

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

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In the Matter of )	Docket Nos.
)	50-247-LR
Entergy Nuclear Operations, Inc. )	and 50-286-LR
(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.

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

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

4  
5 **Q. Please describe your educational and professional background?**

6 A. I have received a B.S. and M.S. in engineering, and a Ph.D. in mechanical engineering  
7 from the University of California in Los Angeles. I am an expert in the field relating to nuclear  
8 power plant aging management. I have 45 years of professional experience in the fields of  
9 nuclear safety regulation and licensing, design basis and severe accidents, thermal-hydraulics,  
10 material/environment interaction, corrosion, erosion, fatigue, cavitation (i.e. fatigue induced  
11 metal degradation), fouling, radioactivity transport, industrial instrumentation, environmental  
12 monitoring, pressurized water reactor (“PWR”) steam generator (“SG”) transient testing and  
13 accident analysis, design, and project management. My *curriculum vitae*, which has previously  
14 been provided in this proceeding as Exhibit RIV000004, fully describes my education,  
15 professional experience, and publications.

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

2 A. I was retained by Riverkeeper as an expert witness in the proceedings concerning the  
3 application by Entergy Nuclear Operations, Inc. (“Entergy”) for the renewal of two separate  
4 operating licenses for the nuclear power generating facilities located at Indian Point on the east  
5 bank of the Hudson River in the Village of Buchanan, Westchester County, New York, for  
6 twenty years beyond their current expiration dates. The purpose of this testimony is to provide  
7 support for, and my views on Contention NYS-38/RK-5, jointly filed by Riverkeeper and the  
8 State of New York, concerning Entergy’s failure to demonstrate that it has programs to  
9 effectively manage the aging of several critical components or systems during the proposed 20-  
10 year extended operating terms. Contention NYS-38/RK-5, which was admitted by the Atomic  
11 Safety & Licensing Board (“ASLB”) on November 20, 2011, identified three aging management  
12 programs (“AMP”) at Indian Point that Entergy has failed to demonstrate meet NRC regulations  
13 because they rely upon commitments to take future action: Entergy’s programs for managing (1)  
14 the fatigue of metal components, (2) primary water stress corrosion cracking for the steam  
15 generator divider plates, and (3) reactor vessel internals.<sup>1</sup> My testimony specifically addresses  
16 the first of these programs, that is, Entergy’s failure to demonstrate that metal fatigue of reactor  
17 components will be adequately managed during the proposed periods of extended operation at  
18 the plant as required by 10 C.F.R. § 54.21(c).

19

20 **Q. Please describe your professional experience specifically as it relates to metal**  
21 **fatigue.**

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

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<sup>1</sup> 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.

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2 The major fields of study I pursued to obtain my Doctorate degree were heat transfer and mass  
3 transfer, fluid dynamics, and electrochemistry. This educational background qualifies me to  
4 opine on issues related to metal fatigue because a major element in fatigue analysis is water  
5 chemistry and mass transfer since such issues are related to the calculations of environmental  
6 correction factors, or “F<sub>en</sub>.” The numerous peer reviewed articles I have published, as listed in  
7 my *curriculum vitae*, include a professional article related to such issues.<sup>2</sup>

8

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

19

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

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<sup>2</sup> 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).

<sup>3</sup> 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.

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

5  
6 While at NRC, I also managed an international multimillion dollar cooperative program  
7 involving NRC, Westinghouse, the United Kingdom and EPRI, called the MB2, for the testing of  
8 a full size section of Westinghouse Model F steam generator under steady state and operational  
9 transients, the data from which was extensively used to validate computer codes and analytical  
10 models. Understanding the temperature response of components under transients is crucial to  
11 understanding fatigue analyses.

12  
13 Over the past several decades, I have been a sought after source of expertise about tube  
14 degradation. For example, in the early 1990s, I testified in Congress in connection with material  
15 degradation at the Trojan Nuclear Power Plant; a major law firm representing Indian Point  
16 retained me as a consultant in connection with the tube accident that occurred at Indian Point;  
17 and most recently, I was extensively sought after for an expert opinion by the media in relation to  
18 the discovery of severe tube degradation at the San Onofre Nuclear Generating Station in  
19 January 2012.<sup>5</sup>

20

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

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

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<sup>4</sup> 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 5 <http://pbadupws.nrc.gov/docs/ML0107/ML010750315.pdf>, ADAMS Accession No. ML010750315 (Exhibit RIV000105).

<sup>5</sup> *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).

1 management program for addressing metal fatigue; all of the pleadings involving Riverkeeper  
2 Contentions TC-1, TC-1A, TC-1B, and TC-5; Entergy's submission to the ASLB on August 10,  
3 2010 entitled, "Notification of Entergy's Submittal Regarding Completion of Commitment 33  
4 for Indian Point Units 2 and 3," NL-10-082; Entergy's "refined" Environmental Fatigue  
5 Evaluations for Indian Point Units 2 and 3 generated by Entergy's vendor Westinghouse in June  
6 2010; NRC Staff's "Request for Additional Information for the Review of the Indian Point  
7 Nuclear Generating Unit Numbers 2 and 3, License Renewal Application," dated February 10,  
8 2011; Entergy's "Response to Request for Additional Information (RAI), Aging Management  
9 Programs, Indian Point Nuclear Generating Unit Nos. 2 & 3, Docket Nos. 50-247 and 50-286,  
10 License Nos. DPR-26 and DPR-64," NL-11-032, dated March 28, 2011; NRC Staff's "Safety  
11 Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos.  
12 2 and 3, Supplement 1" dated August 2011 ("SER Supplement 1"); and Entergy's Letter to the  
13 ASLB dated May 15, 2012 pertaining to the timing additional metal fatigue evaluations to be  
14 performed by Entergy. In addition, I have reviewed hundreds of documents identified by  
15 Entergy as relevant to Riverkeeper's technical safety contentions, numerous relevant NUREG  
16 reports, scientific and scholarly reports and articles, industry guidance documents and reports,  
17 and other documents generated by NRC, Entergy, industry groups, and scientific organizations.  
18 I have used the above-referenced documents to inform me of the relevant facts and derive my  
19 conclusions.

20  
21 Numerous documents I have relied upon in forming the opinions contained in this testimony  
22 have been previously submitted in this proceeding in support of Contention NYS-26B/RK-TC-  
23 1B, as follows: Exhibits RIV000036-058, NYS00146A-146C, NYS00147A-147D, NYS000160,  
24 NYS000161, NYS000195, NYS000325, NYS00326A-326F, NYS000346, NYS000349-352,  
25 NYS000354-358B, NYS000361-369B. In addition, I have relied upon certain additional  
26 documents in forming the opinions contained in this testimony; these documents are provided in  
27 support of Contention NYS38/RK-5 as Exhibits RIV000103 to RIV000106 and NYS000395. To  
28 the best of my knowledge, these are all true and accurate copies of each document that I used  
29 and/or relied upon in preparing this testimony.

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31

1 **Q. What is metal fatigue?**

2 A. As I explained at length in written testimony and a report submitted in support of  
3 Riverkeeper Contention RK-TC-1B earlier in this proceeding, metal fatigue is an aging  
4 phenomenon that refers to when a structure or test specimen is subjected to repeated, “cyclic,”  
5 loading during plant operation, under which a crack will initiate and the structure will fail under  
6 stresses that are substantially lower than those that cause failure under static loadings.<sup>6</sup> Material  
7 composition, strain rate, temperature and local water chemistry are some of the factors that  
8 contribute to fatigue of metal parts.<sup>7</sup> During each loading cycle, a certain fraction of the fatigue  
9 life of a component is used up depending on the magnitude of the applied stress, and eventually,  
10 after the number of allowable cycles, N, the structure will use all its fatigue life.<sup>8</sup> The number of  
11 cycles actually experienced at any given stress amplitude, n, divided by the corresponding  
12 number of allowable cycles, N, is called the usage fatigue factor (“CUF”).<sup>9</sup> The maximum  
13 number of cycles that should be experienced by any structure or component should always result  
14 in a CUF that does not exceed 1.0, or unity. Section III of the American Society of Mechanical  
15 Engineers (“ASME”) Code provides fatigue curves in air for various materials which specify the  
16 allowable number of cycles for a given stress intensity.<sup>10</sup> The ASME Code requires that the  
17 CUF at any given location be maintained below one.<sup>11</sup>

18

19 **Q. Does metal fatigue have safety implications?**

20 A. Yes. As I discussed in previous submittals support of Riverkeeper Contention RK-TC-  
21 1B, metal fatigue may result in small leaks or cracks that could lead to pipe ruptures and/or other  
22 equipment malfunctions.<sup>12</sup> Such failures can interfere with the safe operation of the plant and  
23 have serious consequences to public health and safety.<sup>13</sup>

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<sup>6</sup> RIV000034 at 4:23-21, 5:1-6; RIV000035 at pp.1-3.

<sup>7</sup> RIV000034 at 4:23-21, 5:1-6; RIV000035 at pp.1-3.

<sup>8</sup> RIV000034 at 4:23-21, 5:1-6; RIV000035 at pp.1-3.

<sup>9</sup> RIV000034 at 4:23-21, 5:1-6; RIV000035 at pp.1-3.

<sup>10</sup> RIV000034 at 4:23-21, 5:1-6; RIV000035 at pp.1-3.

<sup>11</sup> RIV000034 at 4:23-21, 5:1-6; RIV000035 at pp.1-3.

<sup>12</sup> RIV000034 at 5:8-17; RIV000035 at p.3.

<sup>13</sup> RIV000034 at 5:8-17; RIV000035 at p.3.

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

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

12

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

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

21

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

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

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<sup>14</sup> RIV000034 at 5:19-31, 6:1-4; RIV000035 at pp.1-3.

<sup>15</sup> RIV000034 at 5:19-31, 6:1-4; RIV000035 at pp.1-3.

<sup>16</sup> RIV000034 at 5:19-31, 6:1-4; RIV000035 at pp.1-3.

<sup>17</sup> RIV000034 at 5:19-31, 6:1-4; RIV000035 at pp.1-3.

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

1 reported in revised LRA Tables 4.3-13 and 4.3-14 in August 2010, indicated that the  $CUF_{en}$   
2 values for the locations evaluated, i.e. the same six sample locations prescribed in NUREG/CR-  
3 6260, were all below 1.0.<sup>19</sup> As explained at length in written testimony submitted in support of  
4 Riverkeeper Contention RK-TC-1B earlier in this proceeding, Entergy's "refined" analyses do  
5 not demonstrate that the  $CUF_{en}$  for the components evaluated will not exceed unity (1.0) during  
6 the proposed extended licensing terms, because Entergy employed a flawed methodology that  
7 failed to account for all relevant plant parameters, and which resulted in underestimated fatigue  
8 predictions.<sup>20</sup>

9  
10 **Q. In your opinion, were Entergy's "refined" fatigue analyses an adequate response to**  
11 **Entergy's initial finding that the  $CUF_{en}$  of four reactor components would exceed unity**  
12 **during the period of extended operation?**

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

20  
21 According to regulatory and industry guidance, since the  $CUF_{en}$  for various components were  
22 initially found to exceed the regulatory threshold of 1.0, as presented in original LRA Tables 4.3-  
23 13 and 4.3-14, Entergy is required to identify and investigate additional reactor locations for  
24 potential high susceptibility to metal fatigue. In particular, according to industry guidance  
25 document, MRP-47, Revision 1, Electric Power Research Institute, *Materials Reliability*  
26 *Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal*  
27 *Application* (2005),

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<sup>19</sup> 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).

<sup>20</sup> RIV000034 at pp.6-20; RIV000035 at pp.4-21.

1 plant-unique evaluations may show that some of the NUREG/CR-  
2 6260 [2] locations do not remain within allowable limits for 60  
3 years of plant operation when environmental effects are  
4 considered. In this situation, *plant specific evaluations should*  
5 *expand the sampling of locations accordingly to include other*  
6 *locations where high usage factors might be a concern.*<sup>21</sup>  
7

8 In addition, NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, (“*GALL Report*”)  
9 Revision 1, specifies that “[f]or programs that monitor a sample of high fatigue usage locations,  
10 corrective actions include a review of *additional* affected reactor coolant pressure boundary  
11 locations,” and that sample locations identified in NUREG/CR-6260 are simply the “minimum”  
12 set of components to analyze.<sup>22</sup> Furthermore, Revision 2 of the *GALL Report* (the most recent  
13 version of the report) specifies that the sample set for fatigue calculations that consider the  
14 effects of the reactor water environment “should include the locations identified in NUREG/CR-  
15 6260 and *additional plant-specific component locations* in the reactor coolant pressure boundary  
16 *if they may be more limiting than those considered in NUREG/CR-6260.*”<sup>23</sup> Entergy’s fatigue  
17 analyses to date demonstrate that the components analyzed will likely exceed unity, and were,  
18 therefore, not necessarily the most limiting locations and bounding for the entire plant.  
19

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

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

<sup>21</sup> Exhibit NYS000350 at 3-4 (emphasis added).

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

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

1 require an environmentally-assisted fatigue analysis and the actions that will be taken for these  
2 additional locations.”<sup>24</sup> Thus, NRC Staff has now conceded that there may be more limiting  
3 components, and that the  $CUF_{en}$  values in LRA Tables 4.3-13 and 4.3-14 may not be bounding.  
4

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

6           Entergy will review design basis ASME Code Class 1 fatigue  
7           evaluations to determine whether the NUREG/CR-6260 locations  
8           that have been evaluated for the effects of the reactor coolant  
9           environment on fatigue usage are the limiting locations for the  
10          Indian Point 2 and 3 plant configurations. If more limiting  
11          locations are identified, the most limiting location will be  
12          evaluated for the effects of the reactor coolant environment on  
13          fatigue usage.<sup>25</sup>  
14

15 NRC Staff’s SER Supplement 1 memorializes NRC Staff’s acceptance of this commitment.<sup>26</sup>  
16

17 My review of recent correspondence from Entergy to the ASLB dated May 15, 2012, indicates  
18 that “[t]he due date for completion of this Commitment [43] is prior to September 28, 2013 for  
19 Indian Point Energy Center (“IPEC”) Unit 2 and prior to December 12, 2015 for IPEC Unit 3”  
20 and that “Entergy has determined that the initial screening review of design basis ASME Code  
21 Class 1 fatigue evaluations, as described in Commitment 43, to determine whether the  
22 NUREG/CR-6260 locations are the limiting locations for IPEC, is expected to be completed  
23 within approximately the next four months” and that “[i]f more limiting locations are identified,  
24 then Entergy will take further actions as necessary in accordance with Commitment 43.”<sup>27</sup>  
25

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<sup>24</sup> 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).

<sup>25</sup> 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).

<sup>26</sup> 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).

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

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

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

9  
10 Entergy’s initial EAF analyses, as memorialized in Entergy’s original LRA Tables 4.3-13 and  
11 4.3-14, as well as Entergy’s “refined” EAF analyses, demonstrate that the components analyzed  
12 will likely exceed unity, and that, therefore, those components were not necessarily the most  
13 limiting locations and bounding for the entire plant. Likewise, NRC Staff has now  
14 acknowledged that there may be more limiting components and that the CUF<sub>en</sub> values in LRA  
15 Tables 4.3-13 and 4.3-14 may not be bounding. NRC Staff has rightly questioned Entergy’s  
16 claim that LRA Tables 4.3-13 and 4.3-14 represent limiting conditions for the entire Indian Point  
17 plant. The most recent version of the *GALL Report* clearly specifies that fatigue calculations  
18 “should include . . . *additional plant-specific component locations* in the reactor coolant pressure  
19 boundary *if they may be more limiting than those considered in NUREG/CR-6260.*”<sup>28</sup> Therefore,  
20 it was not appropriate for NRC Staff to accept Entergy’s vague commitment to determine at  
21 some point in the future what additional locations must be analyzed. An actual analysis to  
22 determine the most limiting locations must be performed *before* a determination is made about  
23 license renewal.

24  
25 Entergy’s commitment to “review design basis ASME Code Class 1 fatigue evaluations” is not  
26 confirmation or justification that the locations selected for environmentally-assisted fatigue  
27 analyses in LRA Tables 4.3-13 and 4.3-14 consist of the most limiting locations for the plant.  
28 Entergy has, to date, not provided any analysis that would support a conclusion that the CUF<sub>en</sub>  
29 values in LRA Tables 4.3-13 and 4.3-14 bound all other components at the plant. Entergy has

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

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

11

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

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

16

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

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

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

14  
15 Despite the complexity involved in determining the most limiting locations at Indian Point,  
16 Entergy has failed to provide any information about how this analysis will be performed to allow  
17 for meaningfully comment upon the adequacy of the analysis. This leaves Entergy’s AMP  
18 insufficient, as it does not comply with the directive in the *GALL Report* or demonstrate that  
19 metal fatigue will be appropriately monitored, managed and corrected during the period of  
20 extended operation.

21

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

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

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<sup>29</sup> 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).

<sup>30</sup> LRA Tables 4.3-9, 4.3-10.

1

<i>Component</i>	<i>Location</i>	<i>CUF on Record</i> <sup>31</sup>
IP2 Reactor Vessel	Inlet Nozzles at weldments	0.050
IP3 Reactor Vessel	Inlet Nozzles at weldments	0.049
IP2 Reactor Vessel	Outlet Nozzles at weldments	0.281
IP3 Reactor Vessel	Outlet Nozzles at weldments	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 weldments 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

<sup>31</sup> 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

1

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

3

4 **Q. Are you familiar with Entergy's use of the computer model WESTEMS™ in**  
5 **relation to metal fatigue at Indian Point?**6 A. I am aware that Entergy has indicated that it relies upon the WESTEMS™ computer  
7 model in performing  $CUF_{en}$  calculations.

8

9 **Q. Do you have an opinion regarding NRC Staff's approval of Entergy's commitment**  
10 **to provide explanations and justifications of any user intervention in future calculations**  
11 **using the WESTEMS™ computer model at some point in the future, prior to the expiration**  
12 **of the current Indian Point reactor operating licenses, but not prior to a decision on license**  
13 **renewal, as memorialized in NRC Staff's SER Supplement 1?**<sup>32</sup>14 A. Entergy must specify the criteria and assumptions upon which it will rely to modify the  
15 WESTEMS™ computer model for the calculation of  $CUF_{en}$  *prior* to a decision on license  
16 renewal. Varying criteria and assumptions to prospectively be employed may affect the validity  
17 and robustness of the analysis. Thus, without specifying the modifications to be made to the  
18 model, or the process for deciding when and how to have user intervention in the use of the

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<sup>32</sup> 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).

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

3

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

7 A. In light of NRC Staff's acceptance of vague commitments to perform necessary metal  
8 fatigue investigations, analyses, and justifications, in the future, Entergy has failed to  
9 demonstrate that the aging effects of metal fatigue will be adequately managed for the proposed  
10 periods of extended operation, and has, thus, failed to comply with 10 C.F.R. § 54.21(c) or  
11 regulatory guidance, including the *GALL Report*. Entergy has failed to make the affirmative  
12 demonstration that it has a program to sufficiently monitor, manage, and correct metal fatigue-  
13 related degradation at Indian Point.

14

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

16 A. Yes.

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

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In the Matter of )  
 )  
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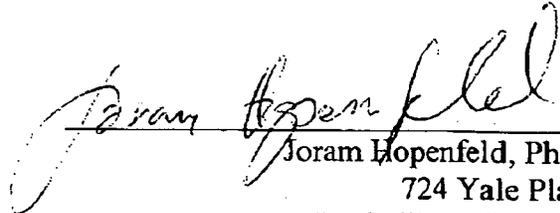
Entergy Nuclear Operations, Inc. )  
(Indian Point Nuclear Generating )  
Units 2 and 3) )  
\_\_\_\_\_ )

Docket Nos.  
50-247-LR  
and 50-286-LR

**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)

  
\_\_\_\_\_  
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June 19 2012