**RIV000034**
**Submitted: December 22, 2011**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
ATOMIC SAFETY AND LICENSING BOARD

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| In the Matter of | ) | |
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| Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3) | ) | Docket Nos. 50-247-LR and 50-286-LR |
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# Riverkeeper, Inc. provisionally designates the attached Testimony of Dr. Joram Hopenfeld dated December 20, 2011 as containing Confidential Proprietary Information Subject to Nondisclosure Agreement

## REDACTED, PUBLIC VERSION

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**PREFILED WRITTEN TESTIONY OF DR. JORAM HOPENFELD REGARDING RIVERKEEPER CONTENTION TC-1B – METAL FATIGUE**

On behalf of Riverkeeper, Inc. ("Riverkeeper"), Dr. Joram Hopenfeld submits the following testimony regarding Riverkeeper Contention TC-1B.

**Q. Please state your name and address.**

A. My name is Dr. Joram Hopenfeld and my business address is 1724 Yale Place, Rockville Maryland, 20850.

**Q. What is your educational and professional background?**

A. I have received the following degrees from the University of California in Los Angeles: a B.S. and M.S. in engineering, and a Ph.D. in mechanical engineering. I am an expert in the field relating to nuclear power plant aging management. I have 45 years of professional experience in the fields of nuclear safety regulation and licensing, design basis and severe accidents, thermal-hydraulics, material/environment interaction, corrosion, fatigue, radioactivity transport, industrial instrumentation, environmental monitoring, pressurized water reactor steam generator transient testing and accident analysis, design, and project management, including 18 years in the employ of the U.S. Nuclear Regulatory Commission ("NRC"). My education and professional experience are described in my *curriculum vita*, which is provided as Exhibit RIV000004.

**Q. What is the purpose of your testimony?**

A. The purpose of my testimony is to provide support for, and my views on, Riverkeeper's Contention TC-1B related to the aging effects of metal fatigue at Indian Point Generating Unit Nos. 2 and 3 during proposed 20-year extended operating terms. This contention was admitted

by the Atomic Safety & Licensing Board ("ASLB") on November 4, 2010.[1] Riverkeeper asserts that Entergy Nuclear Operations, Inc. ("Entergy"), the owner of Indian Point, has failed to demonstrate that metal fatigue of reactor components will be adequately managed during the proposed periods of extended operation at the plant as required by 10 C.F.R. § 54.21(c).

**Q. Have you prepared a report in support of your testimony?**

A. Yes, I prepared an expert report, provided as Exhibit RIV000035, which reflects my analysis and opinions.

**Q. What materials have you reviewed in preparation for your expert report and testimony?**

A. I have reviewed numerous documents in preparation of my expert report and testimony, including the following: all of the pleadings involving Riverkeeper Contention TC-1B, the relevant section of Entergy's License Renewal Application ("LRA"), Entergy's Amendment 2 to the LRA, dated January 22, 2008, Entergy's submission to the ASLB on August 10, 2010 entitled, "entitled "Notification of Entergy's Submittal Regarding Completion of Commitment 33 for Indian Point Units 2 and 3," "refined" Environmental Fatigue Evaluations for Indian Point Units 2 and 3 generated by Entergy's vendor Westinghouse in June 2010, relevant requests for additional information from the NRC and responses thereto by Entergy concerning metal fatigue, NRC Staff's Safety Evaluation Report, and Supplement 1 thereto, hundreds of documents identified by Entergy as relevant to Riverkeeper's metal fatigue contentions, numerous relevant NUREG reports, scientific and scholarly reports and articles, industry guidance documents and reports, and other documents generated by NRC, Entergy, industry groups, and scientific organizations. I have used such documents to inform me of the relevant facts and derive my conclusions.

[1]Riverkeeper initially filed Contention TC-1, followed by Amended Contention TC-1A, concerning metal fatigue, which were admitted by the ASLB on July 31, 2008. *See* In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, Memorandum and Order (Ruling on Petitions to Intervene and Requests for Hearing) (July 31, 2008), at 161-62. In response to new metal fatigue evaluations performed by Entergy, Riverkeeper and NYS jointly filed an amended contention, NYS-26B/RK-TC-1B, which the ASLB admitted as superseding the previous contentions. *See* In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, Ruling on Motions for Summary Disposition of NYS-26/26A/RiverkeeperTC-1/1A (Metal Fatigue of Reactor Components) and Motion for Leave to File New Contention NYS-26B/Riverkeeper TC-1B) (November 4, 2010) at 29.

A list of the particular documents that I reference in my expert report is included at the end of the report. Those references have been provided as Exhibits NYS000146A-146C, NYS00350-352, NYS000354-357, NYS000361-362, and RIV000036-058, in support of my testimony. To the best of my knowledge, these are true and accurate copies of each document that I referred to, used and/or relied upon in preparing my report and this testimony. In some cases where the document was extremely long and only a small portion is relevant to my testimony, an excerpt of the document is provided. If it is only an excerpt, that is noted on the cover of the Exhibit.

**Q. What conclusions have you reached about metal fatigue at Indian Point?**

A. In my professional judgment, and as I describe in more detail below and in my report, Entergy has failed to demonstrate that the serious aging mechanism of metal fatigue will be adequately managed throughout the proposed extended licensing terms at Indian Point. Though Entergy has proffered "refined" analyses purporting to demonstrate that certain components will remain within allowable acceptance criteria for metal fatigue, these analyses are flawed and inaccurate. Consideration of all relevant factors reveals that many components may become susceptible to metal fatigue and pose safety risks during the proposed periods of extended operation. In light of this eventuality, Entergy should have, but did not, expand the scope of components to be assessed for metal fatigue. Entergy has otherwise not provided sufficient details concerning an Aging Management Program ("AMP") to ensure that the degradation effect of metal fatigue would be adequately handled during the license renewal periods.

**Q. What is metal fatigue?**

A. Metal fatigue is an aging phenomenon that refers to when a structure or test specimen is subjected to repeated, "cyclic," loading during plant operation. Under such cyclic loading a crack will be initiated and the structure will fail under stresses that are substantially lower than those that cause failure under static loadings. Material composition, strain rate, temperature and local water chemistry are some of the factors that contribute to fatigue of metal parts. During each loading cycle, a certain fraction of the fatigue life of a component is used up depending on the magnitude of the applied stress. Eventually, after the number of allowable cycles, N, the structure will use all its fatigue life. The number of cycles actually experienced at any given

stress amplitude, n, divided by the corresponding number of allowable cycles, N, is called the usage fatigue factor ("CUF"). The maximum number of cycles that should be experienced by any structure or component should always result in a CUF that does not exceed 1.0, or unity. Section III of the American Society of Mechanical Engineers ("ASME") Code provides fatigue curves in air for various materials which specify the allowable number of cycles for a given stress intensity. The Code requires that the CUF at any given location be maintained below one.

**Q. What are the safety implications of metal fatigue?**

A. Fatigue may result in small leaks, which, if not detected in time, could lead to a pipe ruptures or other equipment failures. Fatigue may also create small cracks that propagate and cause a given component to malfunction or break up and form loose parts which can interfere with the safe operation of the plant. Such failures may have serious consequences to public health and safety. For example, if one of the feed water distribution nozzles (J tubes) were to fail from fatigue, pieces from the broken nozzle could be lodged between steam generator tubes, causing the tubes to rupture and leading to a potential core melt. Components which are susceptible to fatigue must, therefore, have a planned management program to ensure that the plant functions efficiently and safely.

**Q. Please explain how component susceptibility to metal fatigue is predicted.**

A. Crack growth rate for a given stress intensity can be predicted using an equation that includes empirical constants that were derived from laboratory tests in air under controlled conditions. However, this equation can predict crack growth reliably only as long as the equation is used under the conditions that were used to calibrate the empirical constants. In order to account for crack propagation in the actual reactor environment, the individual usage factor in air is multiplied by a corresponding environmental correction factor, "$F_{en}$." $F_{en}$ is the ratio of the fatigue life in air at room temperature to the fatigue life in water at the local temperature. The environmentally corrected CUF is expressed as $CUF_{en}$.

Laboratory tests were conducted under controlled conditions at the Argonne National Laboratory ("ANL"), to generate Fen factors. Because laboratory tests were not prototypic of the reactor environment, ANL provided a detailed discussion of the required adjustments to be made to the

laboratory data. The ANL equations describe Fen in terms of the temperature (T), dissolved oxygen (DO), sulfur content (S), and strain rate (e): Fen = f(T, DO, S, e). This equation cannot be used without knowing the value of the four variables at the surface of a component at any given time, during both steady state and transient operations.

**Q. Entergy's LRA contained the results of an analysis of the effects of environmentally assisted fatigue on certain reactor components during the proposed period of extended operation. What was the outcome of this $CUF_{en}$ analysis?**

A. LRA Tables 4.3-13 and 4.3-14 indicated that the $CUF_{en}$ of four risk significant reactor components would exceed unity during the period of extended operation. Due to these results, Entergy committed to performing a refined fatigue analysis in order to lower the predicted $CUF_{en}$ values to less than 1.0. The results of this "refined" environmentally assisted fatigue ("EAF") analysis, reported in revised LRA Tables 4.3-13 and 4.3-14 in August 2010, indicated that the $CUF_{en}$ values for all locations evaluated were below 1.0.

**Q. Have you reviewed Entergy's refined environmentally assisted fatigue analyses?**

A. Yes, I have reviewed two reports generated by Westinghouse, both dated June 2010, pertaining to Entergy's refined fatigue analyses for Unit 2 and Unit 3, as well as other documents identified by Entergy as relevant to the refined analyses.

**Q. What is your opinion regarding the validity of the results of Entergy's refined analyses?**

A. Based on my review of the June 2010 metal fatigue evaluations and related documents, I believe that the methodology employed to calculate Entergy's new $CUF_{en}$ values is highly suspect, and that the validity of the results is questionable. I believe that there is a wide margin of error due to many critical underlying assumptions that Entergy's refined analyses have failed to properly address. Entergy's new calculations have likely grossly under-predicted the $CUF_{en}$ values for the components evaluated for several reasons in particular: [redacted]

[redacted]

**Q. Please explain why the use of laboratory data must be adjusted in order to determine appropriate $F_{en}$ factors in the calculation of $CUF_{en}$.**

A. The equations for determining $F_{en}$ factors were derived from laboratory tests. Due the significant differences that exist between the laboratory and reactor environment, there are numerous uncertainties in applying the Fen equations to actual reactor components. In, NUREG/CR-6909, *Effect of LWR Coolant Environment on Fatigue Life of Reactor Materials*, ANL identifies numerous such uncertainties, which include material composition, component size and geometry, surface finish, loading history, strain rate, mean stress, water chemistry, dissolved oxygen levels, temperature, and flow rate.[2] Such uncertainties can have a significant affect upon fatigue life and ignoring them will result in underestimated $CUF_{en}$ calculations. For example, variations of temperature when temperature is below 150°C can reduce fatigue life by a factor of two; increased water conductivity due to the presence of trace anionic impurities in the coolant, which has already been documented to cause stress corrosion cracking at several nuclear plants, may decrease fatigue life of austenitic stainless steels; variation in sulfide morphology at a low strain rate may result in a difference by an order of magnitude in fatigue life; and surface temperature fluctuations and non-uniform temperature distributions during stratification can increase the potential for crack initiation and growth, thereby reducing fatigue life. So, to appropriately apply the Fen equations to actual reactor components, the user must consider all of the relevant uncertainties, and the results must be adjusted to account for the varying parameters.

In NUREG/CR-6909, *Effect of LWR Coolant Environment on Fatigue Life of Reactor Materials*, ANL specifies that appropriate bounding Fen values of 12 for stainless steel and 17 for carbon and low alloy steel to account for the numerous uncertainties in using the Fen equations.[3] These values are based on a review of data from different laboratory tests covering a wide range of parameters. These bounding Fen factors are not necessarily conservative, and it is reasonable to

[2] *See* NUREG/CR-6909 at 72.

[3] *See* NUREG/CR-6909 at iii, 3.

expect even higher Fen values in the actual reactor environment, especially for those components that experience stratified flows and thermal striping.

**Q. How did you reach the conclusion that Entergy's refined fatigue evaluations** [redacted]

**Q. Please explain how** [redacted]

A. [redacted] Entergy's assessment that no $CUF_{en}$ for the evaluated components exceeds unity remains unsubstantiated. The use of the bounding $F_{en}$ values recommended by ANL, [redacted] would increase the $CUF_{en}$ values beyond unity for eight of the components analyzed, as I calculated in my expert report.

**Q. Entergy has previously claimed that using the bounding Fen values recommended in NUREG/CR-6909 "would actually yield less conservative $CUF_{en}$ values, because the ASME Code design air curves for carbon steel and low-alloy steels contained in air [which is what Entergy considered] . . . are more conservative than the newer air curves in NUREG/CR-6909."[4] How would you respond to this assertion?**

A. Entergy's apparent position that its reliance on the ASME code curves automatically results in a more conservative $CUF_{en}$ value than would be reached if Entergy had individually evaluated all the uncertainties identified by ANL, is completely unfounded and simply wrong. While the current ASME code does incorporate a fatigue margin design of 2 on stress and 20 on cycles for carbon and low alloy steels in air, this conservatism was not intended to provide a margin of safety. The ASME code curves are valid in relation to fatigue life in air, and do not reflect the specific effects of the reactor coolant environment. In contrast, NUREG/CR-6909 provides guidance on appropriate $F_{en}$ factors to account for uncertainties inherent in $CUF_{en}$ due to the reactor, i.e. water, environment. The uncertainties discussed in NUREG/CR-6909 are not reflected anywhere in the current ASME code. The difference between the NUREG/CR-6909 air curve and the ASME code air curve is small in comparison to the many uncertainties associated with the actual reactor environment that must be considered to calculate an appropriate $F_{en}$.

**Q. Please explain how DO levels affect the determination of appropriate $F_{en}$ values for calculating $CUF_{en}$.**

A. One of the largest uncertainties in determining appropriate $F_{en}$ values is the concentration of dissolved oxygen ("DO") in the water at the surface of each component during the transient. The $F_{en}$ varies exponentially with the DO level, and is therefore sensitive to uncertainties in DO concentrations. The equations for determining $F_{en}$ were experimentally derived under conditions where the temperature and DO at the surface of the specimen were known. In contrast, in a reactor plant, the DO in many cases is unknown. This is in particularly true during startup and shutdown transients. During these transient the oxygen content at the surface of the component varies significantly due to oxygen incursions from external sources and because DO has a negative solubility coefficient in water. The level of DO, therefore, increases significantly

[4] Declaration of Nelson F. Azevedo in Support of Applicant's Motion for Summary Disposition of Contentions NYS-26/26A and Riverkeeper TC-1/1A (August 20, 2010), at ¶ 48.

during shutdown transients. During startup transients, DO will be at a maximum at the beginning of the transient and then decrease towards its steady state value. Oxygen excursions occur during heatups. Oxygen concentrations can vary with changes in temperature by more than an order of magnitude in comparison to oxygen levels during normal operating conditions. DO levels during transients are not measured at the surface of the reactor components at operating plants. As a result, the actual DO levels, and resulting $F_{en}$, are subject to uncertainties. For example, an uncertainty of five in DO levels at the surface of a given component could lead to under-predicting the $F_{en}$ by a factor of five at a minimum. This uncertainty can be accounted for by using bounding oxygen values during each transient. These values are different for different materials. The bounding oxygen value for stainless steel is the lowest value of oxygen during the transient, while the opposite is true for carbon and low alloy steels.

Due to the uncertainties related to DO levels during transients, ANL, as well as the Electric Power Research Institute ("EPRI"), specifically instruct that in determining $F_{en}$, DO can be conservatively taken as the maximum value for the transient.[5] ANL indicates that for carbon and low-alloy steels and austenitic stainless steel, values of 0.4 ppm and 0.05 ppm, respectively, can be used to perform a conservative evaluation.[6]

**Q. How did you reach the conclusion that Entergy's refined fatigue evaluations** [redacted]

A. [redacted]

[5] NUREG/CR-6583, ANL-97/18, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels* at 78; EPRI's Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, MRP-47 at 4-19.

[6] NUREG/CR-6909 at A-5.

**Q. Has Entergy and/or Westinghouse provided any explanation regarding the treatment of DO levels in the refined EAF analyses?**

A. [redacted]

[redacted]

**Q. Please explain how [redacted] affects the results of the refined EAF analysis.**

[redacted]

[7] Environmental Fatigue Evaluation for Indian Point Unit 2, WCAP-17199-P, Revision 0 (Westinghouse, June 2010), IPECPROP00056486, a p.5-24; Environmental Fatigue Evaluation for Indian Point Unit 3, WCAP-17200-P, Revision 0 (Westinghouse, June 2010), IPECPROP00056577, at p.5-24.

[8] NUREG/CR-6909 at 40.

[redacted] EPRI's Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, MRP-47, Rev. 1, calculates $F_{en}$ values as high as 130 at high DO levels, [redacted]

[redacted]

**Q. Can you please explain what a heat transfer coefficient is in relation to determining $CUF_{en}$.**

A. A heat transfer coefficient is a parameter used to determine the rate of heat transfer during transients, in order to calculate thermal stress and its impact on fatigue life. Heat transfer is a major factor in the determination of $CUF_{en}$ because it controls the cyclic thermal stresses during transients. Thermal stresses arise when there is a change in the local fluid temperature like during heat-ups or cool-downs or due to local mixing of hot and cold fluids. Failures result from either low stress at high cycle or high stress at low cycle. Stresses from thermal fatigue pose a serious risk of damage and have caused cracks in pipes and leakage at several nuclear reactors. This trend is expected to increase with time.

To calculate the rate at which heat is transferred to the reactor component surface during a transient (or in other words the temperature distribution of a component during a transient) in order to determine thermal stress, thermal-hydraulic computer codes are used with plant data to perform a heat transfer analysis. Heat transfer coefficients (h), water temperature, cycling period, and interface motion are all important inputs to this heat transfer analysis, and the consequent determination of the $CUF_{en}$ values. The $CUF_{en}$ value will vary greatly depending on the inputs used to perform the heat transfer analysis. The heat transfer coefficient h is the most important parameter in this regard. The heat transfer coefficient is commonly expressed in terms

[9] EPRI, MPR 47 at 4-22.

of geometry (G), fluid properties (P), flow rates (Q), and temperature difference between the coolant and the surface of the component ($\Delta T$): $h = f(G, P, Q, \Delta T)$. h is an experimental parameter and has been measured and determined for many different geometries, flow rates and rates of temperature change, and is known for well-defined, controlled conditions. However, the local flow at the surface of many reactor components during transients is not well defined and, therefore, approximations and assumptions are required in calculating the proper h for a given set of conditions. Such approximations lead to uncertainties in the $CUF_{en}$ because uncertainties in h directly impact the errors in the calculated stress. Typical variations in h could increase stress by a factor of 2. Increase in turbulence due to local discontinuities, and increase in the rate of the local temperature change increases h. Increase in h increases the corresponding stress and reduces fatigue life. For example, h along nozzles and bends varies in intensity because of the large variation in turbulence along their surface. This leads to non-uniform heat loads and introduces larger uncertainty in the stress distribution in comparison to simpler flow configurations. Stratified flow in the pressurizer surge line, is another example where non-uniform heat loads exist. Such flows occur when a warmer fluid flows on top of a cooler fluid, with a temperature difference between the two fluids as high as 350°F. Instabilities at the interface between the two fluids are known to produce high frequency temperature fluctuations on the surface of the component, with the potential for accelerating crack initiation and growth. Another factor that would lead to non-uniform stress distributions is preferential wall wear due to flow accelerated corrosion ("FAC") in low alloy steel components. For example, my review of Entergy ultrasonic examination reports indicates that in components where flow is not fully developed, component wall thickness can vary by more than 400% at Indian Point.

**Q. How did you reach the conclusion that Entergy's refined fatigue evaluations [redacted]**

A. [redacted] For other components, Entergy has failed to provide sufficient information to justify that heat transfer was appropriately considered, also casting doubt on the accuracy of Entergy's "refined" $CUF_{en}$ calculations.

**Q. Can you please explain how you reached the conclusion that Entergy** [redacted]

[10] *See* WCAP-14950, Westinghouse, Mitigation and Evaluation of Pressurizer Insurge/Outsurge Transients (February 1998), IPECPROP00059247.

[redacted] Stratified flow is a major source of error in calculating $CUF_{en}$. In fact, when the flow is stratified, an uncertainly of at least two in the heat transfer coefficient can be expected. While the accuracy of the heat transfer coefficient for natural convections under controlled conditions is on the order of +/-30%, for a stratified flow with an unstable mixing at the interface between moving hot and cold fluid, higher uncertainties can be expected. [redacted][11]

[redacted]

I also reviewed Entergy documentation related to the heat transfer analysis for the inlet and outlet reactor vessel nozzles.[12] The temperature distributions for these components were based on over 40-year old Combustion Engineering calculations. These calculations did not use a

[11] Kwang-Chu Kim et.al., Thermal fatigue estimation due to thermal stratification in the RCS branch line using one-way FSI scheme, Journal of Mechanical Science and Technology, 22 (2008) 2218-2227.

[12] Combustion Engineering, Inc., Nuclear Components Engineering Department, C.E. Contract No. 17765, "Analytical Report for Indian Point Reactor Vessel Unit No. 2," C.R. Crockrell and J. C. Lowry, CENC-1110 (April 22, 1968); Combustion Engineering, Inc., Nuclear Components Engineering Department, C.E. Contract No. 3366, "Analytical Report for Indian Point Reactor Vessel Unit No. 3," C.R. Crockrell and J. C. Lowry, CENC-1122 (June 1969).

finite element analysis and were based on a simplified 2-D model where the heat transfer coefficient was assumed constant along the flow and thermal properties were taken as independent of the temperature. The calculations were based on "as installed" nozzle dimensions. Due to nozzle geometry, the heat transfer is not uniform along the nozzle and can vary by 20%-30% depending on flow velocity location along the nozzle and flow direction. Such variation would result in axial thermal stress, which is not included in the 2-D analysis. Additionally, it is apparent that this analysis neglected to account for the observed fact that low alloy steels are subjected to wall thinning due to FAC. The reduction in wall thickness after 60 years of operation is expected to reduce fatigue life. Based on these considerations, it is apparent that Entergy's calculations used inappropriate heat transfer coefficients in the calculation of CUFen for the inlet and outlet reactor vessel nozzles

**Q. Can you please explain how you reached the conclusion that Entergy failed to justify that heat transfer was appropriately considered in the calculation of $CUF_{en}$ for certain components.**

A. For the other components evaluated in Entergy's "refined" $CUF_{en}$ analyses, meaning the RCS charging system, RCS injection nozzles, and RHR class 1 piping, Entergy did not provide the actual equations employed to determine the heat transfer coefficients. [redacted] Though Entergy has purported to provide sufficient information concerning the calculation of heat transfer coefficients for the rest of the components evaluated, Entergy's documentation and analyses do not specify the heat transfer coefficients used, or how h was determined, for these components. To assess the uncertainty of h, it is imperative to know the component geometry, the piping geometry upstream of the component, the flow velocities, and the corresponding expressions for h, none of which are specified by Entergy for the components at issue. [redacted] Given the uncertainties associated with determining the heat transfer coefficient, h, it is imperative to ensure that the methodology and assumptions employed were adequate to account for such uncertainties. Entergy's calculations have failed to do this. Without an understanding of the

values of h and the assumptions used to arrive at such values, the methodology employed by Entergy to re-calculate $CUF_{en}$ for the three relevant components remains questionable. Thus, based on the information relied upon by Entergy to support the "refined" evaluation, it is simply impossible to conclude that the new $CUF_{en}$ values for these other components are accurate.

**Q. How did you reach the conclusion that Entergy's refined fatigue evaluations [redacted]**

A. To evaluate the remaining fatigue life of a given component, it is necessary to consider the past as well as future loading during all transients. In other words the fatigue status of the component must be known from the time it was installed to the time it is removed from service. An accurate number of past and anticipated future transients is essential to the calculation of the $CUF_{en}$ values. [redacted]

To assess the severity of past transients, each transient that has occurred must be described to determine its contribution to the $CUF_{en}$. Yet considering past transients can be difficult because of the fact that degradation of some components was not included in the fatigue analysis when nuclear power plants were originally built. For example, the pressurizer surge line falls into this category. Historical records in such cases are incomplete and insufficient to provide adequate inputs for the number of heat-up and cool-down transients, stratification frequency, and system $\Delta T$. These parameters are required for thermal hydraulic and stress calculations. It is apparent from my review of Entergy's documents [redacted]

[redacted]

[redacted]

Justification of an appropriate number of future transients is critical in light of the fact that the useful life of most engineering components and structures follow a "bathtub curve" whereby component failure toward the end of operating life will occur at a very high frequency. Because Indian Point will be entering an extended period of operation, [redacted]

[redacted]

**Q. Please summarize your conclusions about Entergy's refined EAF analyses.**

A. Informed judgment is required in order to determine whether Entergy has adequately accounted for the significant degree of uncertainty associated with calculating $CUF_{en}$. In other words, it is critical to understand all of the underlying assumptions employed to arrive at the new EAF calculations. [redacted]

Entergy's failure to sufficiently account for all relevant parameters has resulted in predictions that are non-conservative. Given the large uncertainties in the input parameters and other assumptions used to generate the revised metal fatigue calculations, the methodology employed by Entergy suggests the likelihood of a wide margin of error, and the detrimental effects of the environment on fatigue strength of the components evaluated are likely grossly underestimated. In fact, many of the revised $CUF_{en}$ values remain very close to unity. The fact that Entergy's refined $CUF_{en}$ values are reported to the fifth significance figure (i.e. to a ten-thousandth of a decimal point), with several just a hair below unity, clearly shows that Entergy does not appreciate the uncertainties inherent in calculating the $CUF_{en}$ values. With a margin of error to account for varying input data and other undisclosed assumptions, such numbers could be considerably higher than the 1.0 regulatory threshold. In any event, without an error analysis, the claimed high degree of accuracy of the results remains questionable at best. Entergy repeatedly applies the term "bounding" to its analyses and results, implying that such results are conservative, and that no error analysis is necessary. However, there is no explanation to show that the results are in any respect bounding or conservative.

Based on all these considerations, it is my professional opinion that Entergy's refined $CUF_{en}$ calculations cannot be used as the basis for concluding that the aging effects of metal fatigue will be adequately managed at Indian Point during the PEO.

**Q. In your opinion, does Entergy otherwise have an adequate program for managing the effects of metal fatigue during the proposed period of extended operation?**

A. No. Entergy has chosen to rely upon its refined EAF analyses to demonstrate that metal fatigue will be managed throughout the period of extended operation. This is not adequate for several reasons.

First, for the foregoing reasons contained in this testimony, Entergy's new calculations do not demonstrate that the $CUF_{en}$ for the components evaluated will not exceed unity during the proposed extended licensing terms.

Second, in order for Entergy to have an effective AMP to monitor for metal fatigue, it must expand the scope of the fatigue analysis beyond simply representative components, to identify other components whose $CUF_{en}$ may be greater than 1.0. This is necessary because Entergy's initial findings presented in in Tables 4.3-13 and 4.3-14 of the April 2007 LRA indicated that the $CUF_{en}$ values for various components exceeded the regulatory threshold of 1.0, and under such circumstances, applicable regulatory and industry guidance required Entergy to identify additional reactor locations for potential high susceptibility to metal fatigue. Entergy's refined EAF evaluations did not expand the scope of components analyzed, but rather only assessed those locations identified in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (1995). In addition, the most recent version of NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, Revision 2 specifies that the sample set for fatigue calculations should include additional plant-specific component locations if they may be more limiting than those considered in NUREG/CR-6260. Entergy's fatigue analyses to date demonstrate that the components analyzed will likely exceed unity, and were, therefore, not necessarily the most limiting locations and bounding for the entire plant. In fact, NRC Staff has now conceded this point, as demonstrated by a request for information issued to Entergy seeking confirmation and justification that the locations selected for EAF

analyses consisted of the most limiting locations for the plant. However, in a supplement to the Safety Evaluation Report pertaining to the Indian Point license renewal proceeding, NRC Staff accepted only a vague commitment from Entergy to determine at some point in the future whether the locations assessed were the most limiting for Indian Point. This was not appropriate and Entergy's failure to actually confirm and justify bounding and limiting locations for Indian Point leaves Entergy's AMP insufficient, as it does not comply with the directive in the *GALL Report* or demonstrate that metal fatigue will be appropriately monitored, managed and corrected during the period of extended operation. Entergy must identify the locations that may be more limiting, and which will be the subject of $CUF_{en}$ calculations, *now*, and not just articulate a plan to determine such locations later. Such a determination is complex and would require consideration of thermal striping during stratification, an assessment of experience at other PWRs, and identifying and ranking of all components susceptible to thermal fatigue in terms of numerous key parameters, including the ratios of the local heat transfer coefficient and the local material conductivity, wall thickness, fluid temperature, $\Delta T$, dissolved oxygen levels, flow velocities, number of transients, magnitude and cycling frequency of surface temperatures and loads, and surface discontinuities, and flow discontinuities in each component. Entergy has not provided any information about how the analysis to determine the most limiting locations as Indian Point will be performed to allow for meaningfully comment upon the adequacy of the analysis.

Lastly, Entergy does not have an adequate AMP for metal fatigue because, in the absence of a reliable and complete assessment of $CUF_{en}$ values for susceptible plant components, Entergy has failed to define specific criteria concerning component inspection, monitoring, repair, and replacement. Entergy's plans for correcting metal fatigue related degradation depend initially upon calculating the vulnerability of plant components. Entergy intends to rely upon future $CUF_{en}$ calculations throughout the period of extended operation to manage metal fatigue. Entergy's calculations are meant to signify when components require inspection, monitoring, repair, or replacement, and, according to Entergy, will trigger when such actions are taken. As such, the validity of Entergy's monitoring program depends upon the accuracy of the calculations of the $CUF_{en}$. When a fatigue monitoring program is entirely based on a predictive analysis and not on actual measurements, and the analysis is flawed, the monitoring program is

invalid. Thus, Entergy's flawed methodology for calculating $CUF_{en}$, which Entergy ostensibly intends to employ throughout the period of extended operation, as well as Entergy's failure to expand the scope of components to be assessed, renders Entergy's vague commitments to inspect, repair, and replace affected locations insufficient to ensure proper management of metal fatigue during the proposed PEO. Without accurate metal fatigue calculations to properly guide Entergy's aging management efforts, Entergy has failed to define specific criteria to assure that susceptible components are inspected, monitored, repaired, or replaced in a timely manner. Once components with high $CUF_{en}$ values have been properly identified, Entergy must describe a fatigue management plan for each such component that should, at a minimum, rank components with respect to their consequences of failure, establish criteria for repair versus defect monitoring, and establish criteria for the frequency of the inspection (considering, for example defect size changes and uncertainties in the stress analysis and instrumentation), and allow for independent and impartial reviews of scope and frequency of inspection. Entergy has not done this.

**Q. Please summarize your opinions regarding whether or not Entergy has demonstrated that metal fatigue of reactor components will be adequately managed during the proposed periods of extended operation as required by 10 C.F.R. § 54.21(c).**

A. Based on my review of Entergy's submissions concerning metal fatigue and other relevant documents, in my professional opinion, Entergy has failed to demonstrate that the aging effects of metal fatigue will be adequately managed for the period of extended operation, and has, thus, failed to comply with 10 C.F.R. § 54.21(c) or regulatory guidance, including the *GALL Report*. Entergy's refined EAF $CUF_{en}$ analyses did not account for numerous critical factors and as a result have a wide margin of error and have likely underestimated the refined $CUF_{en}$ values. As many of the values derived by Entergy's new evaluation are very close to unity, the regulatory threshold, it is highly likely that if Entergy had accurately considered all relevant factors, many of the $CUF_{en}$ values would, in actuality, exceed 1.0. Entergy also improperly failed confirm that the components evaluated represent the most limiting locations providing a bounding analysis, and failed to expand the scope of components to be subject to a $CUF_{en}$ analysis accordingly. And lastly, in light of the insufficiency of Entergy's CUFen analyses, Entergy has otherwise failed to provide any details concerning the inspection, monitoring, repair, and replacement to ensure that the degradation effects of metal fatigue would be sufficiently handled.

**Q. Does this conclude your initial testimony regarding Riverkeeper Contention TC-1B?**

A. Yes.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
ATOMIC SAFETY AND LICENSING BOARD

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| In the Matter of | ) | |
| | ) | |
| Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3) | ) | Docket Nos. 50-247-LR and 50-286-LR |

## DECLARATION OF DR. JORAM HOPENFELD

I, Joram Hopenfeld, 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.

Executed in Accord with 10 C.F.R. § 2.304(d)

Joram Hopenfeld

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