarding the use of environmental fatigue data for license renewal evaluations. Clarify the method in which the staff recommendation in SECY 95-245 is addressed by the program.

Response:

The report will be modified to incorporate the revised proposed industry position on fatigue. This revised position considers environmental effects for an extended period of operation.

The revised industry position has expanded the first step to clarify that environmental effects will be considered, as appropriate. Specifically, the first step in the revised process (identifying fatigue sensitive sub-components) includes consideration of reactor water environmental effects. If sub-components are identified as fatigue sensitive based on this preliminary screening, the second step quantitatively addresses the significance of environmental effects. For sub-components that are not identified as fatigue sensitive in step 1, the CLB cyclic duty is checked to confirm it envelopes the number of cycles expected through the license renewal term.

The appropriate changes to sections 3.3, 3.4, 4.2.1, Tables 3-2 thru 3-16, 4-4, 4-5, 4-6, and 4-7, and Figures 4-1and 4-2 will be made to incorporate the revised industry position that considers environmental effects

The report requires utilities to follow current industry activities to completion until the final NRC position is given in this area.

Request for Additional Information

2d. Table 4-8 lists parameters developed by the PVRC to identify components where the environmental effect on fatigue life are not considered significant. Describe the test data used to establish the criteria for water flow velocity.

Response:

The criteria for water flow velocity in Table 4-8 will be corrected to show >3 m/sec (9.843 ft/sec) instead of >3 m/sec (3.3 ft/sec). The test data used to establish the criteria for water flow velocity should be described in the references listed in reference 43 (attached).

Request for Additional Information

3. Are there aging management programs (other than the ASME Code examinations that you cite) that you want the staff to generically credit to participating WOG Plants? If so, identify each program and provide more detail about actions taken, results, and validity for the period of extended operation. For example, the report does not describe programs related to generic communications and technical specifications other than to list the documents in a table. Existing augmented examinations should also be described and justified to demonstrate that the effects of aging will be adequately managed so that the intended function will be maintained for the period of extended operation.

Response:

Section 4 of the report contains all of the generic aging management programs for which the WOG wants the staff's review and approval. The WOG uses six attributes to provide the details that are applicable to all domestic WOG plants. Plant-specific License Renewal applications will provide additional details consistent with their CLB and as deemed necessary by the utility.

Current commitments (those that are part of the CLB), which are credited for aging management, will be addressed in plant-specific applications. As stated in the Rule, 10 CFR 54.33, CLB requirements will continue during the extended period of operation unless otherwise justified by the utility and approved by the NRC.

Request for Additional Information

4. Are there any relevant components in areas inaccessible for maintenance and inspection? If so, how will their aging be managed?

Response:

There may be relevant components, for the aging management programs described in Sections 4.1 and 4.2, that are located in areas inaccessible for maintenance and inspection. It is the individual plant's responsibility to identify their own inaccessible areas relative to RCS piping and associated components during preparation of their LR application. The plant specific inaccessible areas cannot be addressed in a generically applicable report. The possibility of components located in areas that are considered to be inaccessible for maintenance and inspection is discussed below for the AMPs in Sections 4.1 and 4.2.

The evaluation has determined that the aging effects that require management are identified in sections 4.1 and 4.2. Section 4.1 provides current industry practices and section 4.2 provides additional activities and attributes required to manage aging effects.

Section 4.1 of the report describes the attributes for current aging management programs for wear of closures and stress relaxation of bolts. Current industry practices for these AMPs should already account for the possibility of RCP and Class 1 valve closure flanges and bolting located in areas that are considered to be inaccessible for maintenance and inspection.

Section 4.2 of the report describes additional activities and program attributes for fatigue and thermal aging. Both of these programs include analyses, in addition to inspections, as options to manage the aging effects from fatigue and thermal aging for license renewal. If relevant components are located in areas inaccessible for maintenance and inspection, the analyses options could be considered as acceptable alternatives to the inspection options.

Request for Additional Information

5. Describe how the owner's group reviewed applicable generic communications and associated licensee commitments. The staff found generic communications of the aging effects of the RCS not discussed in the report, for example Bulletin 82092 on bolting, and Generic Letter 85-20 on thermal sleeves.

Response

Section 3.1 will be revised to describe the process used by the WOG to review Generic Communications. An updated review will be performed to capture any additional items that occurred, or were missed, since the original review was done three years ago.

The following information was provided to the authors in the WOG GTR template:

"Identify plant-specific operating experience which identifies aging effects. Review operating and maintenance history. This should include, but is not limited to: plant maintenance data, inservice inspection data, industry experience, NPRDS data, vendor data, EPRI reports, NUREGs, Licensee Event Reports (LERs), DOE Aging Management Guidelines (AMGs), NUMARC License Renewal Industry Reports, NRC generic letters/bulletins/notices, the Westinghouse Information Delivery System (IDS), and the internet. Many of these sources are readily available in the technical library. When using the internet, as any other reference, ensure that the information is timely. Identify any unresolved issues, see the Westinghouse technical lead for the latest EPRI memorandum."

Request for Additional Information

6. Your report states that current activities are sufficient to manage boric acid corrosion. Are current activities, as referenced in your report, consistent with the programs developed and implemented in response to Generic Letter 88-05? If no consistent with GL 88-05, describe current activities and provide a basis for how your current programs provide reasonable assurance that the aging effect will be managed during the period of extended operation.

Response:

The WOG feels that the current activities referenced (leakage monitoring and walkdowns) are consistent with responses to Generic Letter 88-05. Additional details will not be provided in this generically applicable report. Plant-specific License Renewal applications will provide these details as consistent with their CLB and as deemed necessary by the utility.

The WOG supports the position that boric acid corrosion of external surfaces is not related to aging. This type of corrosion is caused by an event, in this case, degradation of the reactor coolant pressure boundary. The degradation causes an abnormally harsh environment that can cause degradation. Since current activities monitor for the event (leakage) and the degradation it causes (loss of material) and repair degradation as necessary, the effects of events are managed by current activities and do not require separate aging management programs.

Request for Additional Information

7. Discuss why the program with the set of attributes identified would be an effective aging management program (i.e., provide reasonable assurance that a program with the attributes described would be able to detect and correct the effects of aging before the component would reach a condition in which it could not perform its intended function under all CLB design conditions.) Explain why all six attributes identified in your report may not be necessary for a program.

Response:

The purpose of the six WOG attributes is to describe the generic aging management programs in sufficient detail for use by the utility and review/approval by the NRC. These descriptions, as contained in section 4, explain how the program manages an aging effect(s) to maintain the appropriate intended function(s) for an extended period of operation. Section 4 also contains text explaining why these programs will remain effective during an extended period of operation.

All six attributes may not be necessary based on the type of the activity performed by the program. For example, a program that uses analytical techniques to ensure intended functions are maintained do not have a surveillance technique. An analysis does not inspect anything. The analysis (and specifically the results) would be part of the acceptance criteria used to

determine further actions: acceptance, further analysis with less conservative assumptions, or replacement.

Request for Additional Information

8. Will continuing commitments be addressed in plant specific applications for license renewal (rather than generically)?

Response:

Current commitments (those that are part of the CLB), which are credited for aging management, will be addressed in plant-specific applications. As stated in the Rule, 10 CFR 54.33, CLB requirements will continue during the extended period of operation unless otherwise justified by the utility and approved by the NRC.

Request for Additional Information

9. NUREG-1557 (pages B-66 and B-67) lists stress corrosion cracking as an aging effect, for a number of components in the reactor coolant system requiring aging management. For some of the identified components the issue was unresolved. Provide an aging management program for these components.

Response:

It appears that the two unresolved issues in NUREG-1557, on pages B-66 and B-67, are identified as follows.

#1) IGSCC can occur under the operating conditions (water chemistry) during shutdown because oxygen is introduced to primary coolant during cool down to control CRUD-bursts, and coolant is exposed to air during many shutdowns (S-V-38).

#2) The potential of cracking in cladding remote from welds should be addressed. SS cladding may have regions of low delta ferrite that have been sensitized during PWHT and thus susceptible to IGSCC; ASME Sect. XI requires inspection of weld and weld regions (SI S-1).

Open issue #1, (S-V-38), is judged to be resolved for the class 1 piping and associated components. For IGSCC to occur in austenitic stainless steel, three things must be present: a susceptible material, stress approaching or exceeding yield strength, and an aggressive environment such as an oxidizing environment. In the absence of one of the three above conditions, IGSCC will not occur. The stress in the class 1 piping and associated components will not approach or exceed yield strength during shutdown. The report cites steps taken by Westinghouse (pages 92 and 93) to eliminate or reduce the susceptibility of class 1 piping and associated component materials to sensitization and from coming in contact with an aggressive environment. The efficiency of this practice in the prevention of IGSCC has been demonstrated by years of operating experience without exhibiting IGSCC in the class 1 piping and associated components. Therefore, an additional program to manage the aging effects from IGSCC is not necessary for the class 1 piping and associated components because IGSCC should not be an issue.

Open issue #2, (SI S-1), does not apply to the WCAP-14575 report. The scope of WCAP-14575 does not include any class 1 piping and associated components that have cladding material.

Request for Additional Information

10. The industry has experienced cracking of thermal sleeves. Provide an assessment for the cracking of thermal sleeves.

Response:

There are five designs for the thermal sleeves which were installed in some of the branch connection nozzles in the Reactor Coolant Loop. These five designs are numbered 0 thru 4 where 0 is referred to as the original design, and 1 thru 4 are referred to as design generations 1 thru 4, respectively. Figures 2-11 and 2-12 show the thermal sleeve design configurations for large and small bore nozzles. Table 2-1 identifies the thermal sleeve design generations used for each plant and also identifies the plants that did not use thermal sleeves.

In 1977, the attachment welds anchoring the thermal sleeve to the 3" charging nozzle failed on the Farley Unit 1 plant during hot functional testing. The thermal sleeve attachment weld failure was through the attachment or anchor welds on the thermal sleeve which was a design generation 3. This industry issue is identified in Table 3-1 by document IN 82-30 on page 78 of the report.

Westinghouse investigated this problem and concluded that no safety concerns relative to loose or missing thermal sleeves were identified. Based on the evaluations performed, the probable cause, of the operating plant thermal sleeve attachment weld cracks, was high cycle fatigue resulting from flow induced vibration. The generic analyses indicated that the nozzle integrity was not expected to be compromised by the loss or removal of the sleeves. And, for the original design and design generations 1 and 2, there was no obvious cause for concern.

While no safety concern had been identified, the potential financial and plant availability exposure which could result from the existence of migrating thermal sleeves in the primary reactor coolant system were recognized. Therefore, Westinghouse suggested that those utilities with thermal sleeve design generations 3 and 4, should remove them at the next convenient opportunity. For plants under construction, Westinghouse issued field change notices to remove the thermal sleeves.

Thermal transient stresses are considerably higher in the nozzles with thermal sleeves removed, and in each case, the design basis for the plant was revised to show acceptability. Typically, Westinghouse has been able to show that the nozzles are still acceptable with thermal sleeves removed.

Since this issue has been resolved in the past, and included in the design basis, it is not judged necessary to re-address the issue for the aging management report.

PVP-Vol. 306, Fatigue and Crack Growth: Environmental Effects,
Modeling Studies, and Design Considerations
ASME 1995

STATUS OF PVRC EVALUATION OF LWR COOLANT ENVIRONMENTAL EFFECTS ON THE S-N FATIGUE PROPERTIES OF PRESSURE BOUNDARY MATERIALS

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ABSTRACT

The Pressure Vessel Research Council (PVRC) has made a concerted effort in the past several years to compile and evaluate test data related to the effects of light-water reactor (LWR) coolant environments on the fatigue behavior of structural materials used in LWR pressure boundary applications. This paper presents the status and findings-to-date of the part of this PVRC effort concerned with effect of the LWR environment on S-N fatigue behavior. The overall purpose of this activity is to formulate recommendations to the ASME Code for methods and procedures to include any needed considerations of the coolant environment in LWR design.

A large amount of test data has been collected and analysis of this database shows that some combinations of environmental and mechanical test conditions can result in reduced S-N fatigue life of ferritic and austenitic steels compared to an air environment. The extent of reduction depends on the values of the influencing variables which include the dissolved oxygen content, the temperature and possibly the flow velocity of the coolant water, and the amplitude and the rate of cyclic straining. In addition, for ferritic steels, the sulfur content of the material may be another factor. Independent "screening" values of these variables for which environmental effects are deemed acceptable have been defined and are discussed.

An important need is the modeling and characterization of the environmental effects when conditions exceed the independent screening values and some examples of this effort are presented. In spite of the large amount of collected data, there are several areas of incomplete definition and the need for test data are noted. In addition, ASME Code implementation includes need for other analyses and test data to formulate design methods for conditions such as variations in strain rate and temperature during a cyclic transient.

INTRODUCTION

Beginning in early 1992, the Pressure Vessel Research Council (PVRC) has been engaged in the compilation, analysis, and evaluation of S-N fatigue data for tests conducted in water that is similar to or simulates light-water reactor (LWR) coolant water chemistries. This activity was prompted by results from a number of tests from laboratories in several countries which were reported in the few years preceding 1991. Although there had been papers in the preceding 25 years or so discussing the effect of LWR-type coolant water on fatigue behavior, the more recent results tend to show fairly large reductions in S-N cyclic life for some combinations of mechanical and water chemistry conditions. It may be noted that the earlier investigations focussed on the effects of the water environment on fatigue crack growth properties with less emphasis on S-N fatigue life behavior.

Because of the potential impact that the more recent S-N fatigue results could have on the fatigue design basis in Section III of the ASME Boiler and Pressure Vessel Code (ASME Code) and consequently, on pressure boundary integrity of LWRs, the ASME Code's Board on Nuclear Codes and Standards (BNCS) requested PVRC for an evaluation of the available information. Specifically, the BNCS request to PVRC in June 1991 was in summary: "*BNCS looks to PVRC to obtain, characterize and report in sufficient detail to ASME such data as may be useful to ASME in its evaluation of the fatigue curves of Sections III and XI.*"

PVRC's response to the BNCS request initially included several actions. Administratively, a steering committee and three working groups (WGs) were formed to coordinate the activities. The three WGs and their scopes are:

WG on S-N Data Analysis — To collect, compile, and analyze S-N data and make recommendations for changes in fatigue design curves to the Section III of the Code.
WG on da/dN Data Analysis — To collect, compile, and analyze da/dN data and prepare fatigue crack growth curves for Section XI and other sections of the Code.
WG on Evaluation Methods — To conduct in-depth review of fatigue design criteria and methods in Section III of the Code and make recommendations for changes and improvements.

This paper focuses on the status and preliminary findings of the WG on S-N Data Analysis. The WG on da/dN Data Analysis was formerly a Materials Properties Council (MPC) activity and the results of their effort have been regularly presented to ASME Code's Section XI and in ASME PVP Conference papers. The activities of the WG on Evaluation Methods are somewhat more longer range and it is currently formulating preliminary findings and position statements on various items related to fatigue design procedures.

SUMMARY AND FEATURES OF THE S-N DATABASE

In early 1992, PVRC organized a workshop consisting of a number of contributions by experts and investigators in various aspects of fatigue and related environmental effects. The information presented has been published in Reference 1 (1992). One of the purposes of the workshop was to determine the potential worldwide sources of relevant S-N test results and information.

As a result of inquiries and solicitation, a large amount of data totaling nearly 2800 S-N test results in air and water environments has been received and compiled. A listing of the sources of the data and some remarks about the data are presented in Table 1. Summaries of this database categorized by material, test environment, and test parameters for carbon and low alloy steels are presented in Table 2 and in Table 3 for austenitic steels and nickel alloys. The water environments used in these investigations varied substantially. For the purposes of this evaluation, all environments which contained boric acid, lithium hydroxide, and less than 10 ppb oxygen were considered to represent a pressurized-water reactor (PWR) environment. The environments which contained high-purity water were considered to represent boiling-water reactor (BWR) environments. There is not a single water composition which could be called typical of a PWR or a BWR environment. For this reason, environmental variables that have been shown to be important in describing the environmental effects on fatigue are being considered in the development of models.

Although a large amount of data has been collected as indicated by Tables 2 and 3, detailed information about test specimens, test condition, and test materials are missing in several instances. Additionally, it will be evident in later discussions that test data for some vital ranges of variables are not covered in spite of the large database.

EVALUATION OF AIR ENVIRONMENT CARBON AND LOW ALLOY STEEL TEST DATA

The original Section III fatigue design curves were based on a relatively small amount of test data. In the case of carbon and low alloy steels, the data were limited to room temperature tests utilizing either cantilever bending or axial hourglass specimens. Plots of

TABLE 1
SOURCES OF TEST DATA AND REMARKS

COUNTRY	FACILITY	REMARKS
Japan	17 plus laboratories	Nearly all axial specimens tests, some hour-glass specimens; test results compiled in "JNUFAD" database (Reference[2])
U.S.A.	Argonne Lab	Axial specimens
	Mat. Eng. Assoc.	Axial specimens
	General Electric	Cantilever bend specimens tested in autoclave connected to Dresden 1 BWR plant
	Babcock & Wilcox	Tubular specimens; as-milled processed ID test surface, fossil plant water chemistry
Germany	Siemens KWU	Test material and water chemistry information incomplete
	MPA Stuttgart	Test material and water chemistry information incomplete
Russia	Promotey Institute	Test material and water chemistry information incomplete

TABLE 2
SUMMARY OF PVRC S-N DATABASE FOR CARBON AND LOW ALLOY STEELS

Material	Test Environment	Number of Data Points	Temperature, °C	Strain Amp., %	Strain Rate, %/sec	Oxygen Content, ppm
Carbon Steel	Air	191	25 to 288	0.11 to 1.78	9.8 to 0.004	
	PWR	45	300	0.11 to 1.27	5.1 to 0.4	
	BWR	221	240 to 300	0.13 to 1.8	20 to 0.0004	0.001 to 8
Low Alloy Steel	Air	425	25 to 350	0.09 to 8.6		
	PWR	28	288	0.17 to 0.4	0.4 to 0.004	0.003 to 0.009
	BWR	369	50 to 290	0.15 to 1.22	2.4 to 0.0004	0.005 to 8
Weld Metal	Air	20	25	0.14 to 1.5	1 Hz	
	BWR	41	290	0.16 to 0.8	6.4 to 0.0004	0.2 to 8

TABLE 3
PVRC S-N DATABASE FOR AUSTENITIC STEELS
AND NICKEL ALLOYS

Material	Test Environment	Number of Data Points	Temperature, °C	Strain Amp., %	Strain Rate, %/sec	Oxygen Content, ppm
Inconel 600	Air	100	25 to 288			
	BWR	90	288	0.25 to 0.8	0.04 to 0.06	
Inconel Weld	Air	13	25 to 280	0.175 to 0.8	0.4	
	BWR	18	288	0.25 to 0.9	0.4 to 0.04	0.2
304 SS	Air	234	25 to 300	0.11 to 2.82		
	BWR	182	260 to 300	0.12 to 1.3	0.03 to 0.06	0.2 to 8.0
316 SS	Air	295	21 to 650	0.13 to 1.52	0.04 to 4.0	
	BWR	89	288	0.13 to 1.5	0.4 to 0.004	0.2 to 8.0
Alloy 800	Air	24	20 to 427	0.2 to 3.0		
310 SS	Air	8	26	0.3 to 4.0		
348 SS	Air	15	427	0.7 to 1.67		
347SS	Air	118	20 to 350	0.2 to 1.3		
Inconel [illegible]	Air	206	20 to 427	0.15 to 4.2		

the data and the derivation of the mean life or "best fit" curves can be found in the Code Criteria Document (Reference 3; 1969). Since only room temperature test data were available, adjustment for temperature effects were made through the temperature variation of the elastic modulus in the calculation of the so-called "fictitious stress". The net result of this procedure is that when these mean curves are adjusted to derive total strain amplitude (or range) versus cyclic life at higher temperatures, the curves are shifted upwards with increasing temperature. More recently, the Code curves for austenitic steels and nickel base alloys have been revised based on a expanded database. However, the fictitious stress procedure is still used in Code design analysis.

The PVRC database contains a considerable amount of air environment baseline tests on test materials utilized for water environment tests; these air data have been compared to the Code mean "best fit" curves as well as to other fitted curves. Figure 1 shows a S-N plot of the air test data for carbon steels in the PVRC compilation. (NOTE: For this and other plots in this paper, cyclic life is generally based on life at 25% load drop from a stable hysteresis loop.) It can be noted that in the life range below 100,000 cycles, most of the values are below the ASME mean curve. Part of the reason is that a considerable number of the tests in the PVRC compilation are at higher temperatures up to 288 C (550 F), and the ASME mean curve when adjusted to higher temperatures results in an upward shift as discussed above.

Recently, Argonne National Laboratory (ANL) reported a statistical analysis of much of the database in the PVRC compilation (Keisler, et al., 1994). The analysis provides mean fit results for carbon and low alloy steels for air and water environments and includes the effect of temperature, sulfur content of the test material, and water chemistry variables. The ANL mean for carbon steels in air at 288 C is shown in Figure 1 and it can be seen that it is slightly lower than the ASME mean and provides a better fit to all of the data. However, it may be noted that the low side scatter from the ANL mean can range up to about a factor of 3 on cyclic life. The scatter factor is about 4 to 5 in relation to the ASME mean. These factors will be referred to later, in connection with water environment tests.

Figure 2 presents a similar comparison of air test data in the PVRC compilation for low alloy nuclear pressure vessel steels with ASME and ANL mean curves. In this case, the ASME and the ANL mean curves are very similar in the life range below 10,000 cycles. At higher cycles, the ANL mean is lower than the ASME mean. Similar to the carbon steel data, the low side scatter ranges up to a factor of 4 on life. One purpose of general plots such as Figures 1 and 2 is to identify potential outliers; this has been an active task in the PVRC activity.

Although the data shown in Figures 1 and 2 are quite extensive, they are limited to tests conducted to provide a baseline for materials tested in water environments. Air environment S-N fatigue tests have been conducted by a number of other investigators (Conway and Sjodahl, 1991; Yoshida, et al., 1978; General Electric, 1966 and 1968) in the case of carbon steels. These additional carbon steel data have been examined in the PVRC work and appear to fall into approximately the same scatterband as the data shown in Figure 1. Examination of air test data for low alloy steels, not in the present PVRC compilation, remains to be done.

In summary, a good database of air environment S-N fatigue test results for carbon and low alloys steels at temperatures of interest to LWR applications has been compiled. A remaining task is to determine the best representation of these data for ASME Code purposes.

EVALUATION OF WATER TEST DATA FOR CARBON AND LOW ALLOY STEELS

The results of the examination and analysis of the compiled data for S-N fatigue tests on carbon and low alloy steels conducted in LWR-type water environments show that the severest detrimental effects on cyclic life occurs when the test conditions involved a combination of certain factors:

- High test temperatures but in the range of normal LWR coolant temperatures
- High dissolved oxygen content in the water, higher than normal LWR operating conditions
- Slow cyclic strain rate, i.e., low frequency
- Strain amplitudes (range) involving plastic strains
- Relatively high sulfur content in the test material

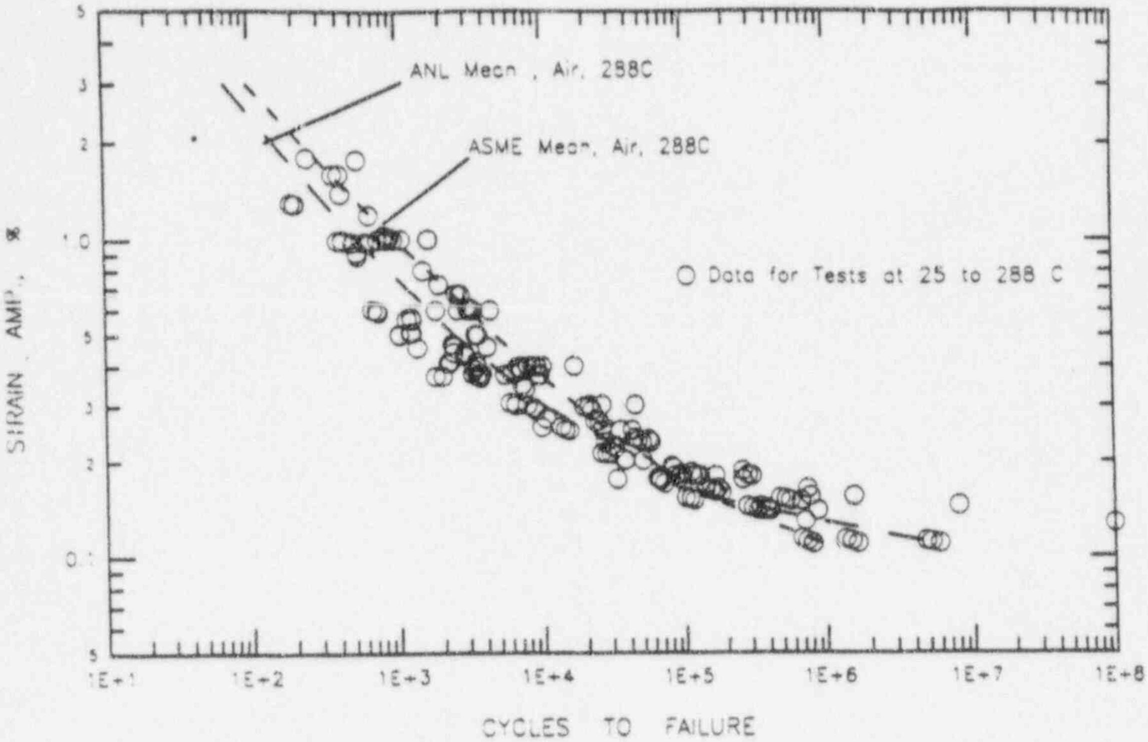

FIGURE 1. COMPILATION OF AIR TEST DATA FOR CARBON STEELS AND COMPARISON TO ASME AND ANL MEAN FIT CURVES

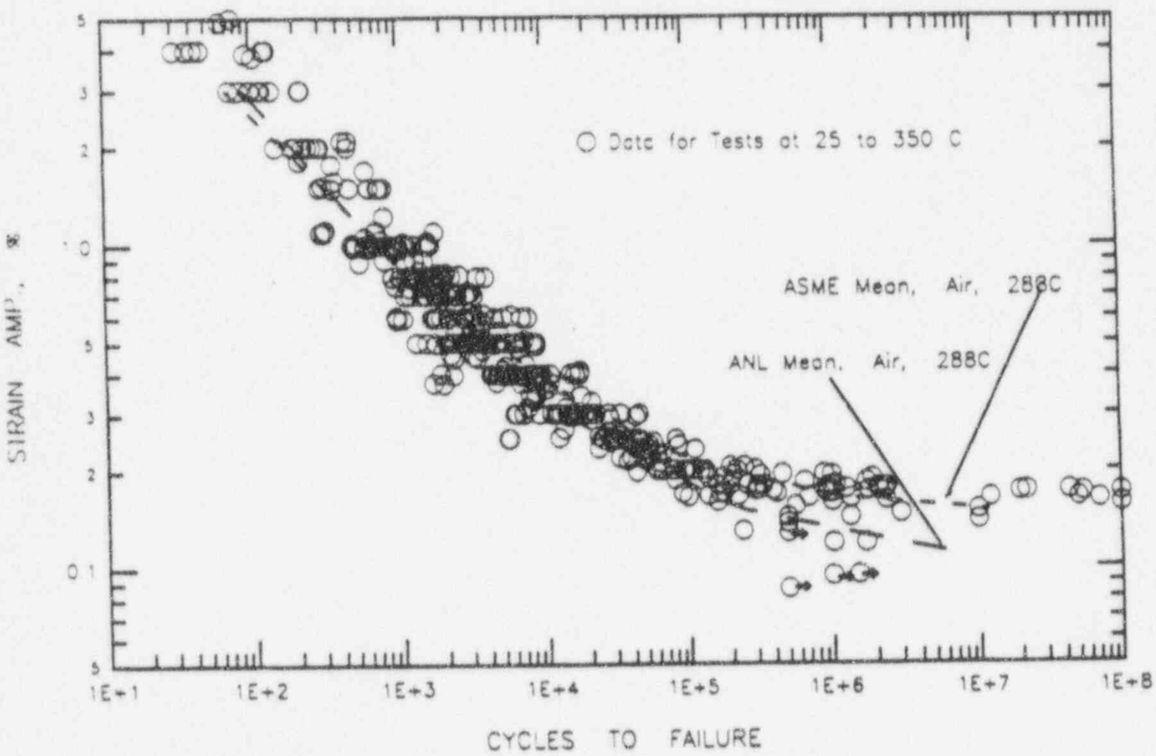

FIGURE 2. COMPILATION OF AIR TEST DATA FOR LOW ALLOY STEELS AND COMPARISON TO ASME AND ANL MEAN FIT CURVES

Although the detrimental effect on S-N life can be large for the worst combinations of these test conditions, these worst-case combinations are generally not typical of LWR operating conditions. The combination of very low strain rates and the relatively large strain ranges that result in large environmental effects do not seem to be typical of events in operating plants. In addition, the high oxygen levels at which much of the data have been obtained are above the levels typical of BWR plants. Therefore, one of the tasks in the PVRC activity consisted of defining a tentative set of criterion values for test and material parameters where the environmental effects would be expected to be moderate or acceptable. This required quantification of "moderate" or "acceptable" environmental effects with respect to the air environment data used in the development of the ASME Code fatigue design curves. Recalling that the analysis of the collected air environment test data indicated a factor of about 4 for temperature and data scatter effects, a factor of 4 on the ASME mean life was chosen as a working definition of "moderate" or "acceptable" water environment effect.

Based on examination of the database, it was determined that values of independent parameters as listed in Table 4 should result in only a moderate detrimental effect on cyclic life. It should be noted that independent means that only one criterion needs to be satisfied, regardless of the values of the other parameters. It has been observed that in order to have a large effect of the environment on the S-N fatigue life, a critical combination of conditions is necessary. If any one of the conditions is missing, the effect of the environment on the fatigue life will be moderate. For example, if the strain rate is greater than 0.1% per second, only a moderate environmental effect is expected even if the dissolved oxygen is high, the temperature is 288 C (550 F), and the material has a high sulfur content. Additional discussion of the selection of the values for the independent criterion has been presented by Van Der Sluys (1993). A major task of the PVRC WG on S-N data analysis is the validation of each of the criterion listed in this table.

TABLE 4
VALUES OF INDEPENDENT PARAMETERS FOR ACCEPTABLE OR MODERATE ENVIRONMENTAL EFFECTS ON THE S-N FATIGUE LIFE OF CARBON AND LOW ALLOY STEELS

Strain Amplitude	≤0.1%
Strain Rate	≥0.1%/sec
Dissolved Oxygen	≤0.1 ppm
Temperature	≤150°C
Sulfur in Steel	≤0.003%
Water Flow Velocity	>3 m/sec

A plot which examines the validity of the values derived for each of the independent criterion for moderate environmental effects for carbon and low alloy steels is shown in Figure 3. It can be seen that a factor of 4 on the ASME mean curve encompasses a large portion of the data for tests which meet any one of the independent criterion value. Although not shown in the figure, a factor of 5 would encompass virtually all of the data. It may be noted that although Figure 3 shows only the ASME mean curve for carbon steel reduced by a factor of 4, Reference 3 shows that the ASME mean curves for carbon steels and for low alloy steels are very similar. Thus, the carbon steel curve with the factor of 4 in Figure 3 would also apply to low alloys steels.

Another consistency check of the criterion values for moderate environmental effects can be made using the ANL statistical analysis model (Keisler, et al.; 1994) mentioned earlier. This model when calculated for parameter values of 0.1 ppm dissolved oxygen, 288 C, 0.015% sulfur, and 0.001%/sec strain rate results in a mean life curve which is approximately a factor of 4 on life reduced below the ANL mean air curve at 288 C. In this case, the 0.1 ppm dissolved oxygen is the governing independent criterion.

With two exceptions, an adequate validation of the values of independent criterion for moderate environmental effects has been found. The exceptions involve the sulfur content and the flow velocity criteria. Definitive experimental data for the effects of these two variables on S-N behavior are lacking. However, results of fatigue crack growth tests involving these variables indicate that low sulfur content in the test material or high flow velocities significantly diminish crack growth rate. The values adopted for the independent criterion for S-N behavior are based on fatigue crack growth results and do require confirmation.

EVALUATION OF WATER ENVIRONMENT TEST DATA FOR AUSTENITIC STEELS AND NICKEL ALLOYS

Table 3 shows that the PVRC database contains quite a large number of tests in LWR-type water environment on austenitic steels and nickel-base alloys. Evaluation of the results for these materials has lagged in the PVRC effort because of the greater apparent concerns about the behavior of carbon and low alloy steels.

Figure 4 provides an overview of the water environment test results for base metal and welds of nickel-base Alloy 600 and base metals of Types 304 and 316NG (a low carbon, nitrogen-added 316) austenitic stainless steels. Except for a few results, the data are above the ASME design curve despite the fact that the test conditions include high oxygen contents and slow strain rates. Also, the Alloy 600 weld metal appears to behave similar to the base metal. The evaluation of the data has not proceeded to the point of establishing

independent criterion values where only moderate environmental effects are observed as for ferritic steels.

In austenitic stainless steels, a metallurgical phenomenon of sensitization can occur when the material is held at intermediate temperatures or in weld heat affected zones (HAZs). Sensitization is known to aggravate stress corrosion cracking of austenitic stainless steels in high oxygen LWR water. The PVRC compilation includes water environment S-N test results for sensitized austenitic stainless steels and the data are shown in Figure 5. The results for sensitized 304 SS in this figure clearly show that sensitization can aggravate the environmental effects of high temperature water. In contrast, the results for 316NG shows very little effect and are within the scatterband of results in Figure 4. There are at least two metallurgical reasons for the difference. These are that 316NG has a lower carbon content and the sensitizing heat treatment applied was a 2-hour holding time compared to 10 hours for the 304 material. Both factors would result in greater sensitization in the 304 material and presumably greater sensitivity to a water environment. As mentioned before, weld HAZs can also become sensitized but there are no data available to determine the behavior of typical weld HAZ in LWR-type water environments.

STATISTICAL ANALYSIS OF ENVIRONMENTAL EFFECTS

The Argonne National Laboratory (ANL) statistical analysis (Keisler, et al.; 1994) discussed earlier predicts mean S-N fatigue life curves for carbon and low alloy steels for air and water environments and includes the effect of temperature, sulfur content of the test material, and water chemistry variables. The mean curve predicted for a BWR environment (300 C, 200 ppb oxygen) is compared with data from the database at four strain rates on carbon steel in Figures 6 through 9. The PVRC database includes Babcock & Wilcox (B&W) test results that are not in the database used by ANL in the statistical analysis. These data are included in the data plotted in these figures. There are 28 data points from B&W in these four figures. They do not stand out or represent outliers. The ANL predicted mean curve in Figure 6 fits the low cycle fatigue data well. It underpredicts the high cycle fatigue limit however similar to the effect observed in the air data. For data at a lower loading frequency presented in Figure 7, the predicted curve is a good representation of the data. For the data at an even lower loading frequency of 0.001%/sec, Figure 8 shows that the predicted curve appears to be closer to a lower bound than a prediction of the mean. In this figure, most of the data are the B&W data in the PVRC database. The prediction at the lowest strain rate for which data are available is shown in Figure 9. The predicted curve fits the data well for data at the low strain rate of 0.0004%/sec. In this figure, one of the three data points is a B&W test result.

In general, the Argonne model appears to predict the mean S-N curves of carbon steel in the BWR water environment quite well. It is a ongoing project to compare the model with other data sets in the PVRC data base.

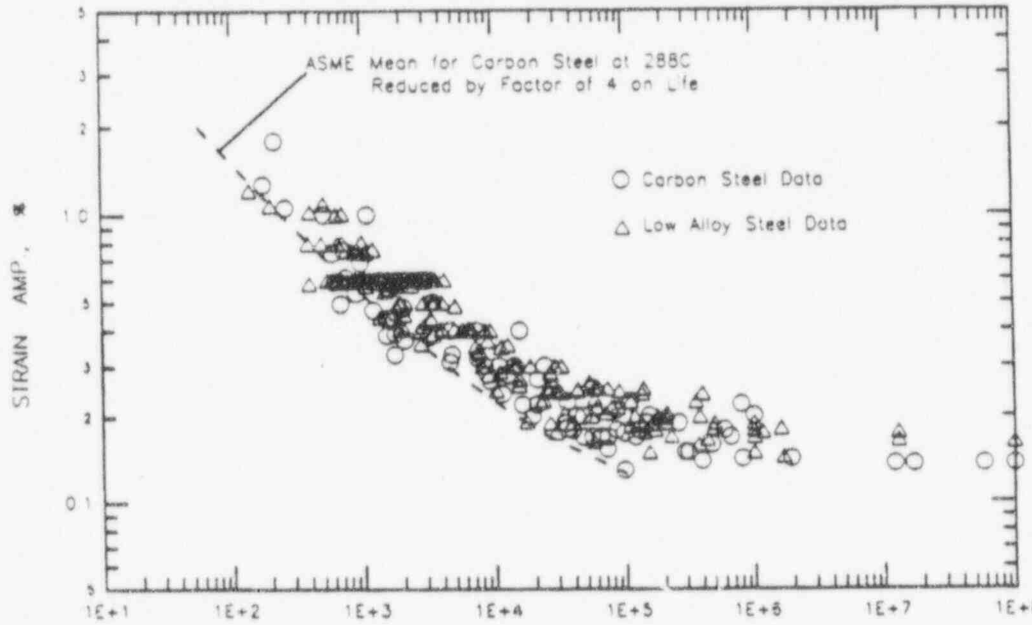

FIGURE 3. COMPILATION OF LWR-TYPE WATER ENVIRONMENT TEST DATA SATISFYING ANY OF THE INDEPENDENT CRITERION FOR MODERATE ENVIRONMENTAL EFFECT AND COMPARISON TO ASME MEAN CURVE REDUCED BY A FACTOR OF 4 ON LIFE

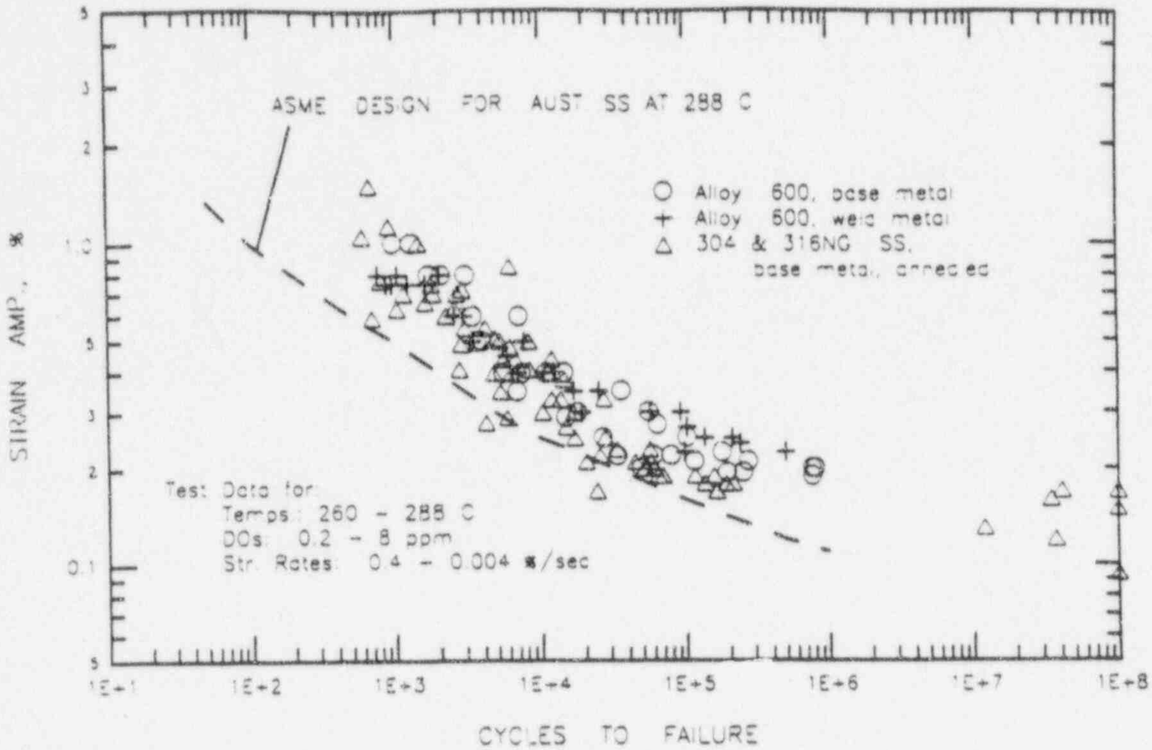

FIGURE 4. LWR-TYPE WATER ENVIRONMENT TEST DATA FOR ANNEALED AUSTENITIC STEELS AND NICKEL-BASE ALLOY 600

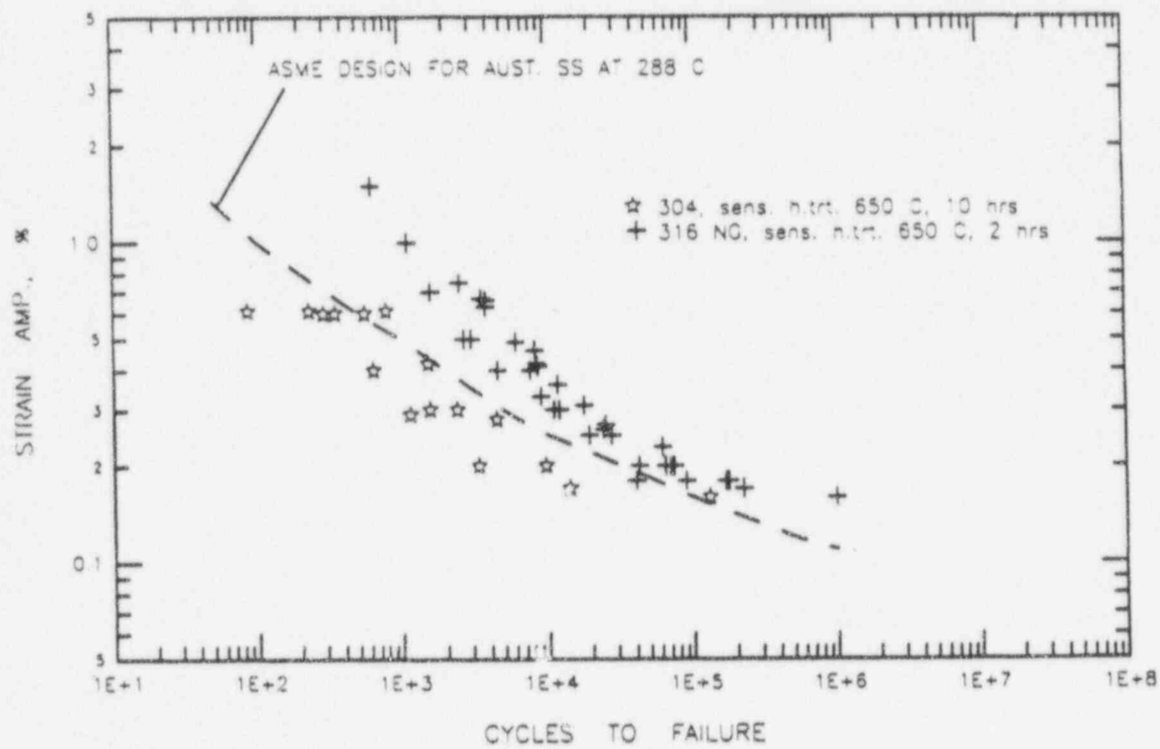

FIGURE 5. LWR-TYPE WATER ENVIRONMENT TEST DATA FOR ANNEALED AUSTENITIC STEELS GIVEN SENSITIZING HEAT TREATMENTS

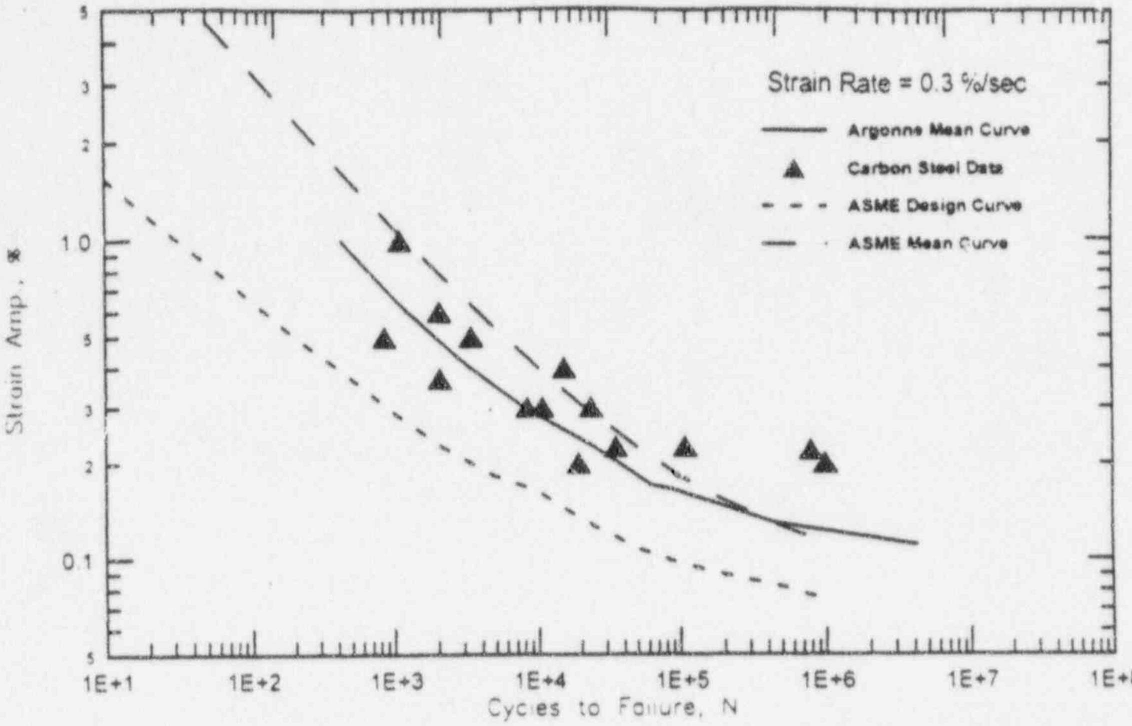

FIGURE 6. DATA FOR CARBON STEEL IN HIGH-TEMPERATURE WATER WITH 200 PPB OXYGEN CONTENT COMPARED WITH THE ASME MEAN AND DESIGN CURVES AND THE ANL MEAN FIT CURVE

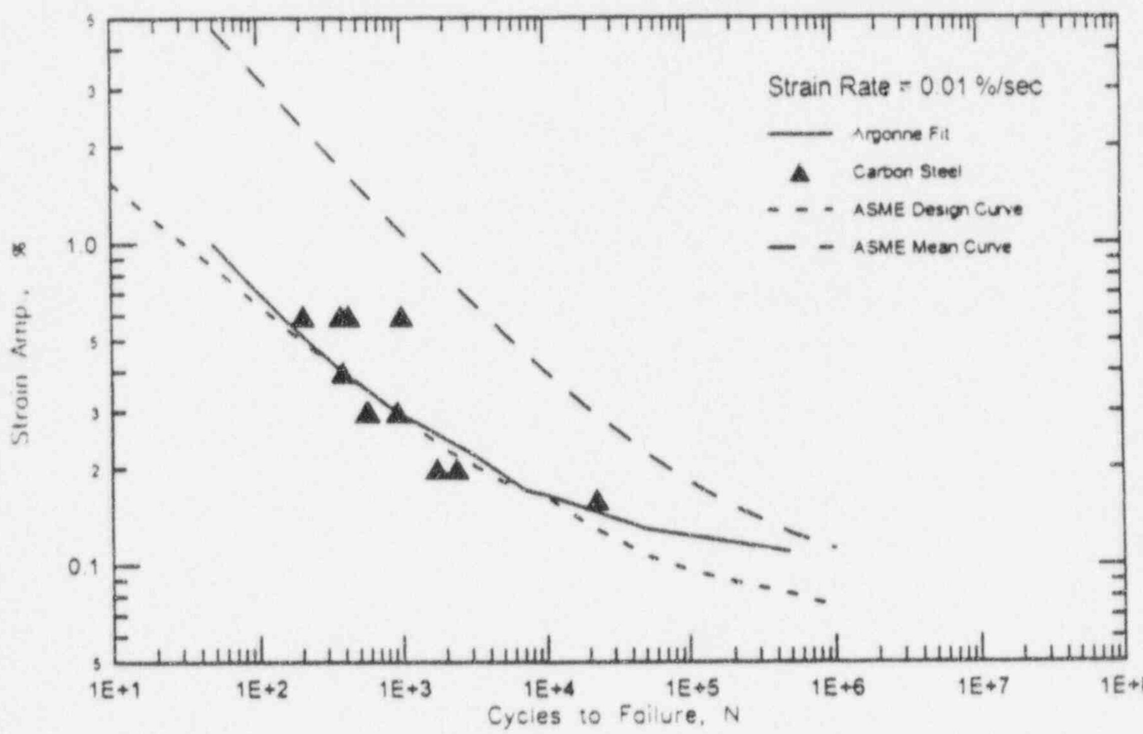

FIGURE 7. DATA FOR CARBON STEEL IN HIGH-TEMPERATURE WATER WITH 200 PPB OXYGEN CONTENT COMPARED WITH THE ASME MEAN AND DESIGN CURVES AND THE ANL MEAN FIT CURVE

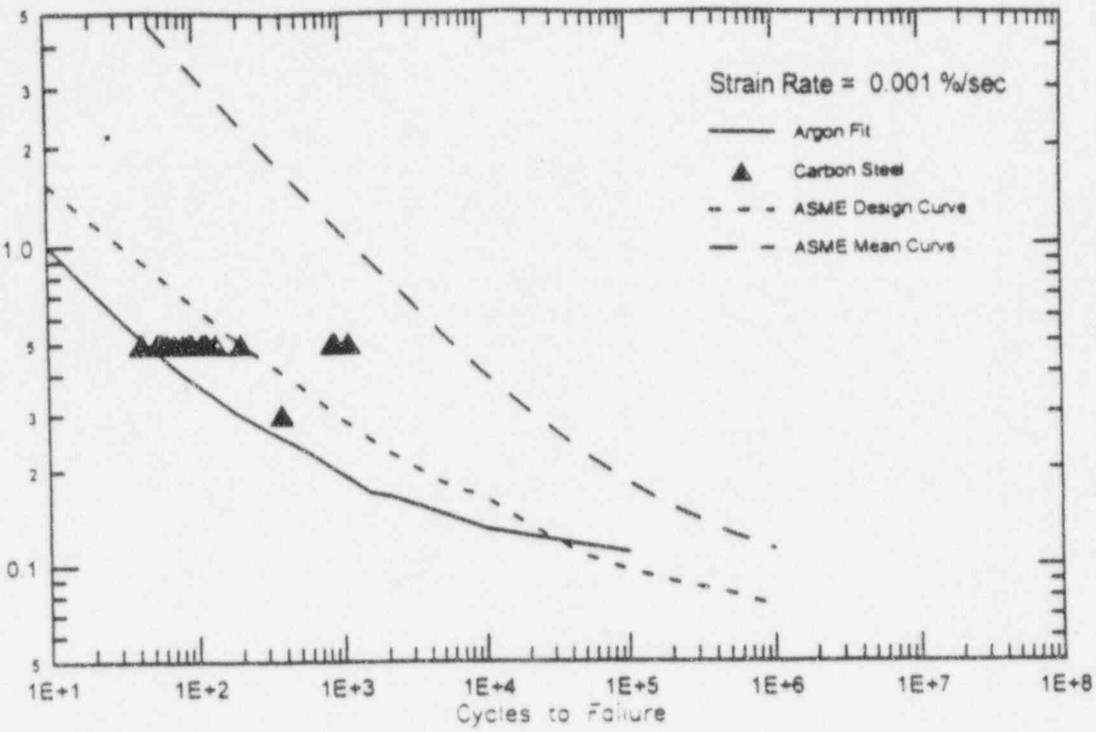

FIGURE 8. DATA FOR CARBON STEEL IN HIGH-TEMPERATURE WATER WITH 200 PPB OXYGEN CONTENT COMPARED WITH THE ASME MEAN AND DESIGN CURVES AND THE ANL MEAN FIT CURVE

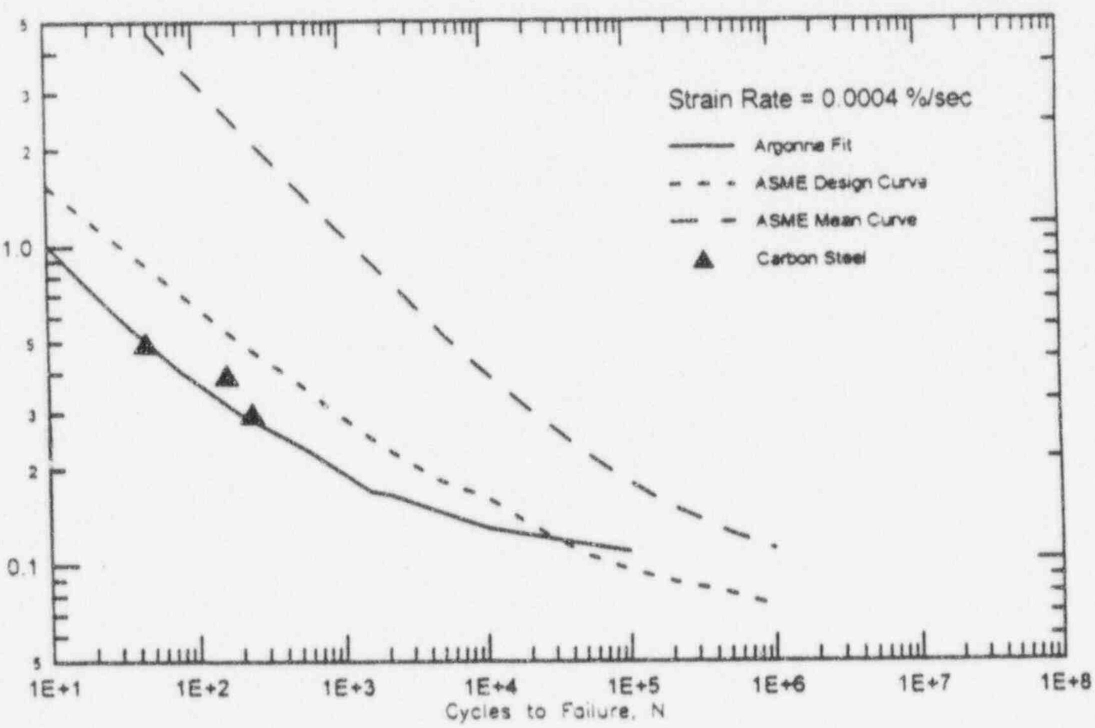

FIGURE 9. DATA FOR CARBON STEEL IN HIGH-TEMPERATURE WATER WITH 200 PPB OXYGEN CONTENT COMPARED WITH THE ASME MEAN AND DESIGN CURVES AND THE ANL MEAN FIT CURVE

ASME CODE IMPLEMENTATION CONSIDERATIONS

As mentioned earlier, the BNCS request to PVRC was to evaluate the results and make recommendations for ASME Code implementation if warranted. The following discusses some of the considerations that will need to be included in the recommendations.

"Crack Initiation" and Code Design

PVRC has had extended discussions of "crack initiation" in S-N fatigue tests especially in the low cycle regime and its relevance to the ASME Code design curve. Although the design curve includes a factor of 20 on mean life in the low cycle regime, some hold the view that a crack is "initiated" when the imposed number of cycles exceeds the design curve cycles. Other opinions contend that the mean cyclic life is an indicator of crack initiation. This difference is compounded in a water environment test where the question is whether the decrease in cyclic life is attributable to earlier crack initiation or to faster crack growth, or both.

Some information about the influence of the water environment on this point can be inferred from cyclic stress-strain data. This refers to the peak tension and compression loads in the cycle in a strain controlled fatigue test. The usual practice is to use the loads at one-half of the cyclic life to construct a plot of the stress at half-life as a function of the strain amplitude. Figure 10 shows such data for three carbon steels tested at 288 C in the PVRC compilation. A couple of features can be noted. For the results shown in the figure, the A106B material shows higher stresses than the A333, Grade 6 or the A508, Cl 1 materials. This is an indication of greater cyclic strain hardening in the A106B test material relative to the other two test materials. More important however, there seems to be no systematic indication that the stresses in a water environment test are different from those in an air environment. This suggests that the "crack initiation" event is not markedly different in cycles between the two environments. If it were a difference in the stresses could be expected.

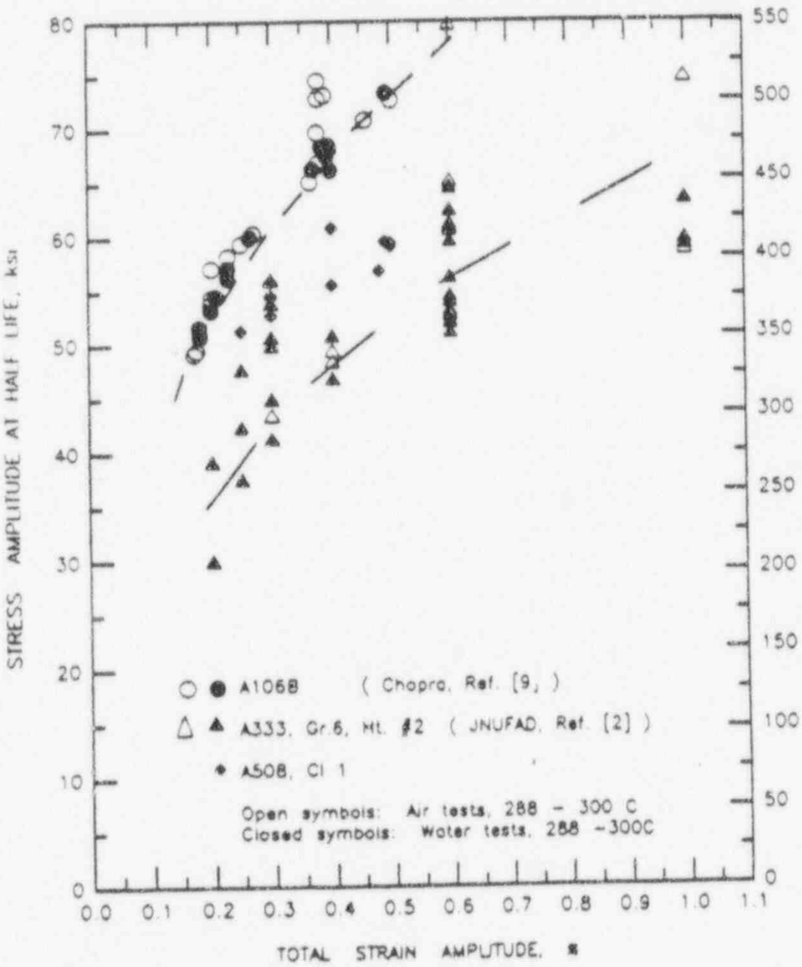

FIGURE 10. CYCLIC STRESS-STRAIN DATA AND TREND CURVES FOR SEVERAL CARBON STEELS IN AIR AND LWR-TYPE WATER ENVIRONMENTS

Design Margins and Statistical Considerations

The fatigue design curves for ferritic steels in Section III of the ASME Code contain the well-known factors of 2 on stress or 20 on cyclic life relative to the mean life curves. However, for the results of statistical analyses such as the ANL formulation, the relation between the traditional factors and the statistical results requires additional consideration. The assessment needs to include the fact that the factor of 20 includes a factor of 4 intended to account for "surface finish, atmosphere, etc." according to Cooper in Reference 1 as well as the fact that environmental effects are a multivariable situation where the significance of global variance and confidence limits is not clear. The PVRC effort in the Working Group on Evaluation Methods is developing guidance on this issue.

Test Data for Design and Operational Transients

All of the test results discussed so far were obtained under relatively constant test conditions such as constant strain rate during the cycle and constant water temperature. However, transient events in operating plants often involve time-varying conditions resulting in varying strain rates and temperatures. For Code implementation, it will be necessary to develop rational procedures to define the applicable parameter values for time-varying actual and design conditions. Several investigators have recently initiated environmental fatigue tests to study these questions and PVRC will assist in the evaluation of the results.

Another design consideration is the evaluation of mean stress effects. The ASME Code provides for this effect in the current design curves. However, S-N behavior in a water environment for conditions of relatively small cyclic stresses but in the presence of high mean stresses is unknown. Currently, it is assumed that the effect is small because the cyclic strain amplitudes are small and in the range where environmental effects are small to moderate without the mean stress, but this assumption requires verification tests when mean stresses are present.

DATA NEEDS

Although a large amount of test data has been collected and evaluated, the environmental effects cannot be definitively characterized due to the lack of some critical data. This situation is understandable in view of the number and range of variables that are involved. Several instances of need for additional tests and data have already been noted. For convenience, these and other areas where additional information would result in more definitive conclusions are listed below:

- Effect of flow velocity on S-N behavior; if verified, this effect would be very significant in defining the relevance of laboratory test results to operating plants
- S-N properties of austenitic stainless steels in PWR primary coolant chemistry water
- S-N properties of carbon steels in BWR coolant water containing 100 to 200 ppb oxygen and at 150 to 250 C
- S-N properties of austenitic, carbon, and low alloy steel weld metals in representative LWR coolant water; also, the properties of weld HAZ for austenitic steels
- More information on the relationship between sulfur content and environmental effects for low alloy and carbon steels
- Environmental effects for high mean stress (R ratio) at low strain amplitudes (or range).

Studies on some of these needs are underway at several laboratories but a long time will be required for results considering long times required for many of the tests and the complexity of the data needs.

SUMMARY AND CONCLUSIONS

The PVRC effort over the past several years on evaluating the effect of LWR-type coolant water on the S-N fatigue properties of pressure boundary materials has resulted in the following accomplishments and tentative findings:

- A large number of S-N fatigue results for tests conducted in baseline air environment and in water environments of various chemistries have been collected and compiled.
- Moderate to large reductions in S-N life relative to life in air environment tests can occur for some combinations of water chemistry, mechanical test parameters, and material characteristics; however, the range of combinations resulting in large effects are generally not typical of operating LWR plants.
- A set of independent "screening" values which define conditions where the environmental effects would be moderate or acceptable have been formulated and partially validated by the available test data.
- The statistical model recently developed by Argonne National Laboratory (ANL) appears to have reasonably good capability of correlating the results of laboratory tests conducted on carbon and low alloy steels in various combinations of water chemistries, mechanical test parameters, and material sulfur content, and for predicting the mean S-N life for these test conditions.
- Design evaluation of the environmental effects of LWR coolants will require additional studies and testing to relate statistical analysis results to design margins, to develop design procedures for design and plant operating events that have varying strain rates and temperature conditions during cyclic transients, to define the effect of high mean stresses, and to obtain additional S-N

data in certain critical ranges of water chemistries and temperatures.

ACKNOWLEDGMENTS

The authors wish to acknowledge the fruitful and informative discussions with many PVRC members and participants on the topic of this paper. We would also like to thank the Japanese Thermal and Nuclear Power Engineering Society for the donation of their test results. However, it should be noted that the interpretations and discussions presented in the paper are those of the authors and do not necessarily represent PVRC statements.

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