UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR VERMONT YANKEE, LLC	)	Docket No. 50-271-LR
AND ENTERGY NUCLEAR OPERATIONS, INC.	)	
	)	
(Vermont Yankee Nuclear Power Station)	)	

AFFIDAVIT OF KENNETH C. CHANG
CONCERNING NEC CONTENTIONS 2A & 2B (Metal Fatigue)

Q1. Please state your name, occupation, and by whom you are employed.

A1. My name is Kenneth C. Chang. I am employed by the U.S. Nuclear Regulatory Commission ("NRC") as Chief, Engineering Review Branch 1, Division of License Renewal of the Office of Nuclear Reactor Regulation ("NRR"). A statement of my professional qualifications is attached hereto.

Q2. Please explain your duties in connection with the Staff's review of the License Renewal Application ("LRA") submitted by Entergy and Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. ("Entergy," "Vermont Yankee," or "VYNPS").

A2. As the Branch Chief of RER1, I have the overall responsibility for the safety reviews, in mechanical and materials engineering disciplines, of aging management programs ("AMPs"), aging management reviews ("AMRs"), and time-limited aging analysis ("TLAA"), including the metal fatigue to address the environmentally-assisted fatigue ("EAF"), associated with the license renewal applications. With regard to the Vermont Yankee LRA, I directed the staff of RER1's performance of the safety review of the AMPs, AMRs, and TLAAs in Vermont Yankee's LRA. I was with the audit team while performing the on-site audit. I personally reviewed the sections associated with the metal fatigue and the fatigue monitoring program along with my assigned staff because I am a known expert in these areas. After the audit, I wrote Section 4.3.3 of the SER with information gathered by the audit team.

Q3. What is the purpose of your testimony?

A3. The purpose of this testimony is to present the Staff's position regarding NEC Contentions 2A & 2B (Recalculation of CUFs). As admitted by the Board in LBP-07-15, 66 NRC 261 (2007) and Order (Granting Motion to Amend Contention 2A) (April 24, 2008) (unpublished) ("April 24, 2008 Order"), NEC contends that Entergy's analyses of environmentally corrected cumulative usage factors (CUFens) is "flawed by numerous uncertainties, unjustified assumptions, and insufficient conservatism, and produced unrealistically optimistic results. Entergy has not by these analyses, demonstrated that the reactor components assessed will not fail due to metal fatigue during the period of extended operation." I have read relevant portions of LPB-06-20, 64 NRC 131 (2006) (admitting NEC Contention 2 (Metal Fatigue)); NEC's "Petition for Leave to Intervene, Request for Hearing and Contentions" (May 26, 2006); NEC's "Motion to File a Timely New or Amended Contention (July 12, 2007), NEC's "Motion to File a Timely New or Amended Contention (September 4, 2007), LBP-07-15, 66 NRC 261 (2007) (admitting NEC Contention 2A); NEC's "Motion to File a Timely New or Amended Contention" (March 17, 2008); and the April 24, 2008 Order.

Q4. Describe the Staff's review of Entergy's reanalysis of environmentally adjusted cumulative usage factors ("CUFen").

A4. In a letter dated September 17, 2007 (Staff Exh. 22), the applicant submitted Amendment 31 with the results of its "refined" fatigue analyses, as specified in Commitment 27, for all the locations identified in NUREG/CR-6260 (Staff Exh. 6). In this letter, the applicant also provided additional information on Fatigue Monitoring Program ("FMP") as specified in Commitment 5. The staff reviewed the additional information on the FMP and found it acceptable because the program is now consistent with the Section X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary" of the GALL Report (NUREG-1801, Revision 1) (Staff Exh. 7). In order to perform a detailed review of the refined fatigue analysis, specifically in the area of EAF, the staff performed an audit on October 9-10, 2007 at VYNPS. In a letter dated November 14, 2007, the applicant submitted its responses to questions relating to EAF identified through this audit.

The staff reviewed the applicant's response and determined that shear stresses can not be neglected in calculating stress intensities, which were used to determine the allowable cycles and the CUFs at all NUREG/CR-6260 locations, because it is difficult to determine the threshold for when shear stresses are small enough to be negligible. Therefore, the applicant's method for calculating stress intensities using a special purpose computer code could be invalid. The staff concluded that the way the software calculates the stress intensity through a simplified 1-dimensional ("1-D") stress input to Green's Function may not be valid because it simplifies the six stress components discussed in the ASME Code rules into one component of stress. Since the staff could not confirm the validity of Vermont Yankee's September 2007 refined fatigue analysis and CUF calculation, Request for Additional Information ("RAI") 4.3.3-2 was issued. RAI 4.3.3-2 asked Entergy to provide the following:

> Please identify the exceptions where maximum component stress difference with time did not match the maximum stress intensity calculated by ANSYS. In addition, please justify the exceptions, based on quantities evaluations, that the shearing stresses are negligible and the maximum component stress difference is the maximum stress intensity for the branch nozzle blend radius (nozzle corner) locations with geometrical

> discontinuities for the applicable thermal transients. Your response should cover the shearing stress differences at the 0-180 degree axis and the 90-270 degree axis to the pipe run axis.

In a letter dated December 11, 2007 (Staff Exh. 8), Entergy submitted Amendment 33 and provided its response to RAI 4.3.3-2, which included results from the new fatigue analysis ("reanalysis") of Feedwater ("FW"), Reactor Recirculation Outlet ("RR") and Core Spray ("CS") nozzles. The staff reviewed Entergy's response as well as the additional calculations and determined that the applicant did not resolve the staff's concerns. Specifically, the staff noted that it was reported in Entergy's response that component stress differences could be 10% to 50% lower than the maximum stress intensity calculated by ANSYS, Inc. software using all six stress components. In addition, the staff noted that Entergy utilized a simplified 1-D stress as part of the computer software input to calculate stresses due to temperature transients. The staff found there was not enough information to assure the validity of the simplified Green's function input. The concerns identified above were related to Entergy via a telephone conference call on December 18, 2007. Entergy and the staff were unable to resolve the issues raised, and the applicant requested to have a public meeting to further discuss the EAF analysis performed for the plant.

On January 8, 2008, the staff and Entergy held a public meeting to discuss the response to the RAI. Following Entergy's presentation and discussion, Entergy agreed to perform an additional EAF analysis on the reactor pressure vessel FW nozzle to confirm that results presented in Amendment 33 are conservative. Entergy defined this analysis as the "confirmative analysis." In a letter dated January 30, 2008 (Exh. NEC-JH_34), Entergy submitted the results of the confirmative analysis for the FW nozzle to the NRC. At the February 7, 2008 ACRS Full Committee Meeting, the staff reiterated its concerns on the reanalysis and informed the ACRS Committee members that the staff did not have sufficient time to evaluate the confirmative analysis. 549th ACRS Meeting Transcript (ML080500208) at 81-82, 87 (Staff Exh. 9). The staff reviewed Entergy's response, which included an audit on February 14, 2008, and found that for this analysis of the FW nozzle, the stress intensities and the CUFs were calculated in accordance with the ASME Code requirements and the CUF met the Code limit. However, it also showed that the previous analysis was not bounding for the feedwater nozzle using all the same inputs, including Fen values. Therefore, the staff requested that Entergy define this analysis as the "analysis of record" for the FW nozzle. By letter dated February 21, 2008 (Staff Exh. 23), Entergy stated that it considers the January 2008 analysis the analysis of record for the FW nozzle. As explained below, the FW nozzle is Vermont Yankee's most FAC-susceptible nozzle. Nevertheless, because the CUF value from the analysis of record does not bound the CUF value from Entergy's December 11, 2007 Amendment 33 for the FW nozzle, the staff questioned whether the CUF values for CS and RR outlet nozzles from December 2007, which also used the simplified 1-D stress input, are bounding. Thus, the staff imposed a license condition requiring Vermont Yankee to perform ASME Code NB-32003200 analysis for CS and RR outlet nozzles without using simplified stress inputs.

Q5. In Table 4.3-3 of Vermont Yankee's LRA (Staff Exh. 10), the CUFens for some of the listed components are greater than 1.0. Explain why it is possible to "refine" predicted CUFens to less than 1.0?

A5. When a calculated CUFen for a component is greater than the allowable value of 1.0, it is possible to reduce the predicted value of CUFen. This is done by analyzing the actual transients cycles experienced by the plant to obtain CUFen instead of using original design cycles. In general, actual plant transients are less severe than the design transients, which are defined on a generic basis for all similar plants for the design of the component, and therefore, typically result in a CUF value that is lower than that of the original design calculation. In addition, transients may occur less frequently than specified by the original design, which may lead to a lower CUF value for the component. The ASME Code allows performance of a more detailed analysis as a way to demonstrate code compliance.

Q6. Describe how Entergy performed the refined analysis submitted September 17, 2007 (Staff Exh. 22)?

A6. In the September 17, 2007 letter, Entergy submitted its refined analysis for the following locations: Reactor Pressure Vessel ("RPV") vessel shell/bottom head, RPV shell at shroud support, FW nozzle forging blend radius, RR Class 1 piping, RR inlet nozzle forging, RR inlet nozzle safe end, RR outlet nozzle forging, CS nozzle forging blend radius and FW piping riser to RPV. Existing stress analyses were used for the controlling locations on the vessel shell and RR inlet nozzles. New fatigue analysis for the Class 1 portions of the FW and RR piping were performed per ASME III, NB-3600. In addition, new stress analyses were performed for the FW, RR outlet, and CS nozzles, which Entergy stated to be in accordance with ASME III, NB-3222. The stress intensities for the thermal transients were calculated using a simplified 1-D stress as part of the input to a computer code. Fen values, calculated by formulas presented in NUREG/CR-6583 (Staff Exh. 11) and NUREG/CR-5704 (Staff Exh. 12), were conservatively used in the calculation of CUFens. The Fen values were conservative because the parameters used for the calculations bound VYNPS operating data and maximized the Fen values for higher CUFens.

Q7. How was the revised analysis submitted on December 11, 2007 (Staff Exh. 8) different from the refined analysis submitted on September 17, 2007 (Staff Exh. 22)?

A7. The analysis submitted on December 11, 2007 provides the results for the following locations: FW - blend radius, FW - safe end, CS - blend radius, CS - safe end, CS - piping, RR Outlet - blend radius, and RR Outlet safe-end. New locations for FW, CS, and RR Outlet nozzles were added because the staff found that Entergy did not have sufficient basis to conclude that the locations stated in the September 17, 2007 analysis are controlling locations. In addition, the December 11, 2007 analysis contains a comparison of maximum component stress difference and maximum stress intensity. This additional information was submitted at the staff's request so that a determination could be made on the validity and conservatism of the calculation method using simplified 1-D stress input, for VYNPS locations.

Q8. NEC has questioned the underlying assumptions of Entergy's September 17, 2007 refined analysis. Does Entergy's refined analysis make any assumptions? If yes, describe those assumptions and explain the effect of those assumptions on the analysis.

A8. The assumptions referred to by NEC and its expert are not really assumptions. They are conditions of the analysis. The purpose is to make the analysis more conservative. The first area being questioned is the use of a simplified 1-D Green's Function to calculate stresses from temperature transients for FW, RR outlet, and CS nozzles. This relates to the method of analysis, and not the assumptions. This approach assumes that the maximum component stress differences with time match the maximum stress intensity, and therefore shear stress is negligible. Based on its review of these analyses, the staff has found that this assumption is not always valid as the consideration of shear stress components could account for more than 10 percent of maximum stress intensity in cases where the component geometry does not have an axis of symmetry. The second area being questioned is the locations selected for analysis for the following components: FW, RR outlet and CS nozzles. The staff found that Entergy did not have sufficient basis to conclude that locations selected were in fact controlling locations.

The staff noted that in calculating CUF for NUREG/CR-6260 components, Entergy accounted for all design transients and linearly projected the accrued cycles for these transients. The staff finds it adequate when the FMP will be used to track these transients. In addition, the staff found through the February 14, 2008 audit that the ranges of different parameters used to calculate Fen bound the actual data accrued by the plant, and therefore are considered to be conservative for the Fen calculation. During that audit, the staff also questioned Entergy about whether these parameters would be bounding for the period of extended operation ("PEO"). Entergy responded that all inputs except dissolved oxygen will remain valid for the PEO but that dissolved oxygen has been included in the scope of the Water Chemistry (evaluated in SER Section 3.0.3.1.11 (Staff Exh. 1)) and Fatigue Monitoring Programs, which will ensure that the dissolved oxygen value remains below the dissolved oxygen value used as input to the Fen factor. SER Section 4.3.3.2 (Staff Exh. 1).

Q9. Does Entergy's January 30, 2008 analysis of record (Exh. NEC-JH_34) make any assumptions?

A9. The analysis of record uses the same water chemistry, the same set of transients, and the same projection of transient cycles as the reanalysis to calculate

CUFen. These are analysis inputs and conditions, not assumptions. These inputs have been reviewed by the staff as documented in SER Section 4.3.3 (Staff Exh. 1). The staff found the inputs acceptable. No new assumptions were made in the analysis of record.

Q10. Are Entergy's assumptions about the number of transients in the analyses submitted in September and December 2007 and the analysis of record conservative?

A10. VYNPS submitted its refined analysis on September 17, 2007 (Staff Exh. 22), and a revised analysis on December 11, 2007 (Staff Exh. 8). On January 30, 2008, VYNPS submitted its confirmative analysis (Exh. NEC-JH_34), which later became the analysis of record (Staff Exh. 23). All three analyses used the same transients and the same number of cycles. The staff reviewed the method of transient projections for the PEO during its audit and reviews through the question and answer process. As stated in their September 17, 2007 letter (Staff Exh. 22), VYNPS will track the transients and associated cycles as part of the FMP to ensure the validity of the analysis. Therefore, although the staff cannot determine the level of conservatism regarding the number of transient cycles at this time, FMP, which includes cycle counting, will ensure that the cycle projection is valid and that the fatigue analysis results are conservative, because the results of FMP will be periodically reviewed and cycle projections updated when necessary. Thus, NEC's concern that Entergy's assumptions about the number of transients Vermont Yankee will experience during the PEO are not sufficiently conservative is addressed by the FMP, which ensures that the predicted number of transients is not exceeded.

Q11. Did Entergy perform an error analysis to show the error range for each variable in its CUFen analyses? If not, explain why an error analysis was not necessary.

A11. Entergy did not perform an error analysis for each variable in the CUFen analyses. Error analysis is not necessary because conservatism is built into the ASME Code (the code used by the analysis of record as well as by the refined analysis for the reactor vessel and recirculation nozzle), the equations used to calculate Fen values in NUREG/CR-6583 (Staff Exh. 11) and NUREG/CR-5704 (Staff Exh. 12) (which have been adjusted for uncertainties in life), and the parameters affecting the Fen values (which had been adjusted for uncertainties that are associated with material and operating conditions). Fen values were maximized as practicable consistent with plant conditions. In addition, the FMP and the Water Chemistry Program will track the transients and chemistry conditions in the analyses to ensure their validity as it relates to transient cycles and Fen values. Specifically, water chemistry will be monitored to verify that dissolved oxygen concentration values are below the values used in the analysis.

Q12. Dr. Hopenfeld proposed his own recalculation of CUFen values based on the CUF values originally presented in the LRA and what he asserts are "bounding" values for Fens. See Fourth Hopenfeld Declaration at 10. Do you agree with Dr. Hopenfeld's analysis? Why or why not.

A12. I do not agree with Dr. Hopenfeld's recalculation of CUFen values. The key to the CUFen values is in the calculation of Fen. The Fen values used in Dr. Hopenfeld's recalculation are maximum Fen values for low-alloy steel and stainless steel. Their usage assumes the worst-case scenarios for reactor conditions, which Entergy has proved otherwise based on data VYNPS has collected over its operating history. For Dr. Hopenfeld's recalculation, the design basis CUF values are multiplied by these maximum Fen values which yielded CUFen values greater than 1.0 for almost all of the components' locations. While these CUFen values may seem to be un-acceptable, the CUF values used to calculate CUFen do not pertain to VYNPS. This is because during its audit, the staff noted that some CUF values in LRA Tables 4.3-1 and 4.3-3 were not Vermont Yankee-specific CUF values, but rather were representative values from old vintage nuclear steam supply system ("NSSS") for BWR plants taken directly from NUREG/CR-6260 (Staff Exh. 6). In other words, VYNPS used CUF values from NUREG/CR-6260 in its LRA. These values represented CUFs calculated for components of a plant of the same vintage as VYNPS but did not use VYNPS-specific design information and transients. The staff requested that plant-specific CUF values be calculated through a refined fatigue analysis using plant specific design and transient information.

Q13. Did the Staff conclude that Entergy's analyses (September 2007, December 2007, and January 2008) used appropriate equations to calculate Fens?

A13. Yes. The staff reviewed the Fen calculations for components in all three analyses and noted that Entergy used the equations specified in NUREG/CR-5704 and NUREG/CR-6583 (Staff Exh. 11), which are referenced by the GALL Report. Therefore, the staff concludes that appropriate equations were used by Entergy for license renewal applications. It is important to note that while the equations used to calculate Fen may be different in other applications, they were developed for specific applications. For license renewal, equations in NUREG/CR 5704 and 6583 are endorsed by USNRC and are appropriate for use in the application.

Q14. Dr. Hopenfeld has asserted that Entergy should have, but did not, obtain CUFen values by calculating the partial usage factor for each stress cycle, multiply it by the corresponding Fen value, and then summing the individual products for all stress cycles. Did Entergy calculate CUFens values correctly?

A14. In reviewing the VYNPS calculations for CUFen values for vessel shell, FW nozzle, RR outlet nozzles and CS nozzle, the staff has determined that these values were calculated correctly. The stress analyses and the CUF calculation were completed in accordance with ASME NB-3200, which defines the procedure for analysis for cyclic loadings. The Fen values were calculated by the equations defined in NUREG/CR-5704 (Staff Exh. 12) and NUREG/CR-6583 (Staff Exh. 11) as recommended by the GALL Report. CUFen values are calculated by multiplying the design CUF values by these Fen values. Hence, the CUFen values are calculated correctly.

In his sixth declaration, Dr. Hopenfeld questioned the Fen values on the basis that Fen was proportioned for two different water chemistries, hydrogen water chemistry ("HWC") and normal water chemistry ("NWC"), and that he does not agree that Entergy used bounding Fen values. Fen represents a fatigue life correction factor for the effects of reactor coolant environment. In the absence of plant data, the staff would agree that using proportional values from two different water chemistries is not appropriate and that dissolved oxygen ("DO") value under NWC should be used. However, VYNPS has implemented HWC as part of its Water Chemistry Control – BWR program (evaluated in SER Section 3.0.3.1.11 (Staff Exh. 1)). This implementation is aimed to limit the potential for intergranular stress corrosion cracking by reducing the dissolved oxygen content in the reactor coolant. The dissolved oxygen content was monitored by VYNPS for years prior to HWC implementation and will continue to be monitored as part of Water Chemistry Control - BWR Program. The DO values used in the Fen calculations are the average DO values plus one standard deviation, which bounds almost all the data points in normal plant operation. The staff noted that excursions where oxygen content increase do occur during heatup, however, no significant thermal transients occur during this period so that practically no fatigue usage factor is accrued during this period.

Therefore, the staff does not agree with Dr. Hopenfeld's statement.

Q15. Did the Staff conclude that Entergy's September 2007, December 2007, January 2008 analyses appropriately considered water chemistry (oxygen content) and temperature?

A15. Yes. The Staff verified through the February 14, 2008 audit that all three reanalyses appropriately included the following four parameters used to determine the environmental correction factors (F_{en}) calculated by formulas defined NUREG/CR-6583 (Staff Exh. 11) and NUREG/CR-5704 (Staff Exh. 12): dissolved oxygen, strain rate, temperature and sulfur content. The Staff verified that the values of strain rate, temperature, and sulfur content used in the calculation of Fen would remain valid for the period of extended operations. The dissolved oxygen will be maintained below the level input into the F_{en} factor calculation through the use of the Water Chemistry and Fatigue Monitoring Programs, which the staff have found adequate to manage aging.

Q16. Is Entergy's September 17, 2007 refined analysis overly optimistic? Why or why not? What about the December 11, 2007 revised analysis?

A16. No, Entergy's refined analysis submitted on September 17, 2007 (Staff Exh. 22) and revised analysis submitted on December 11, 2007 (Staff Exh. 8) are not overly optimistic. The staff reviewed the September 2007 refined analysis, the December 2007 reanalysis, and the January 2008 analysis of record for the FW nozzle, and found that the applicant used the same: axisymmetric finite element model (FEM), transient definitions and cycles, ANSYS computer code, ASME elastic-plastic correction factor, water chemistry input, formula for calculating Fen, and alternating stress values corrected for modulus of elasticity (E) values for both analyses. They are correct per ASME Section III Code and the GALL Report recommendations, but not overly optimistic. The only differences between the analyses are that the analysis of record used the ASME NB-3200 methodology, all six stress components, and the appropriate, but not the bounding Fen, for all the transient pairs being evaluated. The use of appropriate Fen values for each transient pair is correct. The use of one bounding Fen value is overly conservative and acceptable, but not necessary.

Q17. Did Entergy's analyses (September 2007, December 2007, and January 2008) use outdated statistical equations (referring to NUREG/CR 6583 and NUREG/CR 5704) to perform its reanalysis instead of NUREG/CF-6909?

A17. The staff guidance for evaluating metal fatigue of components is provided in the GALL Report and in NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants." These documents specify NUREG/CR-6583 (Staff Exh. 11) and NUREG/CR-5704 (Staff Exh. 12) for the calculation of the environmental correction factors. The staff specified NUREG/CR-6583 and NUREG/CR-5704 in Revision 0 of its guidance documents while Argonne National Laboratory (ANL) was still refining the equations used to calculate the environmental correction factor in order to provide regulatory stability given the large number of license renewal applications that were under development at the time. The final ANL equations were provided in NUREG/CR-6909 (Exh. NEC-JH_26). The license renewal guidance is generally more conservative than the guidance in NUREG/CR-6909, especially for carbon and low-alloy steels. The staff endorsed NUREG/CR-6909 in Regulatory Guide (RG) 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors," Revision 0, March 2007 (Staff Exh. 13). RG 1.207 states that the regulatory guide only applies to new plants. Therefore, Entergy followed the staff guidance for license renewal applications and used the correct equations in its reanalysis.

Q18. Did Entergy's January 2008 analysis, which became the analysis of record, use outdated statistical equations (referring to NUREG/CR 6583 and NUREG/CR 5704) to perform its reanalysis instead of NUREG/CR-6909?

A18. The analysis of record used the formulae of NUREG/CR-6583 (Staff Exh. 11) and NUREG/CR-5704 (Staff Exh. 12). The formulae are not outdated. The formulae are the current agency and industry standard for license renewal applications as recommended in NUREG-1800 and NUREG-1801. The staff endorsed NUREG/CR-6909 in Regulatory Guide (RG) 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors," Revision 0, March 2007 (Staff Exh. 13). RG 1.207 stated that the regulatory guide only applies to new plants.

Q19. Did the Staff find Entergy's CUF September and December 2008 analyses acceptable? If yes, explain why. If not, describe the staff's concerns.

A19. The Staff did not find the results of the Entergy's analysis submitted by the September 17, 2007 and the December 11, 2007 letters acceptable. The staff was concerned that Entergy used a simplified stress input to generate the Green's function to calculate stresses from temperature transients. Entergy attempted to use a 1-D stress input instead of using six stress components as input to generate the Green's function. This process can result in inaccurate stress intensities as defined in the ASME Code. For NUREG/CR-6260 components having an axis of symmetry, the simplified Green's function method can fairly yield accurate stress results since all stresses can be accounted for by the simplified stress inputs. However for components without an axis of symmetry, the stresses predicted by the simplified Green's function could be inaccurate.

Q20. On January 30, 2008, Entergy submitted what the Staff now considers the "analysis of record" of the CUF for the feed water nozzle. What was the purpose of that analysis?

A20. The analysis of record, which Entergy designated as "confirmative analysis," was performed in accordance with ASME Code Section III, Subsection NB-3200 with the same design and transient inputs as the refined analysis. The analysis of record was performed because the staff did not find the previous analyses acceptable in all cases. In addition, it allowed the staff to see the amount of change in CUF values resulting from application of the simplified Green's function and determine the details of the FMP (such as preventive actions and corrective actions) that must be implemented such that Vermont Yankee can use the FMP to manage the aging of the components for the PEO. In addition, the CUFen value from the FW nozzle analysis of record serves as a guide as to whether additional analyses based on NB-3200 methodology need to be performed for the other two affected nozzles to provide more accurate values of CUFs and a sound basis for the FMP. The analysis submitted for staff review on January 30, 2008 (Exh. NEC-JH_34), showed that the simplified 1-D stress input approach was not conservative for FW nozzle and does not validate the analysis submitted on December 11, 2007 for CS and RR outlet nozzles. Therefore, the staff imposed a license condition for VYNPS to perform ASME Code analysis for CS and RR outlet nozzles.

Q21. Was Entergy's analysis of record necessary for the Staff to conclude that environmentally assisted fatigue will be adequately managed during the period of extended operation? Why?

A21. Yes. The Staff could not make its safety determination based on the refined analysis or the reanalysis. The analysis of record used the correct methodology, design information and transient data; and therefore ensured that the resultant CUFen is valid.

Q22. Describe the similarities and differences between Entergy's September and December 2007 analyses and Entergy's analysis of record. Explain the significance of each similarity or difference.

A22. There are a lot of similarities and even the same input data between the two analyses. The biggest difference between the two is the method of calculating stress intensities. The September and December analyses incorporated the so-called simplified Green's function, which used the simplified 1-D stress as input. The analysis of record was performed using the ASME Code, Section III rules and the software used for previously approved applications. No Green's function was used, which meant there was no simplification in stress inputs. All six stress components were accounted for in the analysis of record. The transient and water chemistry data used to calculate the Fen values were the same. These inputs were reviewed and approved by the staff when it was reviewing the refined analysis and therefore were carried forward into the analysis of record. However, the September and December 2007 analyses used the bounding Fen value while the analysis of record used the appropriate Fen value calculated for operating conditions.

Q23. Entergy only did an ASME code analyses using all six stress components for the reactor vessel feed water nozzle. Explain why the reactor feedwater nozzle was selected and why performing the analysis of only the FW nozzle was acceptable to the Staff.

A23. The FW nozzle is the most limiting NUREG/CR-6260 component for VYNPS based on CUF calculated from the refined and reanalysis submitted on September 17, 2008 (Staff Exh. 22) and December 11, 2007 (Staff Exh. 8) respectively. From the review of VYNPS operating history as well as operating experiences from other BWR plants, the FW nozzle experiences the most severe temperature transients and the highest number of transient cycles for similar configurations in comparison to other NUREG/CR-6260 components. The FW nozzle results will act as the bounding case, giving the Staff confidence that the CUF values for the RR outlet and CS nozzles will remain below the allowable limit if similar analyses are preformed. The other locations in NUREG/CR-6260 do not need to be reanalyzed using the ASME-code method because the maximum component stress difference with time matches the maximum stress intensity calculated by ANSYS and therefore shear stress is negligible.

It is reasonable to believe that if the CUF values for the components having the highest stresses and CUF remain within the code limit, the components with lower stresses and CUF should also remain within the code limit as well if the analysis is performed. Nevertheless, because the September and December 2007 analysis for the FW nozzle cannot demonstrate the conservatism of the previous analyses, the previous analyses on FW nozzle are voided and the confirmative analysis becomes the analysis of record. Similar analyses need to be performed for the other two nozzles because the analysis of record showed that the CUF value for the FW nozzle in the December 11, 2007 submittal is not conservative, and therefore, the December 11, 2007 CUF values for the RR outlet and CS nozzles may not be conservative as well. A license condition is in effect for VYNPS, requiring the ASME Code Subsection NB-3200 analysis for the core spray and recirculation nozzle components. These analyses will be submitted to the staff for review, and, upon approval, will become the analysis of record for these two nozzles.

Q24. Did the Staff conclude that Entergy's September 2007, December 2007, and January 2008 analyses appropriately considered water chemistry (oxygen content) and temperature?

A24. Yes, Entergy's September 17, 2007 and December 11, 2007 analyses as well as its January 30, 2008 analysis of record adequately account for the water chemistry effects in the evaluation of environmentally-assisted fatigue to determine the value of Fen. Oxygen content, temperature, strain rate, and sulfur content are appropriately considered. During the February 14, 2008 audit, Entergy allowed the staff to review its operating data from the date when monitoring first began to show that the parameters selected for water chemistry and temperature are conservative for the determination of Fen. Based upon its review of Entergy's operating data, the Staff concluded that Entergy appropriately considered oxygen content, temperature, strain rate, and sulfur content.

Q25. Does Entergy's analysis of record appropriately address the expected number of transients? Explain why or why not.

A25. Yes. The expected number of transients is no different from the number used in the refined analysis or the reanalysis. Please see the answer to Q10 of this testimony for the staff's comment on expected number of transients for the refined analysis.

Q26. Why did the Staff conclude that Entergy's analysis of record demonstrated that the CUFs for key components will not reach unity during the period of extended operations?

A26. The staff concluded that the revised feedwater nozzle analysis is consistent with the rules of the ASME Code, Section III and yielded a CUF value less than the code limit of 1.0 for the PEO. However, since the FW nozzle analysis of record did not demonstrate that the previous analyses were conservative, Entergy will submit an analysis summary as part of its license condition analyses for core spray and recirculation outlet nozzles. Nevertheless, since the FW nozzle bounds the CUF for these two nozzles, it is reasonable to believe that these two components' locations will not reach the limit of 1.0 as well when the analysis is completed and therefore the Staff has reasonable assurance that CUFs for key components will not reach unity during the PEO. It should also be noted the Vermont Yankee's FMP assures that the CUFs of key components do not reach unity.

Q27. Describe Entergy's fatigue monitoring program.

A27. The FMP tracks the number of critical thermal and pressure transients for selected reactor coolant system pressure boundary components so they do not exceed the design limit of 1.0 on fatigue usage. The transients tracked by the FMP are done either through manually counting or real-time monitoring with existing software and hardware. The program validates analyses that explicitly assume a specified number of thermal and pressure fatigue transients for each component by assuring that the actual number of transients does not exceed the assumed number of transients used in the fatigue analyses. As stated above the FMP is consistent with the guidance in GALL Section X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary (Staff Exh. 7).

Q28. Explain how Entergy's analysis of record fits into Vermont Yankee's fatigue monitoring program.

A28. The Fatigue Monitoring Program will track the number of transients used in the fatigue analysis of all RCS pressure boundary components including the analysis of record for the FW nozzle and future analyses to be completed and considered as analyses of record for the RR outlet and CS nozzles under the license condition. The number of transients will be continuously trended against the assumed number of transients used to calculate the CUF values. The FMP, therefore, serves to validate the analysis of record. Corrective actions will be taken if the tracked number of transients approaches the projected number of transients in the analysis of record. Corrective actions include either additional refinement of the analysis, aging management, or repair/replacement. These actions are consistent with the GALL Report, and are therefore acceptable to the staff.

Q29. Why did the Staff conclude in Section 4.3.3.4 of its SER that "the application has demonstrated, as required by 10 C.F.R. § 54.21(c)(1)(iii) that the effects of aging on the intended function(s) will be adequately managed during the period of extended operation"?

A29. Entergy has developed an aging management program, FMP, which the Staff finds adequate to manage the aging effects of SSCs to ensure that the intended functions will be maintained in the PEO. An element of the program requires VYNPS to take corrective actions if CUF values approach the allowable limit. Attachment 1 of Amendment 31 submitted on September 17, 2008 (Staff Exh. 22) described the details of the FMP which was found to be consistent with the program described in NUREG-1801, Section X.M1 (Staff Exh. 7). By tracking the number of cycles analyzed in the analysis of record, the FMP will manage the effects of environmentally assisted fatigue on reactor coolant system pressure boundary components through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

Kenneth C. Chang
Statement of Professional Qualifications

CURRENT POSITION:

Chief, Engineering Review Branch 1	Division of License Renewal (DLR), Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Rockville, MD

EDUCATION:

B.S., National Taiwan University, 1963, Civil Engineering
M.S., Washington University, St. Louis, Mo., 1966, Applied Mechanics and Structures
Ph.D., University of California, Berkeley, Ca., 1970, Applied Mechanics

PROFESSIONAL REGISTRATION:

Professional Engineer, Mechanical, Pennsylvania

SUMMARY:

Dr. Chang has over 38 years of design, analysis and qualification experience in the commercial nuclear power industry of which, 21 years were with Westinghouse Electric Corporation serving as fellow engineer, advisory engineer, manager for Reactor Coolant System Analysis Group, and manager for Piping Systems Engineering Section. Dr. Chang provided technical services to seven utilities through Chang Engineering Services from 1995 to 2002 before joining the USNRC. He is currently serving as chief of Engineering Review Branch 1 responsible for the safety technical review of license renewal applications. One of Dr. Chang's strength is the ability to combine complex NSSS engineering, systems, structural, mechanical and materials issues into a single project and develop resolutions that is understandable by stake holders. His technical expertise includes:

- License renewal safety audit of aging management programs (AMPs) and aging management reviews (AMRs)
- Time Limited Aging Analysis(TLAA)
- Leak-before-break demonstration
- ASME fatigue evaluation
- NSSS design and analysis
- Piping system analysis and operability demonstration
- Design and operating transient definition
- Seismic engineering
- Water hammer & thermal hydraulics
- High energy line breaks (HELB)
- Mechanical and flow induced vibration

Dr. Chang is a member of ASME Section XI, Special Working Group on Plant Life Extension and was Member of PVRC Committee on Dynamic Analysis and Testing and ASME B&PV Code Section III, Working Group on Piping. Dr. Chang authored approximately thirty (30) technical papers, presentations, and workshops to the nuclear industry worldwide in areas of fatigue analysis, seismic engineering, thermal hydraulic and structural dynamic analyses, piping design, aging management, and time limited aging analyses.

EXPERIENCE:

U.S. Nuclear Regulatory Commission

GG-13, Mechanical Engineer	March 2002 to April 2003
GG-14, Project Manager	April 2003 to February 2004
GG-15, Senior Mechanical Engineer	December 2003 to January 2006
Chief, Engineering Review Branch 1, DLR	January 2006 to Present

- Responsible for AMP and AMR safety audits of all LRAs.
- Metal fatigue emphasizing environmental assisted fatigue.
- Support foreign regulators in Safe Long-Term Operation.
- Davis-Besse RV head degradation and CRDM nozzle cracking.
- AP1000 Design Certification Document-review of piping design.
- Power uprate.
- North Anna Unit 2 RV head replacement program.
- LRA guidance documents updating.
- Technical Monitoring of contractor performance.
- Team leader of LRA AMP and AMR audits.
- Prepared, coordinated, and performed the "Orientation on Revised LRA Review process."

Consulting Services (Chang Engineering Services) May 1995 to March 2002

- Demolition of containment vessel (CV) and internal concrete.
- Reactor vessel buckling analysis and design of the protection system.
- Provide technical training to foreign utilities.
- Follow the development of USNRC license renewal activities.
- MOV weak link - owner's acceptance review.
- High energy line break (HELB) program owner.
- Equipment seismic qualification review.
- Thermal cycle monitoring program and fatigue issues.
- Technical lead - seismic qualification of SSCs.
- Condition report (CR) evaluation and corrective action development.
- Outage support in areas of piping, supports, mechanical, structure, and NSSS.
- Implementation of auxiliary piping snubber reduction program.
- Modified the USAR for SG snubber elimination, LBB, etc.
- Participated in 10 CFR 50.54(f) Recovery Program for several NPPs.
- Performed independent assessment of the design/licensing basis.
- Developed and reviewed licensing position papers and topical reports.

Altran Corporation November, 1993 to April, 1995

Senior consultant and project manager responsible for technical and business development of engineering services to nuclear utilities including the feasibility study of Steam Generator Snubber Elimination for Farley 1 and 2, and Millstone 3.

Westinghouse Electric Corporation February, 1973 to October, 1993

Dr. Chang held the following technical and managerial positions at Westinghouse:

Advisory Engineer
Technical manager responsible for the technical development for piping, support, and equipment qualification. Specific tasks included steam generator snubber elimination (including project development, marketing, feasibility study and implementation) for Wolf Creek/Callaway, thermal hydraulic methods, reactor vessel head vent analysis, NRCB 88-08 and 88-11 related issues, outage support, snubber reduction, equipment qualification, and licensing interface with the USNRC.

Manager, Business and Technology Development
The staff manager responsible for Structural mechanics Division's technology programs for piping, supports, structural analysis, pre-operational testing, and as-built reconciliation (IEB 79-14).

Manager, Piping Systems Engineering
The second level engineering manager responsible for the analysis and qualification of the reactor coolant loop and auxiliary system piping. This includes the management of three engineering groups responsible for the seismic and structural programs and providing licensing coordination for structural engineering. This Section has the overall responsibility of Class 1 piping fatigue, HELB and snubber reduction, including Maanshan 1 and 2, Shearon Harris, Diablo Canyon 1 and 2, V.C. Summer and Vogtle Unit 1.

Manager, Systems Structural Analysis
The group was responsible for the design and analysis of the primary coolant loop and ASME Class 1 piping systems. This includes the methods development, participation in the industrial committees, and implementation of thermal hydraulic and structural dynamics analyses for the pressurizer safety and relief valve discharge (NUREG 0737) and the feed-water check valve slam transients.

Fellow Engineer, Mechanics and Materials Technology
Lead technical staff responsible for the development of technology through leading two task forces. One task force assumed responsibility for the analysis of the pressurizer safety and relief lines for all loading including thermal hydraulic loads. The other was responsible for the development and completion of the generic fatigue analysis.

Principal Engineer, Mechanics and Materials Technology
Lead Engineer responsible for analysis of reactor coolant loop, Class 1 auxiliary lines and generic fatigue. Responsibility included interfacing with fluid systems group, reviewing existing system transients and defining new design transients for the qualification of standard Westinghouse 2, 3, and 4-loop PWR plants.

EDS Nuclear, INC January, 1970 to February, 1973

Dr. Chang was affiliated with EDS Nuclear, Inc. as a Project Engineer. In this capacity, he was responsible for the analysis of piping and components for Fast Flux Test Facility in accordance with ASME Boiler and Pressure Vessel Code Section III and high temperature Code cases. His responsibility also included the development of methodology for implementing Class 1 fatigue analysis and performing response spectra and time history seismic analysis for piping of various light water reactor plants.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of ENTERGY NUCLEAR VERMONT YANKEE, LLC AND ENTERGY NUCLEAR OPERATIONS, INC. (Vermont Yankee Nuclear Power Station)	))))))	Docket No. 50-271-LR

AFFIDAVIT OF KENNETH C. CHANG

I, Kenneth C. Chang, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

KENNETH C. CHANG

Executed at Gaithersburg, MD
this 12th day of May, 2008