PSEG
P.O. Box 236, Hancocks Bridge, NJ 08038-0236

PSEG
Nuclear LLC

APR 28 2011

10 CFR 50.90

LR-N11-0122

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

Hope Creek Generating Station
Facility Operating License No. NPF-57
NRC Docket No. 50-354

Subject: **RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION, LICENSE AMENDMENT REQUEST – OPERATION WITH FINAL FEEDWATER TEMPERATURE REDUCTION AND FEEDWATER HEATERS OUT-OF-SERVICE**

References: (1) PSEG to NRC letter, "License Amendment Request – Operation With Final Feedwater Temperature Reduction and Feedwater Heaters Out-Of-Service," dated September 22, 2010.

In Reference 1 PSEG Nuclear LLC (PSEG) requested an amendment to Facility Operating License No. NPF-57 for Hope Creek Generating Station (HCGS). The amendment request proposed changes to paragraph 2.C (11) of the Facility Operating License (FOL). The proposed change would allow HCGS to operate with Final Feedwater Temperature Reduction (FFWTR) for the purposes of extending the normal fuel cycle. In addition, the analysis provided would also allow operation with Feedwater Heaters Out-Of-Service (FWHOOS) at any time during the operating cycle.

The NRC provided PSEG a Request for Additional Information (RAI) related to the Reference 1 request. Attachment 1 to this submittal provides the responses to the RAI. Also provided in Attachment 1 is supplemental information the NRC staff requested related to RAI question 5.

No new regulatory commitments are established by this submittal.

If you have any questions or require additional information, please do not hesitate to contact Mr. Paul Duke at (856) 339-1466.

A001
LIR

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 4/28/11
(Date)

Sincerely,

John F. Perry
Site Vice President
Hope Creek Generating Station

Attachments (1)

cc:

W. Dean - NRC Region I
R. Ennis, Project Manager - USNRC
NRC Senior Resident Inspector – Hope Creek (X24)
P. Mulligan, Manager IV, NJBNE
Commitment Coordinator – Hope Creek
PSEG Commitment Coordinator - Corporate

REQUEST FOR ADDITIONAL INFORMATION (RAI)

REGARDING PROPOSED LICENSE AMENDMENT

OPERATION WITH FINAL FEEDWATER TEMPERATURE REDUCTION

AND FEEDWATER HEATERS OUT-OF-SERVIICE

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

By application dated September 22, 2010 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML102790111), PSEG Nuclear LLC (PSEG or the licensee) submitted a license amendment request for the Hope Creek Generating Station (HCGS). The proposed amendment would modify the Facility Operating License (FOL) and Technical Specifications (TSs) to allow HCGS to operate at a reduced feedwater temperature for purposes of extending the normal fuel cycle. The amendment would also allow operation with feedwater heaters out-of-service at any time during the operating cycle.

The Nuclear Regulatory Commission (NRC) staff has reviewed the information the licensee provided that supports the proposed amendment and would like to discuss the following issues to clarify the submittal. The following questions are from the Mechanical and Civil Engineering Branch (EMCB):

EMCB-RAI-1
It appears that, as proposed in the license amendment request (LAR), a final feedwater temperature reduction (FFWTR) of 102 °F and 60 °F temperature reduction for feedwater heaters out-of-service (FWHOOS) would reduce the feedwater (FW) design temperature from 431.6 °F to 329.6 °F and from 431.6 °F to 371.6 °F, respectively. Please provide similar information which shows how the minimum operating FW temperature will be reduced by the LAR and verify whether (or not and why) the minimum operating temperatures reflecting the FFWTR and FWHOOS have been utilized in the evaluations described in the LAR.

RAI-1 Response

FFWTR and FWHOOS are flexibility options to the normal operating condition. Therefore, the normal operating condition and the normal feedwater design temperature remain unchanged with the implementation of these flexibility options. When operating in the FFWTR or FWHOOS flexibility option conditions, the feedwater temperature at rated power conditions shall not be less than the limits of 329.6°F or 371.6°F, respectively as these are the values that were used in the evaluations.

EMCB-RAI-2
For the structural evaluations performed for the proposed LAR, please provide references for the codes utilized. If different than the original code(s) of construction that the evaluated systems, structures, and components (SSCs) were ordered or built to, please discuss how the

later codes have been reconciled to the original codes and address whether the original code allowable values have been utilized for these analyses.

RAI-2 Response

Hope Creek original design basis Code allowable values were used in Hope Creek FFWTR/FWHOOS Reactor Internals Structural Integrity evaluation (Section 5.3 of NEDC-33506P). No allowable value was changed for FFWTR/FWHOOS evaluation. The original design basis evaluation was performed based on the original design basis ASME Boiler and Pressure Vessel Code, Section III 1968 Edition with Addenda 1969.

The Reactor Coolant Pressure Boundary (RCPB) piping components and supports have been evaluated for the Annulus Pressurization (AP) amplified response spectra (ARS) from the effects of the FFWTR/FWHOOS operating modes (Section 5.4 of NEDC-33506P). The results were compared with the original design basis AP ARS evaluations. It was concluded that the results of the original design basis AP ARS bounded the results from the AP ARS evaluations associated with the FFWTR/FWHOOS, as discussed in the response to RAI-11. The comparison was based on the calculated piping moments, stresses, support loads and the valve accelerations. The year of the Code has no effect on these comparisons.

The structural evaluations for the reactor pressure vessel (RPV) pedestal and the biological shield wall (BSW) were performed using structural acceptance criteria from the original codes of construction. For concrete, American Concrete Institute (ACI) 318, Building Code Requirements for Reinforced Concrete, 1971 with the 1974 supplement, including the 1973 and 1974 revisions, is the original code of construction that provides structural acceptance criteria. For structural steel, American Institute of Steel Construction (AISC) Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, February 12, 1969 with Supplement 1 of November 1, 1970, Supplement 2 of December 8, 1971 and Supplement 3 of June 12, 1974 is the original code of construction that provides structural acceptance criteria

For code discussion related to the FW nozzles refer to the response to RAI-14.

EMCB-RAI-3

Annulus pressurization (AP) loads are determined from breaks postulated in the nuclear steam supply system (NSSS) piping. Attachment 4 to the LAR, GE-Hitachi Nuclear Energy Americas LLC (GEH) report NEDC-33506P[1], states in a number of places that, "The AP loads remain bounded by the original design basis AP loads." Please specify which pipe breaks the AP loads in the first part of the quote are referring to and which pipe breaks the AP loads in the second part of the quote are referring to.

RAI-3 Response

The Pipe Breaks considered in this analysis were the Recirculation Suction Line Break (RSLB) and the Feed Water Line Break (FWLB) as discussed in section 4.1 of NEDC-33506P. Likewise, the pipe breaks considered in the original design basis were the RSLB and the FWLB.

1 Attachment 3 to the LAR, GEH report NEDO-33506 (ADAMS Accession No. ML102790106), is a publicly available version of Attachment 4 to the LAR, GEH report NEDC-33506P which is proprietary and non-publicly available.

EMCB-RAI-4
As discussed in Section 4.1 of Attachment 1 to the LAR, GEH issued a 10 CFR Part 21 Safety Information Communication, SC09-01, titled, "Annulus Pressurization Loads Evaluation" on June 8, 2009. As described by the licensee, SC09-01 identified a potential issue with the methodology that developed the AP loads. As a corrective action, SC09-01 recommended that affected plants review their licensing and design bases in light of the issues presented and consider reevaluating the AP loads to ensure that they are consistent with the plant's design and licensing basis. Please provide a list of reference(s) of these reviews and or evaluations and discuss the results and conclusions.

RAI-4 Response

The reviews and evaluations referenced and the results obtained are those discussed in Section 4.1 of Attachment 1 of the LAR and in Sections 4 and 5 of Attachment 4 of the LAR (NEDC-33506P). There are no other lists of references or results. The plant design and licensing basis events for AP loads are the RSLB and the FWLB events and both were evaluated for the FFWTR and FWHOOS conditions consistent with SC 09-01. This involved the expanded evaluation of ARS frequencies for the bounding loads versus using the limiting mass/energy release case only. This included an expanded evaluation of the FWLB. Prior to SC09-01 the FWLB was believed to be bounded by the mass and energy release of the RSLB. Other line breaks are outside of the existing HCGS design and licensing basis.

As discussed in Attachment 1, Section 4.1.3 of PSEG Letter dated September 22, 2010 (ADAMS ML102790111); for FWTR other break locations were assessed considering the RSLB and FWLB results and recent EPU/MELLLA analyses. Based on the more rigorous and updated EPU/MELLLA analyses methods, and the location and size of the Recirculation and Feedwater lines, it is not expected that there would be any adverse structural impact due to other break locations. This is consistent with the SC 09-01 conclusions that it is expected that there is sufficient conservatism in the current analysis to accommodate any impact. Additional actions related to SC 09-01 are being addressed by the BWROG.

EMCB-RAI-5
In reference to the issues identified in GEH SC09-01, Section 4.1 of Attachment 1 of the LAR states that GEH and PSEG evaluated the effect of FWTR on AP loads. In addition, this section of the LAR states that:

> *The structural responses and amplified response spectra (ARS) of the RPV [reactor pressure vessel], reactor internals, piping, biological shield wall (BSW), and RPV pedestal (drywell inner skirt) were evaluated due to the application of Recirculation Suction Line Break (RSLB) and Feedwater Line Break (FWLB) loads.*

These evaluations are further discussed in Attachment 4 to the LAR, NEDC-33506P, which only considers RSLB and FWLB. Part of the issue identified by GEH SC09-01 is that plants had not analyzed all major break locations from their design and licensing basis but only considered the recirculation line and feedwater line breaks.

a) Please provide a technical justification for the omission of the loads due to breaks in the recirculation discharge, low pressure coolant injection, core spray, and main steam for the AP analyses.

b) Breaks from the omitted lines mentioned in (a) above are included in the HCGS Updated Final Safety Analysis Report (UFSAR) Section 3.6. Please provide an explanation for the apparent inconsistency between the UFSAR and Section 4.1.3 of Attachment 1 of the LAR which indicates that breaks other than RSLB and FWLB are outside of the existing HCGS design basis.

c) Please provide a discussion which addresses how the ARS was developed.

RAI-5 Response

a) See the response to RAI-4 above.

b) UFSAR Section 3.6 addresses all High Energy Line Break (HELB) and Moderate Energy Line Break (MELB) breaks inside and outside containment. AP load design and licensing bases are in UFSAR Appendices 6B and 3C, which only includes both FWLB and RSLB.

c) Beginning with the nodal Acceleration Time History (ATH) data, the Amplified Response Spectra (ARS) were generated for each break through the following method:

1) The Level-II Engineering Computer Program, SPECA05V, was used to translate the ATH data into Response Spectra. Note that Table 1-3 of NEDC-33506A contains a typographical error that indicates SPECA version 04V, however, the correct version is 05V.

2) These spectra were then peak-broadened by ±15%. This was done to account for uncertainty in the analysis and corresponds to the maximum required peak broadening given in NRC Regulatory Guide 1.22.

3) The peak-broadened spectra were then compared across the various power/flow points and bounding spectra were created.

Finally, a scale factor of 1.6 was applied to these bounding spectra.

<u>EMCB-RAI-5, Supplemental Information Request.</u>
During a conference call on March 17, 2011, the NRC staff requested the following additional information:

A description of the LAMB and COMPARE codes and how they are used; and identify and describe any other codes used in the FFWTR evaluation not in NEDC-33506P Table 1-3.

RAI-5 Supplemental Information Response:

In addition to LAMB and COMPARE, PISYS08P and RELAP5 were also used in the FFWTR evaluation, as discussed below:

LAMB

LAMB is a thermal-hydraulic code to analyze the short-term blowdown phase of a LOCA where there is a rapid system depressurization. The typical LAMB calculation extends only into the first 30 to 45 seconds of the LOCA transient before initiation of the ECCS flows. LAMB was approved as identified in Table 1-3 of NEDC-33506P. The same code was used for AP loads in HCGS MELLLA (NEDC-33066P) and was accepted by the NRC (ADAMS ML060620470).

PISYS08

PISYS08P was inadvertently left out of NEDC-33506P, Table 1-3. PISYS08P is a GEH proprietary, Level 2, NRC reviewed code, (ADAMS Accession No. ML100890011) and was used to evaluate the MS Line B and Recirculation Loop B piping response in terms of piping stresses and support loads.

COMPARE

COMPARE is used for sub-compartment pressurization analysis and was used in the FFWTR Evaluation for the determination of Annulus Pressurization pressure-time histories.

Using mass/energy releases from the GEH LAMB code, sub-compartment pressure time histories were calculated using COMPARE MOD 1-PC (COMPARHS), Revision 0A, code. The results of COMPARE were used directly in determining the net forces and moments on the reactor vessel and the biological shield wall for AP loads. Previously the ARTS/MELLLA and EPU Hope Creek design bases Annulus Pressurization pressure-time histories were also generated using COMPARE.

RELAP5

RELAP5 is used for determining the mass and energy (M/E) release transients in a pipe break and was used in the FFWTR evaluation for RWCU pipe breaks

RELAP5, Mod 3.2 was used to determine mass and energy releases for four RWCU pipe break locations (HELB) and four different initial conditions. This was used to show that the current design basis M/E releases bound the M/E releases at FFWTR conditions. RELAP5 has been widely used by the nuclear industry and by the NRC.

EMCB-RAI-6

In reference to RSLB and FWLB, NEDC-33506P Sections 4.1.1 and 4.1.2 state that, "the current analysis structural loads on the RPV and reactor internals are bounded by the design basis loads." This statement is not clear. Please explain what the term "current analysis structural loads..." means and whether it includes the proposed 102 °F FFWTR effects.

RAI-6 Response

The current analysis structural loads are the results of the loads during the high energy breaks due to the FFWTR and FWHOOS operating conditions. The proposed 102°F FFWTR effects are included in the analysis scope.

EMCB-RAI-7
The LAR indicates that the AP loads have been evaluated for their affect on the primary and containment structures, but it only discusses the effects on the RPV, RPV internals, BSW and RPV pedestal.

a) Have the effects of the AP loads, when considering GEH SC09-01 for the proposed amendment, been evaluated for all of the containment structures including the drywell structure and its attachments?

b) If the answer to (a) above is affirmative, please discuss the results and conclusions of these evaluations. If not, provide a technical justification for not performing these evaluations.

RAI-7 Response

The AP loads have not been evaluated for all of the containment structures.

Response spectra at the drywell location for the FFWTR/FWHOOS conditions (this analysis) for RSLB and FWLB were compared with the design basis response spectra. It was concluded that the results of the original design basis AP ARS bounded the results from the AP ARS evaluations associated with the FFWTR/FWHOOS, as discussed in the response to RAI-11.

Other structures are addressed in SC 09-01 as excerpted below:

> "The potential increase in the annulus pressurization load term was included in the load combinations for the major RPV and reactor internal components. The increases in stresses for these components were then determined and compared against the allowable stress limits. The RPV and internal components were shown to have sufficient margin to accommodate the increase in AP forces.
>
> The shield wall structure is usually evaluated using a bounding static pressure load. The forces and moments calculated by the structural dynamic model are not used in these evaluations; therefore, the concern raised is not applicable to the shield wall analysis. In addition, because the shield wall analysis is a static analysis, the maximum mass/energy release assumption is appropriate for this analysis.
>
> The same conclusions for the RPV and internals apply to the containment structures, piping and equipment. This conclusion is consistent with the physical geometry of the plant and the distribution of the AP loading. That is, the AP loading on the primary structure corresponds to an applied nodal load vector applied to a local region of the primary structure on the inner surface of the shield wall and the outer surface of the RPV. Furthermore, globally the AP loading tends to be self-equilibrating. Also, the response of the primary structure will tend to not build-up away from the local region of application of the AP load.

Consequently, the response of the primary structure in the balance of plant (BOP) due to the AP loading will be relatively small compared to the seismic response. That is, the seismic SSE response in the BOP will tend to dominate the AP response. Also, if the BOP bounding faulted load combination contains other dynamic inertia load terms in addition to SSE and AP or does not contain AP; this will further reduce the impact of the AP loads. It then follows that the potential increase in the AP asymmetric pressurization load will have a smaller effect on the BOP responses than on the RPV and internals responses."

The AP analysis is a forced response analysis. The heavier the structure, the less excitation is created by the applied forces. Because containment is a large heavy structure, the response due to the AP forces will be greatly attenuated (as will the spectra). Therefore, the AP response in the primary and secondary containment structures is small compared to other events such as the Safe Shutdown Earthquake (SSE) in the overall load contribution.

EMCB-RAI-8

PSEG's letter LR-N10-0356 dated September 20, 2010 (ADAMS Accession No. ML102660024), provided an RAI response supporting the HCGS license renewal application. Table 4.3.1-2, "Fatigue Monitoring Locations for HCGS Reactor Pressure Vessel Components and Estimated CUFs," on page 10 of Enclosure A to the letter shows that the design basis 40-year fatigue cumulative usage factor (CUF) for the shroud support is 0.672. The table also shows that the estimated 60-year CUF for the shroud support is 0.465.

As discussed in Section 5.3 of NEDC-33506P, for the proposed amendment, the fatigue life of the reactor internals (including the shroud support) was evaluated based on 60-years, taking into consideration the proposed plant life extension (PLEX). Please provide additional information to address how the 60-year CUF for the shroud support was determined.

RAI-8 Response

The 60-year CUF for the shroud support identified in Section 5.3 of NEDC-33506P was determined via HCGS license renewal application calculations. The value of 0.47 in NEDC-33506P is a rounded value and is the same as 0.465 value in Table 4.3.1-2 provided in PSEG's letter LR-N10-0356 dated September 20, 2010 (ADAMS Accession No. ML102660024). As stated in NEDC-33506P, the effect of FWTR on the shroud support fatigue life is insignificant; therefore the HCGS license renewal application calculations are still applicable and were not effected by FWTR.

The HCGS license renewal application calculations provide an estimated 60-year CUF. This estimate is based on 60-year cycle projections provided in Table 4.3.1-1, "HCGS Reactor Pressure Vessel Design Transients and 60-Cycle Projections" in PSEG's letter LR-N10-0356. Estimated CUF is computed via a design-basis fatigue calculation where the projected number of cycles is substituted for the assumed design basis number of cycles using the governing stress methodology.

The cycle projections or number of occurrences expected for 60 years of operation were obtained by extrapolating the numbers of occurrences actually incurred to-date (as of 12/31/2007), and using the rate of occurrence experienced during the previous twelve years of operation (nine operating cycles). The frequency of events, such as scrams and shutdowns, experienced in the previous twelve years is significantly less than that experienced during the

first ten years of operation, and is expected to remain equal to or less than the trend over the previous twelve years through the period of extended operation by maintaining careful attention to good operating practices. Conservatism was added beyond the mathematically projected number of cycles to accommodate potential variation in plant performance late in plant life, as well as to allow for additional events where the projected number of cycles was very low and the likelihood of additional events could not be ruled out.

EMCB-RAI-9
List the references which approve the exclusion of the FW Sparger from the licensing renewal evaluation, as implied on pages 5-4 through 5-5 of NEDC-33506P.

RAI-9 Response

The scoping assessment for all reactor internals was included in Section 2 of the Hope Creek's License Renewal Application, submitted by PSEG to the NRC on August 18, 2009 (ADAMS ML092430375). The feedwater sparger was evaluated against the License Renewal scoping criteria in 10 CFR 54.4(a). As stated in the scoping evaluation in the LRA, "A safety assessment for these components [including the FW Sparger] has been performed and reported in BWRVIP-06. The evaluation concluded that these components do not perform a safety related function. This report also concluded that failure of these components will not result in consequential failure of any safety related equipment."

The NRC's Safety Evaluation dated March 2011 (ADAMS ML110690244) concludes that PSEG has appropriately identified the reactor internals mechanical components within the scope of license renewal as required by 10 CFR 54.4(a)

EMCB-RAI-10
Page 9 of Attachment 1 of the LAR indicates that an analysis has demonstrated that HCGS operation with the proposed feedwater temperature reductions (FWTR) will not exceed the design limits for the design basis accident (DBA) loss-of-coolant accident (LOCA) peak drywell pressure and temperature. Please state the design limits for the DBA-LOCA peak drywell pressure and temperature and compare these limits with those derived from the analysis for proposed FWTR. In addition, please list the loads considered and governed for the existing design limits and for the FFWTR analysis values.

RAI-10 Response

The design limits for the DBA-LOCA peak drywell pressure and temperature are 62 psig and 340°F, respectively. The HCGS LOCA analysis of record was performed at 102% of the original targeted Extended Power Uprate (EPU) power (100% = 3952 MWt) in support of the EPU submittal (ADAMS Accession number ML081230540). The current or actual EPU power level is 3840 MWt and this value was used (at 102%) in the current FFWTR/FWHOOS DBA-LOCA peak drywell pressure and temperature analysis. That analysis demonstrated FFWTR/FWHOOS to have negligible impact with respect to the EPU analysis and was bounded by the EPU values of 50.6 psig and 298°F reported in the EPU submittal. The peak DW pressure of 50.6 psig and peak DW temperature of 298°F are within the Design Limits of 62 psig and 340°F respectively. The FFWTR/FWHOOS containment hydrodynamic loads evaluations included vent thrust, pool swell, condensation oscillation and chugging loads

(Reference Section 3.3.2 of NEDC-33506P). These are the loads considered in the existing design limits and for the FFWTR/FWHOOS analysis values.

EMCB-RAI-11
With respect to NEDC-33506P, Section 5.4, "Reactor Coolant Pressure Boundary Piping," please discuss how the AP loads have changed based on the proposed amendment. In addition, provide summaries for the reactor coolant pressure boundary pipe stresses and fatigue CUFs and pipe supports/restraints and compare these values to the Code allowable values to support your statement that, "The results of those evaluations showed with the change in AP loads, the stresses on the piping, supports, and restraints will continue to meet the applicable ASME Code requirements."

RAI-11 Response

General Description

The effects on the AP loads due to the FFWTR/FWHOOS conditions on the AP amplified response spectra (ARS) are shown on Figures C-2, C-4, C-6, C-7, C-8, C-10, C-11, C-13, C-18, and C-19 compared to the original design AP ARS for various locations inside the containment. The original design AP ARS with 2% damping are in black color and the FFWTR/FWHOOS condition AP ARS with 2% damping are in green color.

Two of the critical piping systems were selected to perform analysis to assure that the calculated piping moments, stresses, support loads, and valve accelerations in the reactor coolant pressure boundary (RCPB) piping from the FFWTR/FWHOOS AP ARS are less than the values based on the original design AP ARS.

The two piping systems selected were (1) HCGS main steam line B (MS) and (2) HCGS recirculation system loop B (RRS). The main steam piping system is selected because the main steam nozzle is at the higher elevation of the Reactor Pressure Vessel (RPV), and is supported by the biological shield wall and the containment wall and it has the most severe effect of the AP load. The main steam piping included the 26-inch main loop piping and all the 10-inch safety relieve valve piping to the suppression pool diaphragm floor. The recirculation system is selected since the recirculation RPV nozzle is at the lower elevation of the RPV and has lesser effect of the AP load. The recirculation piping included the 28-inch main loop piping, the riser pipes to the RPV nozzles, the RHR supply line and RHR return line. These two piping systems include various pipe sizes, various RPV nozzles; various pipe supports, and various natural frequencies in the range for the AP loads analyses.

Two dynamic analysis cases were performed. Case (1) used the original design AP ARS. Case (2) used the FFWTR/FWHOOS AP ARS. If the calculated piping moments, stresses, support loads and the valve accelerations due to Case (2) are less than the values based on the Case (1) for both piping systems, then the original design AP results envelop the FFWTR/FWHOOS AP results. This conclusion can be extended to all the other piping systems in the RCPB inside the containment because these two piping systems includes various pipe sizes, various RPV nozzles; various pipe supports, and various natural frequencies in the range for the AP loads analyses.

Note that fatigue analysis is performed for Normal/Upset conditions and is not applicable to this condition (Emergency/Faulted), as stated in NEDC-33506P, Section 5.3 (also see the response to RAI#12).

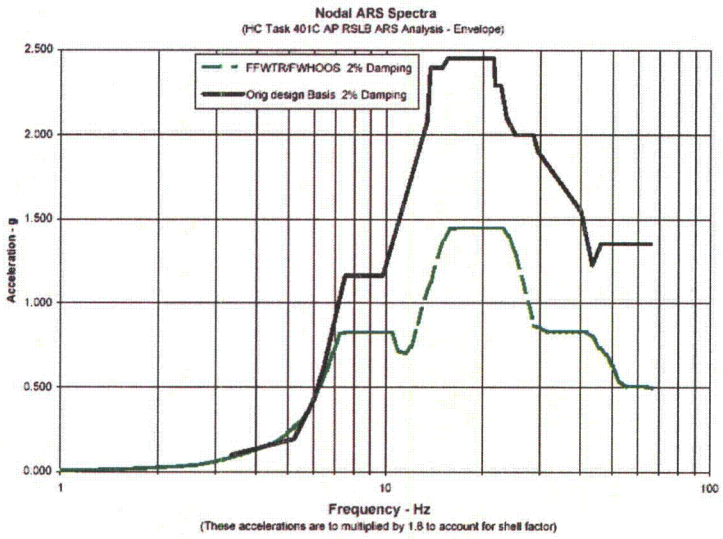

Figure C-2: **RSLB 15% broadened** **Node** **31** **@ Elev.** **2094.93**
RPV (near MSL)

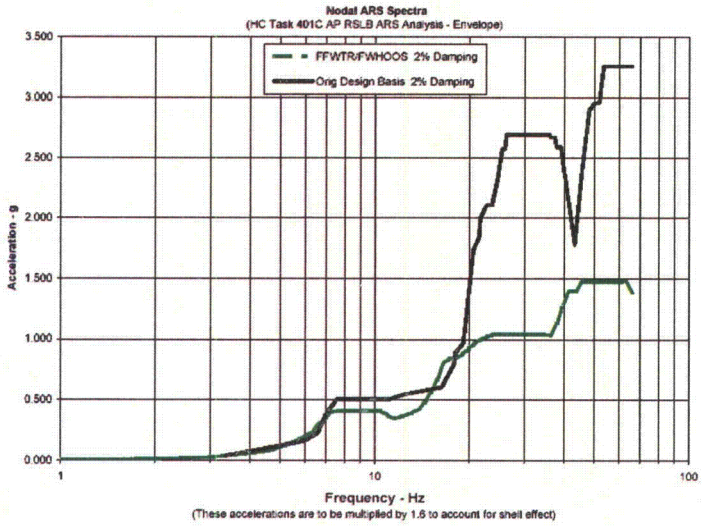

Figure C-4: **RSLB 15% broadened** **Node** **39** **@ Elev.** **1940.04**
Shield Wall (top of Shield Wall)

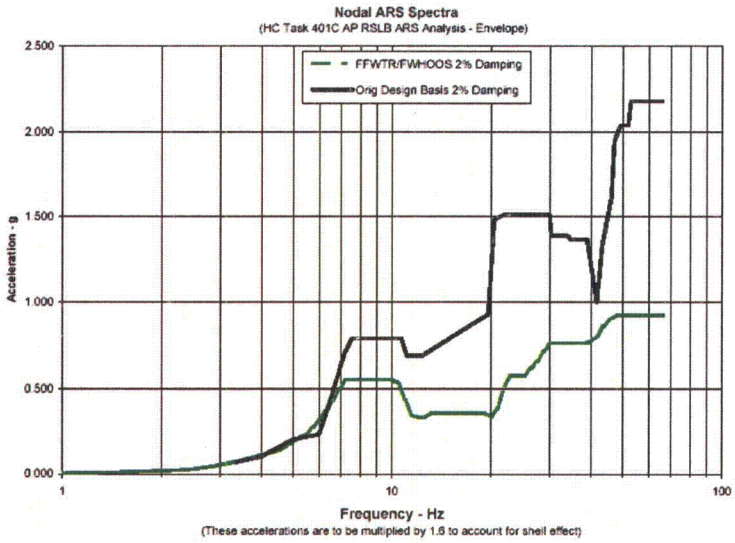

Figure C-6: **RSLB 15% broadened** **Node** **51** **@ Elev.** **1825.63**
RPV (near FWL)

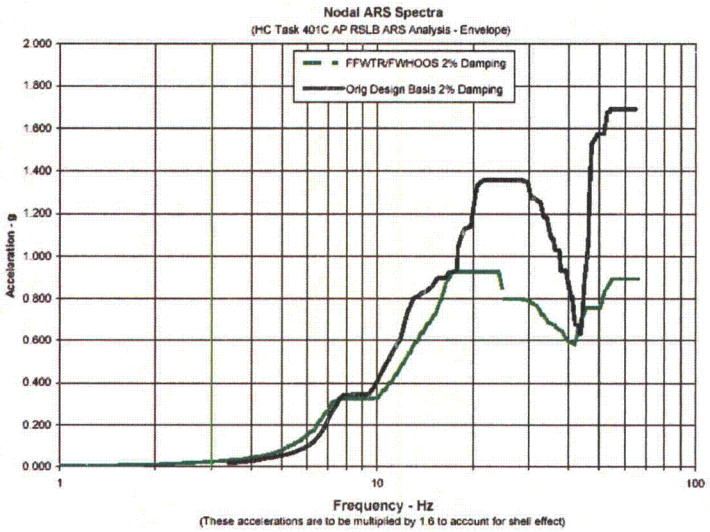

Figure C-7: **RSLB 15% broadened** **Node** **58** **@ Elev.** **1704.00**
Shield Wall

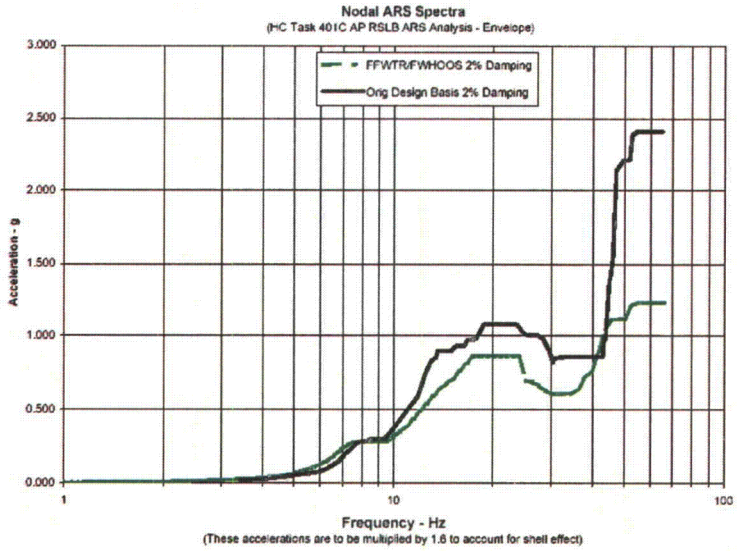

Figure C-8: **RSLB 15% broadened** **Node** **65** **@ Elev.** **1609.44**
Shield Wall

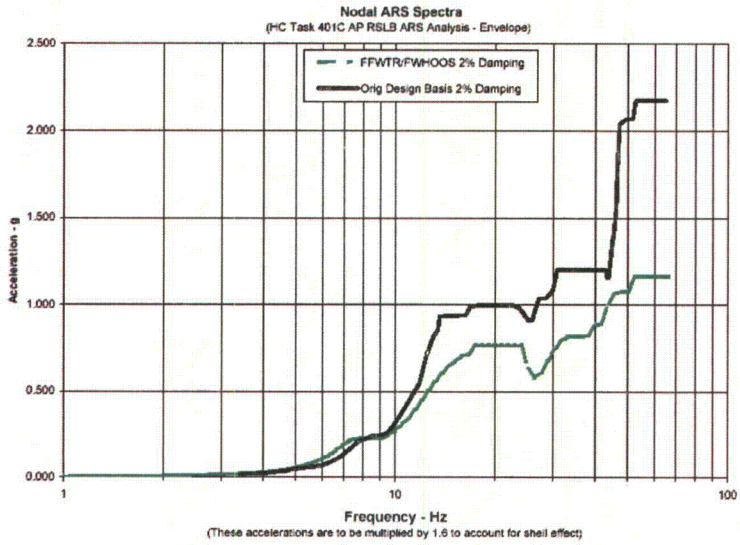

Figure C-10: **RSLB 15% broadened** **Node** **73** **@ Elev.** **1515.00**
Shield Wall (near middle)

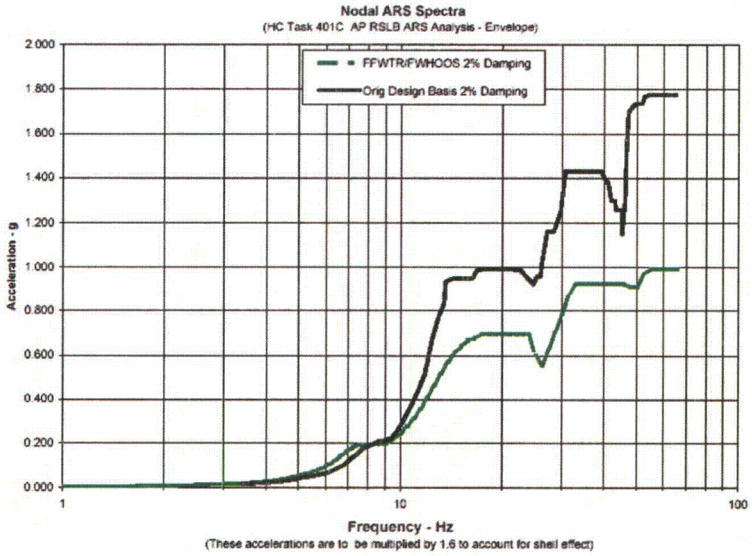

Figure C-11: RSLB 15% broadened Node 82 @ Elev. 1459.56
Shield Wall (near bottom)

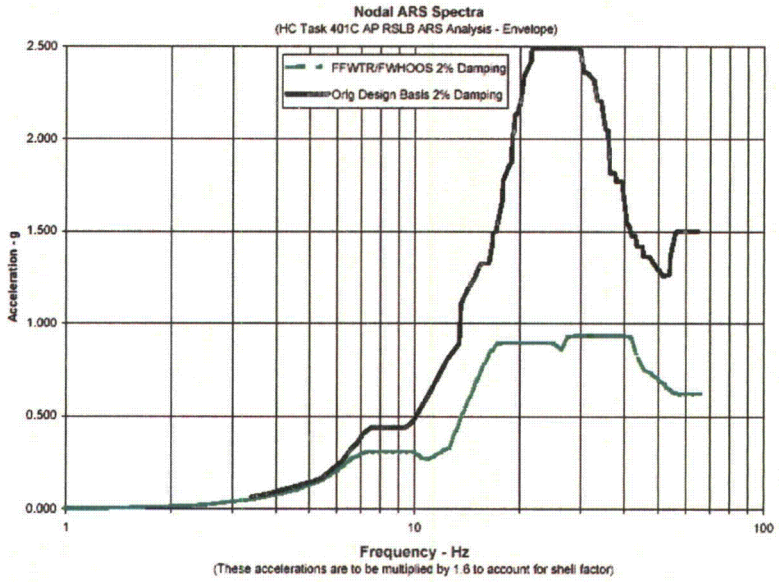

Figure C-13: RSLB 15% broadened Node 114 @ Elev. 1574.12
RPV (near RSL)

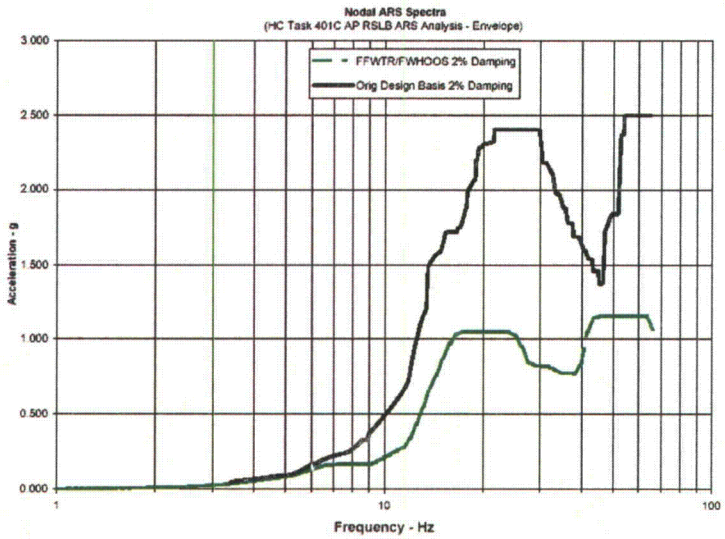

Figure C-18: RSLB 15% broadened Node 137 @ Elev. 1383.00
RPV Bottom

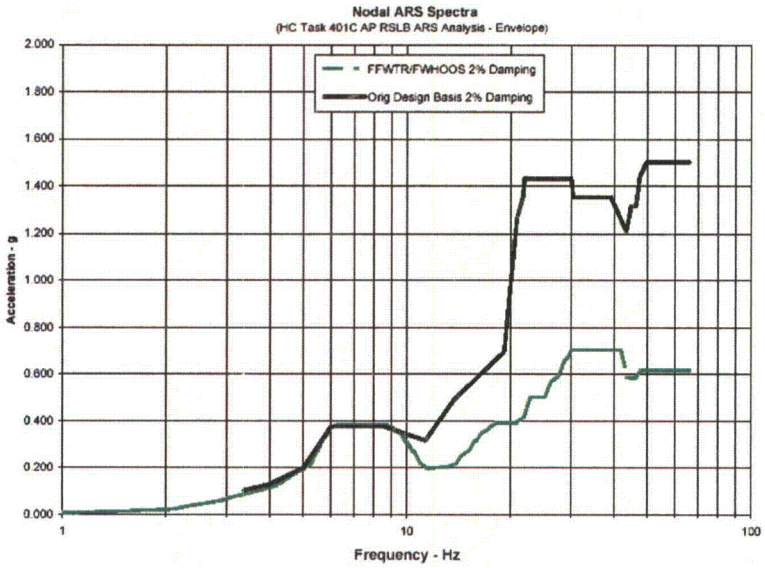

Figure C-19: RSLB 15% broadened of Node 60 @ Elev. 1787.25
Shroud

Method of Analysis

The design AP ARS and the FFWTR/FWHOOS AP ARS are input into the same piping models to calculate the piping moments, pipe stresses, piping support loads, and the valve accelerations for comparison. Therefore, the year of the Code had no affect to these comparisons. Normally, the dynamic analysis for AP load only applies the load in the direction of the postulated pipe break. In order to assure that the comparison results can be applied to all the piping in the RCPB inside the containment, the horizontal directional ARS of original design, and FFWTR/FWHOOS ARS, were applied in X and Z direction. The results were SRSS (square root sum of the square). A summary of the analysis load types and load cases are shown below:

Main Steam piping (MS):

(1) Postulated recirculation line break (RSLB)

(a) Original design AP (ORIG)

(a-1) Horizontal X direction (APX)

(a-2) Horizontal Z direction (APZ)

$(ORIG)=(APX^2 + APZ^2)^{0.5}$

(b) FFWTR/FWHOOS AP (FFWTR)

(b-1) Horizontal X direction (APX)

(b-2) Horizontal Z direction (APZ)

$(FFWTR)=(APX^2 + APZ^2)^{0.5}$

(2) Postulated feedwater line break (FWLB)

(a) Original design AP (ORIG)

(a-1)Horizontal X direction (APX)

(a-2) Horizontal Z direction (APZ)

$(ORIG)=(APX^2 + APZ^2)^{0.5}$

(b) FFWTR/FWHOOS AP (FFWTR)

(b-1)Horizontal X direction

(b-2) Horizontal Z direction

$(FFWTR)=(APX^2 + APZ^2)^{0.5}$

Comparisons:

RSLB ratio = (FFWTR)/(ORIG)

FWLB ratio = (FFWTR)/(ORIG)

The Reactor Recirculation System (RRS) piping analysis load types and load cases are the same as MS piping system.

The MS Line B and Recirculation Loop B piping response in terms of piping stresses and support loads were evaluated using the GEH proprietary program PISYS08P. The damping ratio used for the original design ARS, 2%, was used for the FFWTR/FWHOOS AP ARS. RG 1.61 Revision 1, 2007, increased the damping ratio to 4% for all pipe sizes for Level D events. The RG 1.61 4% value is not used for this analysis. All the analyses use multiple support response spectrum analysis. All the input parameters are the same for both cases.

Results
Evaluations were performed to determine the effect of the RSLB and FWLB with FFWTR/FWHOOS AP ARS in comparison to original design AP ARS to determine the comparative effect on the RCPB piping.

MS Loop B piping analysis:
The MS B piping analysis results of FFWTR/FWHOOS AP ARS were compared with those of the original design basis. Based on the comparison, it is observed that the original design basis loads bound current FFWTR/FWHOOS AP ARS for damping values of 2% analysis for piping stresses and pipe supports loads. The summary of the comparison are as follows.

Main Steam Piping	RSLB Ratio = (FFWTR)/(ORIG)	FWLB ratio = (FFWTR)/(ORIG)
Average MS piping stress ratio	0.666	0.701
Maximum MS piping and branch stress ratio	0.759	0.757
Average Support load ratios	0.682	0.696
Maximum Support load ratios	0.814	0.715

The table shows that the stresses and support loads at the FFWTR/FWHOOS conditions are less than the results from the original AP load. The minimum margin is 18.6% (100 – 81.4)

RRS Loop B piping analysis:
The RRS Loop B piping analysis results of FFWTR/FWHOOS AP ARS were compared with those of the original design basis. Based on the comparison, it is observed that the original design basis loads bound the current FFWTR/FWHOOS AP ARS for damping values of 2% analysis for piping stresses and pipe supports loads. The summary of the comparisons are as follows.

RRS Piping	RSLB Ratio = (FFWTR)/(ORIG)	FWLB ratio = (FFWTR)/(ORIG)
Average RRS piping stress ratio	0.654	0.660
Maximum RRS piping and branch stress ratio	0.632	0.641
Average Support load ratios	0.586	0.635
Maximum Support load ratios	0.645	0.670

The table shows that the stresses and support loads at FFWTR/FWHOOS conditions are less than the results from the original AP load. The minimum margin is 33% (100 – 67).

CONCLUSIONS

The results of the piping evaluation at the FFWTR/FWHOOS conditions for the AP ARS for the RSLB and FWLB indicate that the piping stresses and piping support loads (forces and moments) are bounded by the original design basis loads of the representative piping systems (MS Line B and RRS Loop B) stresses and support loads. Therefore, the design basis AP ARS responses are bounded for all the piping systems in the RCPB. The conclusions and results support the statement in NEDC-33506P, "The results of those evaluations showed with the change in AP loads, the stresses on the piping, supports, and restraints will continue to meet the applicable ASME Code requirements."

EMCB-RAI-12
NEDC-33506P, Section 5.2, discusses flow-induced and acoustic loads resulting on key RPV internals for the FWTR of 102 °F and states that "The acoustic and flow-induced loads will be used for further structural evaluation." Please provide stress and fatigue summaries compared to Code allowable values which demonstrate how these loads have affected the structural integrity of the RPV internal structures. In addition, please explain why these loads have not been developed for the FWHOOS condition.

RAI-12 Response

Flow-induced and acoustic loads were used in Reactor Internals Structural Integrity evaluation (Section 5.3 of NEDC-33506P). The results presented in Section 5.3 of NEDC-33506P were calculated based on the limiting load combination (including flow-induced and acoustic load). In most cases the acoustic load was used in the evaluation because the flow-induced load is bounded by the acoustic load and the loads do not exist simultaneously. Flow-induced and acoustic loads were not used in the fatigue assessment, because they are Faulted condition loads. Fatigue assessment is performed only for Normal/Upset (service level A/B) condition.

Note that the statement "The acoustic and flow-induced loads will be used for further structural evaluation." – is referring to the subsequent task in Section 5.3 of NEDC-33506P.

The comparative summary information is provided in Section 5.3 of NEDC-33506P as a narrative discussion with specific values and allowables provided for the shroud, the core plate, the jet pump assembly, and the Low Pressure Coolant Injection (LPCI) Coupling. Also the fatigue usage factors are provided for the shroud support, the core plate, top guide, the jet pump assembly, the core spray line and sparger, and the Core DP and Liquid Control Line. All fatigue usage factors are compared to the allowable value of one. The specific values presented in the NEDC are not repeated here to preclude this response having to be proprietary.
The feed water temperature reduction due to FWHOOS (60° F reduction) is bounded by that of FFWTR (102° F reduction) due to higher subcooling at lower feed water temperature. Therefore, flow-induced and acoustic loads were not developed for the FWHOOS.

EMCB-RAI-13
Please discuss the limiting conditions/events that bound the FFWTR and FWHOOS conditions for the system cycling fatigue usage for the FW nozzles and FW piping.

RAI-13 Response

Any transient corresponding to FFWTR conditions would be bounded by the reduction to 0% power (from rated power) transient, and any transient corresponding to FWHOOS conditions would be bounded by the daily power reduction transient. The reduction to 0% power transient is a reactor power reduction from rated power conditions to 0% power prior to a complete shutdown. It consists of essentially the same feedwater temperature reduction as is done during FFWTR, except more rapidly. The daily power reduction transient is a reduction in power level from rated to 75% and then a return to rated power. This includes a 66°F reduction in feedwater temperature followed by a return to the original temperature.

EMCB-RAI-14
NEDC-33506P, Section 6.5, shows the 40-year rapid cycling fatigue usage for the FW nozzle safe end for the FFWTR duty, the FWHOOS duty and the original duty. For the FFWTR and FWHOOS duties, a newer ASME Code (2001 with 2003 Addenda) has been utilized for the values of the instantaneous coefficient of thermal expansion in the fatigue evaluation.

a) Please explain why the original design basis Code was not used for the fatigue evaluation and provide a technical justification which reconciles the newer Code to the original Code values used for the FW nozzles.

b) Please describe the methodology used in the NEDC-33506P Reference 25 for the feedwater nozzle rapid cycling analysis.

c) In addition, to the safe end values shown in NEDC-33506P please provide 40-year and 60-year max CUF values for the FW nozzle safe end, blend radius area and the FW sparger. Include system cycling and rapid cycling.

RAI-14 Response

a) For the rapid cycling analysis, the newer ASME Code (2001 Edition with Addenda to 2003) was used to be consistent with most recent feedwater nozzle system cycling analysis, and the only information used was the instantaneous coefficient of thermal expansion for carbon steel as a function of temperature. Since the values from the newer Code are larger than those from the older Code, and since the alternating stress is proportional to this coefficient, it is conservative to use the newer Code.

b) The feedwater nozzle rapid cycling methodology used is consistent with the original design analysis and is as follows:

1. The feedwater nozzle metal surface peak-to-peak temperature range ($\Delta T_{p\text{-}p}$) is calculated.

   $\Delta T_{p\text{-}p} = \bar{A}\ C_3\ C_4\ (T_A - T_{FW})$, where

   $\bar{A}$ = amplitude coefficient
   C_3 = location-dependent coefficient
   C_4 = leakage-dependent coefficient
   T_A = Region A (vessel) temperature

T_{FW} = feedwater temperature

2. Next, the alternating stress (S_{alt}) produced by rapid cycling is calculated.

   $S_{alt} = E\ \alpha\ \Delta T_{p\text{-}p}/[2(1 - \nu)]$, where

   E = Young's modulus, taken from the fatigue curve = $30(10)^6$ psi for carbon steel
   α = instantaneous coefficient of thermal expansion, evaluated at the average surface temperature
   ν = Poisson's ratio, assumed as 0.3
   $\Delta T_{p\text{-}p}$ = metal surface peak-to-peak temperature range

3. Finally usage is determined by fatigue curves from S_{alt}.

c) The table below provides the projected 40-year and 60-year usage factors and the allowable CUF for the feedwater safe-end and the blend radius. The projected values are not maximum values. The fatigue monitoring program will ensure that the actual fatigue usage will remain below the allowable value.

Location	40-Year Projected Usage Factor (1)	60-Year Projected Usage Factor (1)	Allowable
Safe-End	0.192	0.305	1.0
Blend Radius	0.143	0.210	1.0

1. Includes both system cycling and rapid cycling. System cycling CUF is based on CUF as of 12/31/2007 and the trends from the most recent twelve years (nine operating cycles) of actual plant operation data.

As stated in RAI-9, the feedwater spargers were not included in the License Renewal evaluation. Therefore 40-year and 60-year projections are not available.

EMCB-RAI-15
For FFWTR and FWHOOS please provide the following:

a) Cycle summaries for the FFWTR and FWHOOS which show the plant life associated permissible frequencies of occurrence. Also, if different, please provide the number of cycles used for the fatigue evaluations.

b) Discuss the permissible number of days per year that the plant can operate with the proposed FFWTR and FWHOOS and how this was derived.

c) Also, please address how the FFWTR and FWHOOS limits (temperature and cycles) will be implemented and protected or monitored.

RAI-15 Response

a) As stated in RAI-13 the cycles that bound FFWTR and FWHOOS are Reduction to 0% Power and Daily Power Reduction.

The Daily Power Reduction has insignificant impact on fatigue and is not included in the fatigue monitoring program. Therefore cycle summaries are not available for Daily Power Reduction. The design number of cycles is 6,667 (minimum number from all UFSAR sources).

The below is the cycle summary and 60-year projection for Reduction to 0%. The 60-year projection is based on 12 years prior to 12/31/2007.

Design Number of Cycles	111
Cycles as of 12/31/2007	79
Cycles as of 12/31/2010	82
60-year projected number of cycles	174

The number of design cycles does not represent a design limit. The fatigue for a component is normally the result of several different thermal and pressure transients. Exceeding the number of cycles for one transient does not necessarily imply the fatigue usage will exceed an acceptance limit. The fatigue monitoring program will monitor all necessary plant transients to ensure the fatigue usage remains less than the allowable limit. In the event that the monitored usage factor is predicated to exceed the allowable value for any component, appropriate corrective action will be taken in accordance with the corrective action program.

b) The number of days assumed used in the rapid cycling fatigue evaluation for days per year operation with FFWTR and FWHOOS is 128 days and 20 days respectively. As with design cycle limits, the number of days with FFWTR and FWHOOS is not a design limit and not regarded as a permissible number of days. Rapid cycling fatigue usage is included in the fatigue monitoring program. The fatigue monitoring program will ensure the total fatigue usage, cyclic plus rapid, remains less than the allowable limit.

These values were derived from 192 days of FFWTR per operating 18-month cycle and FWHOOS for 30 days per 18-month operating cycle.

c) As discussed previously, cycles will be monitored by the fatigue monitoring program. The fatigue monitoring program will monitor all necessary plant transients to ensure the fatigue usage remains less than the allowable limit. The temperature limits will be monitored and be constrained by operating procedures.