From: "Limberger, Wayne" <wlimber@entergy.com>
To: <JGR@nrc.gov>
Date: 11/14/2007 4:06:49 PM
Subject: Responses to Open AMP/AMR Audit Questions

<<Q&A Open Questions.PDF>>
Jonathan,

Attached are the last 10 pages of the AMP/AMR Audit Q&A database report containing the Vermont Yankee updated responses to the six remaining open questions from the EAF portion of the audit (Questions #387 through #392). Also attached are the signed submittal letter and the Attachment cover page. The hard-copy package with the full database report is due to arrive there tomorrow morning.

CC: "Mannai, David" <dmannai@entergy.com>, "Metell, Mike" <hmetell@entergy.com>, "Lach, David J" <DLach@entergy.com>, "YOUNG, GARRY G" <GYOUNG4@entergy.com>, "COX, ALAN B" <acox@entergy.com>

Mail Envelope Properties (473B6361.539 : 10 : 38201)

Subject:	Responses to Open AMP/AMR Audit Questions
Creation Date	11/14/2007 4:05:47 PM
From:	"Limberger, Wayne" <wlimber@entergy.com>
Created By:	wlimber@entergy.com

Recipients
nrc.gov
TWGWPO03.HQGWDO01
JGR (Jonathan Rowley)

entergy.com
acox CC (ALAN B COX)
GYOUNG4 CC (GARRY G YOUNG)
DLach CC (David J Lach)
hmetell CC (Mike Metell)
dmannai CC (David Mannai)

Post Office	**Route**
TWGWPO03.HQGWDO01	nrc.gov entergy.com

Files	**Size**	**Date & Time**
MESSAGE	435	11/14/2007 4:05:47 PM
BVY 07-079 Submittal Letter.pdf		57882
BVY 07-079 Attachment Page.pdf		63922
Q&A Open Questions.PDF	679412	
Mime.822	1099928	

Options

Expiration Date:	None
Priority:	Standard
ReplyRequested:	No
Return Notification:	None
Concealed Subject:	No
Security:	Standard

Junk Mail Handling Evaluation Results
Message is eligible for Junk Mail handling
This message was not classified as Junk Mail

Junk Mail settings when this message was delivered
Junk Mail handling disabled by User
Junk Mail handling disabled by Administrator
Junk List is not enabled
Junk Mail using personal address books is not enabled
Block List is not enabled

Entergy Nuclear Operations, Inc.
Vermont Yankee
P.O. Box 0500
185 Old Ferry Road
Brattleboro, VT 05302-0500
Tel 802 257 5271

Entergy

November 14, 2007

BVY 07-079

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

Reference: 1. Aging Management Program/Aging Management Review (AMP/AMR) Audit Q&A Database, Revision 5, dated March 27, 2007

Subject: Vermont Yankee Nuclear Power Station
License No. DPR-28 (Docket No. 50-271)
<u>Update of Aging Management Program Audit Q&A Database</u>

This letter provides Revision 6 of the AMP/AMR audit question-and-answer (Q&A) database. In addition to providing a complete current version of the database, this update addresses Staff questions from the Environmentally Assisted Fatigue (EAF) audit portion of the NRC's Vermont Yankee License Renewal Application Aging Management Program review process. Questions # 387 through #392 of this database remain open and are submitted herewith for Staff review. All other responses were previously reviewed and are considered closed. The enclosed database (Attachment 1) updates and supersedes Reference 1.

This letter contains no new commitments.

Should you have any questions concerning this matter, please contact Mr. David Mannai at (802) 258-5422.

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

Executed on November 14, 2007.

Sincerely,

For T. SULLIVAN

Ted A. Sullivan
Site Vice President
Vermont Yankee Nuclear Power Station

Attachment
cc listing (next page)

cc: Mr. James Dyer, Director
U.S. Nuclear Regulatory Commission
Office O5E7
Washington, DC 20555-00001

Mr. Samuel J. Collins
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. Jack Strosnider, Director
U.S. Nuclear Regulatory Commission
Office T8A23
Washington, DC 20555-00001

Mr. Jonathan Rowley, Senior Project Manager
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
MS-O-11F1
Rockville, MD 20853

Mr. Mike Modes
USNRC RI
475 Allendale Road
King of Prussia, PA 19406

Mr. James Kim, Project Manager
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O 8 C2A
Washington, DC 20555

USNRC Resident Inspector
Entergy Nuclear Vermont Yankee, LLC
PO Box 157
Vernon, Vermont 05354

Mr. David O'Brien, Commissioner
VT Department of Public Service
112 State Street – Drawer 20
Montpelier, Vermont 05620-2601

Diane Curran, Esq.
Harmon, Curran, Spielberg & Eisenberg, LLP
1726 M Street, N.W., Suite 600
Washington, DC 20036

Attachment 1

Vermont Yankee Nuclear Power Station
License No. DPR-28 (Docket No. 50-271)

AMP/AMR Audit
Q&A Database
Revision 6

Item	Request	Response
387	The ASME Code defines that stress intensity (SI) from two temperature transients is calculated from the stress components from the two conditions. Please explain how it could be calculated from stress intensities of the two conditions derived from Greens Functions, especially at locations of geometric discontinuity. Also, please justify the validity of combining the thermal transient stress intensities with the stress intensities from the external loads and pressure loading.	To address Environmentally Assisted Fatigue (EAF) for the NUREG/CR-6260 locations at Vermont Yankee, the stress inputs for the reactor vessel and nozzles were either taken from the design basis stress analyses or new stress analyses were performed. Existing stress analyses were used for the controlling locations on the vessel shell and for the Recirculation Inlet nozzles. New stress analyses were performed for the Feedwater, Reactor Recirculation Outlet, and Core Spray nozzles per ASME III, NB-3200. Updated fatigue analyses for the reactor vessel and nozzles were performed per ASME III, Subsection NB-3222. New fatigue analyses for the Class 1 portions of the Feedwater and Reactor Recirculation/RHR piping were performed per ASME III, NB-3600. Finite element models (FEM) using ANSYS were developed for the new fatigue analyses of the Reactor Recirculation Outlet and Core Spray nozzles. The FEM for each nozzle is 2-D axisymmetric about the centerline of each nozzle. The radius of the vessel in the FEM was multiplied by a factor of two (2) to account for variation in pressure stress for a nozzle oriented normal to the cylindrical vessel shell. For the Feedwater nozzles, a previously developed, 2-D axisymmetric ANSYS FEM was used. The vessel radius used in this model was 1.5 times the radius of the vessel. Pressure stresses from this model were factored by (2.0/1.5) = 1.333 to account for variation in pressure stress for a nozzle oriented normal to the cylindrical vessel shell. For the new fatigue analyses of the Feedwater, Reactor Recirculation Outlet, and Core Spray nozzles, stress intensities due to internal pressure were calculated directly using the ANSYS FEM model. The controlling location for thermal stresses at the safe end of each FEM was determined using a 500ºF to 100ºF temperature step transient at 100% flow conditions. The controlling location in the blend radius of each FEM was taken as the location of maximum stresses due to internal pressure. Stress intensities for each thermal transient were determined using Green's function (GF) methodology. The GF at each controlling location was developed from the FEM stress results for the 500ºF to 100ºF temperature step transient. At each controlling location, absolute values of the component stress differences, (SZ-SX, SY-SX, SZ-SY), were compared to the maximum stress intensity calculated from ANSYS. For ease of calculation, the stress difference which most closely matched the total stress intensity calculated by ANSYS was used to determine the GF at each location. In most cases the maximum component stress difference with time matched the maximum stress intensity calculated by ANSYS. This shows that shearing stresses are negligible for the thermal transient at that location and the maximum component stress difference is the maximum stress intensity. Stress components from attached piping loads at the controlling thermal stress locations were calculated separately using standard strength of materials equations. Stress intensities were calculated from the stress components per ASME Code, Section III, Subsection NB-3215. These stress intensities are referred to as the "hand calculation method" as described below. To show that the GF approach used to calculate alternating stress intensities for the

thermal transients obtains results comparable to results from an ASME Code, Section III, Subsection NB-3222 calculation, a comparison with the results a previous fatigue calculation was conducted. This comparison used the identical FEM constructed for the VY Feedwater nozzle.

The VY ASME Code design fatigue calculation (VY-10Q-303) which was performed directly using ANSYS, was compared to the EAF calculation (VY-16Q-302) performed using the GF methodology for the turbine roll transient. This is the most severe design basis transient for VY Feedwater nozzle. To ensure a consistent comparison between the two calculations, the same stress path locations were selected. The Code fatigue calculation alternating stresses (using the limiting Sz-Sx stress difference) were extracted from the ANSYS model at the same paths used in the EAF calculation. To be consistent the Code FEM analysis was re-run with the same heat transfer coefficients and material properties used for the GF calculation. The comparison showed that the alternating stress intensities calculated using the FEM with the Code methodology and those calculated with the GF methodology are within 1% at both the safe end and blend radius locations.

Although this comparison was for the feedwater nozzle, the results are considered to be equally applicable to all other nozzle locations based on a BWR Vessel and Internals Project (BWRVIP) study (EPRI Report No. 1003557, "BWRVIP-108: BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," Final Report, October 2002.

In BWRVIP-108, 3-D models of four different nozzles were developed and analyzed that bounded all nozzle geometries for the BWR fleet. The results of this study showed that for a range of vessel nozzles modeled using the same technique, the ratio of maximum pressure stress intensity at the blend radius to the primary membrane stress intensity at the vessel wall away from the nozzle is nearly a constant with an average ratio that varies by +/- 3%. This indicates that all different sized BWR vessel nozzles have the same geometric characteristics for calculating peak stresses in the blend radius regions.

Figures 4-30 to 4-33 in BWRVIP-108 show the nozzle blend radius stress profiles for pressure and steady state thermal stresses for the four (4) different BWR nozzles. The figures show a significant variation of pressure stress around the centerline of the nozzle with the peak hoop pressure stresses occurring at the +90º (top) and -90º (bottom) azimuths. This is due to the differences in hoop and axial stresses in a cylindrical vessel. The new FEM models used in the Vermont Yankee EAF evaluations were 2-D axisymmetric about the centerline of each nozzle. The radius of the vessel in the FEM was multiplied by a factor of two (2) to account for variation in pressure stress for a nozzle oriented normal to the cylindrical vessel shell.

Figures 4-30 to 4-33 in BWRVIP-108 also show no significant variance in steady state thermal stresses at the nozzle. The figures show the magnitude of axial stress at the 0º & 180º azimuths is equal to the magnitude of the hoop stress at +90º and -90º azimuths. This shows that the thermal stress in the blend radius oriented normal to the axis of the nozzle is nearly constant. Thermal transients used in the EAF evaluations are localized to the nozzle safe end, bore, and blend radius regions. Therefore, the use of 2-D axisymmetric modeling vs. the use of a 3-D FEM is adequate to determine thermal transient stresses in both the safe end and blend radius locations.

The adequacy of the hand calculations used to calculate mechanical load stresses is addressed as follows:

Stress intensities from the attached piping loads at the controlling thermal stress locations were calculated from stress components per ASME Section III, Subsection NB-3215. For the feedwater nozzle of another BWR plant, hand calculations were performed for stresses due to mechanical loads. The hand calculations were performed using the same methodology as used for VY. These were compared to the results from a finite element model which included the mechanical loads applied directly to the model.

The finite element model was an axisymmetric two-dimensional (2-D) finite element model. This model was constructed and meshed in a very similar manner to the VY nozzle FEMs. Non-symmetric loading elements were used and the shear, moment, axial, and torsional loads were applied to the model.

A comparison of the stresses from the hand calculations vs. the FEM is as follows:

Location: Safe-End - Linearized Membrane + Bending Stress
Stress from Hand Calculations (psi): 8863
Stress from FEM (psi): 5852
Difference, Hand Calc vs. FEM: +51.45%

Location: Safe-End - Total Stress
Stress from Hand Calculations (psi): 8863
Stress from FEM (psi): 7855
Difference, Hand Calc vs. FEM: +12.83%

Location: Nozzle Forging - Linearized Membrane + Bending Stress
Stress from Hand Calculations (psi): 1042
Stress from FEM (psi): 769
Difference, Hand Calc vs. FEM: +35.50%

Location: Nozzle Forging - Total Stress
Stress from Hand Calculations (psi): 1042
Stress from FEM (psi): 554
Difference, Hand Calc vs. FEM: +88.09%

As shown by these results, use of the hand calculations is conservative compared to the FEM results. Stress intensities from attached piping loads are larger in the safe end section of each nozzle and are significantly reduced for the blend radius section due to the larger section thickness provided by the nozzle reinforcement. As shown in Table 3 of calculation VY-16Q-302 for the VY Feedwater nozzles, the maximum stress intensity from attached piping loads is 5708 psi for the safe end and 265 psi for the blend radius.

A comparison of the maximum stress intensity from the attached piping loads with the total stress intensity from the significant transients from Tables 4 and 5 of calculation VY-16Q-302 for the VY Feedwater Nozzle follows:

Location: Table 5: Safe End
Attached Piping Maximum Stress Intensity (psi): 5708.

Transient: No. 3: Startup, t = 16,328 sec.
Total Stress Intensity (psi): 14396.
Attached Piping Stress Intensity is 39.6 % of Total

Transient: No. 4: Turbine Roll, t = 4 sec.
Total Stress Intensity (psi): 53379.
Attached Piping Stress Intensity is 10.7% of Total

Transient: No. 11: Scram – LOFP, t = 2168 sec.
Total Stress Intensity (psi): 70223.
Attached Piping Stress Intensity is 8.1% of Total

Transient: No. 20A: Hot Standby, t = 4 sec.
Total Stress Intensity (psi): 53379.
Attached Piping Stress Intensity is 10.7% of Total

Location: Table 4: Blend Radius
Attached Piping Maximum Stress Intensity (psi): 265.

Transient: No. 3: Startup t = 16,782 sec
Total Stress Intensity (psi): 34282.
Attached Piping Stress Intensity is 0.8% of Total

Transient: No. 4: Turbine Roll t = 1802 sec.
Total Stress Intensity (psi): 67667.
Attached Piping Stress Intensity is 0.4% of Total

Transient: No. 11: Scram – LOFP t = 195 sec.
Total Stress Intensity (psi): 74567.
Attached Piping Stress Intensity is 0.4% of Total

Transient: No. 20A: Hot Standby, t = 183 sec.
Total Stress Intensity (psi): 66298.
Attached Piping Stress Intensity is 0.4% of Total

As shown in table above, the contribution to the total stress range from the attached piping loads is more significant for the safe end location and could effectively be ignored for the blend radius location.

Combining the thermal transient stress intensities directly with the stress intensities calculated from the external loads and pressure loading essentially combines the maximum principal stresses calculated for each load case. This allows for combination of stress results where different methods or models are used to calculate the stresses, and typically produces conservative results compared to combining all stress components and then determining a stress intensity.

The practice of combining stress intensities from thermal transient load cases directly with the stress intensities from mechanical loads vs. combining all stress components and then calculating a combined stress intensity was used by CB&I for the original design analyses for Vermont Yankee and by GE in the analyses for the replacement safe ends for the Reactor Recirculation Inlet and Outlet Nozzles.

For the evaluation of EAF for Vermont Yankee, the combination of thermal stress intensities with stress intensities from external loads and pressure was performed as follows:

The stress intensity calculated from external loading is added to the maximum calculated thermal transient stress intensity using the same sign to increase the stress range. This is necessary because the direction of applied loading from external loads is not known. The stress intensity from the external loads for each transient is scaled for the temperature of the transient assuming no stress occurs at 70ºF and full values are reached at reactor design temperature of 575ºF. This maximizes the stress range pairings for the fatigue analysis. The pressure stress intensity value is added to the stress intensity from the combined thermal and external loadings directly as a positive value, since pressure is always positive due to the known direction of loading.

388 Provide justification for statement on page 5 of 34 of Calculation No. VY-16Q-302, that "The Greens Function methodology provides identical results compared to running the input transient through the finite element model."

A calculation was performed by the vendor as part of the generic verification for the Green's function approach. The calculation compared the results from an ANSYS FEM analysis of a feedwater nozzle for a turbine roll transient with the results using Green's functions for the same transient. The results showed the stress range differences between the Green's function approach and the ANSYS FEM for the safe end location were between -0.06% and 3.43% and for the blend radius location were between -1.73% and 1.56%. These differences are considered well within the accuracy range of the analysis.

In addition, a VY specific comparison was made for the Feedwater nozzle as described in the response to question 387 above. The comparison showed that the alternating stress intensities calculated using the ANSYS FEM with the ASME Code methodology and those calculated with the GF methodology for the same transient inputs are within 1% at both the safe end and blend radius locations.

Further discussion of Green's functions and how they are used in a fatigue monitoring system is available in two papers. The papers are titled "An On-Line Fatigue Monitoring System for Power Plants: Part I – Direct Calculation of Transient Peak Stress Through Transfer Matrices and Green's Functions" and "An On-Line Fatigue Monitoring System for Power Plants: Part II – Development of a Personal Computer Based System for Fatigue Monitoring", ASME Pressure Vessel and Piping Conference, Vol.112, 1986 (Kuo, Tang, and Riccardella).

The intent of the statement in this and other calculations was to indicate that equivalent stress history results are obtained from each method (Green's function vs. FEM) for a given thermal transient.

Item	Request	Response
389	For the blend radius for the feedwater nozzle in Calculation No. VY-16Q-302, Table 4, Page 16: Why are the Total & M+ B stresses for Thermal Transient 3 shown in columns 3 & 4 high at t=0 sec. (zero stress state?) This question also applies to: Transient 4 at t =1801.9 sec. Transient 9 at t = 2524 sec. Transient 21-23 at t= 20144 sec. This question may also apply to transients 11, 12, and, 14.	The Green's functions are based on constant material properties and heat transfer coefficients. Therefore, parameters were chosen to bound the transients that result in the majority of the fatigue usage. The temperatures in the design transients range from 100ºF to 549ºF. Material properties and heat transfer coefficients at 300ºF were used. These bound the cold water injection events. In addition, the instantaneous value of the coefficient of thermal expansion is used instead of the mean value. To maximize stresses in the blend radius, the Green's function was based on a fluid temperature shock of 500ºF to 100ºF in the nozzle flow path while the vessel wall portion of the model was exposed to a constant fluid temperature of 500ºF. Therefore the reference point stress estimated from this Green's function at an ambient nozzle fluid temperature is non-zero due to the vessel wall being held at 500ºF. The resulting stress ranges from the thermal transient analysis using the Green's function methodology are accurate regardless of the reference point used as long as the material properties used are consistent with the transient temperature range. With the above in mind, Table 4 of VY-16Q-302 was set up to yield stress pairings which ensure the calculated stress ranges would be maximized. These temperature conditions are appropriate for Green's function integration of all feedwater nozzle transients which contribute to fatigue, since they occur with feedwater flow injecting through the nozzle into a hot vessel. In reality, these conditions are conservative for transients where there is no flow in the nozzle or for transients where the reactor temperature drops below 500°F. This is due to a large temperature gradient induced into the nozzle structure due to the temperature difference between the reactor and nozzle flow path portions of the model. This temperature difference leads to the high stress values observed for Transients 3, 14 and 21-23 at ambient temperatures.

Item	Request	Response
390	Explain why there are differences in the calculated CUF values a between Rev. A and Rev. 0 of the Structural Integrity Calculations. Also, why are the CUFs calculated by Structural Integrity different from the CUFs shown in Tables 4.3.1 & 4.3.3 of the Vermont Yankee License Renewal Application?	Calculations VY-16Q-301 through VY-16Q- 311 performed for VY and issued as Revision A have the Revision Description on each calculation cover sheet labeled as "Initial Draft for Review". These draft calculations were issued for client review and comment. The draft versions of the calculations were never intended to be the issued version until all inputs were finalized and all external and internal reviews and comments were incorporated. The Revision A calculations were provided under Entergy's obligation to provide all documents related to Environmentally Assisted Fatigue for NEC Contention 2. The Revision 0 versions of these calculations were subsequently issued after comments on the draft calculations were resolved and all design inputs finalized and verified. Revision 0 (or later) versions of the calculations are the Calculations of Record. The Revision A drafts are no longer applicable. The Revision 0 calculations and reports incorporated reviewer comments which included; expanded descriptions of the methodologies and analyses, additional references, typographical corrections, and component specific technical comments which affected the final CUF values. Referring to Table 3-10 in the Summary Report, SIR-07-132-NPS (VY-16Q-404), the most significant changes from Revision A to Revision 0 for 60 year environmental CUFs greater than 0.50 were for the Feedwater nozzle and the Reactor Recirculation / RHR piping. Specific inputs which affected the CUF for the Feedwater nozzle included increased pressure stresses and reduced thermal stress inputs for isothermal events. The Revision 0 calculation for the Feedwater nozzle (VY-16Q-302) shows the blend radius as the limiting section vs. the safe end as the limiting section in Revision A. This resulted in an increase in the 60 year CUF from 0.0127 to 0.0636 and an increase in the 60 yr environmental CUF from 0.127 to 0.639 at the Feedwater nozzle blend radius. Corrections to the transient temperature inputs for the Reactor Recirculation / RHR piping model resulted in the maximum calculated CUFen = 0.7446 at the RHR return tee location. The draft Revision A of the calculation had the RHR suction tee controlling. The CUFs shown in Tables 4.3.1 & 4.3.3 of the Vermont Yankee License Renewal Application were based on the design basis fatigue evaluations factored to account for the effects of the 120% Extended Power Uprate. For locations with no plant specific CUFs, representative values from NUREG/CR-6260 were used. The CUFs calculated for the Environmentally Assisted Fatigue evaluation are different from the CUFs shown in the VY LRA due to a number of factors specific to each location. These include: •updated finite element modeling (FEM) vs. the shell analysis techniques used in the original design analysis, •direct thermal transient analysis using the FEM vs. the separate thermal analyses to determine temperature distributions used in the original design analyses, •use of updated transient definitions for 60 years of operation. The updated transient definitions are shown in Design Input Record (DIR) for EC No. 1773, Rev. 0, "Environmental Fatigue Analysis for Vermont Yankee Nuclear Power Station"

Revision 1, dated 7/26/07.

For the NUREG/CR-6260 locations without existing fatigue analyses, new VY plant-specific ASME III fatigue analyses were performed.

391 On page 1-1 of Report VY-16Q-401 it indicates that refined transient definitions 60 years are used in the computation of the CUF including EAF effects. Please explain the refinements in the transient definitions.

The original design transients for the VY Reactor Vessel are given in Section 5.1.8 and Attachment D to General Electric Purchase Specification No. 21A1115, "Reactor Pressure Vessel", Revision 4, 10/21/69 and certified on 10/23/69 as contained in the Reactor Pressure Vessel Design Report. This document is the Vermont Yankee Reactor Vessel Design Specification. Additional clarifications and descriptions for the design transients were provided by General Electric in GE Letter W. J. Zarella to D.W. Edwards - Yankee Atomic, Subject: "V. Y. R.P.V. Temperature Transient / Cycling Events", No. G-HB-5-124, dated November 5, 1975.

Earlier versions of the specification made reference to a GE Thermal Cycle Drawing No. 885D941. The final version of the design specification relocated this cycle information to Attachment D of the specification and deleted references to GE drawing No. 885D941.

Comparisons were made between the VY Design Specification transients and the design transients shown on Thermal Cycle Drawings from other GE BWR 4 plants of the same and later vintage. The later plants have more detailed thermal cycle descriptions based on the experience from the earlier GE BWRs.

In general, VY is designed for a smaller spectrum of the most severe transients as compared to the full spectrum of transients used for the later units. As described in General Electric Letter No. G-HB-5-124, the number of cycles for each VY design transient exceeds the number of cycles for the same transient from the typical GE thermal cycles diagram listed in the original VY FSAR. For example; the single severe design transient for the VY Feedwater nozzle of 1500 cycles exceeds the 518 Start-up, Loss of Feedwater Heater, Scram, and Shutdown events listed in the original VY FSAR.

To insure a realistic projection of design thermal transient cycles and events for 60 years of operation, the Thermal Cycle Diagrams used at a number of BWR 4 plants were used as a starting point. The VY Design Specification transients were mapped onto the typical BWR 4 Transient Diagram. Then projections for 60 years were made based on the numbers of cycles in the VY Design Specification, the numbers actually analyzed in the VY Design Certified Stress Report for Vermont Yankee Reactor Vessel, Chicago Bridge & Iron, Contract 9-6201, and the number of cycles experienced by VY in approximately 35 years of operation. For all Service Level A & B events, the 60-year projected cycles for each transient used in the EAF evaluations exceed the actual number of cycles experienced by VY projected to 60 years of operation. The basis for the 60 year transient definitions is documented in Appendix C of calculation VYC-378 Revision 2.

Item	Request	Response
392	For the Feedwater Nozzles there are large differences in the CUFs without the Fen factors shown in shown in Table 4.3.1 of the Vermont Yankee License Renewal Application and those shown in calculation VY-16Q-302. Section 2.0 of the calculation on page 4 of 32 states, "...several of the conservatisms originally used in the original feedwater evaluation (such as grouping of transients) are removed ...". Please explain what conservatisms were removed.	The original Design Transient for the VY Feedwater nozzle is given in Attachment D to GE Specification No. 21A1115, "Reactor Pressure Vessel", Revision 4, 10/21/69. It is a single severe design transient intended to envelop all Start-up, Loss of Feedwater Heater, Scram, and Shutdown events. It consists of 1500 cycles of: • Feedwater nozzle at 546ºF steady state and 0% feedwater flow, followed by, • Step change from 546ºF to 100ºF with 25% feedwater flow, followed by, • Feedwater nozzle at 100ºF steady state and 25% feedwater flow, followed by, • Step change from 100ºF to 260ºF, followed by, • A ramp from 260ºF to 376ºF at 250ºF/Hr concurrent with increasing feedwater flow 25% to 100% rated flow. This transient is equivalent to a Startup and Turbine Roll event combination specified on newer BWR plant Thermal Cycle Diagrams. As described in GE Letter No. G-HB-5-124, dated November 5, 1975, the 1500 such events considered in the design fatigue evaluation of the Feedwater nozzle exceed the 518 Start-up, Loss of Feedwater Heater, Scram, and Shut- down events listed in the original FSAR. The CUF for the Feedwater nozzle shown in Table 4.3.1 of the Vermont Yankee License Renewal Application is based on the design basis fatigue evaluations factored to account for the effects of the 120% Extended Power Uprate (EPU). Changes in temperatures for EPU are from GE Nuclear Energy Certified Design Specification No. 26A6019, "Reactor Vessel - Extended Power Uprate", Rev. 1, 8/29/03. The evaluation of EPU effects on the feedwater nozzle and safe end stress and fatigue analysis is contained in VY Engineering Report, VY-RPT-05-00100, Rev. 0, "Task T0302 Reactor Vessel Integrity-Stress Evaluation EPU Task Report for ER-04-1409". Section 3.3.1.1 of GE Report for Task 302, shows the value for the feedwater nozzle safe end EPU CUF for 40 years = 0.75. This is the value shown in Table 4.3.1. The 0.75 CUF value is based on the original design report. The original design analysis was performed for "loose fit" feedwater spargers where the annular cold gap between the stainless steel thermal sleeve and the nozzle safe end was 0.020 inch. The feedwater spargers and thermal sleeves were replaced in 1976 with new "interference fit" thermal sleeves. The interference fit thermal sleeves significantly reduce leakage flow past the thermal sleeve into the bore region of the nozzles. This reduces the heat transfer from the process fluid to the nozzle base metal, thereby reducing thermal stresses during system thermal transients. Subsequent to the GE report, a re-analysis of the Feedwater nozzle was performed. Report No. SIR-04-020 Revision 0, March 2004. "Updated Stress and Fatigue Analysis for the Vermont Yankee Feedwater Nozzles" documents a revised ASME III Stress and Fatigue Analysis for the Feedwater nozzle and safe end. This

analysis included effects of the interference fit thermal sleeve. The analysis was performed for both the original licensed power and system flow rates using the enveloping design transient, "Startup, Loss of Feedwater Heaters, Scram & Shutdown", from the original Design Specification and for EPU power and flow conditions as modified per the EPU Design Specification. For the nozzle safe end, the 40-year CUF using 1500 cycles of the enveloping transient and including EPU effects = 0.4513 (as compared to the 0.75 value used in the LRA). The primary reason for the decrease in CUF was a result of the improved heat transfer coefficients resulting from the interference fit thermal sleeve.

The calculated fatigue usage for the safe end prior to the installation of the interference fit thermal sleeve using the actual number of startup and shutdown cycles and the allowable number of cycles from the original CB&I design report is 0.02. As documented in Appendix D to calculation VYC-378, Rev. 2, this has a negligible effect on the revised CUF for the safe end including EPU effects shown above. This is primarily due to conservatisms in the Updated Feedwater Fatigue Analysis for the number of cycles operating under EPU conditions.

For the Environmentally Assisted Fatigue (EAF) evaluation, (VY-16Q-302), a realistic projection of Design Thermal Transient Cycles and Events for 60 years of operation based on the Feedwater Nozzle Thermal Cycle Diagram from a typical BWR 4 was used. A described in the response to Question 391, the enveloping design transient was mapped to the "Turbine Roll & Increase to Rated Power" transient. Other transients including Loss of Feedwater Heaters and Scram events were taken directly from the typical BWR 4 Feedwater Nozzle Thermal Cycles Diagram using VY specific EPU design pressures and temperatures. The projections for 60 years were based on the number of events in the VY Design Specification, the numbers analyzed in the VY Design Certified Stress Report for VY Reactor Vessel, and the number of cycles experienced by VY in approximately 35 years of operation.

The design transients used in the EAF evaluation for the VY Feedwater nozzle are shown in Attachment 1 to Design Input Record (DIR) for EC No. 1773, Rev. 0, "Environmental Fatigue Analysis for Vermont Yankee Nuclear Power Station" Revision 1, dated 7/26/07.