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OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

December 26, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Ashok C. Thadani, Director [Original /s/ by A. Thadani]
Office of Nuclear Regulatory Research

SUBJECT: CLOSEOUT OF GENERIC SAFETY ISSUE 190, "FATIGUE EVALUATION OF METAL COMPONENTS FOR 60-YEAR PLANT LIFE"

The staff has completed all actions planned for the resolution of Generic Safety Issue (GSI) 190, and will close out the issue without imposition of new or additional generic action for licensed plants up to 60-year plant life. This GSI was identified in 1996 by a memorandum from T. Speis to A. Thadani, dated 08-26-1996, to supplement the ongoing work on GSI-78, "Monitoring of Fatigue Transient Limits for Reactor Coolant system," and residual work on GSI-166, "Adequacy of Fatigue Life of Metal Components," by addressing fatigue of metal components for 60-year plant life.

The conclusion to close out this issue is based upon the low core damage frequencies from fatigue failures of metal components estimated by technical studies making use of recent fatigue data developed on test specimens. The results of these probabilistic analyses and associated sensitivity studies led the staff to conclude that no generic regulatory action is required. However, calculations including environmental effects, that were performed to support resolution of this issue, and the nature of age-related degradation indicate the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, the staff concludes that, consistent with existing requirements in 10 CFR 54.21, licensees should address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. The background information and the basis for the closeout conclusion are presented in Attachment 1.

The advanced light water reactors (ALWRs) that have been certified under 10 CFR Part 52 were designed for a 60-year life expectancy. The associated fatigue analyses accounted for the design cycles based on a 60-year plant life but did not account for the environmental effects as addressed in GSI-190. However, the staff has concluded that there is sufficient conservatism in the fatigue analyses performed for the generic 60-year ALWR plant life to account for environmental effects. The staff therefore concludes for ALWRs that there is not a need for imposing a revision of these fatigue analyses on a generic basis or to impose additional monitoring requirements.

On December 3, 1999, the staff met with the ACRS, and presented the proposed resolution of this issue. The ACRS reviewed and accepted the proposed resolution (Attachment 2).

Interaction with the industry is an important aspect of the resolution of this issue. Therefore, the staff has had several meetings with the Nuclear Energy Institute (NEI) and the Electric Power

U.S. NUCLEAR REGULATORY COMMISSION
In the Matter of Entergy Nuclear Vermont Yankee LLC
Docket No. 50-271 Official Exhibit No. E2-03-VY
OFFERED by: Applicant/Licensee Intervenor
NRC Staff Other
IDENTIFIED on 7/21/08 Witness/Panel NEC 2
Action Taken: ADMITTED REJECTED WITHDRAWN
Reporter/Clerk MAC

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DS-03

Research Institute (EPRI) regarding this issue. Most recently, on November 17, 1999, the staff held a workshop with the industry to discuss the resolution of GSI-190 and plans for addressing the broader range of fatigue issues affecting nuclear plants. In addition, the staff has reviewed several EPRI reports on subject of fatigue. Appendix C to Attachment 1 briefly summarizes the contents and staff review of these reports. The staff is aware that the industry continues to be concerned about what will constitute an acceptable aging management program for fatigue. The industry and NRC are maintaining an ongoing dialogue on this subject.

Attachments:
1. Background Information
2. Letter to W. Travers from D. Powers, dated December 10, 1999

cc: S. Collins
C. Paperiello
F. Miraglia

CONTACT: Khalid Shaukat, MEB/DET/RES
301-415-6592

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Attachment 1

Resolution of GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life"

Background:

The ASME Boiler and Pressure Vessel Code (B&PV) Section III, Subsection NB contains design requirements for cyclic loading conditions. Appendix I to Section III specifies the code design fatigue curves that are based on strain-controlled tests of small polished specimens at room temperature in air. To obtain these design fatigue curves, best-fit curves to the experimental test data were lowered by a factor of 2 on stress or 20 on cycles, whichever was more conservative, at each point on the best-fit curve. This was intended to account for uncertainties in relating fatigue life of laboratory test specimens to those of actual reactor components. More recent fatigue strain-vs-life (S-N) data from the United States and Japan show that light water reactor (LWR) environments can have potentially significant effects on the fatigue life of carbon steel, low-alloy steel and austenitic stainless steel. Test specimen fatigue life in simulated LWR environments was found to be shorter than that determined by corresponding tests in air (Appendix A). This implies that the factor of 2 or 20 applied to the original best-fit- data curve may not be adequate. Under an NRC/RES-sponsored project, Argonne National Laboratory (ANL) developed interim design fatigue curves which addressed the environmental effects on fatigue life of the materials (NUREG/CR-5999).

GSI-190 Prioritization and Related GSIs:

The effects of the interim design curves, developed by ANL, on fatigue life of selected components were studied under two generic issues; GSI-78, "Monitoring of Fatigue Transient Limits for Reactor Coolant system," and GSI-166, "Adequacy of Fatigue Life of Metal Components."

GSI-78 was developed to determine whether fatigue monitoring was necessary at operating plants, and later included the calculation of risk due to through-wall cracking of metal components due to fatigue. GSI-166 was developed to assess the significance of more recent fatigue test data on the fatigue life of a sample of components in plants where a code fatigue design analysis had been performed. A Fatigue Action Plan (FAP) was developed to coordinate the efforts on fatigue life estimation and addressed the ongoing issues under GSI-78 and GSI-166 for 40-year plant life.

In resolving GSI-166, the staff completed the following studies regarding the concerns related to design basis fatigue transients for the operating plant life of 40-years.

- Under the FAP, a study was made (NUREG/CR-6260) by Idaho National Engineering Laboratory (INEL) to evaluate the design code fatigue cumulative usage factor (CUF) for selected components in primary system environments using the interim fatigue curves of NUREG/CR-5999.

- Existing fatigue data were compiled on specimens cycled to failure in simulated LWR conditions, and statistical models were developed (NUREG/CR-6335) by ANL for estimating the effects of service conditions on fatigue life of selected components.

In SECY 95-245 (Completion of Fatigue Action Plan), dated 09/25/95, the staff concluded that no immediate actions were required to deal with the fatigue issue for the current plant design life of 40 years. This conclusion was based on work indicating that the calculated fatigue usage factors for operating plants were conservative and that a cost/benefit analysis per the GSI process would not support imposing any new requirements for the 40-year operating license period. However, it was recognized that environmental effects could result in fatigue still being an issue for plants operating an additional 20-years under a renewed license . On completion of the FAP, the ACRS letter to the EDO (03/14/96) agreed with the staff's conclusion that the risk from fatigue failure of components in the reactor coolant pressure boundary is very small for a 40-year plant life.

Since the procedures for the resolution of GSIs require consideration of a license renewal period of 20 years, the environmental effects of fatigue on pressure boundary components for 60 years of plant operation were examined. By a memorandum to A. Thadani, from T. Speis, dated 08/26/96, RES documented that this further study for the 20-year life extension of the concerns of GSI-78, and GSI-166, would now be carried on under a new GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life." Therefore, GSI-78 and GSI-166 were considered resolved by a closeout memorandum to H. Thompson, from D. Morrison, dated 02/05/97.

GSI-190 addressed the environmental effects on design basis fatigue transients, studying the probability of fatigue failure, either leakage or pipe failure, and associated core damage frequency (CDF) for 60-year plant life. It did not address all aspects of fatigue-related degradation, including those that are outside the design basis.

Other fatigue-related degradation such as recent cracking problems at Oconee, and in France, which were related to unanticipated operating conditions involving thermal/mechanical stresses, are outside the scope of GSI-190. Several of these more recent fatigue events are discussed in Appendix B. These events are the results of unanticipated cyclic loads and thus are clearly outside the scope of the design basis transients used for fatigue design governed by the Section III of the ASME, B&PV Code. Situations like this may continue to occur.

Interaction with the industry is an important aspect of the resolution of this issue. Therefore, the staff has had several meetings with the Nuclear Energy Institute (NEI) and the Electric Power Research Institute (EPRI) regarding this issue. Most recently, on November 17, 1999, the staff held a workshop with the industry to discuss the resolution of GSI-190 and plans for addressing the broader range of fatigue issues affecting nuclear plants. In addition, the staff has reviewed several EPRI reports on the subject of fatigue. Appendix C to this Attachment briefly summarizes the contents and staff review of these reports.

Technical Analysis:

The approach to resolving GSI-190 built upon the approaches used in resolving GSI-166 for 40-year plant life and in the Fatigue Action Plan studies. The Pacific Northwest National Laboratory (PNNL) performed calculations of the probability of component failure and the Core Damage Frequency (CDF) associated with these failures. PNNL made use of the previous and most recent testing performed to develop fatigue design curves for stainless steel in simulated LWR environmental conditions.

PNNL then performed probabilistic fatigue calculations on 47 sample components from 6 locations in five PWR- and two BWR-plants using the pc-PRAISE code. During this work the staff identified that the pc-PRAISE code could not model the large aspect ratios of the initial cracks (ratio of crack length to crack depth), nor could it model the joining of several small cracks to make a large crack which would subsequently propagate through the wall thickness of the component. The staff concluded that the effect of large aspect ratios was an important factor in fatigue analyses and there was a need to modify the pc-PRAISE code to model these things.

PNNL modified the pc-PRAISE code with the help of a subcontractor who had originally written the code, and performed the fatigue calculations for several small cracks to grow and link together. Testing of the modified code demonstrated that, under strong thermal gradient loading, there was a high likelihood of long cracks being produced which is consistent with service experience. This was viewed as a major improvement in the performance of the pc-PRAISE code and was used in performing the subsequent through-wall cracking frequency calculations.

Using the modified pc-PRAISE code, PNNL performed a probabilistic analysis for the crack initiation and through wall crack growth in the components mentioned above for 40- and 60-year plant life considering both air and LWR environments. Calculations for the air environment and the 40-year life were done to confirm that the effect of the revised fatigue curves, coupled with the modified pc-PRAISE code, provided results that were consistent with the previous studies. An evaluation was performed to estimate the conditional CDF from the fatigue failure of these components. Given a through wall crack, the objective was to estimate the conditional probabilities of a small leak, of a large leak, and of a pipe break. Data on pipe failure events indicate that only a small fraction of through-wall flaws result in large leaks or breaks, and the types of failures are dependent on the particular failure mechanism involved. Probabilistic fracture mechanic models, like the one contained in the pc-PRAISE code, predict that fatigue failures will usually be in the mode of small leaks rather then large leaks or breaks. From the conditional probabilities of small leak, large leak, and pipe break, the conditional CDF was estimated for the seven example plants based on results extracted from probabilistic risk assessment (PRA). The major findings from the calculations include the following points:

- Many of the components have cumulative probabilities of crack initiation and cumulative probabilities of through-wall cracks that approach unity within the 40 to 60-year time period. However, some components, often with similar values of fatigue usage factors, show much lower failure probabilities. This is a result of the ANL statistical fit to the test data which indicates that the probability of crack initiation for a constant CUF value is a function of the strain amplitude of the applied loading.

- The maximum failure rate (through-wall cracks per reactor year) is in the range of 10^{-2}, and those failures were associated with high CUF locations.

- Failure rates for other components having much lower failure probabilities are changed by as much as an order of magnitude from 40 to 60-years, but these components make relatively small overall contributions to the cumulative CDF estimates.

- The maximum CDF based on these calculated failure rates is about 10^{-8} per year. These maximum values correspond to components with very high cumulative failure probabilities, and the failure rates do not change much from 40 to 60 years. The range of CDF was between 10^{-6} to 10^{-15}.

Based upon these low CDFs, the staff concludes that they would not be used as a basis for a cost/benefit backfit analysis to justify imposition of a new regulatory requirement on operating reactors. However, the calculations supporting resolution of this issue, which included consideration of environmental effects, indicates the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, with the consideration of risk informed perspective, the staff concludes that, consistent with existing requirements in 10 CFR 54.21, licensees should address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

Sensitivity Study:

Sensitivity studies were performed by varying important input parameters to the pc-PRAISE code that could significantly alter the results. The parameters chosen for the sensitivity studies were wall thickness, stress gradient, initial flaw (crack) depth, flaw length, multiple crack initiation, correlation between crack initiation and crack growth, start of fatigue crack growth, oxygen content of reactor water, sulfur content of metal, and strain rate.

These studies were primarily aimed at increasing confidence in the results of the above studies, and to better understand the limitations of the assumptions made in the analyses using the revised pc-PRAISE code, which incorporated capabilities for crack linking. The staff concluded that the results from the code were consistent with physical expectations for fatigue crack growth and consistent with observations from actual fatigue failures. Therefore, the results of PNNL report were considered to provide a reasonable basis for our conclusions.

Conclusions:

The estimated contribution to CDF from fatigue failures of the evaluated components is on the order of 10^{-6} , or lower, per year. Recognizing the uncertainties in the calculations, the contribution to CDF from fatigue failures could be of the same order of magnitude as the contribution to CDF from a small LOCA (10^{-5} per year), but still below the threshold normally used to justify additional regulatory requirements for operating reactors.

It should be noted that based on analyses performed for GSI-190, it was expected that the cumulative usage factor (CUF) for certain components would exceed the ASME Code design limit of CUF $\leq$ 1 when evaluated for design basis transients for 60 years. The calculated CDFs

for the components with the highest failure frequencies show essentially no increase in CDF from 40 to 60 years.

The results of the probabilistic analyses, along with the sensitivity studies performed, the interactions with the industry (NEI and EPRI), and the different approaches available to the licensees to manage the effects of aging, lead to the conclusion that no generic regulatory action is required, and that GSI-190 is resolved. This conclusion is based primarily on the negligible calculated increases in CDF in going from 40 years 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 the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

APPENDIX A

Recent Technical Information on Fatigue in the U.S. and Japan

Under an NRC/RES-sponsored project, fatigue tests have been conducted by ANL on Type 304 and 316NG stainless steel to evaluate the effects of various loading variables under water environments. The results confirm decreases in fatigue life in low dissolved oxygen water. The formation and growth of fatigue cracks in air and water environments are discussed in two letter reports by ANL; 1) "Effects of LWR Coolant Environment on Fatigue Lives of Austenitic Stainless Steels," dated November 1997, and 2) "Updated Fatigue Design Curves for Austenitic Stainless Steels in LWR Environments," dated January 19, 1998. The latter report contains the revised fatigue S-N curves developed by ANL and a comparative ASME Mean curve for stainless steel in air and water environments. The report also gives equations and explains a methodology for calculating fatigue life of components based on tests and other databases.

An independent evaluation of fatigue life of metal components has been ongoing in Japan. Dr. Higuchi of the industry and Prof. Iida of the University of Tokyo initiated a research project on this issue in 1991 which is expected to be completed in 2006. Their work involves the study of a fatigue database (254 data points for carbon steels and 319 data points for low alloy steels) and further fatigue tests of specimens under simulated LWR environments. They studied the fatigue behavior of carbon and low alloy steels in high temperature water environments for different dissolved oxygen contents and varying strain rates. Their study thus far shows results similar to those observed by ANL and they have reported their findings to the appropriate ASME Code Committees.

Results from the work described above and other studies have, for a significant time, indicated that the ASME Code design curves for fatigue life need to be modified to account for the effects of light water reactor environments. Since the cognizant Code Committee has not taken actions in this regard, RES/DET has sent a letter to request expeditious ASME Code action on this issue (Letter to J. Ferguson from J. Craig, dated December 1, 1999).

APPENDIX B

Recent Fatigue-related Events

Oconee: This event is discussed in IN 97-46. On April 22, 1997, at 12:50 p.m., Oconee Unit 2 was shut down because of unidentified reactor coolant system (RCS) leakage exceeding the technical specification limit of 1 gpm. Until reactor pressure was sufficiently reduced, the leakage rate rose to approximately 12 gpm. A subsequent containment entry identified a non-isolable leak from a through-wall crack in the weld connecting the MU/HPI pipe and the safe-end of the 2A1 reactor coolant loop (RCL) nozzle.

Preliminary analysis indicates that crack initiation and propagation in the weld was caused by high-cycle fatigue due to a combination of thermal cycling and flow induced vibration. Although the root cause of the cracking is not well understood, the licensee has identified a number of thermal/mechanical conditions that may have contributed to the crack propagation of the 2A1 pipe to safe-end weld. The licensee has hypothesized that, in addition to the thermal cycling experienced at the nozzle during heat up/cool down and other plant transients, a likely contributor to the fatigue may have been the alternate heating and cooling of the weld by intermittent mixing of the hot reactor coolant leaking through the gap in the contact area between the loose thermal sleeve and the safe-end, and the cooler normal makeup water flowing through the associated MU/HPI line. Although the precise contribution of the gap is unknown, it is believed that a gap may be a prerequisite for cracking in the piping since the cracked pipes also had gaps between the thermal sleeve and the safe end.

This phenomenon was identified as the probable cause for similar safe-end cracking observed at Crystal River and other B&W plants (including Oconee) in the early 1980's. This issue was previously addressed in Information Notice 82-09 and Generic Letter 85-20.

Dampierre-1, France: A similar event occurred on December 14, 1996, when a non-isolable leak on piping connecting the safety injection system to the reactor coolant system was found in Dampierre Unit 1 in France. The damaged pipe length was examined and a through wall crack located on an uninterrupted portion of straight piping (not on a stressed area such as a weld or a bend). The licensee has not identified the root cause of the cracking, but concluded that the most probable cause was temperature variations produced by cold water coming from leaking valves located upstream in the safety injection system. The licensee also concluded that the presence of a through-wall defect on a straight portion of a pipe is likely to raise questions about previous assumptions made regarding the root cause of the cracking.

Civaux-1, France: On May 12, 1998, a 30 cubic meter/hour leak from an RHR train occurred in Civaux-1 operating at 50% power less than five months into the reactor's operation. It took several hours to detect the source of the leak: a crack 180 millimeters long on a 250-millimeter-diameter section of pipe that allows bypass of the heat exchanger on the RHR train A. About 300 cubic meters of spilled primary coolant was recovered in the sump.

The probable cause was considered to be a pipe defect in a longitudinal weld between two sections of the elbow and possible related defects elsewhere. In-depth investigations are underway to pinpoint the cause.

APPENDIX C

Interaction with the Industry

During the course of resolving GSI-190, the NRC has had extensive interactions with the Nuclear Energy Institute (NEI) and the Electric Power Research Institute (EPRI). EPRI also submitted five Topical Reports on the subject of environmental effects of fatigue on components from PWRs and BWRs. The staff reviewed these reports and the assessment is briefly discussed below. The staff believes that Aging Management Programs for fatigue must be consistent with the spirit of GDC-14, in that components affected by fatigue should maintain "an extremely low probability of abnormal leakage." Discussion is continuing with industry as to how aging management programs will demonstrate that this current licensing basis requirement will continue to be satisfied during the renewal phase

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

The objective of this report was to develop simplified but not overly conservative procedures for estimating reactor water environmental effects on the reactor vessel and piping in light of new test data. The background is that pressure retaining components in LWRs are designed to meet ASME Code Section III which requires a fatigue evaluation for transient stresses. The laboratory data from various test programs indicate that fatigue lives shorter than the Code design values are possible under water environments. This report proposes use of an environmental correction factor to the current Code fatigue evaluation procedures. The report cites a need for simplified, but not overly conservative, procedures for ASME Section III, NB-3600- and NB-3200-type analyses to account for reactor water environment effects. EPRI suggested introduction of an environmental correction factor F_{en} to obtain fatigue usage reflecting the environmental effects. EPRI also suggested proposed changes to ASME Section III fatigue evaluation procedures for possible consideration by ASME Code Committee to include environmental effects in fatigue evaluations conducted according to NB-3600 and NB-3200. The evaluation procedures could be added as a non-mandatory appendix to the Code.

The staff agrees with the concept of using an environmental correction factor (F_{en}) to obtain fatigue usage reflecting environmental effects. This information and other recommendations mentioned above have been forwarded to the appropriate ASME Code committee (Letter to J. Ferguson from J. Craig, dated December 1, 1999).

- EPRI TR-107515, "Evaluation of Thermal Fatigue Effects on Systems Requiring Aging Management Review for License Renewal for the Calvert Cliffs Nuclear Power Plant," December 1997.

The objective of this report was to demonstrate the current industry technical position on fatigue evaluation for license renewal, through the application of existing methodology, and criteria on selected systems/components in a pilot plant. The report develops a technical evaluation methodology for determining the fatigue life adequacy of these selected locations to be used as a guide for managing fatigue effects for the license renewal period. EPRI's conclusion is that the effects of reactor water environments are already compensated by two existing conservatisms in Class 1 ASME Code fatigue analysis procedures - (1) the low cycle portion of

the design fatigue curve margin factor of 20 on cycles or a factor of 2 on stress, and 2) the design basis definitions of thermal transients. Therefore, the explicit treatment of reactor water environmental effects in fatigue analysis is not considered necessary for license renewal.

The staff assessment of this report was that the latest ANL fatigue data reported in "Updated Fatigue Design Curves for Austenitic Stainless Steels in LWR Environment (March 1998)," were not considered.

- EPRI TR-110043, "Evaluation of Environmental Fatigue Effects for a Westinghouse Nuclear Power Plant," April 1998.

In addition to the main objective of this report evident in its title, another objective was to supplement similar conclusions made by BG&E for license renewal work on Calvert Cliffs.

Fatigue reactor water environmental effects were evaluated at 14 specific components in a Westinghouse PWR plant. The components addressed were: (1) the pressurizer shell, (2) the pressurizer spray nozzle, (3) the pressurizer surge nozzle, (4) the pressurizer water temperature instrument nozzle, (5) RCS hot leg surge nozzle, (6) charging nozzle, and (7 through 14) steam generator feedwater nozzles (4 locations per steam generator for 2 steam generators). They include components fabricated from carbon steel and austenitic stainless steels. They encompass specific component locations for which actual plant transient data were available for at least three years of plant operation from fatigue monitoring.

A selective environmental fatigue methodology was implemented which uses effective environmental fatigue multipliers, F_{en} lower than those produced by applying environmentally adjusted fatigue curves (NUREG/CR-5999) in NUREG/CR-6260. The conclusion was that environmental effects are already compensated by two existing conservatisms in ASME Code Class 1 fatigue analysis procedures: 1) the low cycle portion of the design fatigue curve margin of 20, and 2) the design basis definitions of thermal transients. The combination of these two conservatisms is such that explicit treatment of reactor water environmental effects in fatigue design analysis is not considered necessary for a 60-year plant life.

The staff assessment of this report was that the latest ANL fatigue data reported in "Updated Fatigue Design Curves for Austenitic Stainless Steels in LWR Environment (March 1998)," were not considered.

- EPRI TR-110356, "Evaluation of Environmental Thermal Fatigue Effects on Selected Components in a Boiling Water Reactor Plant," April 1998.

This report evaluated environmental effects at three specific locations in a newer vintage BWR plant; 1) the feedwater nozzle safe ends, 2) the feedwater nozzle forgings, and 3) the control rod drive penetrations. The evaluations included components fabricated from carbon steels and austenitic stainless steels.

A selective environmental fatigue methodology was implemented which used effective environmental fatigue multipliers, F_{en} lower than those produced by applying environmentally adjusted fatigue curves (NUREG/CR-5999) in NUREG/CR-6260.

The conclusion was that environmental effects are already compensated by two existing conservatisms in ASME Code Class 1 fatigue analysis procedures: 1) the low cycle portion of the design fatigue curve margin of 20, and 2) the design basis definitions of thermal transients. The combination of these two conservatisms is such that explicit treatment of reactor water environmental effects in fatigue design analysis is not considered necessary for a 60-year plant life.

The staff assessment of this report was that the latest ANL fatigue data reported in "Updated Fatigue Design Curves for Austenitic Stainless Steels in LWR Environment (March 1998)," were not considered.

- EPRI TR-107943, "Environmental Fatigue Evaluation of Representative BWR Components," May 1998.

This topical report by EPRI considers the latest environmental fatigue data from ANL. In this report, EPRI presented a comparative study done by INEL and by EPRI for calculating cumulative usage factors (CUFs) at six components of a newer vintage BWR, presenting results of the evaluation of environmental fatigue CUFs at the same six components using EPRI/GE methodology. The components are: (1) reactor vessel shell and lower head, (2) reactor vessel feedwater nozzle, (3) reactor recirculating piping, (4) core spray line reactor vessel nozzle and associated Class 1 piping, (5) residual heat removal (RHR) nozzles and associated piping, and (6) feedwater line Class 1 piping. Additionally, a review of available Argonne statistical models and their impact on evaluations is also provided.

The report concludes that even after accounting for environmental effects, the CUFs for analyzed components in a sample BWR plant are predicted to be less than 1.0 for both 40-year and 60-year operating periods. Therefore, a plant specific analysis for environmental fatigue effects is not considered necessary.

The staff assessment of this report was that, although the latest ANL fatigue data for environmental effects were considered, the staff still has some technical concerns with the application of the data. It appears from the results of this report that further plant-specific evaluation will be required.

Attachment 2

December 10, 1999

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE-190, "FATIGUE EVALUATION OF METAL COMPONENTS FOR 60-YEAR PLANT LIFE"

During the 468th meeting of the Advisory Committee on Reactor Safeguards, December 2-4, 1999, we reviewed the proposed resolution of Generic Safety Issue (GSI)-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life." During our review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

RECOMMENDATIONS

- We agree with the staff's proposal that GSI-190 be resolved without any additional regulatory requirements.
- The staff should ensure that utilities requesting license renewal consider the management of environmentally assisted fatigue in their aging management programs.

BACKGROUND

The effects of fatigue for the 40-year initial reactor license period were studied and resolved under GSI-78, "Monitoring of Fatigue Transient Limits for Reactor Coolant System," and GSI-166, "Adequacy of Fatigue Life of Metal Components."

The staff concluded that risk from fatigue failure of components in the reactor coolant pressure boundary was very small for 40-year plant life. In our March 14, 1996 letter, we agreed with the staff's conclusion.

GSI-190 was established to address the residual concerns of GSI-78 and GSI-166 regarding the environmental effects of fatigue on pressure boundary components for 60-years of plant operation. The scope of GSI-190 included design-basis fatigue transients, studying the probability of fatigue failure and its effects on core damage frequency (CDF) of selected metal components for 60-year plant life.

DISCUSSION

Resolution of GSI-190 was based on the results of an NRC-sponsored study performed by the Pacific Northwest National Laboratory (PNNL). In that study, PNNL examined design-basis fatigue transients and the probability of fatigue failure of selected metal components for 60-year plant life and the resulting effects on CDF.

The PNNL study showed that some components have cumulative probabilities of crack initiation and through-wall growth that approach unity within the 40- to 60-year period. The maximum failure rate (through-wall cracks per year) was in the range of 10^{-2} per year, and those failures were associated with high cumulative usage factor locations and components with thinner walls, i.e., pipes more vulnerable to through-wall cracks. There was only a modest increase in the frequency of through-wall cracks in major reactor coolant system components having thicker walls. In most cases, the leakage from these through-wall cracks is small and not likely to lead to core damage. Therefore, the projected increased frequency in through-wall cracks between 40- and 60-years of plant life does not significantly increase CDF. Based on the low contributions to CDF, we agree with the proposed resolution of GSI-190.

Environmentally assisted fatigue degradation should be addressed in aging management programs developed for license renewal. Minimization of leakage is important for operational safety, occupational doses, and for continued economic viability of the plants.

Dr. William J. Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

Signed

Dana A. Powers
Chairman

References:

- Memorandum dated November 12, 1999, from Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Generic Safety Issue-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life."
- Letter dated March 14, 1996, from T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, to James M. Taylor, Executive Director for Operations, NRC, Subject: Resolution of Generic Safety Issue-78, "Monitoring of Fatigue Transient Limits for the Reactor Coolant System."
- Letter dated October 16, 1995, from T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, to Shirley Ann Jackson, Chairman, NRC, Subject: Fatigue Action Plan.