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 JOHN R. FAIR
CONCERNING NEC CONTENTIONS 2A & 2B (Metal Fatigue)

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

A1. My name is John R. Fair. I am employed by the U.S. Nuclear Regulatory Commission ("NRC"), as a Senior Mechanical Engineer in the Office of Nuclear Reactor Regulation's ("NRR") Division of Engineering. 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. Although not specifically assigned to review the Vermont Yankee LRA, I advised my colleagues in the NRR Division of License Renewal ("DLR") in reviewing Entergy's metal fatigue submissions. I attended the January 8, 2008 public meeting between the Entergy and the NRC Staff as well as the February 7 and March 6, 2008 ACRS Meetings discussing the Vermont Yankee LRA. I was also involved in preparing Regulatory Issue Summary 2008-10 Fatigue Analysis of Nuclear Power Plant Components (April 11, 2008) (Exh. NEC-JH_23).

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by the Board in LBP-07-15, 66 NRC 261 (2007) and Order (Granting Motion to Amend Contention 2A) (April 24, 2008) (unpublished) ("April 24, 2008 Order"), NEC contends that Entergy's analyses of environmentally corrected cumulative usage factors (CUFens) is "flawed by numerous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. In Table 4.3-3 of Vermont Yankee's LRA, the CUFens for some of the listed components are greater than 1.0. (Staff Exh. 10) Explain why it is possible to "refine" predicted CUFens to less than 1.0?

A4. Many design calculations used conservative assumptions in the calculation of the CUF. The analyst did not refine the analysis if the calculated CUF was less than 1.0. 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 by refining the conservative assumptions used in the CUF calculation. For example, 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. The use of actual transients experienced by the plant would 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.

Q5. Did Entergy's January 2008 analysis (Exh. NEC-JH_34), 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?

A5. 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. Vermont Yankee is not a

new plant. Therefore, Entergy followed the correct staff guidance for its Vermont Yankee LRA.

Q6. 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.

A6. No. Dr. Hopenfeld asserts that environmental correction factors of 17 for carbon and low alloy steels and 12 for stainless steel should have been used to calculate the VY CUFs. Dr. Hopenfeld states that these values were bounding values in NUREG/CR-6909. Dr. Hopenfeld cited selected information taken from NUREG/CR-6909 without addressing the entire procedure. The recommended procedure for incorporating environmental effects into the fatigue equations is provided in Appendix A of NUREG/CR-6909. The procedure includes equations for calculating the environmental correction factor, Fen. The equations are a function of several variables: sulfur content of the component, temperature of the component, oxygen level in the fluid, and strain rate resulting from the applied loading. The procedure does not recommend that bounding values be used for the fatigue evaluations. In fact, for strains below the listed threshold levels, the procedure specifies a Fen value of 1.0. The Appendix A fatigue evaluation procedure also allows for the use of an average temperature during the transient to calculate the Fen. This can have a significant impact on the calculated Fen for transients that involve a large change in temperature. In addition, the Appendix A procedure contains new air fatigue curves that can be used in conjunction with the Fen values for calculating the CUF. The new air fatigue curves for carbon and low-alloy steels are less conservative than the current ASME fatigue curves

as illustrated in Figures A.1 and A.2. The austenitic stainless steel air curve is also less conservative than the ASME air curve in the low cycle region (less than 500 cycles) and more conservative in the high cycle region as illustrated in Figure A.3. Dr. Hopenfeld did not consider the impact of the new ANL air fatigue curves in his calculations.
Dr. Hopenfeld did not consider the entire fatigue evaluation procedure in NUREG/CR-6909, and the Fen values cited by Dr. Hopenfeld are not applicable to the design CUFs reported by Entergy.

Q7. Did the Staff find Entergy's CUF September and December 2008 analyses acceptable? If yes, explain why. If not, describe the Staff's concerns.

A7. The Staff did not find the results of the Entergy's analysis submitted by the September 17, 2007 (Staff Exh. 22) and the December 11, 2007 (Staff Exh. 8) 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 used one value of stress input instead of using six stress components as input to generate the Green's function. This process requires a great deal of judgment by the analyst to ensure that the simplification still provides a conservative result. The Staff was unable to make a judgment regarding the conservatism of the Entergy approach and requested that Entergy perform a confirmatory analysis of the reactor pressure vessel feedwater nozzle.

Q8. 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?

A8. As stated in the NUREG-1907 "Safety Evaluation Related to the License Renewal of Vermont Yankee Nuclear Power Station," Section 4.3.3.2. at 4-43 (Staff Exh. 1), 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 period of extended operation ("PEO"). However, since the feedwater ("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.

John R. Fair
Statement of Professional Qualifications

CURRENT POSITION:

Senior Mechanical Engineer: Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Rockville, MD

EDUCATION:

B.S. Mechanical Engineering, University of Maryland, 1972
M.S. Mechanical Engineering, University of Maryland, 1973
Advanced Graduate Studies, Engineering Mechanics, University of Maryland, 1975-76

SUMMARY:

Over 35 years of experience in the nuclear power industry, including 31 years at the Nuclear Regulatory Commission. Significant experience in the following areas:

- Development of staff technical position regarding fatigue evaluation of ASME Code components
- Member of ASME Code working groups on seismic design environmental fatigue
- Review of topics related to the mechanical design of ASME Code components
- Review of fatigue TLAA evaluations for several license renewal applications
- Design analysis of ASME Code and ANSI B31.1 piping systems

EXPERIENCE:

U.S. Nuclear Regulatory Commission, 1977 - Present

1990-present Senior Mechanical Engineer - Office of Nuclear Reactor Regulation

- Responsible for review and preparation of safety evaluation reports on topics related to the mechanical design of components at nuclear power plants
- The primary areas of review include ASME Code analyses of components (including the fatigue analyses of Class 1 components) and the seismic analysis of piping systems
- Participated as a member of ASME special working groups developing piping seismic design criteria and component fatigue design criteria
- Developed a Commission paper to address technical concerns related to the fatigue analysis of nuclear power plant components (SECY-95-245)
- Presented and defended NRC staff positions regarding mechanical design criteria at numerous ACRS and public meetings
- Developed NRC review criteria for license renewal fatigue evaluations

- Provided technical input for the update of licensing guidance documents related to the design of mechanical components, including development of a new SRP section to address piping design acceptance criteria

1987-1990 Senior Mechanical Engineer - Office of Special Projects

- Responsible for review and preparation of safety evaluation reports related to the restart and licensing of TVA nuclear power plants
- Lead several team inspections of TVA's mechanical and civil/structural design calculation reconstitution effort at the Sequoyah, Browns Ferry and Watts Bar nuclear power plants

1981-1987 Senior Mechanical Engineer - Office of Inspection and Enforcement

- Responsible for review of events reported at nuclear power plants in the area of mechanical engineering
- Developed bulletins and address safety concerns identified at operating nuclear power plants
- Provided technical support to regions and other NRC offices in the area of mechanical component and piping design

1978-1981 Senior Mechanical Engineer - Office of Nuclear Reactor Regulation

- Responsible for review and preparation of safety evaluation reports related to issues identified at operating nuclear power plants
- Developed the criteria for the evaluation of pipe supports using concrete expansion anchor bolts (NRC Bulletin 79-02)

1977-1978 Mechanical Engineer - Office of Standards Development

- Responsible for the development of rules and regulatory guides for nuclear power plants in the area of mechanical engineering

Bechtel Power Corporation 1974-1977, Senior Mechanical Engineer

- Responsible for ASME and ANSI B31.1 design and evaluation of nuclear power plant piping systems
- Developed a design guide for the routing and evaluation of small bore piping
- Performed as-built inspections of installed piping systems
- Resolved thermal expansion measurement discrepancies identified during the Hatch Nuclear Plant, Unit 1 startup thermal monitoring program

University of Maryland 1972-1973, Graduate Teaching Assistant

- Responsible for teaching fluid mechanics laboratory courses

MPR Associates, 1971-1973, Engineering Aide (part-time)

- Performed structural analysis of nuclear power plant components

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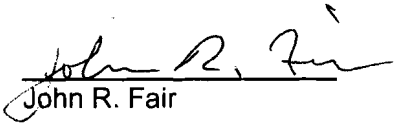
AFFIDAVIT OF JOHN R. FAIR

I, John R. Fair, 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.



John R. Fair

Executed at Rockville, MD
this 13th day of May, 2008