

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

**Riverkeeper, Inc. provisionally designates
the attached Testimony of Dr. Joram Hopenfeld
dated December 20, 2011 as containing
Confidential Proprietary Information
Subject to Nondisclosure Agreement**

REDACTED, PUBLIC VERSION

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

**PREFILED WRITTEN TESTIMONY OF DR. JORAM HOPENFELD
REGARDING NYS-26-B/RK-TC-1B - METAL FATIGUE**

On behalf of Riverkeeper, Inc. ("Riverkeeper"), Dr. Joram Hopenfled submits the following testimony regarding NYS-26-B/RK-TC-1B.

Q. Please state your name and address.

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

Q. What is your educational and professional background?

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

Q. What is the purpose of your testimony?

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

1 by the Atomic Safety & Licensing Board (“ASLB”) on November 4, 2010.¹ Riverkeeper asserts
2 that Entergy Nuclear Operations, Inc. (“Entergy”), the owner of Indian Point, has failed to
3 demonstrate that metal fatigue of reactor components will be adequately managed during the
4 proposed periods of extended operation at the plant as required by 10 C.F.R. § 54.21(c).

5
6 **Q. Have you prepared a report in support of your testimony?**

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

9
10 **Q. What materials have you reviewed in preparation for your expert report and**
11 **testimony?**

12 A. I have reviewed numerous documents in preparation of my expert report and testimony,
13 including the following: all of the pleadings involving Riverkeeper Contention TC-1B, the
14 relevant section of Entergy’s License Renewal Application (“LRA”), Entergy’s Amendment 2 to
15 the LRA, dated January 22, 2008, Entergy’s submission to the ASLB on August 10, 2010
16 entitled, “entitled “Notification of Entergy’s Submittal Regarding Completion of Commitment
17 33 for Indian Point Units 2 and 3,” “refined” Environmental Fatigue Evaluations for Indian Point
18 Units 2 and 3 generated by Entergy’s vendor Westinghouse in June 2010, relevant requests for
19 additional information from the NRC and responses thereto by Entergy concerning metal fatigue,
20 NRC Staff’s Safety Evaluation Report, and Supplement 1 thereto, hundreds of documents
21 identified by Entergy as relevant to Riverkeeper’s metal fatigue contentions, numerous relevant
22 NUREG reports, scientific and scholarly reports and articles, industry guidance documents and
23 reports, and other documents generated by NRC, Entergy, industry groups, and scientific
24 organizations. I have used such documents to inform me of the relevant facts and derive my
25 conclusions.

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

1
2 A list of the particular documents that I reference in my expert report is included at the end of the
3 report. Those references have been provided as Exhibits NYS000146A-146C, NYS00350-352,
4 NYS000354-357, NYS000361-362, and RIV000036-058, in support of my testimony. To the
5 best of my knowledge, these are true and accurate copies of each document that I referred to,
6 used and/or relied upon in preparing my report and this testimony. In some cases where the
7 document was extremely long and only a small portion is relevant to my testimony, an excerpt of
8 the document is provided. If it is only an excerpt, that is noted on the cover of the Exhibit.
9

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

11 A. In my professional judgment, and as I describe in more detail below and in my report,
12 Entergy has failed to demonstrate that the serious aging mechanism of metal fatigue will be
13 adequately managed throughout the proposed extended licensing terms at Indian Point. Though
14 Entergy has proffered “refined” analyses purporting to demonstrate that certain components will
15 remain within allowable acceptance criteria for metal fatigue, these analyses are flawed and
16 inaccurate. Consideration of all relevant factors reveals that many components may become
17 susceptible to metal fatigue and pose safety risks during the proposed periods of extended
18 operation. In light of this eventuality, Entergy should have, but did not, expand the scope of
19 components to be assessed for metal fatigue. Entergy has otherwise not provided sufficient
20 details concerning an Aging Management Program (“AMP”) to ensure that the degradation effect
21 of metal fatigue would be adequately handled during the license renewal periods.
22

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

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

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

7
8 **Q. What are the safety implications of metal fatigue?**

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

18
19 **Q. Please explain how component susceptibility to metal fatigue is predicted.**

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

28
29 Laboratory tests were conducted under controlled conditions at the Argonne National Laboratory
30 ("ANL"), to generate F_{en} factors. Because laboratory tests were not prototypic of the reactor
31 environment, ANL provided a detailed discussion of the required adjustments to be made to the

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

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

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

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

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

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

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

■ [REDACTED]
2 [REDACTED]
3
4 **Q. Please explain why the use of laboratory data must be adjusted in order to**
5 **determine appropriate F_{en} factors in the calculation of CUF_{en} .**

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

23
24 In NUREG/CR-6909, *Effect of LWR Coolant Environment on Fatigue Life of Reactor Materials*,
25 ANL specifies that appropriate bounding F_{en} values of 12 for stainless steel and 17 for carbon
26 and low alloy steel to account for the numerous uncertainties in using the F_{en} equations.³ These
27 values are based on a review of data from different laboratory tests covering a wide range of
28 parameters. These bounding F_{en} factors are not necessarily conservative, and it is reasonable to

² See NUREG/CR-6909 at 72.

³ See NUREG/CR-6909 at iii, 3.

Q. How did you reach the conclusion that Entergy's refined fatigue evaluations

_____S

Q.

1 **Q. Entergy has previously claimed that using the bounding F_{en} values recommended in**
2 **NUREG/CR-6909 “would actually yield less conservative CUF_{en} values, because the ASME**
3 **Code design air curves for carbon steel and low-alloy steels contained in air [which is what**
4 **Entergy considered] . . . are more conservative than the newer air curves in NUREG/CR-**
5 **6909.”⁴ How would you respond to this assertion?**

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

18
19 **Q. Please explain how DO levels affect the determination of appropriate F_{en} values for**
20 **calculating CUF_{en} .**

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

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

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

12
13 Due to the uncertainties related to DO levels during transients, ANL, as well as the Electric
14 Power Research Institute ("EPRI"), specifically instruct that in determining F_{en} , DO can be
15 conservatively taken as the maximum value for the transient.⁵ ANL indicates that for carbon and
16 low-alloy steels and austenitic stainless steel, values of 0.4 ppm and 0.05 ppm, respectively, can
17 be used to perform a conservative evaluation.⁶

18
19 **Q. How did you reach the conclusion that Entergy's refined fatigue evaluations**

20
21 **A.**

22 [REDACTED]
23 [REDACTED]
24 [REDACTED]
25 [REDACTED]
26 [REDACTED]
27 [REDACTED]
28 [REDACTED]

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

⁶ NUREG/CR-6909 at A-5.

1

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

4 A. [REDACTED]
5 [REDACTED]
6 [REDACTED]
7 [REDACTED]
8 [REDACTED]
9 [REDACTED]
10 [REDACTED]
11 [REDACTED]
12 [REDACTED]
13 [REDACTED]
14 [REDACTED]
15 [REDACTED]
16 [REDACTED]
17 [REDACTED]
18 [REDACTED]
19 [REDACTED]
20 [REDACTED]

21

22 **Q. [REDACTED]**
23 [REDACTED]

24 A. [REDACTED]
25 [REDACTED]
26 [REDACTED]
27 [REDACTED]
28 [REDACTED]

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

⁸ NUREG/CR-6909 at 40.

[REDACTED]
[REDACTED]
3 [REDACTED] Notably, EPRI's Materials Reliability Program: Guidelines for Addressing Fatigue
4 Environmental Effects in a License Renewal Application, MRP-47, Rev. 1, calculates F_{en} values
5 as high as 130 at high DO levels, [REDACTED]
6 [REDACTED]
7

8 By failing to appropriately account for DO levels during each transient in the calculation of F_{en} ,
9 it is impossible to conclude that the claimed CUF_{en} values, which Westinghouse and Entergy's
10 "refined" analysis purport to predict to a ten-thousandth of a decimal point, are accurate.
11

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

14 A. A heat transfer coefficient is a parameter used to determine the rate of heat transfer
15 during transients, in order to calculate thermal stress and its impact on fatigue life. Heat transfer
16 is a major factor in the determination of CUF_{en} because it controls the cyclic thermal stresses
17 during transients. Thermal stresses arise when there is a change in the local fluid temperature
18 like during heat-ups or cool-downs or due to local mixing of hot and cold fluids. Failures result
19 from either low stress at high cycle or high stress at low cycle. Stresses from thermal fatigue
20 pose a serious risk of damage and have caused cracks in pipes and leakage at several nuclear
21 reactors. This trend is expected to increase with time.
22

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

⁹ EPRI, MPR 47 at 4-22.

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

23
24 **Q. How did you reach the conclusion that Entergy's refined fatigue evaluations**

25
26 **A.**

27
28
29 For other components, Entergy has failed to provide sufficient information to justify
30 that heat transfer was appropriately considered, also casting doubt on the accuracy of Entergy's
31 "refined" CUF_{en} calculations.

1

Q. Can you please explain how you reached the conclusion that Entergy

4

A.

29

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

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

7 [REDACTED]
8 [REDACTED]
9 [REDACTED]
10 [REDACTED]

11 [REDACTED]
12 [REDACTED]
13 [REDACTED]
14 [REDACTED]
15 [REDACTED]
16 [REDACTED]
17 [REDACTED]
18 [REDACTED]
19 [REDACTED]
20 [REDACTED]
21 [REDACTED]

22 [REDACTED]

23
24 I also reviewed Entergy documentation related to the heat transfer analysis for the inlet and
25 outlet reactor vessel nozzles.¹² The temperature distributions for these components were based
26 on over 40-year old Combustion Engineering calculations. These calculations did not use a

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

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

1 finite element analysis and were based on a simplified 2-D model where the heat transfer
2 coefficient was assumed constant along the flow and thermal properties were taken as
3 independent of the temperature. The calculations were based on “as installed” nozzle
4 dimensions. Due to nozzle geometry, the heat transfer is not uniform along the nozzle and can
5 vary by 20%-30% depending on flow velocity location along the nozzle and flow direction.
6 Such variation would result in axial thermal stress, which is not included in the 2-D analysis.
7 Additionally, it is apparent that this analysis neglected to account for the observed fact that low
8 alloy steels are subjected to wall thinning due to FAC. The reduction in wall thickness after 60
9 years of operation is expected to reduce fatigue life. Based on these considerations, it is apparent
10 that Entergy’s calculations used inappropriate heat transfer coefficients in the calculation of
11 CUF_{en} for the inlet and outlet reactor vessel nozzles.
12

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

16 A. For the other components evaluated in Entergy’s “refined” CUF_{en} analyses, meaning the
17 RCS charging system, RCS injection nozzles, and RHR class 1 piping, Entergy did not provide
18 the actual equations employed to determine the heat transfer coefficients. [REDACTED]
19 [REDACTED]
20 [REDACTED]

21 [REDACTED] Though Entergy has purported to provide sufficient information concerning the
22 calculation of heat transfer coefficients for the rest of the components evaluated, Entergy’s
23 documentation and analyses do not specify the heat transfer coefficients used, or how h was
24 determined, for these components. To assess the uncertainty of h , it is imperative to know the
25 component geometry, the piping geometry upstream of the component, the flow velocities, and
26 the corresponding expressions for h , none of which are specified by Entergy for the components
27 at issue. [REDACTED]
28 [REDACTED]

29 Given
30 the uncertainties associated with determining the heat transfer coefficient, h , it is imperative to
31 ensure that the methodology and assumptions employed were adequate to account for such
uncertainties. Entergy’s calculations have failed to do this. Without an understanding of the

1 values of h and the assumptions used to arrive at such values, the methodology employed by
2 Entergy to re-calculate CUF_{en} for the three relevant components remains questionable. Thus,
3 based on the information relied upon by Entergy to support the “refined” evaluation, it is simply
4 impossible to conclude that the new CUF_{en} values for these other components are accurate.
5

6 **Q. How did you reach the conclusion that Entergy’s refined fatigue evaluations** [REDACTED]
7 [REDACTED]

8 A. To evaluate the remaining fatigue life of a given component, it is necessary to consider
9 the past as well as future loading during all transients. In other words the fatigue status of the
10 component must be known from the time it was installed to the time it is removed from service.
11 An accurate number of past and anticipated future transients is essential to the calculation of the
12 CUF_{en} values. [REDACTED]
13 [REDACTED]
14 [REDACTED]
15 [REDACTED]
16 [REDACTED]
17

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

[REDACTED]

2 [REDACTED]

3

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

17 [REDACTED]

18

19 Justification of an appropriate number of future transients is critical in light of the fact that the

20 useful life of most engineering components and structures follow a “bathtub curve” whereby

21 component failure toward the end of operating life will occur at a very high frequency. Because

[REDACTED] Indian Point will be entering an extended period of operation, [REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

26 [REDACTED]

27 [REDACTED]

[REDACTED]

30 [REDACTED]

31

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

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

6 [REDACTED]
7 [REDACTED]
8 [REDACTED]
9 [REDACTED]
10 [REDACTED]
11 [REDACTED]
12 [REDACTED]
13
14 Entergy's failure to sufficiently account for all relevant parameters has resulted in predictions
15 that are non-conservative. Given the large uncertainties in the input parameters and other
16 assumptions used to generate the revised metal fatigue calculations, the methodology employed
17 by Entergy suggests the likelihood of a wide margin of error, and the detrimental effects of the
18 environment on fatigue strength of the components evaluated are likely grossly underestimated.
19 In fact, many of the revised CUF_{en} values remain very close to unity. The fact that Entergy's
20 refined CUF_{en} values are reported to the fifth significance figure (i.e. to a ten-thousandth of a
21 decimal point), with several just a hair below unity, clearly shows that Entergy does not
22 appreciate the uncertainties inherent in calculating the CUF_{en} values. With a margin of error to
23 account for varying input data and other undisclosed assumptions, such numbers could be
24 considerably higher than the 1.0 regulatory threshold. In any event, without an error analysis,
25 the claimed high degree of accuracy of the results remains questionable at best. Entergy
26 repeatedly applies the term "bounding" to its analyses and results, implying that such results are
27 conservative, and that no error analysis is necessary. However, there is no explanation to show
28 that the results are in any respect bounding or conservative.
29

1 Based on all these considerations, it is my professional opinion that Entergy's refined CUF_{en}
2 calculations cannot be used as the basis for concluding that the aging effects of metal fatigue will
3 be adequately managed at Indian Point during the PEO.
4

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

7 A. No. Entergy has chosen to rely upon its refined EAF analyses to demonstrate that metal
8 fatigue will be managed throughout the period of extended operation. This is not adequate for
9 several reasons.
10

11 First, for the foregoing reasons contained in this testimony, Entergy's new calculations do not
12 demonstrate that the CUF_{en} for the components evaluated will not exceed unity during the
13 proposed extended licensing terms.
14

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

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

20
21 Lastly, Entergy does not have an adequate AMP for metal fatigue because, in the absence of a
22 reliable and complete assessment of CUF_{en} values for susceptible plant components, Entergy has
23 failed to define specific criteria concerning component inspection, monitoring, repair, and
24 replacement. Entergy's plans for correcting metal fatigue related degradation depend initially
25 upon calculating the vulnerability of plant components. Entergy intends to rely upon future
26 CUF_{en} calculations throughout the period of extended operation to manage metal fatigue.
27 Entergy's calculations are meant to signify when components require inspection, monitoring,
28 repair, or replacement, and, according to Entergy, will trigger when such actions are taken. As
29 such, the validity of Entergy's monitoring program depends upon the accuracy of the
30 calculations of the CUF_{en} . When a fatigue monitoring program is entirely based on a predictive
31 analysis and not on actual measurements, and the analysis is flawed, the monitoring program is

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

15
16 **Q. Please summarize your opinions regarding whether or not Entergy has**
17 **demonstrated that metal fatigue of reactor components will be adequately managed during**
18 **the proposed periods of extended operation as required by 10 C.F.R. § 54.21(c).**

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

1 and replacement to ensure that the degradation effects of metal fatigue would be sufficiently
2 handled.

3

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

5 A. Yes.

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

In the Matter of)
)
)

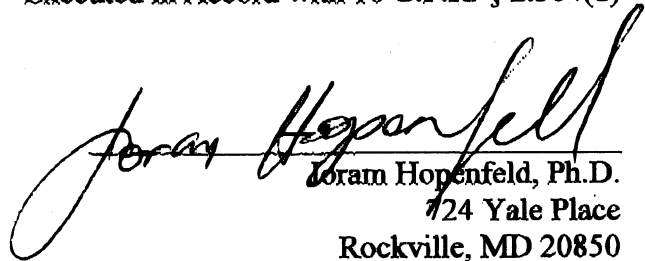
Entergy Nuclear Operations, Inc.)
(Indian Point Nuclear Generating)
Units 2 and 3))
_____)

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

DECLARATION OF DR. JORAM HOPENFELD

I, Joram Hopenfelf, 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 Hopenfelf, Ph.D.
724 Yale Place
Rockville, MD 20850

December 20 2011