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

In the Matter of	Docket Nos. 50-247-LR and 50-286-LR
Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3)	June 19, 2012

PREFILED WRITTEN TESTIMONY OF DR. JORAM HOPENFELD REGARDING CONTENTION NYS-38/RK-TC-5

On behalf of Riverkeeper, Inc. (“Riverkeeper”), Dr. Joram Hopenfeld submits the following testimony regarding the State of New York and Riverkeeper’s Joint Contention NYS-38/RK-TC-5.

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. Please describe your educational and professional background?

A. I have received a B.S. and M.S. in engineering, and a Ph.D. in mechanical engineering from the University of California in Los Angeles. 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, erosion, fatigue, cavitation (i.e. fatigue induced metal degradation), fouling, radioactivity transport, industrial instrumentation, environmental monitoring, pressurized water reactor (“PWR”) steam generator (“SG”) transient testing and accident analysis, design, and project management. My *curriculum vitae*, which has previously been provided in this proceeding as Exhibit RIV000004, fully describes my education, professional experience, and publications.

Q. What is the purpose of your testimony?

A. I was retained by Riverkeeper as an expert witness in the proceedings concerning the application by Entergy Nuclear Operations, Inc. ("Entergy") for the renewal of two separate operating licenses for the nuclear power generating facilities located at Indian Point on the east bank of the Hudson River in the Village of Buchanan, Westchester County, New York, for twenty years beyond their current expiration dates. The purpose of this testimony is to provide support for, and my views on Contention NYS-38/RK-5, jointly filed by Riverkeeper and the State of New York, concerning Entergy's failure to demonstrate that it has programs to effectively manage the aging of several critical components or systems during the proposed 20-year extended operating terms. Contention NYS-38/RK-5, which was admitted by the Atomic Safety & Licensing Board ("ASLB") on November 20, 2011, identified three aging management programs ("AMP") at Indian Point that Entergy has failed to demonstrate meet NRC regulations because they rely upon commitments to take future action: Entergy's programs for managing (1) the fatigue of metal components, (2) primary water stress corrosion cracking for the steam generator divider plates, and (3) reactor vessel internals.[1] My testimony specifically addresses the first of these programs, that is, Entergy's failure 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. Please describe your professional experience specifically as it relates to metal fatigue.

A. My education, experience, extensive knowledge, and public recognition make me well qualified to provide opinions and testimony related to the material degradation phenomenon known as "metal fatigue," that is, the fatigue or "cyclic stress" of metal parts due to repeated stresses during plant operation. Understanding fatigue analyses requires knowledge of temperature distributions and oxygen concentrations during plant transient and demands intimate knowledge of heat and mass transfer. As outlined in my *curriculum vitae*, my education, experience, and publications have afforded me with relevant expertise.

[1] In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-0247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, State of New York and Riverkeeper's New Joint Contention NYS-38/RK-TC-5 (Sept. 30, 2011), ADAMS Accession No. ML11273A196.

The major fields of study I pursued to obtain my Doctorate degree were heat transfer and mass transfer, fluid dynamics, and electrochemistry. This educational background qualifies me to opine on issues related to metal fatigue because a major element in fatigue analysis is water chemistry and mass transfer since such issues are related to the calculations of environmental correction factors, or "F_{en}." The numerous peer reviewed articles I have published, as listed in my *curriculum vitae*, include a professional article related to such issues.[2]

While employed at the Atomic Energy Commission ("AEC") and Energy Research and Development Administration ("ERDA") from 1971 to 1977, I managed a program involving various matters, including the following: experimental and numerical modeling of flow mixing and heat transfer in fuel assemblies and heat exchangers; stratified flow/thermal striping and fatigue analysis; cavitation (a fatigue induced metal degradation) and corrosion in water and in sodium; natural circulation; jet mixing; and effects of the leak environment on fatigue crack growth in sodium. On this last topic, I published a report.[3] Notably, the effects of stratified flow and thermal striping on fatigue was recognized at the AEC years before Westinghouse recognized that it could also exist in PWRs, and now stratification is a major element in fatigue analysis.

For 18 years in the employ of the U.S. Nuclear Regulatory Commission ("NRC"), I worked on assessing and resolving issues relating to PWR SGs including conducting extensive studies on tube degradation (via fatigue, stress corrosion cracking, wall thinning, and denting) and their consequences in PWR SGs. Knowledge of crack formation and detection is directly related to fatigue analysis and fatigue management. In addition, SG components under steady or transient conditions are also related to fatigue analysis. Notably, in the mid 1990s, I formulated and raised new concerns, designated a Differing Professional Opinion ("DPO"), about tube cracking, crack

[2] *See Curriculum Vitae* of Joram Hopenfeld, at p.4 (citing "Experience and Modeling of Radioactivity Transport Following Steam Generator Tube Rupture," Nuclear Safety, 26, 286, 1985).

[3] *See* Hopenfeld, et al., Small Sodium to Gas Leak Behavior in relation to LMFBR Leak Detection (International Conference on Liquid Metal Technology, May 3-6, 1976), http://www.osti.gov/bridge/servlets/purl/7252195-KSuoLF/7252195.pdf (Exhibit RIV000103). I also was the U.S. representative in attendance at the International Conference on Cavitation in Fast Breeder Reactors, where I presented on experimental investigation of cavitation inception in a flowing sodium environment.

detection, and safety consequences following certain transients (such as steam line breaks, tube ruptures, and station blackouts), and, following hearings in 2000, the Advisory Committee on Reactor Safeguards largely agreed with the issues raised in the DPO and the NRC initiated a decade long program to address such concerns.[4]

While at NRC, I also managed an international multimillion dollar cooperative program involving NRC, Westinghouse, the United Kingdom and EPRI, called the MB2, for the testing of a full size section of Westinghouse Model F steam generator under steady state and operational transients, the data from which was extensively used to validate computer codes and analytical models. Understanding the temperature response of components under transients is crucial to understanding fatigue analyses.

Over the past several decades, I have been a sought after source of expertise about tube degradation. For example, in the early 1990s, I testified in Congress in connection with material degradation at the Trojan Nuclear Power Plant; a major law firm representing Indian Point retained me as a consultant in connection with the tube accident that occurred at Indian Point; and most recently, I was extensively sought after for an expert opinion by the media in relation to the discovery of severe tube degradation at the San Onofre Nuclear Generating Station in January 2012.[5]

Q. What materials have you reviewed in preparation for your testimony?

A. I have reviewed numerous documents in preparation of my testimony, including the following: Entergy's License Renewal Application ("LRA") concerning the Indian Point nuclear power plant, submitted to the NRC on or about April 30, 2007; Entergy's Amendment 2 to the LRA, dated January 22, 2008, which contained information regarding Entergy's aging

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[4] Memorandum from S. Collins (NRR) to W. Travers (EDO), Re: Steam Generator Action Plan Revision to Address Differing Professional Opinion on Steam Generator Tube Integrity (WITS ITEM 200100026), May 11, 2011, http://pbadupws.nrc.gov/docs/ML0113/ML011300073.pdf , ADAMS Accession No. ML011300073 (Exhibit RIV000104); *see also* NUREG-1740, Voltage-Based Alternative Repair Criteria, A Report to the Advisory Committee on Reactor Safeguards by the Ad Hoc Subcommittee on a Differing Professional Opinion (March/Feb. 2001), at page 5http://pbadupws.nrc.gov/docs/ML0107/ML010750315.pdf, ADAMS Accession No. ML010750315 (Exhibit RIV000105).

[5] *See, e.g.*, Associated Press, Nuke inspectors focus on 'unusual' wear on tubes, Fox News.com, February 3, 2012, http://www.foxnews.com/us/2012/02/03/nuke-inspectors-focus-on-unusual-wear-on-tubes/ (Exhibit RIV000106).

management program for addressing metal fatigue; all of the pleadings involving Riverkeeper Contentions TC-1, TC-1A, TC-1B, and TC-5; Entergy's submission to the ASLB on August 10, 2010 entitled, "Notification of Entergy's Submittal Regarding Completion of Commitment 33 for Indian Point Units 2 and 3," NL-10-082; Entergy's "refined" Environmental Fatigue Evaluations for Indian Point Units 2 and 3 generated by Entergy's vendor Westinghouse in June 2010; NRC Staff's "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Numbers 2 and 3, License Renewal Application," dated February 10, 2011; Entergy's "Response to Request for Additional Information (RAI), Aging Management Programs, Indian Point Nuclear Generating Unit Nos. 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64," NL-11-032, dated March 28, 2011; NRC Staff's "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Supplement 1" dated August 2011 ("SER Supplement 1"); and Entergy's Letter to the ASLB dated May 15, 2012 pertaining to the timing additional metal fatigue evaluations to be performed by Entergy. In addition, I have reviewed hundreds of documents identified by Entergy as relevant to Riverkeeper's technical safety 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 the above-referenced documents to inform me of the relevant facts and derive my conclusions.

Numerous documents I have relied upon in forming the opinions contained in this testimony have been previously submitted in this proceeding in support of Contention NYS-26B/RK-TC-1B, as follows: Exhibits RIV000036-058, NYS00146A-146C, NYS00147A-147D, NYS000160, NYS000161, NYS000195, NYS000325, NYS00326A-326F, NYS000346, NYS000349-352, NYS000354-358B, NYS000361-369B. In addition, I have relied upon certain additional documents in forming the opinions contained in this testimony; these documents are provided in support of Contention NYS38/RK-5 as Exhibits RIV000103 to RIV000106 and NYS000395. To the best of my knowledge, these are all true and accurate copies of each document that I used and/or relied upon in preparing this testimony.

Q. What is metal fatigue?

A. As I explained at length in written testimony and a report submitted in support of Riverkeeper Contention RK-TC-1B earlier in this proceeding, 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 which a crack will initiate and the structure will fail under stresses that are substantially lower than those that cause failure under static loadings.[6] Material composition, strain rate, temperature and local water chemistry are some of the factors that contribute to fatigue of metal parts.[7] 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, and eventually, after the number of allowable cycles, N, the structure will use all its fatigue life.[8] 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").[9] 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.[10] The ASME Code requires that the CUF at any given location be maintained below one.[11]

Q. Does metal fatigue have safety implications?

A. Yes. As I discussed in previous submittals support of Riverkeeper Contention RK-TC-1B, metal fatigue may result in small leaks or cracks that could lead to pipe ruptures and/or other equipment malfunctions.[12] Such failures can interfere with the safe operation of the plant and have serious consequences to public health and safety.[13]

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[6] RIV000034 at 4:23-21, 5:1-6; RIV000035 at pp.1-3.

[7] RIV000034 at 4:23-21, 5:1-6; RIV000035 at pp.1-3.

[8] RIV000034 at 4:23-21, 5:1-6; RIV000035 at pp.1-3.

[9] RIV000034 at 4:23-21, 5:1-6; RIV000035 at pp.1-3.

[10] RIV000034 at 4:23-21, 5:1-6; RIV000035 at pp.1-3.

[11] RIV000034 at 4:23-21, 5:1-6; RIV000035 at pp.1-3.

[12] RIV000034 at 5:8-17; RIV000035 at p.3.

[13] RIV000034 at 5:8-17; RIV000035 at p.3.

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

A. Once again, as I explained in previous submittals in support of Riverkeeper Contention RK-TC-1B, crack growth rate for a given stress intensity can be predicted by multiplying an individual usage factor by a corresponding environmental correction factor, or "F_{en}," to account for the actual reactor environment.[14] F_{en} is the ratio of the fatigue life in air at room temperature to the fatigue life in water at the local temperature, and the environmentally corrected CUF is expressed as CUF_{en}.[15] Argonne National Laboratory ("ANL") has developed equations for determining F_{en} factors in terms of temperature (T), dissolved oxygen (DO), sulfur content (S), and strain rate (e): $F_{en} = f(T, DO, S, e)$.[16] ANL conducted laboratory tests under controlled conditions to generate F_{en} factors, and described required adjustments to be made to the laboratory data to account for the actual reactor environment.[17]

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. Entergy's LRA included the results of assessment of the effect of the reactor water environment on fatigue life for a sample of six components prescribed in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (1995).[18] LRA Tables 4.3-13 and 4.3-14 indicated that the CUF_{en} of four of these risk significant reactor components would exceed unity during the period of extended operation.

Q. Did Entergy undertake any steps in response to these findings?

A. To purportedly demonstrate that metal fatigue will be managed throughout the period of extended operation in light of these findings, Entergy committed to performing a refined fatigue analysis, in relation to the same components, in order to lower the predicted CUF_{en} values to less than 1.0. The results of this "refined" environmentally assisted fatigue ("EAF") analysis,

[14] RIV000034 at 5:19-31, 6:1-4; RIV000035 at pp.1-3.

[15] RIV000034 at 5:19-31, 6:1-4; RIV000035 at pp.1-3.

[16] RIV000034 at 5:19-31, 6:1-4; RIV000035 at pp.1-3.

[17] RIV000034 at 5:19-31, 6:1-4; RIV000035 at pp.1-3.

[18] NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (1995), at 4-1 (Exhibit NYS000355).

reported in revised LRA Tables 4.3-13 and 4.3-14 in August 2010, indicated that the CUF_{en} values for the locations evaluated, i.e. the same six sample locations prescribed in NUREG/CR-6260, were all below 1.0.[19] As explained at length in written testimony submitted in support of Riverkeeper Contention RK-TC-1B earlier in this proceeding, Entergy's "refined" analyses do not demonstrate that the CUF_{en} for the components evaluated will not exceed unity (1.0) during the proposed extended licensing terms, because Entergy employed a flawed methodology that failed to account for all relevant plant parameters, and which resulted in underestimated fatigue predictions.[20]

Q. In your opinion, were Entergy's "refined" fatigue analyses an adequate response to Entergy's initial finding that the CUF_{en} of four reactor components would exceed unity during the period of extended operation?

A. No, because Entergy did not expand the scope of the fatigue analysis beyond simply representative components, to identify other components whose CUF_{en} may be greater than 1.0. Since the CUF_{en} of several NUREG/CR-6260 components exceeded unity, and the non-environmentally corrected CUFs of other components were very close to unity, Entergy should have expanded their fatigue analysis. However, Entergy's refined EAF evaluations did not expand the scope of components analyzed, but rather only assessed those locations identified in NUREG/CR-6260.

According to regulatory and industry guidance, since the CUF_{en} for various components were initially found to exceed the regulatory threshold of 1.0, as presented in original LRA Tables 4.3-13 and 4.3-14, Entergy is required to identify and investigate additional reactor locations for potential high susceptibility to metal fatigue. In particular, according to industry guidance document, MRP-47, Revision 1, Electric Power Research Institute, *Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* (2005),

[19] NL-10-082,Completion of Commitment #33 Regarding the Fatigue Monitoring Program, Entergy Nuclear Operations, Inc., Indian Point Nuclear Generating Unit Nos. 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64 (August 9, 2010) (Exhibit NYS000352).

[20] RIV000034 at pp.6-20; RIV000035 at pp.4-21.

> plant-unique evaluations may show that some of the NUREG/CR-6260 [2] locations do not remain within allowable limits for 60 years of plant operation when environmental effects are considered. In this situation, *plant specific evaluations should expand the sampling of locations accordingly to include other locations where high usage factors might be a concern*.[21]

In addition, NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, ("*GALL Report*") Revision 1, specifies that "[f]or programs that monitor a sample of high fatigue usage locations, corrective actions include a review of *additional* affected reactor coolant pressure boundary locations," and that sample locations identified in NUREG/CR-6260 are simply the "minimum" set of components to analyze.[22] Furthermore, Revision 2 of the *GALL Report* (the most recent version of the report) specifies that the sample set for fatigue calculations that consider the effects of the reactor water environment "should include the locations identified in NUREG/CR-6260 *and additional plant-specific component locations* in the reactor coolant pressure boundary *if they may be more limiting than those considered in NUREG/CR-6260*."[23] 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.

Q. Are you aware of whether or not Entergy ever plans to expand the scope of its fatigue analysis to identify other components at Indian Point whose CUF_{en} may exceed unity during the proposed periods of extended operation?

A. My review of an NRC Staff RAI dated February 10, 2011, Entergy's response thereto dated March 28, 2011, and NRC Staff's SER Supplement 1, dated August 2011, indicates that Entergy may expand the scope of its fatigue analysis at some point in the future before entering the period of extended operation. In particular, NRC Staff's RAI requested that Entergy "[c]onfirm and justify that the locations selected for environmentally assisted fatigue analyses in LRA Tables 4.3-13 and 4.3-14 consist of the most limiting locations for the plant (beyond the generic components identified in the NUREG/CR-6260 guidance)" and "clarify which locations

[21] Exhibit NYS000350 at 3-4 (emphasis added).

[22] NUREG-1801, *GALL Report*, Rev. 1 § X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary, ¶¶ 5, 7 (emphasis added)) (Exhibit NYS00146A-146C).

[23] NUREG-1801, *Gall Report*, Rev. 2 § X.M1, Fatigue Monitoring, ¶ 1 (emphasis added) (Exhibit NYS00147A-147D).

require an environmentally-assisted fatigue analysis and the actions that will be taken for these additional locations."[24] Thus, NRC Staff has now conceded that there may be more limiting components, and that the CUF_{en} values in LRA Tables 4.3-13 and 4.3-14 may not be bounding.

In response, Entergy provided a vague commitment (Commitment 43), as follows:

> Entergy will review design basis ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 locations that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting locations for the Indian Point 2 and 3 plant configurations. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor coolant environment on fatigue usage.[25]

NRC Staff's SER Supplement 1 memorializes NRC Staff's acceptance of this commitment.[26]

My review of recent correspondence from Entergy to the ASLB dated May 15, 2012, indicates that "[t]he due date for completion of this Commitment [43] is prior to September 28, 2013 for Indian Point Energy Center ("IPEC") Unit 2 and prior to December 12, 2015 for IPEC Unit 3" and that "Entergy has determined that the initial screening review of design basis ASME Code Class 1 fatigue evaluations, as described in Commitment 43, to determine whether the NUREG/CR-6260 locations are the limiting locations for IPEC, is expected to be completed within approximately the next four months" and that "[i]f more limiting locations are identified, then Entergy will take further actions as necessary in accordance with Commitment 43."[27]

[24] U.S. NRC Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Numbers 2 and 3, License Renewal Application (February 10, 2011), at 13 (Exhibit RIV000057).

[25] Entergy Response to Request for Additional Information (RAI), Aging Management Programs Indian Point Nuclear Generating Unit Nos. 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64 (March 28, 2011), at p.26 of 27 (Exhibit RIV000058).

[26] Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286, NUREG-1930, Supplement 1 (August 2011), at 4-2 (Exhibit NYS000160).

[27] Correspondence from K. Sutton, P. Bessette (Counsel for Entergy) to (L. McDade, R. Wardwell, M. Kennedy (ASLB) (May 15, 2012) (Exhibit NYS000395).

Q. In your opinion, is Entergy's Commitment 43 adequate to demonstrate that Entergy has a program to effectively manage metal fatigue during the proposed periods of extended operation at Indian Point?

A. No, because Entergy has failed to identify the locations that may be more limiting, and which will be the subject of CUF_{en} calculations, *now* (that is, during the license renewal proceeding), and, has instead, only articulated a plan to determine all critical component locations later, at some point before entering the proposed extended periods of operation. This approach disallows meaningful review by NRC and the public of a critical safety issue.

Entergy's initial EAF analyses, as memorialized in Entergy's original LRA Tables 4.3-13 and 4.3-14, as well as Entergy's "refined" EAF analyses, demonstrate that the components analyzed will likely exceed unity, and that, therefore, those components were not necessarily the most limiting locations and bounding for the entire plant. Likewise, NRC Staff has now acknowledged that there may be more limiting components and that the CUF_{en} values in LRA Tables 4.3-13 and 4.3-14 may not be bounding. NRC Staff has rightly questioned Entergy's claim that LRA Tables 4.3-13 and 4.3-14 represent limiting conditions for the entire Indian Point plant. The most recent version of the *GALL Report* clearly specifies that fatigue calculations "should include . . . *additional plant-specific component locations* in the reactor coolant pressure boundary *if they may be more limiting than those considered in NUREG/CR-6260*."[28] Therefore, it was not appropriate for NRC Staff to accept Entergy's vague commitment to determine at some point in the future what additional locations must be analyzed. An actual analysis to determine the most limiting locations must be performed *before* a determination is made about license renewal.

Entergy's commitment to "review design basis ASME Code Class 1 fatigue evaluations" is not confirmation or justification that the locations selected for environmentally-assisted fatigue analyses in LRA Tables 4.3-13 and 4.3-14 consist of the most limiting locations for the plant. Entergy has, to date, not provided any analysis that would support a conclusion that the CUF_{en} values in LRA Tables 4.3-13 and 4.3-14 bound all other components at the plant. Entergy has

[28] NUREG-1801, *Gall Report*, Rev. 2 § X.M1, Fatigue Monitoring, ¶ 1 (emphasis added) (Exhibit NYS00147A-147D).

not conducted the analysis that would be required to confirm and justify that the components in LRA Tables 4.3-13 and 4.3-14 are bounding. Instead, Entergy has indicated that the process to be used to determine the most limiting locations for which CUF_{en} calculations, and the selection of the locations to be analyzed, will be disclosed in the future, apart from the license renewal review process. Entergy has failed to identify the limiting locations, or describe the methodology to be used to select such locations. This does not affirmatively demonstrate that "the effects of aging on the intended function(s) will be adequately managed for the period of extended operation" as required by 10 C.F.R. § 54.21(c)(1)(iii). Entergy has simply failed to provide sufficient information in order to assess whether Entergy's AMP for metal fatigue is adequate.

Q. Please describe what an analysis to determine the most limiting locations at Indian Point will require.

A. Determining the most limiting locations at Indian Point is not a clearly defined analysis. To the contrary, there are numerous considerations and factors.

The first step for determining the most limiting locations is component identification: selecting and listing all components that are susceptible to fatigue, including but not limited to nozzles, reducers, mixing tees and bends in feed water lines, surge lines, spray lines, and volume control system lines. The second step is component screening: the selected components must be screened and ranked with respect to their most vulnerable locations, considering parameters that are known to effect fatigue life. These include the ratios of the local heat transfer coefficient, the local material conductivity, wall thickness, fluid temperature, ΔT, dissolved oxygen levels, flow velocities, number of transients, magnitude and cycling frequency of surface temperatures, (thermal striping in stratified flows) and loads, and surface discontinuities and flow discontinuities in each component. Moreover, a determination of the most limiting locations should also include an assessment of actual experience at Indian Point as well as at other PWR plants. In addition, thermal striping during stratification should be generally considered as these effect fatigue life, and since the *GALL Report* requires that environmental effects be included in the calculations and does not exclude thermal striping from such requirements. Only after all

such considerations can a detailed numerical fatigue analysis be conducted in relation to the identified areas.

Additionally, an adequate assessment must also consider the synergistic aging effects of primary water stress corrosion cracking ("PWSCC") and thermal fatigue. For example, as discussed in Riverkeeper and the State of New York's joint contention, Entergy has acknowledged a problem with PWSCC for the nickel alloy or nickel-alloy clad SG divider plates exposed to reactor coolant.[29] As stated in Entergy's LRA, the CUF of record for the SG divider plate is already considerably high: 0.683 for Indian Point Unit 2 and 0.789 for Indian Point Unit 3.[30] These CUFs may exceed unity when they are corrected for the effects of PWSCC and the environment. Importantly, the effect of opening the divider plate on the natural convection flow through the steam generator following station blackouts ("SBO") and anticipated transients without scram ("ATWS") is an important consideration in this regard.

Despite the complexity involved in determining the most limiting locations at Indian Point, Entergy has failed to provide any information about how this analysis will be performed to allow for meaningfully comment upon the adequacy of the analysis. This 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.

Q. Do you have an opinion regarding what components Entergy should evaluate to determine whether they may be more limiting?

A. Yes. Since Entergy has not provided any indication to date about how expansive their search for more limiting locations will be, I have prepared the following table consisting of a list of components, as a sample of what Entergy's effort must consider at a minimum to determine whether Indian Point can operate safely during the proposed life extension periods:

[29] In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-0247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, State of New York and Riverkeeper's New Joint Contention NYS-38/RK-TC-5 (Sept. 30, 2011), ADAMS Accession No. ML11273A196; Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286, NUREG-1930, Supplement 1 (August 2011), at 3-18 to 3-19 (Exhibit NYS000160).

[30] LRA Tables 4.3-9, 4.3-10.

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Component	*Location*	*CUF on Record*[31]
IP2 Reactor Vessel	Inlet Nozzles at weldements	0.050
IP3 Reactor Vessel	Inlet Nozzles at weldements	0.049
IP2 Reactor Vessel	Outlet Nozzles at weldements	0.281
IP3 Reactor Vessel	Outlet Nozzles at weldements	0.259
IP2 Reactor Vessel Internals	Lower Core Support Plate	0.521
IP3 Reactor Vessel Internals	Lower Core Plate	0.237
IP2 Pressurizer	Spray Nozzle	0.996
IP3 Pressurizer	Spray Nozzle	0.974
IP2/IP3 Pressurizer	Lower Head at all weldements and at heater penetration	n/a
IP2 Steam Generator, Primary Side	Divider Plate	0.683
IP3 Steam Generator, Primary Side	Divider Plate	0.789
IP2 Steam Generator, Primary Side	Tube to tubesheet welds	0.809
IP2 Steam Generator, Secondary Side	Main feedwater nozzle	0.898
IP3 Steam Generator, Secondary Side	Main feedwater nozzle	1.00
IP2 Steam Generator, Secondary Side	Steam nozzle	0.212
IP3 Steam Generator, Secondary Side	Steam nozzle	0.023
IP2 Steam Generator, Secondary Side	Steam nozzle support ring	0.220

[31] *See* LRA Tables 4.3-3, 4.3-4, 4.3-5, 4.3-6, 4.3-7, 4.3-8, 4.3-9, 4.3-10.

IP3 Steam Generator, Secondary Side	Steam nozzle support ring	0.894
IP2 Steam Generator, Primary Side	Tubes	0.484
IP3 Steam Generator, Primary Side	Tubes	0.161
IP2/IP3 Reactor Pump	Outlet Nozzle	n/a
IP2/IP3 RHR	SI Nozzle	n/a
IP2/IP3 Mixing Tees	RHR system	n/a
IP2/IP3 Piping	Pressurizer Spray line	n/a
IP2/IP3 Piping	Unisolable branches connected to RCS piping	n/a

I derived this list based on the LRA, reactor experience, and extensive literature review.

Q. Are you familiar with Entergy's use of the computer model WESTEMS™ in relation to metal fatigue at Indian Point?

A. I am aware that Entergy has indicated that it relies upon the WESTEMS™ computer model in performing CUF_{en} calculations.

Q. Do you have an opinion regarding NRC Staff's approval of Entergy's commitment to provide explanations and justifications of any user intervention in future calculations using the WESTEMS™ computer model at some point in the future, prior to the expiration of the current Indian Point reactor operating licenses, but not prior to a decision on license renewal, as memorialized in NRC Staff's SER Supplement 1?[32]

A. Entergy must specify the criteria and assumptions upon which it will rely to modify the WESTEMS™ computer model for the calculation of CUF_{en} *prior* to a decision on license renewal. Varying criteria and assumptions to prospectively be employed may affect the validity and robustness of the analysis. Thus, without specifying the modifications to be made to the model, or the process for deciding when and how to have user intervention in the use of the

[32] Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286, NUREG-1930, Supplement 1 (August 2011), at 4-2 to 4-3 (Exhibit NYS000160).

model, Entergy has not demonstrated that the aging effects of metal fatigue will be adequately managed.

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. In light of NRC Staff's acceptance of vague commitments to perform necessary metal fatigue investigations, analyses, and justifications, in the future, Entergy has failed to demonstrate that the aging effects of metal fatigue will be adequately managed for the proposed periods 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 has failed to make the affirmative demonstration that it has a program to sufficiently monitor, manage, and correct metal fatigue-related degradation at Indian Point.

Q. Does this conclude your initial testimony regarding Contention NYS-38/RK-TC-5?

A. Yes.

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

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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