UNITED STATES
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

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In re:	Docket Nos. 50-247-LR; 50-286-LR
License Renewal Application Submitted by	ASLBP No. 07-858-03-LR-BD01
Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Operations, Inc.	DPR-26, DPR-64 February 12, 2015
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Riverkeeper, Inc. provisionally designates

the attached Declaration of Dr. Joram Hopenfeld

dated February 12, 2015 as containing

Confidential Proprietary Information

Subject to Nondisclosure Agreement

REDACTED, PUBLIC VERSION

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DECLARATION OF DR. JORAM HOPENFELD

Joram Hopenfeld, hereby declares under penalty of perjury that the following is true and correct:

1. I have been retained by Riverkeeper, Inc. ("Riverkeeper") as an expert witness in 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 an additional 20 years beyond the expiration of their 40-year operating licenses.

2. I submit this declaration in support of Riverkeeper and the State of New York's additional bases for previously admitted joint contention NYS-38/RK-TC-5, concerning Entergy's failure to demonstrate that it has an adequate aging management programs (AMP) for various reactor components, including one to manage the effects of metal fatigue on reactor vessel internal (RVI) plant components during the proposed periods of extended operation of Indian Point Units 2 and 3.

3. My professional and educational qualifications are described in the *curriculum vitae* appended hereto as Attachment 1. Briefly summarized, I am an expert in the field relating to nuclear power plant aging management. I am a mechanical engineer and hold a doctorate in mechanical engineering. I have 45 years of professional experience in the fields of thermal-hydraulics, material/environment interaction instrumentation, design, project management, and nuclear safety regulation, including 18 years in the employ of the U.S. Nuclear Regulatory Commission (NRC).

4. My extensive professional experience has afforded me with knowledge and expertise regarding 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. Of note, I was a technical consultant and expert witness for the New England Coalition in the Vermont

Yankee license renewal proceeding, where I testified at an adjudicatory hearing concerning metal fatigue.

5. I reviewed the April 30, 2007 License Renewal Application ("LRA") submitted by Entergy to renew the operating licenses for Indian Point Units 2 and 3, and assisted Riverkeeper with the preparation of Contention TC-1, which challenged Entergy's aging management plan for addressing metal fatigue at Indian Point during the proposed period of extended operation.

6. I reviewed Entergy's January 22, 2008 amendment to its original LRA, in which Entergy purported to provide additional information regarding its aging management program for addressing metal fatigue, and assisted Riverkeeper with the preparation of an amended contention (Riverkeeper Contention TC1-A) to articulate the ongoing deficiencies with Entergy's plan to deal with metal fatigue.

7. I reviewed Entergy's August 10, 2010 "Notification of Entergy's Submittal Regarding Completion of Commitment 33 for Indian Point Units 2 and 3," NL-10-082, as well as Entergy's revised metal fatigue evaluations dated June 2010, and assisted Riverkeeper with the preparation of a new and amended consolidated contention (New York State 26-B/Riverkeeper TC-1B), which articulated various deficiencies with Entergy's revised analysis, as well as Entergy's ongoing failure to demonstrate that the affects of metal fatigue would be adequately managed at the Indian Point facilities during the proposed period of extended operation.

8. I reviewed 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 thereto ("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," dated March 28, 2011), and 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"), all of which implicated and discussed metal fatigue, and assisted Riverkeeper with the preparation of joint contention NYS-38/RK-TC-5, concerning, among other things, Entergy's failure to demonstrate that it has an adequate program for managing the aging effects of metal fatigue at Indian Point during the proposed period of extended operation.

9. I have now reviewed NRC Staff's "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Supplement 2" dated November 2014 ("SSER2"), as well as various Entergy documents, including, though not limited to, Entergy's February 2012 "Revised Reactor Vessel Internals Program and Inspection Plan" (NL-12-037), and subsequent communications between Entergy and NRC Staff that developed and amended that plan. After a review of these documents, for the reasons explained more fully below, it is my professional opinion that Entergy has, to date, failed to demonstrate that the effects of metal fatigue on RVI components will be adequately managed at the Indian Point facilities during the proposed period of extended operation.

10. It is my understanding that Entergy intends to rely on its Fatigue Monitoring Program (FMP) to manage the effects of fatigue on RVI components at Indian Point during the period of extended operation, and that in order to account for the effects of the reactor coolant environment on the fatigue of RVI components as required under the ASME Code Section III, Subsections NG-2160 and NG-3121, Entergy has committed, in regulatory commitment 49, to recalculating CUF values for RVI components to include reactor coolant environmental effects. According to Entergy's implementation schedule for regulatory commitment 49, Entergy completed such calculations with respect to Indian Point Unit 2 before September 28, 2013, and will complete such calculations with respect to Indian Point Unit 3 before December 12, 2015. *See, e.g.*, SSER2 at 3-51 to 3-52.

11. Based on my review of the SSER2 and NUREG guidance documents, as well as my extensive professional experience, it is apparent that Entergy's commitment to recalculate the CUF values of RVI components to include reactor coolant environment effects involves a flawed methodology that fails to accurately and fully account for environmental effects, and, thus, assure that fatigue of such components will be adequately managed during the period of extended operation.

12. Entergy's commitment 49 involves calculating CUF values of the limiting RVI components to include reactor coolant environmental effects using factors, known as "F_{en}," that are provided in NUREG/CR-5704 and NUREG/CR-6909. *See* SSER2 at 3-52. However, Entergy's reliance on the F_{en} equations provided in these guidance documents results in a flawed and non-conservative analysis, for several reasons.

13. Entergy's reliance on NUREG/CR-5704 and NUREG/CR-6909 means that the CUF_{en} calculations conducted thus far and those that will be conducted in the future have and will continue to exclude the synergistic effects of neutron irradiation/radiation on metal fatigue of RVI components, thus resulting in non-conservative fatigue life predictions. In particular, the F_{en} equations contained in NUREG/CR-5704 and NUREG/CR-6909 were derived from tests on smooth specimens in the absence of neutron irradiation/radiation effects and under controlled water chemistry. Thus, F_{en} for the fatigue of reactor vessel internals must be corrected to encompass and account for effects of radiation. In fact, Dr. O.K. Chopra, the author of NUREG/CR-5704 and NUREG/CR-6909, has specifically stated that the effects of radiation flux, or fluence, were *not* included in the Argonne National Laboratory (ANL) studies contained in NUREG/CR-6909, and that therefore the user of the F_{en} equations contained therein must account for the effects of fluence on the F_{en} and resulting CUF_{en}. *See* Transcript of Advisory Committee on Reactor Safeguards Subcommittee on Materials, Metallurgy and Reactor Fuels (Dec. 6, 2006), ADAMS Accession No. ML12335A532, at 46-7 (Official Hearing Exhibit RIV000037). However, Entergy has simply used or committed to use the non-conservative equations contained in NUREG/CR-5704 and NUREG/CR-6909, and thus, has failed to, or will fail to, without any justification, include or account for the effects of radiation in their calculation of CUF_{en} for RVI components.[1] [redacted]

[1] There is currently a draft revision to NUREG/CR-6909 (March 2014) out for public comment (Revision 1). As a draft, this report ostensibly does not apply to this proceeding. In any event, the revised report, which memorializes the results of an extensive review that was conducted of the literature on the effects of radiation on fatigue, demonstrates that the Fen as specified by the NUREG/CR-6909 and NUREG/CR-5704 equations include

[redacted]

14. Neutron irradiation is known to affect fatigue life, and, thus, the failure to account for radiation effects in calculating CUF_{en} is significant and results in a highly non-conservative outcome that underestimates fatigue life. In particular, radiation is known to produce lattice defects in individual crystals, thereby causing metals to lose their normal ductility/malleability and reduce their resistance to fatigue crack propagation. Several studies have shown that fluence can have a significant effect on fatigue life reduction depending on the material, number of cycles, hold time, and fluence. For example, Korth and Harper reported a reduction of about one half in fatigue life for 308 stainless steel at cycles below 5000 cycles per second. For example, a reduction of this magnitude to the fatigue life of the lower core support plate, for which the CUF of record as reported in the LRA is 0.521 (*see* LRA at Table 4.3-5), would increase the CUF to 1.042. Thus, when appropriately accounting for water chemistry and the synergistic effects of fatigue and radiation, a given CUF may be higher when multiplied by a F_{en} factor that encompasses degradation effects caused by the reactor's operating environment.

15. The SSER2 does not mention nor does it discuss the fact that the CUF_{en}s that have been or would be determined per Entergy's Commitment 49 could not possibly be conservative because they will be based on non-prototypic tests that were conducted in a radiation free environment.

16. Neither Entergy nor NRC Staff has generated the data necessary regarding radiation effects to support their CUF_{en} analysis. To properly calculate CUF_{en}, it is necessary for Entergy to obtain fatigue life data that accounts for both water chemistry and radiation. Only with such data would it be possible to accurately determine the CUF_{en} values for RVI components.

17. Entergy's reliance on NUREG/CR-5704 and NUREG/CR-6909 for F_{en} equations to determine the CUF_{en} for RVI components is further flawed for the same reasons identified in relation Entergy's metal fatigue analysis of other plant components in Contentions NYS-26/RK-

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unquantifiable uncertainties because the effects of radiation were neglected. In this report, NRC specifically recognizes a data gap with respect to radiation. Such uncertainties in the Fen, together with other uncertainties discussed elsewhere in this declaration, invalidates the position that Entergy's fatigue life predictions are based on the maximum value of the Fen for each component and therefore the CUFens are conservative.

The March 2014 draft revised NUREG/CR-6909 report reasons that not accounting for the effects of radiation on metal fatigue is acceptable. However, such a position is not justified, and it is inappropriate to base CUFen calculations on the equations provided in NUREG/CR-6909 given the significant data gap that exists. The revised NUREG/CR-6909 recommends that the effects of radiation on the Fen be ignored because the existing data cannot be used to quantize these effects and that to do so, it would be required to obtain data on the effects of radiation on the Fen. The revised report contains no indication that such data will be obtained. Such an approach, i.e., to discount the effects of radiation on fatigue life simply because there is no available data is entirely arbitrary and non-scientific. Such an approach of ignoring potential safety problems because of a lack of data is also inconsistent with the ASME Code and common engineering practice. The ASME methodology requires that fatigue life be based on actual calibrated experimental data and conservative assumptions. It is the responsibility of the plant owner to obtain the data in order to be consistent with the ASME code.

TC-1 and NYS-38/RK-TC-5: the use of these equations fails to adequately account for various critical parameters affecting fatigue. For example, the use of the F_{en} equations from NUREG/CR-5704 and NUREG/CR-6909 improperly accounts for variation in dissolved oxygen (DO) during thermal transients. To correct for this deficiency NUREG/CR-5704 and NUREG/CR-6909 specify that maximum DO and temperature values should be used during the transients for conservative calculations. NUREG/CR-6909 provides additional guidance that has not been observed in Entergy's fatigue evaluations to date. The use of the F_{en} equations provided in NUREG/CR-5704 and NUREG/CR-6909 per se, without incorporating the specified instructions attached to these equations, results in significant underestimation of CUF_{en} values, especially for carbon steel, and does not assure that the effects of fatigue will be adequately managed during the PEO.

18. In addition to uncertainties in the CUF_{en} due to incorrect accounting for dissolved oxygen and radiation Entergy's methodology also introduces another major uncertainty by once again apparently relating the CUF_{en} to the CUF of record, i.e. $CUF_{en} = F_{en}$ x CUF (of record). The CUF of record is based on calculations that were valid when the plants were initially designed because all components were presumably in pristine conditions. However, after 40 years of exposure to a hostile light water reactor (LWR) environment most of the components have undergone a change in geometry and surface structure due to erosion/corrosion, stress corrosion, swelling, pitting, and cavitation. Such changes are known to affect fatigue life because they introduce local discontinuities that introduce high local stress concentrations. Such stress concentrations are known to significantly reduce fatigue life. The ASME fatigue curves are based on average stresses only. At least 100 years of experience has been accumulated to show that sharp surface discontinuities introduce high local stress concentrations where cracks are initiated. The ASME code requires that the average stress of a component be multiplied by the appropriate stress intensity factor. For example, the fatigue life of a component with lathe-formed surface is lower by a factor of 10 than if that surface was superfinished. *See* S. McKelvey & A. Fatemi, "Effect of Forging Surface on Fatigue Behavoir of Steels: A Literature Review (University of Toledo) (citing and discussing P. Fluck, 1951, "Influence of surface roughness on the fatigue life and scatter of test results of two steels," *Proceedings of American Society for Testing and Materials*, Vol. 51, pp. 584-592, Am. Soc. of Testing Materials, Philadelphia, PA). It is highly unlikely the stress concentration factors for many reactor components remain the same after 40 years. NUREG/CR-6909 also discusses the importance of surface finish. Effect of LWR Coolant Environment on the Fatigue Life of Reactor Materials, Final Report, NUREG/CR-6909, ANL-06/08 (February 2007), at § 4.1.6. Relying on the CUF of records without allowing for surface changes during service completely ignores the overwhelming importance of surface topography on fatigue life. It would be reasonable to multiply all CUFs of record with potential for surface change by a factor of 10 to account for real life effects of surface deterioration during service. Without such a correction there is no scientific basis for a claim that the CUF_{en}s are conservative.

19. In the absence of the appropriate consideration of radiation effects and other critical parameters, a CUF_{en} value that is less than 1.0 does not necessarily indicate that fatigue issues will not arise during the PEO. Importantly, a CUF_{en} of less than one does not necessarily

demonstrate that fatigue initiation is not expected during the life of the component.[2] The position that a component will not be subject to fatigue because the CUF_{en} for that component is less than one stands in contradiction of the conventional understanding of metal fatigue. NUREG/CR-6909 clearly indicates that cracks (flaws) are present throughout the fatigue life of a specimen under cyclic loads. The report explains that "fatigue life may be considered to constitute propagation of cracks from 10 to 3000 [micro meters] long." NUREG/CR-6909 at 7. Moreover, NUREG/CR-6909 as well as other studies give no indication that fatigue initiation does not exist when $CUF_{en} < 1.0$. In fact, the initiation stage of fatigue involves the growth of microscopic cracks. This stage is not characterized by any specific value of CUF_{en}. Of note, NRC studies at the Oak Ridge National Laboratory show that there is no correlation between a propagating crack size and the CUF as it approaches unity. For example, one study has indicated that when a component experiences a high level of cyclic stress corresponding to a usage factor of less than 1.0, "very small cracks can propagate to sizes that exceed acceptance criteria." G.T. Yahr, et al, *Case Study of the Propagation of a Small Flaw Under PWR Loading Conditions and Comparison with the ASME Code Design Life* (RIV000118), at 2. Notably, fatigue in conjunction with other factors may result in synergistic effects on plant components. For example, fatigue and neutron irradiation may act together and accelerate the individual effects. Thus, the failure to consider radiation effects makes any findings of a CUF_{en} value that is under 1.0 inaccurate and unreliable. The synergistic effect of metal fatigue and irradiation embrittlement are of particular concern during loss of coolant accident (LOCA) transients where very small cracks can propagate to failure under the high intensity LOCA loads.

20. [redacted] The fatigue failure of any one of these components could become accident initiators creating a new accident previously not evaluated. Fatigue fracture of RVI components is considered a serious issue where a fatigue fracture could not be expected to manifest itself in a small detectable leak allowing for a corrective action in a timely manner, unlike in other parts of the plant. NRC safety studies do not include an evaluation of the potential contribution of RVI failures by fatigue to core damage frequency (CDF). *See* Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150 [redacted] which fails to comply with the ASME Code and increases safety risk. Formation of loose parts in the reactor vessel is a realistic consequence of fatigue failures. To demonstrate to the public that CUF_{en}s that exceed one represent an acceptable risk, Entergy must demonstrate that fatigue failures within the reactor vessel will contribute less than 1E-05 to the core damage frequency, CDF. *See* Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission, NUREG/BR -0058, Rev.4.

[2] NRC Staff and Entergy have apparently taken this unconventional position with respect to fatigue analysis that has been conducted for the Indian Point Unit 2 lower support columns, stating in the SSER2 that the CUFen values calculated for the IP2 columns "are less than 1.0, demonstrating that fatigue initiation is not expected during the life of the plant." SSER2 at 3-43.

21. The failure to account for radiation effects, critical plant parameters, and surface finish effects fails to comply with NRC regulations and ASME Code Section III, Subsections NG-2160 and NG-3121 and renders Entergy's CUF_{en} calculations for RVI components non-conservative and inaccurate, and any such future calculations likewise non-conservative and inaccurate, and inadequate to assure that appropriate management of fatigue of such components. By disregarding radiation effects and surface roughness effects on the fatigue of RVI components, the predicted CUF_{en}s are non-conservative and do not comply with the ASME practice of using conservative data to account for uncertainties in the input data.

22. The accuracy of the calculated CUF_{en}s necessarily affects the reliability of the AMP for RVI components. Notably, the difference between $CUF_{en} > 1$ and $CUF_{en} < 1$ can have a major effect on the scope and cost of a fatigue monitoring plan. For CUF>1 Entergy would be required to perform more frequent inspections and increase the number of components for additional analysis as prescribed by NUREG-6260. Since the validity and scope of the FPM depends on the accuracy of the CUF_{en}s, the use of incorrect F_{en} factors would lead to an ineffective inspection regime, including the number of inspected components and the frequency of inspections. Without accurate CUF_{en} evaluations, the number and frequency of inspections will not minimize the risk of failures by metal fatigue. Non-conservative CUF_{en}s would increase the possibility that fatigue susceptible components would remain in service due to inadequately chosen inspection intervals.

23. In summary, Entergy's fatigue analyses for RVI components, as approved in the SSER2, is or will be based on non-conservative CUF_{en}s. [redacted] Thus, the CUF_{en}s represent a creation of numbers that cannot be used to predict fatigue life of reactor components. This renders Entergy's commitment to conduct fatigue analyses for RVI components, any such fatigue analyses conducted to date, as well as NRC Staff's approval of Entergy's commitment and future analyses, as memorialized in the SSER2, insufficient to assure that aging effects will be adequately managed.[3] Not only is the SSER2 and Entergy's analysis and AMP for managing fatigue of RVI components inadequate, it is also misleading, since they purport to employ proven engineering principles of CUF, F_{en}, and statistics, yet does so in a non-conservative, ineffective manner. The purpose of a Safety Evaluation Report (SER) is to provide the public in an open manner all the information about safety risks which are associated with changes in plant conditions. The SSER2 allows Entergy to ignore the effects of CUF_{en} uncertainties on plant safety during normal and accident conditions. Based on the inadequacy of Entergy's CUF_{en}

[3] Notably, the SSER2 was written in a manner that it would be difficult even for an expert in Facture Mechanics to understand many key statements and descriptions that led to the final conclusions that Entergy's AMP for RVI components meets the requirements of 10 CFR § 54.21. Since the SSER2 ostensibly fails to provide all meaningful and necessary references, it is not possible to fully verify the given information or separate facts from fiction. Independent verification is in particularly important in this case because it is apparent that the report was not independently reviewed.

analyses, evaluations of the uncertainties must be discussed and analyzed in terms of the change in core melt frequency.

24. In addition, in Commitment 49 Entergy proposed (and NRC Staff have agreed) that it need not complete the calculation of CUF_{en} values for IP3 RVI components until December 2015. *See* SSER2, at A-15. The schedule for Entergy's resolution of this issue extends beyond the time frame for the hearings in this ASLB proceeding and thus will not allow for a testing of the adequacy of the proposed resolution of these issues in this proceeding. That timeline likely will prevent Riverkeeper from playing any meaningful role in their development or resolution. Entergy's commitment to complete CUF_{en} evaluations for RVI components in the future, without review by the public, the ASLB, and/or NRC Staff, is unacceptable. By merely making a future commitment, it is not possible to fully determine the adequacy of the calculated CUF_{en} values and Entergy's AMP for RVI components. The analysis must be performed *before* a determination is made about license renewal. NRC Staff's acceptance of Entergy's commitment 49 in the SSER2 to conduct environmentally corrected metal fatigue evaluations for RVI components at some time in the future is not warranted or acceptable. By failing to undertake and complete the CUF_{en} analysis now and not in the future, Entergy has failed to demonstrate that metal fatigue of RVI components will be adequately managed during the PEO.

25. Furthermore, Entergy has not yet developed inspection acceptance criteria for baffle former bolts in either IP2 or IP3. *See* SSER2, at 3-20. Instead, it has agreed to develop a technical justification including acceptance criteria for baffle former bolts sometime prior to the first round of anticipated inspections, which might not occur until 2019 for IP2 and 2021 for IP3 and would be after an evidentiary hearing in this proceeding. *See* SSER2, at 3-20. The schedule for Entergy's resolution of this issue extends beyond the time frame for the hearings in this ASLB proceeding and thus will not allow for a testing of the adequacy of the proposed resolution of this issue in this proceeding. That timeline likely will prevent Riverkeeper from playing any meaningful role in their development or resolution.

26. In light of the foregoing, Entergy has failed to demonstrate that it has a program to monitor, manage, and correct metal fatigue related degradation sufficient to comply with 10 C.F.R. § 54.21(c), or the regulatory guidance of NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, or the MRP-227-A guidance developed by EPRI.

In accordance with 28 U.S.C. §1746, I declare under penalty of perjury that the foregoing is true and correct.

Executed on Feb 11, 2015

Joram Hopenfeld

Joram Hopenfeld