January 21, 2004

Mr. Fred A. Emerson
Nuclear Energy Institute
1776 I Street, N.W., Suite 400
Washington, DC 20006-3708

SUBJECT: EVALUATION OF PROPOSED INTERIM STAFF GUIDANCE (ISG)-11: RECOMMENDATIONS FOR FATIGUE ENVIRONMENTAL EFFECTS IN A LICENSE RENEWAL APPLICATION

Dear Mr. Emerson:

By letter dated January 17, 2003, the Nuclear Energy Institute (NEI) submitted information regarding to the subject ISG-11, "Recommendations for Fatigue Environmental Effects in a License Renewal Application," for the Nuclear Regulatory Commission (NRC) staff review (See ADAMS Accession No. ML030300144). The proposed ISG would eliminate evaluation of carbon and low alloy steel components for the effects of the reactor coolant environment for license renewal applicants. The NEI's proposed ISG is based on an assessment documented in Electric Power Research Institute (EPRI) Report, "Materials Reliability Program: Re-Evaluation of Results in NUREG/CR-6674 for Carbon and Low-Alloy Steel Components (MRP-74)," dated November 2002. By letter dated June 30, 2003, the staff issued a request for additional information (RAI) on the proposed ISG (See ADAMS Accession No. ML031810630). The staff provided NEI additional clarification of the technical issues discussed in the RAI during a June 24, 2003, public meeting (See ADAMS NO. ML032340727). NEI responded to the staff's RAI by letter dated September 4, 2003 (See ADAMS Accession No. ML032810473). By letter dated October 3, 2003,(See ADAMS Accession No. ML0332304990), NEI provided a revision to the proposed ISG.

The staff has completed its review of the aforementioned letters and its supporting information submitted by NEI. Enclosed is the staff's evaluation. In the evaluation, the staff identifies several assumptions used in the EPRI report that do not have an adequate technical basis. Consequently, the staff concludes that insufficient technical basis was provided to eliminate evaluation of carbon and low alloy steel components for the effects of the reactor coolant environment for license renewal applications. Therefore, the proposed ISG is disapproved and ISG-11 is closed. If you have any questions concerning this evaluation, please contact Peter J. Kang at (301) 415-2779.

Sincerely,
/RA/
Pao-Tsin Kuo, Program Director
License Renewal and Environmental Impacts
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project 690
Enclosure: As stated
cc w/encl: See next page

January 21, 2004

Mr. Fred A. Emerson
Nuclear Energy Institute
1776 I Street, N.W., Suite 400
Washington, DC 20006-3708

SUBJECT: EVALUATION OF PROPOSED INTERIM STAFF GUIDANCE (ISG)-11: RECOMMENDATIONS FOR FATIGUE ENVIRONMENTAL EFFECTS IN A LICENSE RENEWAL APPLICATION

Dear Mr. Emerson:

By letter dated January 17, 2003, the Nuclear Energy Institute (NEI) submitted information regarding to the subject ISG-11, "Recommendations for Fatigue Environmental Effects in a License Renewal Application," for the Nuclear Regulatory Commission (NRC) staff review (See ADAMS Accession No. ML030300144). The proposed ISG would eliminate evaluation of carbon and low alloy steel components for the effects of the reactor coolant environment for license renewal applicants. The NEI's proposed ISG is based on an assessment documented in Electric Power Research Institute (EPRI) Report, "Materials Reliability Program: Re-Evaluation of Results in NUREG/CR-6674 for Carbon and Low-Alloy Steel Components (MRP-74)," dated November 2002. By letter dated June 30, 2003, the staff issued a request for additional information (RAI) on the proposed ISG (See ADAMS Accession No. ML031810630). The staff provided NEI additional clarification of the technical issues discussed in the RAI during a June 24, 2003, public meeting (See ADAMS NO. ML032340727). NEI responded to the staff's RAI by letter dated September 4, 2003 (See ADAMS Accession No. ML032810473). By letter dated October 3, 2003,(See ADAMS Accession No. ML0332304990), NEI provided a revision to the proposed ISG.

The staff has completed its review of the aforementioned letters and its supporting information submitted by NEI. Enclosed is the staff's evaluation. In the evaluation, the staff identifies several assumptions used in the EPRI report that do not have an adequate technical basis. Consequently, the staff concludes that insufficient technical basis was provided to eliminate evaluation of carbon and low alloy steel components for the effects of the reactor coolant environment for license renewal applications. Therefore, the proposed ISG is disapproved and ISG-11 is closed. If you have any questions concerning this evaluation, please contact Peter J. Kang at (301) 415-2779.

Sincerely,
/RA/
Pao-Tsin Kuo, Program Director
License Renewal and Environmental Impacts
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project 690
Enclosure: As stated
cc w/encl: See next page
Distribution: See next page

Accession no.: **ML040220124**

Document name: C:\MYFILES\Copies\SER Letter (11-03).wpd

OFFICE	LA:RLEP:DRIP	PM:RLEP:DRIP	SC:RLEP:DRIP	EMEB	D: DE	PD:RLEP:DRIP
NAME	MJenkins	PKang	SWest	EImbro (KManoly for)	RBarrett (Bateman for)	PTKuo
DATE	12/3/03	12/3/03	12/5/03	12/13/03	12/16/03	1/21/04

OFFICIAL RECORD COPY

DISTRIBUTION: Letter to F. Emerson, NEI, Dated: January 21, 2004
Accession no.: **ML040220124**
HARD COPY
RLEP RF
Project Manager
E-MAIL:
PUBLIC
J. Craig
D. Matthews
F. Gillespie
C. Grimes
RidsNrrDe
E. Imbro
G. Bagchi
K. Manoly
W. Bateman
J. Calvo
R. Jenkins
P. Shemanski
H. Nieh
J. Fair
S. Black
B. Boger
D. Thatcher
R. Pettis
G. Galletti
C. Li
J. Moore
R. Weisman
M. Mayfield
A. Murphy
S. Smith (srs3)
S. Duraiswamy
Y. L. (Renee) Li
RLEP Staff

A. Thadani
C. Julian
R. Gardner
M. Farber
M. Modes
J. Vora
L. Kozak

NUCLEAR ENERGY INSTITUTE

Project No. 690

cc:

Mr. Joe Bartell
U.S. Department of Energy
NE-42
Washington, DC 20585

Ms. Christine S. Salembier Commissioner
State Liaison Officer
Department of Public Service
112 State St., Drawer 20
Montipelier, VT 05620-2601

Mr. Fred Emerson
Nuclear Energy Institute
1776 I St., N.W., Suite 400
Washington, DC 20006-3708

Mr. William Corbin
Virginia Electric & Power Company
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060

Mr. Frederick W. Polaski
Manager License Renewal
Exelon Corporation
200 Exelon Way
Kennett Square, PA 19348

George Wrobel
Manager, License Renewal
R.E. Ginna Nuclear Power Plant
1503 Lake Rd.
Ontario, NY 14519

Mr. David Lochbaum
Union of Concerned Scientists
1707 H St., NW, Suite 600
Washington, DC 20006-3919

Ronald B. Clary
Manager, Plant Life Extension
V.C. Summer Nuclear Station
Bradham Blvd.
P.O. Box 88
Jenkinsville, SC 29065

Mr. Paul Gunter
Director of the Reactor Watchdog Project
Nuclear Information & Resource Service
1424 16th St., NW, Suite 404
Washington, DC 20036

Mr. Hugh Jackson
Public Citizen's Critical Mass Energy & Environment Program
215 Pennsylvania Ave., SE
Washington, DC 20003

Mary Olson
Nuclear Information & Resource Service
Southeast Office
P.O. Box 7586
Asheville, NC 28802

Talmage B. Clements
Manager - License Renewal
Nuclear Engineering Services
CP&L
410 South Wilmington St.
Raleigh, NC 27602

Mr. Charles R. Pierce
Manager - License Renewal
Southern Nuclear Operating Company
P. O. Box 1295
Birmingham, AL 35201

Mr. Garry G. Young
Manager, License Renewal Services
1448 SR 333, N-GSB-45
Russellville, AR 72802

Richard J. Grumbir
Project Manager, License Renewal
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

Mr. Timothy E. Abney, Manager
Licensing and Industry Affairs
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

EVALUATION
OF
PROPOSED INTERIM STAFF GUIDANCE (ISG)-11
RECOMMENDATIONS FOR FATIGUE ENVIRONMENTAL EFFECTS
IN A LICENSE RENEWAL APPLICATION

1.0 INTRODUCTION

By letter dated January 17, 2003, the Nuclear Energy Institute (NEI) submitted draft Interim Staff Guidance (ISG)-11, “Environmental Assisted Fatigue for Carbon/Low Alloy Steel,” for staff review. The proposed ISG would eliminate evaluation of carbon and low alloy steel components for the effects of the reactor coolant environment for license renewal applicants. The proposed change is based on an assessment documented in Electric Power Research Institute (EPRI) Report, “Materials Reliability Program: Re-Evaluation of Results in NUREG/CR-6674 for Carbon and Low-Alloy Steel Components (MRP-74),” dated November 2002. By letter dated June 30, 2003, the staff issued a request for additional information (RAI) on the proposed ISG. The staff provided NEI additional clarification of the technical issues discussed in the RAI during a June 24, 2003, public meeting. NEI responded to the staff’s RAI by letter dated September 4, 2003. By letter dated October 3, 2003, NEI provided a revision to the proposed ISG.

2.0 REGULATORY EVALUATION

In accordance with 10 CFR 54.21, license renewal applicants must demonstrate that aging effects of passive components will be adequately managed during the period of extended operation. Fatigue cracking of metal components is a potential aging mechanism which must be managed. License Renewal Standard Review Plan (NUREG-1800) Section 4.3, “Metal Fatigue Analysis,” contains the staff position concerning the evaluation of the effect of the reactor water environment on the fatigue life of reactor coolant system components. The staff position recommends that license renewal applicants evaluate a sample of fatigue critical components using environmental life correction factors contained in NUREG/CR-6583, “Effects of Light Water Reactor (LWR) Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels,” and in NUREG/CR-5704, “Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels.” The staff position indicates that the critical components should include, as a minimum, those selected in NUREG/CR-6260, “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components.” NUREG-1801, Chapter X.M1 provides an aging management program which contains the same recommendation to address the components listed in NUREG/CR-6260. The components listed in NUREG/CR-6260 involve six fatigue sensitive locations per plant of which approximately half are carbon and low alloy steel.

3.0 TECHNICAL EVALUATION

Components of the reactor coolant pressure boundary were designed using fatigue curves developed from tests of small polished specimens in air. More recent tests of small polished specimens in reactor coolant system environments indicates that the fatigue life of these components may be substantially less than assumed in the original design. This issue was evaluated for plant license renewal during the resolution of GSI-190, "Fatigue Analysis of Components for 60-year Plant Life." The NRC closed GSI-190 in December, 1999, concluding:

> "*The results of the probabilistic analyses, along with the sensitivity studies performed, the iterations with industry (NEI and EPRI), and the different approaches available to the licensees to mange the effects of aging, lead to the conclusion that no generic regulatory action is required, and that GSI-190 is closed. This conclusion is based primarily on the negligible calculated increases in core damage frequency in going from 40 to 60 year lives. However, the calculations supporting resolution of this issue, which included consideration of environmental effects, and the nature of age-related degradation indicate the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, the staff concludes that, consistent with existing requirements in 10 CFR 54.21, licensees should address the effects of coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.*"

The resolution of GSI-190 relied, in part, on the study performed by Pacific Northwest National Laboratory (PNNL) and published in NUREG/CR-6674, "Fatigue Analysis of Components for 60-Year Plant Life." The study evaluated the probability of fatigue cracking and leakage for the components evaluated in NUREG/CR-6260. The results of the PNNL study indicated a potential for an increase in the frequency of pipe leaks as the plants continue to operate. The PNNL study used the computer program pc-PRAISE to perform probabilistic fatigue crack initiation and fatigue crack growth calculations for the components evaluated in NUREG/CR-6260. Fatigue crack initiation correlations developed by Argonne National Laboratory (ANL) and published in NUREG/CR-6335, "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments," were incorporated into the pc-PRAISE computer code.

The draft ISG is based on re-evaluation of the carbon and low alloy steel components originally evaluated by PNNL and presented in NUREG/CR-6674. This re-evaluation is presented in EPRI Report MRP-74. EPRI claims that more realistic assumptions were used in the re-evaluation of these components, and that the use of these more realistic assumptions results in probabilities of crack initiation and leakage that are significantly less than those published in NUREG/CR-6674. The most significant change EPRI made to PNNL study was in the standard deviation assumed for the endurance limit strain. EPRI replaced the standard deviation used in the PNNL study with a much smaller standard deviation. EPRI cited a typical value of fatigue data scatter proposed by Wirsching (Probabilistic Structural Mechanics Handbook, edited by C. Sundararajan, Chapman & Hall, New York, N.Y., 1995, Chapter 7) as the basis for the change. This handbook value is general in nature and not specifically applicable to carbon and low alloy steels used in nuclear power plants. The standard deviation for the endurance limit strain used in the PNNL study is based on a statistical evaluation of test data relevant to carbon and low alloy steels described in NUREG/CR-6335 and NUREG/CR-6717, "Environmental Effects on Fatigue Crack Initiation in Piping and Pressure Vessel Steels." By RAI dated June 24, 2003, the staff requested that NEI provide the following additional information regarding the EPRI endurance limit strain and its standard deviation:

1. The revised probabilistic fatigue curves do not appear consistent with the data for carbon and low alloy steels. For example, compare probabilistic curves developed using the EPRI assumption for the standard deviation of the endurance limit with the data presented in Figure 14 of Attachment 1 of the submittal.

2. The study does not appear to adjust the endurance limit strain to account for the differences between smooth specimen data and actual components. The ANL correlation used by PNNL was developed to account for this difference. Provide the basis for not adjusting the endurance limit to account for the difference between the specimen data and actual components.

3. The EPRI report indicates that a strain threshold was used in the evaluation but does not show how the threshold was applied. The EPRI Report, page 3-11, references NUREG/CR-6717 for the strain threshold values used for the evaluation. As discussed in NUREG/CR-6717, the thresholds are strain levels at which environmental effects are considered moderate. These thresholds were proposed for use in the development of fatigue design curves. NUREG/CR-6717 also indicates that the threshold strain is approximately 20% higher than the fatigue limit (endurance limit) of the steel. Therefore, the threshold strain should be related to the endurance limit. Additionally, the proposed 0.07% threshold strain for the carbon and low alloy steel design curves has not been universally accepted at this time. For example, some fatigue researchers have proposed using the endurance limit strain of 0.042% as the threshold value. As a consequence, the use of a fixed threshold strain in the probabilistic study is questionable. Explain how the strain threshold values were used in the evaluations presented in Chapter 4 of the EPRI report. Provide the results of the EPRI evaluations without using strain threshold values.

4. The strain thresholds are discussed on page 26 of NUREG/CR-6717. NUREG/CR-6717 indicates that after mean stress effects are taken into account, a threshold strain amplitude of 0.07% provides a 90% confidence level for both carbon and low alloy steels. As discussed previously, the threshold strain is approximately 20% higher than the endurance limit of the steel. Consequently, the 10% probabilistic fatigue curve should approach a strain amplitude of approximately 0.06% at 10E6 cycles. The 10% probability curve shown in Figure 3-11 of the EPRI report is not consistent with a strain of 0.06%. Discuss this discrepancy between Figure 3-11 of the EPRI report and the data assessment contained in NUREG/CR-6717.

NEI's September 4, 2003, submittal contains the response to the staff's RAI. In response to item (a), NEI provided Figure 1 which compares the data contained in Figure 14 of attachment 1 of the original submittal with probabilistic fatigue crack initiation curves based on the EPRI endurance limit assumption. Figure 2 of the submittal provides the same plot with some of the data corrected to account for mean stress effects. Table 1 of the submittal contains the corrected data. The staff checked the first value in Table 1 using the data correction procedure provided in the submittal. Using the NEI procedure, the staff calculated a corrected stress of approximately 40 ksi. This does not match the corrected stress of 46.5 ksi shown in Table 1. Based on the staff's check of one data point, it does not appear that all of the data provided in Table 1 was corrected using the procedure provided in the submittal.

Additional plots comparing the probabilistic fatigue curves to test data are contained in Figures 3 to 10. The submittal states: "In all cases there is no indication of any data that fall below the 5% quantile in the high-cycle region affected mainly by the endurance limit."

A comparison of the actual test data with the 5% probability curve provides a useful check of the validity of the data scatter assumption used in the EPRI study. There is sufficient test data to check the conservatism of the 5% probability curve. Test data falling below the 5% probability curve indicates that the probabilistic curves are not conservative. Figure 3 provides probabilistic curves developed using the ANL data scatter assumption and Figure 4 provides probabilistic curves developed using the EPRI data scatter assumption. The scatter assumption used for the endurance limit affects the shape of the probabilistic fatigue curves throughout the cycle range as is evident by a comparison of Figures 3 and 4. The data scatter assumption for the endurance limit has a significant affect on the shape of the curves in the range of 10E3 to 10E5 cycles. This is significant because most of the fatigue usage for the components evaluated by EPRI occurs for loading conditions consisting of less than10E5 cycles. Therefore, it is important to review the impact of the EPRI assumption in that area of the curves. A review of Figures 2, 4, 6, 8 and 10 indicates that a substantial amount of the test data falls below the 5% probability curve in the range of 10E3 to 10E5 cycles using the EPRI data scatter assumption. In fact, some of the data approach the 0.1% probability curve. In contrast, Figures 3, 5, 7 and 9 show only a few data points fall below the 5% probability curve using the ANL data scatter assumption in the range of 10E3 to 10E5 cycles. The staff's review of the test data plots provided in the NEI response finds that the test data does not support the argument that the EPRI data scatter assumption is more realistic than the ANL data scatter assumption, especially in the cycle range of concern for the component evaluations.

The staff notes that the standard deviation used in the EPRI study was based on a number quoted in the aforementioned Mechanics Handbook. The handbook contains the statement: "Realize that the fatigue process is extremely complicated, and that the scatter in the data depends on a large number of factors, not the least of which is the material itself." The handbook further states: "For default values, coefficient of variation (COV) of 0.5 for cycles to failure at a given stress level and 0.10 for fatigue life at a given cycle life are not unreasonable." The Handbook does not indicate the basis for the COV or provide evidence that these COV values are valid for materials used in reactor water environments. During the June 24, 2003, meeting, the staff questioned why a regression analysis of the actual test data had not been performed. The response in the NEI submittal indicates that it was not practical to perform a regression analysis of the actual test data obtained for reactor water environments because there is insufficient data. Therefore, no rigorous evaluation of actual test data has been performed to support the EPRI data scatter assumption. During the meeting, the staff indicated that the data scatter evaluation of socket welded joints published in pressure vessels and piping (PVP) Volume 383, "Evaluation of Stress Intensification Factors for Circumferential Fillet Welded or Socket Welded Joints," 1999, did not support the EPRI data scatter assumption. The statistical evaluation of the socket weld joint data indicates a standard deviation that is more consistent with the ANL standard deviation. The response in the NEI submittal indicated that the socket welded joint data should not be used because of the large variability in socket/fillet weld geometry. However, the PVP paper contains a statistical evaluation of several data sets in order to eliminate outlier data, such as data on weld joints containing large weld defects. None of these data set evaluations support the EPRI standard deviation assumption. The staff concludes that the statistical evaluation of the socket welded joints, which include actual nuclear power plant components, has greater relevance than a COV value of unknown origin listed in a handbook.

The staff finds that EPRI has not provided an adequate technical basis for changing the standard deviation developed by ANL. The ANL standard deviation is based on a statistical evaluation of the test data. In contrast, EPRI has not performed a statistical evaluation of the test data to support the standard deviation used in its study. Instead, EPRI relies solely on a value cited in a handbook, with no demonstration of its applicability and no supporting data analysis. In addition, the test data plots submitted in response to item (a) of the staff's RAI do not support the assertion that the EPRI data scatter assumption is appropriate in the cyclic loading range of importance in the study.

The NEI September 4, 2003, response to item (b) of the RAI indicated that the endurance limit used in the EPRI study had not been adjusted to account for the difference between smooth test specimens and actual components. As a consequence, the EPRI correlations were revised to account for the adjustment and the calculations were revised accordingly. EPRI used a less conservative method than ANL when it applied the adjustment. ANL applied an adjustment on both cycles and endurance limit when defining the probabilistic curves. EPRI applied either the cycle or endurance limit adjustment depending on which value was controlling as illustrated in Figure A-2 of the response. There is insufficient component test data to establish whether the EPRI procedure provides a more accurate representation of actual component behavior. Furthermore, the staff notes that the EPRI procedure is less conservative in the 10E3 to 10E5 cycle range than the ANL procedure. As discussed previously, the EPRI probabilistic curves are not conservative compared to the specimen test data in that cycle range.

The two remaining staff concerns identified in items (c) and (d) of the staff's RAI involved EPRI's use of a strain threshold in the evaluation. The NEI response indicates that strain threshold values were not used in the revised evaluations. The staff agrees with the elimination of strain threshold values form the evaluations because the use of a fixed threshold strain in the probabilistic study has not been technically justified.

EPRI Report MRP-74, page 3-3, indicates that the ANL cycle adjustment, used to account for the differences between laboratory specimens and actual components, was included in the EPRI study. However, Section 4.7 of the PNNL report indicates that ANL cycle adjustment factor was modified to account for the potential for multiple crack initiation sites. The PNNL report further indicates that the modified adjustment was calibrated against the data from the 9-inch diameter vessel tests described in the ANL report. The staff requested that NEI describe how the PNNL modified adjustment factor was applied in the EPRI study.

The NEI September 4, 2003, response to the RAI indicated that the same adjustment factor used by PNNL was applied in the EPRI study. The response also provided the results of an evaluation performed to determine whether the PNNL adjustment factor was applicable to the EPRI model. NEI concluded that the use of the PNNL adjustment was appropriate because the adjustment factor provided conservative results with the EPRI model. However, staff review of the plots provided in Figures 12 and 14 of the September 4 submittal does not confirm this conclusion. Table A-3 of the submittal provides the fatigue crack initiation probabilities calculated using the EPRI model. The highest reported probability of fatigue crack initiation is 0.0584. Figure 14 clearly shows that the use of the PNNL adjustment factor with the EPRI model is not conservative for fatigue crack initiation probabilities less than 0.2. The staff finds that the use of the PNNL adjustment factor in combination with the EPRI model has not been technically justified.

EPRI Report MRP-74, page 3-11, provides a procedure to account for mean stress effects. The staff requested that NEI discuss the consistency of the mean stress adjustment used in the Chapter 4 evaluations with the mean stress adjustment discussed in NUREG/CR-6717. The response to the RAI indicated that EPRI used values of yield stress and ultimate stress for the mean stress adjustment that are different from the values listed in NUREG/CR-6717. The response further indicated the revised evaluations used the yield and ultimate stress values listed in NUREG/CR-6717 for the mean stress adjustment. The staff agrees that the yield and ultimate stress listed in NUREG/CR-6717 should be used for the mean stress adjustment.

Several of the component evaluations presented in Chapter 4 of the EPRI report use stresses and cycle counts that are different from those used in the PNNL study. The changes affect the calculated environmental fatigue usage factors for these components. The staff requested that NEI provide the environmental fatigue usage factors based on the revised component stress and cycle assumptions. The staff also requested that NEI discuss the actions that would be required by a license renewal applicant to address components with these usage factors. The September 4 response provided the revised environmental fatigue usage factors. However, the response did not discuss the actions that would be required by a license renewal applicant to address components with these environmental usage factors.

The staff reviewed the environmental usage factors associated with the 12 component locations evaluated in the EPRI study. The staff found that seven of the component locations evaluated have environmental usage factors less than 1.0. Therefore, a license renewal applicant would not be required to take any further actions to address these components. In fact, the staff has already reviewed and accepted the industry evaluation of the B&W outlet nozzle, which is one of the component locations evaluated by EPRI. The review is contained in the staff's safety evaluation concerning B&W Owners Group (B&WOG) topical report BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel." The staff notes that similar evaluations could have been performed for the remaining locations. However, the NSSS suppliers did not provide bounding evaluations of these locations for staff review.

The remaining five components have environmental usage factors greater than 1.0. Table A-3 of the submittal provides the revised probability of fatigue crack initiation and leakage using the revised EPRI model assumptions. The highest reported leakage probability is 6.43E-3 for the GE-NEW RHR Straight Pipe. The corresponding probability of fatigue crack initiation is 9.35E-3. The staff selected this location to perform a confirmatory check of EPRI's calculations. The staff notes that some of the assumptions used in the PNNL study were modified in the revised EPRI study whereas other PNNL assumptions were not changed. The revised EPRI model is much more sensitive to these changes in assumptions than the PNNL model. One of the assumptions made by PNNL that was apparently not changed involved the dissolved oxygen concentration for BWR components. PNNL assumed a dissolved oxygen concentration of 0.1 ppm based on the work presented in NUREG/CR-6260. As discussed in NUREG/CR-6260, the BWR environmental fatigue curves were based on dissolved oxygen concentrations of 0.1 ppm or greater. Subsequent ANL correlations provided a more refined model for the dissolved oxygen concentration. Nominal BWR oxygen concentrations are typically 0.2 ppm as discussed in Section 3.3 of EPRI Report TR-105759, "An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Fatigue Evaluations." This is the value that should have been used in the PNNL study. Although this change would not have significantly changed the PNNL results, the change may be significant considering the sensitivity of the EPRI model. Therefore, the staff used an oxygen concentration of 0.2 ppm to evaluate the probability of fatigue crack initiation of the GE-NEW RHR Straight Pipe with the revised EPRI model.

Table 5-125 of NUREG/CR-6260 lists the stresses and cycles used for the fatigue evaluation of the RHR Straight Pipe. Most of the fatigue usage is caused by a thermal stratification transient which produces an alternating stress of 23.98 ksi. The number of cycles is 11,976 for forty years which extrapolates to approximately 18,000 cycles for 60 years. The staff estimated the probability of fatigue crack initiation using the EPRI model for fatigue crack initiation. Equation 3-3 contained in MRP-74, as modified in accordance with the September 4 response to the staff RAI, was used to calculate the probability of fatigue crack initiation of the component. The alternating stress was adjusted using the formula in equation 5 of the NEI response to account for mean stress effects. The adjustment results in an alternating stress of 30 ksi. Using Equation 3-3 with an oxygen concentration of 0.2 ppm, the staff estimates the probability of fatigue crack initiation is approximately 5%, just considering one of the load pairs in Table 5-125 of NUREG/CR-6260. This is a significantly higher probability of fatigue crack initiation than NEI reported in the September 4, 2003, submittal.

The staff evaluation of the RHR Straight Pipe, using the model developed by EPRI, demonstrates the sensitivity of the EPRI model to the input assumptions. EPRI selectively altered some of the assumptions used in the PNNL study to obtain lower estimates of the probability of fatigue crack initiation and leakage. However, EPRI did not rigorously evaluate all of the assumptions used in the PNNL model, especially assumptions which would increase these probabilities. Consequently, the staff concludes that the leakage probabilities calculated by EPRI are no more credible than the leakage probabilities reported by PNNL.

4.0 CONCLUSION

The staff finds that several assumptions used in the EPRI evaluation of component leakage probabilities do not have an adequate technical basis. Consequently, the staff concludes that NEI has not provided a sufficient technical justification to support the proposed ISG that would eliminate evaluation of carbon/low alloy steel components for the effects of the reactor coolant environment for license renewal applicants. On this basis, the proposed ISG is disapproved and ISG-11 is closed.