**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,<br>Entergy Nuclear Indian Point 3, LLC, and<br>Entergy Nuclear Operations, Inc. | DPR-26, DPR-64<br><br>June 9, 2015 |

**Riverkeeper, Inc. provisionally designates**

**the attached Report of Dr. Joram Hopenfeld**

**dated June 8, 2015 as containing**

**Entergy/Westinghouse Designated Confidential Proprietary Information**

**Subject to Nondisclosure Agreement**

**REDACTED, PUBLIC VERSION**

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**SUPPLEMENTAL REPORT OF**
**DR. JORAM HOPENFELD**
**IN SUPPORT OF**

**CONTENTION NYS-26/RK-TC-1B**

**AND**

**AMENDED CONTENTION NYS-38/RK-TC-5**

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## SUMMARY

Refined metal fatigue predictions, as represented by the environmentally adjusted Cumulative Usage Factor ($CUF_{en}$), generated on behalf of Entergy Nuclear Operations Inc. ("Entergy") by Westinghouse Electric Corporation ("Westinghouse"), as first reported in 2010,[1] and as later reported in additional screening and final fatigue analyses,[2] show [redacted]

The analysis herein of the above-referenced most recent results shows that, similar to Entergy's earlier evaluations, Entergy's recent and final fatigue predictions are technically unsustainable, and based on incorrect input data, circular logic, and theories that defy reality. Far from being based on credible science, the results appear to have been manipulated to obtain the desired results. This assessment shows that the application of corrected input data would predict that many key safety components would exhaust their useful fatigue life—$CUF_{en}$ >1.0—sometime during the proposed extended periods of operation. Entergy's fatigue predictions clearly demonstrate that IP2 and IP3 would operate with hundreds of severely weakened components, [redacted], under the most optimistic expectations. Operating with such degraded components represents a major safety issue. For example, components such as the pressurizer spray nozzle and tube sheet welds [redacted],[3] can be expected to fail under design basis accidents ("DBAs"), yet have now been deemed acceptable from a metal fatigue and aging management perspective. Allowing IP2 and IP3 to operate with highly fatigued components is clearly inappropriate and unsafe. Importantly, existing reactor safety studies[4] never addressed scenarios where many key safety components, such as those analyzed by Entergy, were allowed to remain in service without adequate useful fatigue life remaining.

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[1] NL-10-082, "Notification of Entergy's Submittal Regarding Completion of Commitment 33 for Indian Point Units 2 and 3" (August 9, 2010), *available at*, http://pbadupws.nrc.gov/docs/ML1023/ML102300504.pdf, (Exhibit NYS000352), at Table 5-2.1, 5-2.3; Westinghouse, Environmental Fatigue Evaluation for Indian Point Unit 2, WCAP-17199-P, Revision 0 (June 2010), IPECPROP00056486 (Exhibit NYS000361); Westinghouse, Environmental Fatigue Evaluation for Indian Point Unit 3, WCAP-17200-P, Revision 0 (June 2010), IPECPROP00056577 (Exhibit NYS000362).

[2] Westinghouse, Indian Point Unit 2 and Unit 2 EAF Screening Evaluations, Calculation Note Number CN-PAFM-12-35 (November 2012), IPECPROP00072778 (Exhibit NYS000510); Westinghouse, Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, Calculation Note Number CN-PAFM-13-32 (August 2013), IPECPROP00078338, at Tables 2-1, 2-2, 2-3, 2-4 (Exhibit NYS000511).

[3] *See* Westinghouse Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, Calculation Note Number CN, PAFM-13-32 (August 2013), IPECPROP00078338, at Table 2-3 (Exhibit NYS000511).

[4] *See* NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants (1990), *available at*, http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1150/ (Exhibit RIV00146A-Exhibit RIV00146B); NUREG-75/014 (WASH-1400), Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants (1975) (Exhibit RIV000147).

In addition, the results of Entergy's more recent final refined fatigue analyses [redacted]. However, as this assessment demonstrates, the screening evaluations performed for Entergy were likewise flawed, since they [redacted] Entergy's screening evaluation also ignored industry guidance regarding the appropriate scope of a fatigue analysis when such an analysis is expanded beyond the locations identified in NUREG/CR-6260.[5] As a result, the environmentally corrected fatigue life of many safety related components, including steam generator components, remain without adequate fatigue analysis.

As a result of Entergy's flawed and inadequate fatigue evaluations, Entergy has failed to demonstrate that it has an adequate program for managing the effects of metal fatigue on plant components during the proposed periods of extended operation.

## INTRODUCTION

In 2007, Entergy filed a License Renewal Application ("LRA") with the U.S. Nuclear Regulatory Commission ("NRC") applying for 20-year life extensions of the IP2 and IP3 nuclear plants. In the LRA, Entergy reported predictions related to metal fatigue, including the environmentally adjusted CUF values of 6 components identified in NUREG/CR-6260 as "limiting locations."[6] In Riverkeeper Technical Contention RK-TC-1, Riverkeeper, with my expert analysis and support, contested Entergy's Indian Point LRA on the grounds that some of the $CUF_{en}$ values of these locations exceeded unity (1.0) and because the CUF of other components would also be expected to exceed unity if they were corrected to account for environmental effects.[7] Instead of expanding the $CUF_{en}$ analysis to include additional locations, as suggested in Contention RK-TC-1, Entergy, recalculated the $CUF_{en}$ values for the same NUREG/CR-6260 locations and amended its LRA in 2010, claiming that the revised $CUF_{en}$ values of all the NUREG/CR-6260 components were now less than unity (1.0) and such values represented the most limiting locations, i.e., a bounding assessment, for IP2 and IP3.[8] In response, in 2010, in Contention NYS-26B/RK-TC-1B, Riverkeeper and the State of New York

[5] NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components" (February 1995) (Exhibit NYS000335).

[6] *See* Indian Point Energy Center License Renewal Application at Tables 4.3-13, 4.3-14 (Exhibit ENT00015A-B).

[7] *See* Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene in the License Renewal Proceeding for the Indian Point Nuclear Power Plant (November 30, 2007), ADAMS Accession No. ML073410093, at 7-15. Riverkeeper's contention was subsequently consolidated with a similar contention raised by the State of New York.

[8] *See* NL-10-082, License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program (August 9, 2010), *available at*, http://pbadupws.nrc.gov/docs/ML1023/ML102300504.pdf (Exhibit NYS000352).

amended its consolidated contention to challenge Entergy's "refined" fatigue evaluations and the ongoing inadequacy of Entergy's aging management plan for metal fatigue at Indian Point.[9]

Subsequently, in 2011, the NRC acknowledged and conceded that there may be more limiting locations at Indian Point than those identified in NUREG/CR-6260 that were analyzed by Entergy, and requested that Entergy confirm and justify bounding locations for IP2 and IP3.[10] However, as memorialized in Supplement 1 to the Indian Point Safety Evaluation Report dated August 2011, NRC Staff accepted Entergy's vague Commitment 43 to address this issue. In response, Riverkeeper and the State of New York filed an additional contention, Contention NYS-38/RK-TC-5, which, among other bases, contested Entergy's program for managing metal fatigue due to Entergy's failure to expand the scope of its fatigue analysis and conduct a bounding metal fatigue assessment.[11] In connection with this contention, I explained, among other things, that Entergy must expand its analysis to include balance-of-plant and reactor vessel internal (RVI) components. Notably, Entergy justified its failure to conduct fatigue analysis for balance-of-plant components by claiming that the fatigue life of such components had been conservatively analyzed. However, industry guidelines do not specify that balance-of-plant components can be excluded from $CUF_{en}$ analysis, as I have raised in submissions related to Riverkeeper's admitted contentions.

Subsequently, the NRC Staff undertook a supplemental safety review, which culminated in the issuance of Supplement 2 to the Indian Point Safety Evaluation Report in November 2014. In this report, NRC Staff memorialized Entergy's Commitment 49 to manage the effects of fatigue on RVI components at Indian Point during the proposed periods of extended operations by relying on its Fatigue Monitoring Program and recalculating CUF values for RVI components to include reactor coolant environment effects.[12] In response, Riverkeeper and the State of New York successfully raised amended bases to contention NYS-38/RK-TC-5, with support of my expert declaration which criticized Entergy's commitment and flawed methodology for determining $CUF_{en}$ values for RVI components.[13]

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[9] *See* State of New York's and Riverkeeper's Motion for Leave to File a New and Amended Contention Concerning the August 9, 2010 Entergy Reanalysis of Metal Fatigue (Sept. 9, 2010); Declaration of Dr. Joram Hopenfeld in Support of Petitioners State of New York and Riverkeeper, Inc.'s New and Revised Contention Concerning Metal Fatigue (Sept. 9, 2010).

[10] *See* NRC Letter, "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Numbers 2 and 3, License Renewal Application" (February 10, 2011) (Exhibit NYS000199); NUREG-1930, Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Supplement 1 (August 2011) (Exhibit NYS000160).

[11] State of New York and Riverkeeper's New Joint Contention NYS-38/RK-TC-5 (September 30, 2011), ADAMS Accession No. ML11273A196.

[12] *See* NUREG-1930, Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Supplement 2 (November 2014), at 3-51 to 3-52 (Exhibit NYS000507).

[13] State of New York and Riverkeeper's Joint Motion for Leave to Supplement Previously-Admitted Joint Contention NYS-38/RK-TC-5 (February 13, 2015), ADAMS Accession No. ML15044A498; Declaration of Dr. Joram Hopenfeld (February 12, 2015) (Exhibit RIV000148).

In accordance with Entergy's regulatory Commitment 43 to determine the limiting locations for IP2 and IP3, and regulatory Commitment 49 to calculate $CUF_{en}$ values for RVI components, and after years of delay, Entergy vendor, Westinghouse, conducted and issued refined fatigue analyses for Indian Point.[14]

The purpose of this report is to explain how these most recent fatigue evaluations are fundamentally flawed in various respects, and how Entergy continues to lack an adequate aging management program for metal fatigue at Indian Point. Westinghouse and Entergy either ignored important parameters or selected inputs that would minimize the effect of the environment on fatigue life, and was thereby able to obtain $CUF_{en}$ values that were <1. In particular, Entergy's calculations are deficient in the following ways, as will be described in further detail below:

1.) Westinghouse/Entergy failed to properly account for the effects of **dissolved oxygen** on component fatigue;

2.) Westinghouse/Entergy failed to account for **radiation and stress corrosion** effects on metal fatigue;

3.) Westinghouse/Entergy continued the flawed approach of assuming a **CUF of record** in the fatigue analyses; and

4.) Westinghouse/Entergy failed to properly **expand the scope of analysis** to bound the most limiting locations.

In light of these various deficiencies, and given [redacted], an uncertainty analysis should have been conducted, but was not.

Moreover, Westinghouse/Entergy failed to conduct a safety assessment to show that IP2 and IP3 can operate safely during normal operations and DBAs, despite the fact that many of the refined $CUF_{en}$ values are very close to 1 without any uncertainty allowance.

Based on a review of Entergy's latest fatigue evaluations, the conclusion remains that Entergy has failed to demonstrate that the CUFs of components at Indian Point will not exceed unity and/or succumb to metal fatigue during the proposed periods of extended operation, or that it otherwise has an adequate program for managing the effects of metal fatigue at Indian Point during the proposed periods of extended operations for IP2 and IP3.

[14] Westinghouse, Indian Point Unit 2 and Unit 2 EAF Screening Evaluations, Calculation Note Number CN-PAFM-12-35 (November 2012), IPECPROP00072778 (Exhibit NYS000510); Westinghouse, Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, Calculation Note Number CN, PAFM-13-32 (August 2013), IPECPROP00078338 (Exhibit NYS000511).

## DISCUSSION

### 1. The $CUF_{en}$ Equation

Before the degree of inaccuracy and lack of conservatism in Entergy's final predicted $CUF_{en}$ values can be appreciated, it is useful to review key concepts underlying the calculations of the environmentally corrected cumulative usage factors, $CUF_{en}$, and then to describe how Energy improperly selected the input data that misleadingly caused all the most recent final $CUF_{en}$ values to be less than one.

#### *1.1 Argonne National Laboratories Methodology of Accounting for Environmental Effects*

Since the American Society of Mechanical Engineers (ASME) code does not specify a methodology for how to determine the effects of the light water reactor (LWR) environment on metal fatigue, Argonne National Laboratories (ANL) undertook this task in the mid-1990s, with work still in progress. The ANL methodology is based on the following equation (hereinafter referred to as Equation 1):

$$CUF_{en}\text{ (water)} = F_{en}\text{ (lab)} \quad \text{x} \quad CUF\text{ (air)}$$

$CUF_{en}$ is the environmentally corrected cumulative usage factor (CUF), $F_{en}$ is an environmental correction factor to relate laboratory fatigue data in water to laboratory fatigue data in air, and CUF (air) represents a CUF of a given plant component based on data that was obtained in air, commonly specified by the ASME code. Since the $F_{en}$ is an experimental factor, the underlying principle of using the above equation is that the user would not extrapolate the $F_{en}$ to conditions other than those that existed in its derivation.

#### *1.2 ANL Tests and Data Extrapolation*

In the ANL tests, small polished specimens were exposed to cyclic loads in a loop where the temperature and water chemistry at the surface of the specimens were well known and were kept steady during the tests. The tests did not represent prototypic conditions such as would be experienced by actual components during thermal transients. In 2007, ANL published a detailed report, NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials,"[15] which summarized the test results and proposed models and methodologies to explain and account for how water chemistry and material composition affect metal fatigue. The report proposed a model for the transition from crack initiation to crack propagation, and described the role of oxygen and metal composition on crack propagation. The report presented experimental correlation on the effects of oxygen, temperature, and strain rate on the reduction in metal fatigue life when the specimen was exposed to water instead air.

[15] NUREG/CR-6909, ANL-06-08, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials (2007) (Exhibit NYS000357).

Recognizing the lack of prototypical results, NUREG/CR-6909 provided a lengthy list of the differences between the laboratory setting and the actual nuclear plant setting, emphasizing the limitation of the experimental data. In addition, Dr. O.K. Chopra, the principal investigator of the ANL research, discussed the results with the NRC's Advisory Committee on Reactor Safeguards (ACRS) and emphasized that it is the responsibility of the operator to account for the differences between the lab and plant environments when applying the results to draw conclusions regarding fatigue life.[16] Dr. Chopra emphasized that the ANL results may not be conservative.[17] Recognizing that the results of the ANL tests were not prototypical, the Electric Power Research Institute (EPRI) issued a list of guidelines—MRP-47—regarding how the ANL data should be corrected to account for differences between laboratory and plant environments.[18]

As discussed below, throughout the course of the Indian Point license renewal proceeding, Entergy has taken the erroneous position that the ANL data can be applied to IP2 and IP3 directly for accurate metal fatigue predictions without accounting for the known differences between the laboratory and plant environments. Entergy has also incorrectly maintained the position that that CUF in Equation 1 above can be substituted by the CUF of records, without major reanalysis of the CUF of records, which is again discussed in further detail below.

### *1.3 The $F_{en}$ Equation*

The environmental correction factor, $F_{en}$, in Equation 1 above was obtained as a best statistical fit to experimental data. When compared to the experimental data, the ANL best-fit data model, as set forth in NUREG/CR-6909, predicts fatigue life within a factor of 3 for low carbon and low alloy steel and austenitic steels.[19] Equation 1 is an approximate, not an exact equation, and some data points will fall outside of predictions derived from using the equation.

The $F_{en}$ equation provided in NUREG/CR-6909 is as follows:

$$F_{en} = \exp\ (K - K1(\ S^*\ \varepsilon^*\ \ T^*\ O^*))$$

In this equation the different variable represent the following:

| | | |
|---|---|---|
| K, K1 | = | constants depending on material type (carbon steel, low alloy steel stainless steel) |
| S* | = | Transformed Sulphur |

[16] Official Transcript of Proceedings, Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards Subcommittee on Materials, Metallurgy and Reactor Fuels (December 6, 2006), ADAMS Accession No. ML063550058, at 22 (Exhibit RIV000037).
[17] *See generally id.*
[18] EPRI, MRP-47, Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, Final Report, Revision 1 (September 2005) (Exhibit NYS000350).
[19] *See* NUREG/CR-6909, ANL-06-08, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials (2007) at 26, 62 (Exhibit NYS000357).

| | | |
|---|---|---|
| O* | = | Transformed dissolved oxygen near metal surface |
| T* | = | Transformed temperature tem at metal surface: T- 150 |
| ε* | = | Transformed strain rate |
| O or ( DO) | = | dissolved oxygen concentration at the metal surface |
| O* | = | 0 when O <0.04 parts per million (ppm) |
| O* | = | ln(O/.04) for O> 0.04 ppm |

## 2. Inadequate Consideration of Dissolved Oxygen

Dissolved oxygen, DO, is the most important of 12 major variables listed by ANL that must be considered when applying Equation 1 to the LWR environment.

The formation of oxide films on the surface of stainless steel is believed to play a major role in metal fatigue, however that effect is still not completely understood. Until this mechanism is better understood, the application of the $F_{en}$ equation to plant components must employ DO values that resemble those that were used in the development of that equation. Oxygen concentrations in the laboratory tests were uniform in liquid and were conducted at steady state with controlled water chemistry and temperature. On the other hand, in the plant, local oxygen concentrations are not well known during transients because measurements in the plant are made by bulk sampling periodically during steady state operations,[20] usually far removed from the component of interest during thermal transients.

Reactor coolant contains high concentrations of corrosive products that vary in composition around the flow path. Oxygen enters the reactor system usually during heat-up and cool-down operations and is also generated by electrolysis in the core. Hydrazine is used to maintain low DO levels during steady state operating temperatures. Hydrazine is no more than a catalyst which facilitates the formation of metal oxides and hydroxides, $(Fe_2)O_3$, $Fe(OH)_3$, respectively. Hydrazine does not remove oxygen from the reactor system; it only changes its form, which varies with the temperature. Thus, the oxygen could be bound to metal surfaces or be floating crud in the system. It is not possible to arbitrarily assume that the sampled steady state DO concentration in a typical PWR [redacted] would be the same as the DO at the surfaces of all components at IP2 and IP3 during all transients where the temperature undergoes abrupt changes. It is the concentration of DO during temperature transients that must be entered into and accounted for in the $F_{en}$ equation. Such concentrations must be measured, as they can neither be assumed nor calculated. In recognition of the difficulties of determining the DO in real life environments, ANL and EPRI developed guidelines for how to enter the oxygen term in the $F_{en}$ equation for plant application.

[20] *See* EPRI, MRP-47, Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, Final Report, Revision 1 (September 2005) (Exhibit NYS000350).

### *2.1 ANL Guidance*

ANL recommended using 0.4 ppm for the DO value during transients for carbon and low alloy steels and 0.05 ppm for the DO value for stainless steel during all transients.[21]

### *2.2 EPRI Guidance*

EPRI's guidance indicates that the $F_{en}$ depends on the combination of strain rate and DO. Therefore, EPRI's guidelines specify that when the calculations are based on average strain rate, one should use maximum DO values for carbon steel and minimum DO values for stainless steel during the time when the stress increases. Additional guidance is provided when different strain rate methodologies are used.[22]

### *2.3 Entergy's Consideration of Dissolved Oxygen at IP2 and IP3*

As with Entergy's previous fatigue evaluations (as I have discussed in my previous analyses and submissions in relation to Contention NYS-26/RK-TC-1), Westinghouse/Entergy [redacted]

In particular, in conducting the most recent final fatigue calculations, [redacted]

[redacted]

[21] *See* NUREG/CR-6909, ANL-06-08, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials (2007) at A.5 (Exhibit NYS000357).

[22] EPRI, MRP-47, Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, Final Report, Revision 1 (September 2005), at 4-19, 4-20 (Exhibit NYS000350).

[23] Westinghouse, Indian Point Unit 2 and Unit 2 EAF Screening Evaluations, Calculation Note Number CN-PAFM-12-35 (November 2012), IPECPROP00072778, at 28 (Exhibit NYS000510) ("For this screening evaluation, a value of 0.005 ppm is used for the dissolved oxygen (DO) content").

[24] *See generally* Westinghouse, Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, Calculation Note Number CN, PAFM-13-32 (August 2013), IPECPROP00078338 (Exhibit NYS000511).

[redacted]

In addition to the justification provided in the screening evaluation, Entergy has also previously state [redacted] However, it must be noted that neither EPRI nor ANL advanced such a theory even though both realize the importance of oxygen on metal fatigue.

### *2.4 Discussion of Entergy's Incorrect Dissolved Oxygen "Theory"*

In spite of the recognition that the [redacted] Entergy has never provided or acknowledged the existence of plant measurements of DO during transients at key locations. A review of Entergy's documentation of chemistry control at IP2 and IP3 provides no description [redacted]

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[25] Westinghouse, Indian Point Unit 2 and Unit 2 EAF Screening Evaluations, Calculation Note Number CN-PAFM-12-35 (November 2012), IPECPROP00072778, at 26 (Exhibit NYS000510).

[26] *Id.* at 28.

[27] Westinghouse, Indian Point Unit 2 and Unit 2 EAF Screening Evaluations, Calculation Note Number CN-PAFM-12-35 (November 2012), IPECPROP00072778, at 28 (Exhibit NYS000510).

[28] *Id.*

[29] [redacted]

[30] Testimony of Entergy Witnesses Nelson F. Azevedo, Alan B. Cox, Jack R. Strosnider, Robert E. Nickell, and Mark A. Gray Regarding Contention NYS-26B/RK-TC-1B (Metal Fatigue) (March 29, 2012) at 116 (Exhibit ENT0000183).

[31] Westinghouse, Indian Point Unit 2 and Unit 2 EAF Screening Evaluations, Calculation Note Number CN-PAFM-12-35 (November 2012), IPECPROP00072778, at 28 (Exhibit NYS000510).

[32] *See* Material Aging Institute International Conference on Plants Materials Degradations, Chemical conditioning of Light Water reactors Systems (2008), IPEC00265853 (Exhibit RIV000149). This document does not even mention $F_{en}$ equations.

[redacted]

T* and O* have no physical significance other than to provide a best statistical correlation for the $F_{en}$ equation for a given set of data. Since the definitions of T* and O* were formulated in 2006, ANL has obtained additional data and modified the definitions of T* and O* accordingly.[33] To obtain the best statistical fit for the $F_{en}$, the original definitions of T* and O* were changed such that now the second term in the exponent of the $F_{en}$ equation is not zero any more below 150°C. With this change and removal of the discontinuity at T = 150, [redacted]

The effect of the new definition of T* and O* on Entergy's theory is depicted below in Figures 1 and 2. Figure 1 illustrates graphically how [redacted]

[redacted]

[33] *See* ANL, Report on Assessment of Environmentally-Assisted Fatigue for LWR Extended Service Conditions, ANL-LWRS-47 (September 2011) (RIV000150).

Figure 2 shows that [redacted]

[redacted]

Notably, the new definition of T* and O* was published[34] before the revised draft NUREG/CR-6909 was issued,[35] [redacted]

[redacted]

[34] *Id.*

[35] NUREG/CR-6909, Revision 1, ANL-12/60, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials, Draft Report for Comment (March 2014), *available at*, http://pbadupws.nrc.gov/docs/ML1408/ML14087A068.pdf (Exhibit NYS000490).

[36] When the $F_{en}$ equation is expressed explicitly in terms of O*, O* = ( K – lnFen )/(S* ε* T*), O* become infinitely large when T* = 0 and is finite and larger than zero when T* is larger than zero. [redacted] $F_{en}$ is an approximation and is only exact at the point where it intersects the data points.

**Figure 3: Comparison of Bounding $F_{en}$ Factors**

| Dissolved oxygen at the surface (ppm) | Bounding $F_{en}$ (LRA) | Basis | Reference |
|---|---|---|---|
| [redacted] | | | |
| 0.4 | 8.7<br>Carbon steel<br><br>12.25<br>LAS | Industry guidance | NUREG/CR-6909 at A-5; EPRI MPR-47 at 4-21 |

[redacted] The $CUF_{en}$ predictions I previously discussed used the $F_{en}$ equation with oxygen, temperature, strain rate, and Sulphur values of 0.4 ppm, 250°C, 0.005%, and 0.01% respectively. It should be noted that Figures 1and 3 above [redacted] When the effects of strain rate are considered in combination with DO, EPRI calculations show that the $F_{en}$ is about 90 for the ANL recommended value of oxygen concentration of 0.4 ppm.[39] [redacted]

[37] *See* Report of Dr. Joram Hopenfeld in Support of Contention NYS-26-B/RK-TC-1B – Metal Fatigue (December 22, 2011) (Exhibit RIV000035).

[38] Westinghouse, Indian Point Unit 2 and Unit 2 EAF Screening Evaluations, Calculation Note Number CN-PAFM-12-35 (November 2012), IPECPROP00072778, at 28 (Exhibit NYS000510).

[39] EPRI, MRP-47, Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, Final Report, Revision 1 (September 2005), at Figure 4-7, p.4-22 (Exhibit NYS000350).

[redacted]

In conclusion, [redacted] It is also very difficult to understand because it defies all logic. [redacted]

[redacted] NL proposed a realistic alternative of using 0.4 ppm for all transients. [redacted]

[redacted]. Entergy has failed to account for this eventuality and has, thus, continued to fail to demonstrate an adequate program for managing the effects of metal fatigue at Indian Point.

## 3. Inadequate Consideration of Radiation and Stress Corrosion Effects

Reactor vessel internal (RVI) components, for which Entergy undertook fatigue evaluations that are the subject of supplemental bases to contention NYS-38/RK-TC-5 and this report, are subject to radiation exposure. This radiation can effect fatigue life, [redacted]

### *3.1 Radiation and Stress Corrosion Effects*

Metal fatigue, especially in the low-cycle range, depends on ductility. Since neutron irradiation reduces ductility by producing lattice defects in individual crystals, it can be expected that radiation will effect fatigue life. Reduction in fatigue life by radiation was recognized more than 40 years ago.[40] Since then, a number of studies have shown that the main variables that reduce fatigue life are fluence material composition, number of cycles, and hold time. For example, Korth and Harper reported a reduction in fatigue life of about one half for 308 Stainless Steel at

[40] *See* NASA, Nuclear and Space Radiation Effects on Materials, NASA SP-8053 (June 1970), *available at*, http://ntrs nasa.gov/archive/nasa/casi ntrs.nasa.gov/19710015558.pdf (Exhibit RIV000151).

cycles below 5000 per second.[41] Tests in the U.S. and Japan show that irradiation can increase crack growth rates by more than a factor of 5 in low oxygen boiling water reactor (BWR) environments.[42]

Similarly to affecting fatigue, radiation also effects stress corrosion cracking (SCC). The main difference between SCC and metal fatigue is that the former occurs under cyclic loads while the latter occurs at static loads. In a 2012 ANL study, S. Mohanty, et al discussed several deterministic models with empirical crack growth rates to predict SCC and fatigue life.[43] Crack growth rates for alloy 600 were presented in terms of an experimental stress intensity factor and cyclic frequency, stress ratio, and surface temperature. The study concluded that many empirical models are available for reactor component base metals but very few for dissimilar metal welds.[44] The study further concluded that metal fatigue, flow accelerated corrosion (FAC), and SCC can act in combination with each other to magnify their individual effects.[45]

Thus, any analysis of the effects of the LWR environment on fatigue must consider the synergistic effects of radiation SCC and thermal embrittlement. A first step towards this end would be to incorporate the effects of radiation into the $F_{en}$ equation.

Based on extensive literature review of the effects of radiation on fatigue in the LWR environment, a draft revised NUREG/CR-6909 concluded that the limited available data was inconclusive with regard to the impact of irradiation on fatigue in the LWR environment.[46] However, in this revised and still draft report, ANL and NRC inappropriately recommend that the effects of radiation on the $F_{en}$ be ignored because the existing data cannot be used to quantize these effects and that to do so, data on the effects of radiation on the $F_{en}$ would have to be obtained.[47] The draft report gives no indication that such data will ever be obtained, so instead supports ignoring the effects of radiation on fatigue life.

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[41] G. E. Korth & M. D. Harper, *Effects of Neutron Radiation on the Fatigue and Creep/Fatigue Behavior of Type 308 Stainless Steel Weld Metal at Elevated Temperatures* (Seventh ASTM International Symposium on Effects of Radiation on Structural Materials, June 1974), *available at*, http://www.osti.gov/scitech/servlets/purl/4294682 (Exhibit RIV000152).

[42] *See* Argonne National Laboratory, Corrosion and Mechanics of Materials, Light Water Reactors, Irradiation-Induced Stress Corrosion Cracking of Austenitic Stainless Steels, http://www.ne.anl.gov/capabilities/cmm/highlights/ssc_austenic_ss.html (last visited June 2, 2015) (Exhibit RIV000153).

[43] S. Mohanty, S. Majumdar, & K. Natesan, Argonne National Laboratory, *A Review of Stress Corrosion Cracking/Fatigue Modeling for Light Water Reactor Cooling System Components* (June 2012) (Exhibit RIV000154).

[44] *See id.*

[45] *See id.*

[46] *See* NUREG/CR-6909, Revision 1, ANL-12/60, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials, Draft Report for Comment (March 2014), *available at*, http://pbadupws.nrc.gov/docs/ML1408/ML14087A068.pdf (Exhibit NYS000490).

[47] *Id.*

To disregard the effects of radiation on the $F_{en}$ because of lack of data is entirely arbitrary, non-scientific, and inconsistent with the ASME Code. The ASME Code is not prescriptive and does not require precise calculations, but only that the analyst perform calculations that are technically sound and conservative. Given the theoretical and experimental observations on the effects of radiation on fatigue, it would be appropriate to apply the observed factor of 5 in the crack growth rate (CGR)[48] as a bounding multiplier to determine all the $CUF_{en}$ values for all of the RVI components. This approach is supported by experimental observations and is more defendable than the one proposed by ANL and NRC in revised draft NUREG/CR-6909, i.e. to ignore radiation effects on the $F_{en}$ and resulting $CUF_{en}$ for the time being. Notably, neutron fluence varies in the reactor vessel and the CGR can be expected to vary with fluence; accordingly, the factor of 5 for the CGR should be adjusted depending on component location in the reactor vessel.

### 3.2 *Entergy's Theory Why* [redacted]

[redacted]

The SSER Supplement 2 indicates Entergy's rationale for why the synergistic effects of radiation, thermal embrittlement, metal fatigue, and SCC for the support columns in particular can be ignored: because (1) the effects of embrittlement are only significant in the presence of pre-existing flaws where tensile stresses can propagate these flaws; (2) the $CUF_{en}$ of the IP2 support columns is less than one; and (3) a $CUF_{en}$ value less than one demonstrates that fatigue initiation is not expected during the life of the plan.[51]

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[48] *See* Argonne National Laboratory, Corrosion and Mechanics of Materials, Light Water Reactors, Irradiation-Induced Stress Corrosion Cracking of Austenitic Stainless Steels, http://www.ne.anl.gov/capabilities/cmm/highlights/ssc_austenic_ss.html (last visited June 2, 2015) (Exhibit RIV000153).

[49] *See* Westinghouse, Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, Calculation Note Number CN, PAFM-13-32 (August 2013), IPECPROP00078338, at 7-8 (Table 2-1, Table 2-2) (Exhibit NYS000511).

[50] *See* NUREG-1930, Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Supplement 2 (November 2014), at 3-51 to 3-52, A-14, A-15 (Exhibit NYS000490).

[51] *See id.* at 3-40 to 3-45.

### *3.3. Discussion of Entergy's Theory on Radiation and Synergistic Effects*

Entergy's apparent theory as described above disregards scientific facts and the prevailing views of many researchers on the effects of radiation/neutron fluence, and SCC on metal fatigue. Entergy's reliance on the notion that components with CUF values of less than 1 will not be impacted by radiation effects or subject to fatigue initiation or propagation is based on an incorrect perception of the meaning of the CUF. 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 proposed periods of extended operation. Importantly, a $CUF_{en}$ of less than one does not necessarily demonstrate that fatigue initiation is not expected during the life of the component.

The position that a component will not be subject to fatigue cracks because the $CUF_{en}$ for that component is less than one stands in contradiction of the conventional understanding of metal fatigue. The CUF does not represent the absence of cracks or flaws in the material when the CUF or $CUF_{en}$ is less than 1. The CUF is used to assess the possibility of fatigue failure in a given environment and is based on a criterion that the CUF should be kept below one. Large numbers of fatigue tests have shown that a statistically significant number of test specimens would fail under cyclic loading when the CUF exceeded one. These results were based only on the observations of specimen failures, not on crack size history during the tests. The CUF is strictly an empirical criterion that has proven to work well over half a century for full size components in diverse applications.

The common understanding of crack initiation and propagation under cyclic loads stands in contradiction to Entergy's theory. NUREG/CR-6909 schematically depicts crack formation during the fatigue life of specimens under cyclic loads, and clearly indicates that cracks (flaws) are present from the beginning of the test, 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."[52] Schematically, the transition from microscopic cracks to macroscopic cracks occurs about the time it reaches its half-life. The initiation stage of fatigue involves the growth of microscopic cracks and this stage is not characterized by any specific value of $CUF_{en}$. Importantly, NUREG/CR-6909 as well as other studies give no indication that fatigue initiation does not exist when $CUF_{en} < 1.0$, or that there is a correlation between crack size and the $CUF_{en}$. A 2012 ANL study also shows that the time when crack initiation occurs depends on empirical constants and the strain rate, with no mention that cracks do not propagate when $CUF_{en} < 1.0$.[53] In addition, NRC studies at the Oak Ridge National Laboratory (ORNL) 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

[52] NUREG/CR-6909, ANL-06-08, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials (2007) at 7 (Exhibit NYS000357).

[53] *See* S. Mohanty, S. Majumdar, & K. Natesan, Argonne National Laboratory, *A Review of Stress Corrosion Cracking/Fatigue Modeling for Light Water Reactor Cooling System Components* (June 2012) (Exhibit RIV000154).

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."[54] The ANL and ORNL research invalidates the position that when $CUF_{en}$ values are less than one, fatigue initiation of cracks is not expected during the lifetime of the plant.[55]

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.[56] Thus, the failure to consider radiation effects makes any findings of a $CUF_{en}$ value that is under 1.0 inaccurate and unreliable. Entergy's apparent reasoning for why synergistic effects can be ignored employs a circular logic: [redacted] it is still technically impossible to obtain a higher degree of accuracy than the intrinsic accuracy factor of 3 for the best fit $F_{en}$ equation. Clearly, [redacted]

As I have consistently maintained throughout the Indian Point license renewal proceeding, when undertaking fatigue analysis, it is necessary for all assumptions to be revealed and explained in order to allow for an in-depth and accurate review of the results. Once again however, [redacted]

[54] 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*, at 2 (Exhibit RIV000118).

[55] *See* NUREG-1930, Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Supplement 2 (November 2014), at 3-43 (Exhibit NYS000507).

[56] 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.

[57] *See* Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, CN-PAFM-13-32, IPECPROP00078338 (2013) at Table 2-1 (Exhibit NYS000511).

[58] *Id.*

[59] *Id.* at Table 2-1.

[redacted] renders Entergy's assessment of fatigue inaccurate and unreliable for purposes of demonstrating that metal fatigue will be properly managed during the proposed periods of extended operation.

For example, [redacted]

[redacted], all the affected $CUF_{en}$ values for RVI components should be multiplied by a factor of 5, as discussed above, in order to obtain conservative and more accurate estimates that will actually ensure that fatigue will be properly managed during the proposed periods of extended operation.

Finally, [redacted], as discussed in greater detail below.

## 4. Errors From Assuming CUF = CUF of Record

In addition to uncertainties in Entergy's refined and most recent final $CUF_{en}$ values due to [redacted], the methodology employed also introduces another major uncertainty by apparentl [redacted]

The original fatigue analyses for the current licensing basis (CLB) to obtain the CUF of record were conducted more than 40 years ago, and are, thus, based on calculations that were valid when the plants were initially designed because all components were presumably in pristine conditions. The $F_{en}$ was determined by comparing fatigue life of similar test specimens in air and water, and in applying Equation 1 (see above) to the plant; a presumption is that the CUF for each component was calculated on the basis that the geometry was well known.

However, after 40 years of exposure to a hostile 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. The mechanical properties (stress strain curves) of RVI components also changed due to exposure to radiation.

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[60] Westinghouse, Indian Point Unit 2 and Unit 2 EAF Screening Evaluations, Calculation Note Number CN-PAFM-12-35 (November 2012), IPECPROP00072778 at 24-25, 27-28, 33-35 (Exhibit NYS000510).
[61] *Id.* at 28.

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.

Thus, such changes must be applied to the CUF of record if the $F_{en}$ equation (Equation 1 above) is used to calculate $CUF_{en}$ values. However [redacted]

### *4.1 Geometry Changes*

The geometry of many components at IP2 and IP3 have changed over the past 40 years due to flow-accelerated corrosion (FAC). FAC can cause severe wall thinning,[62] that, in addition to reducing metal strength, introduces surface discontinuities. Such discontinuities are well known to introduce local stress concentrations and reduce fatigue life, which has been documented in numerous publications and engineering handbooks.[63]

In spite of such universal acceptance of the importance of stress concentration factors, Entergy has claimed that such effects are not relevant or proven. In addition, during hearings related to FAC, Entergy indicated its position that certain instances of severe wall thinning at Indian Point was not due to FAC, but the result of variations during manufacturing, even though such thinning exceeded the specified tolerances for the impacted components. In any event, given Entergy's position, it is clear [redacted]

### *4.2 Surface Finish*

Similar to wall thinning by FAC, surface roughness introduces local stresses that form nucleation sites for crack initiation. The ASME code requires that the average stress of a component be multiplied by the appropriate stress concentration factor. For example, the fatigue life of a

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[62] *See generally*, Prefiled Written Testimony of Dr. Joram Hopenfeld Regarding Riverkeeper Contention TC-2 – Flow Accelerated Corrosion (December 22, 2011) (Exhibit RIV000003); Prefiled Rebuttal Testimony of Dr. Joram Hopenfeld Regarding Contention NYS-38/RK-TC-5 (November 9, 2012) (Exhibit RIV000134); Entergy Ultrasonic Examination Report, IPEC00020853 (Exhibit RIV000130).

[63] *See* J.S. Kim & J.S. Seo, *Development of Engineering Formulae for Stress Concentration Factors of Local Wall Thinning in CANDU Feeder Pipe Under Pressure*, Proceedings of the ASME 2011 Pressure Vessels & Piping Division Conference, PVP2011, July 17-21, 2011, Baltimore, Maryland, USA, PVP2011-57930 (Exhibit RIV000138).

component with lathe-formed surface is lower by a factor of 10 than if that surface was super finished.[64] 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.[65] Relying on the CUFs of record without allowing for surface changes during service completely ignores the overwhelming importance of surface topography on fatigue life.

At the evidentiary hearings in this proceeding related to FAC, Entergy claimed that some of the most severe surfaces changes on several components were not caused by FAC but were a result of manufacturing defects called ligaments. Apparently, those ligaments were not detected in such components when they were installed 40 years ago. Notably, since the presence of these ligaments was not known, the CUF of record would not have included the increase in peak stress resulting from such rough surfaces. In this respect, the CLB CUF would underestimate the actual CUF. Importantly, FAC is not the only mechanism that would cause changes in surface finish; in addition, pitting and radiation-induced void swelling could also create local discontinuities. Notably, most of the discontinuities due to wall thinning are expected to occur in components that are located on the balance-of-plant side (i.e., the secondary steam generator side), where Entergy claims [redacted] without referencing any analysis that would justify their exclusion from the final refined analysis, as discussed further below.

The effect of surface finish on the synergistic effects of fatigue and SCC could be accounted for by considering that an increase in surface roughness would reduce the crack nucleation or initiation time once a crack has been initiated, whether it would propagate by a steady or a cycling load. The fatigue life would depend on the respective load intensities and cycling frequency.

It would be reasonable to multiply all CUFs of record with potential for surface change by a factor of 10 (in relation to reduction in fatigue life) to account surface roughness and for real life effects of surface deterioration during service. [redacted]

### *4.3 Heat Transfer*

As discussed in previous submissions, a major source for errors in [redacted]

[64] *See* S. McKelvey & A. Fatemi, "Effect of Forging Surface on Fatigue Behavior of Steels: A Literature Review (University of Toledo) (Exhibit RIV000155) (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).

[65] NUREG/CR-6909, ANL-06-08, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials (2007) at § 4.1.6 (Exhibit NYS000357).

[66] *See generally* Report of Dr. Joram Hopenfeld in Support of Contention NYS-26-B/RK-TC-1B – Metal Fatigue (December 22, 2011), at 13-18 (Exhibit RIV000035).

[redacted]

Heat transfer calculations referenced by Entergy have showed that a 2-D model was used for the analysis of the temperature distribution along the reactor vessel nozzles.[67] The model employed assumed an oversimplified uniform heat transfer coefficient along the nozzle. However, the heat transfer is not uniform along the nozzle and can vary by 20%-30% depending on local turbulence along the nozzle. The resulting temperature gradients would require a much more detailed analysis than can be performed with a 2-D model. [redacted] Thermal stratification has a major effect on fatigue life of certain reactor components because of the large temperature differences (300°F) that may exist between different parts of a pipe. Stratification was not considered when PWRs were designed. It was only after plants experienced severe cracking in the late 1980s that fatigue due to stratification and thermal striping became a concern.[68] The pressurizer surge line is most vulnerable to fatigue failure from thermal striping.

[redacted] Thermal striping results from random temperature fluctuations at the interface of the stratified fluid layer. A

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[67] Combustion Engineering, Inc., CENC-1110 (Exhibit RIV000052A-D).

[68] *See* NRC Bulletin No. 88-11: Pressurizer Surge Line Thermal Stratification (December 20, 1988) (Exhibit RIV000115).

[69] *See* Westinghouse, Indian Point Unit 2 and Unit 2 EAF Screening Evaluations, Calculation Note Number CN-PAFM-12-35 (November 2012), IPECPROP00072778 (Exhibit NYS000510); Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, CN-PAFM-13-32, IPECPROP00078338 (2013) (Exhibit NYS000511).

[70] Testimony of Entergy Witnesses Nelson F. Azevedo, Alan B. Cox, Jack R. Strosnider, Robert E. Nickell, and Mark A. Gray Regarding Contention NYS-26B/RK-TC-1B (Metal Fatigue) (March 29, 2012), at A.96 (Exhibit ENT0000183).

[71] *Id.* (Citing Westinghouse, WCAP-9693, Investigation of Feedwater Line Cracking in Pressurized Water Reactor Plants (June 1980) (Exhibit ENT000217)).

[72] *See id.*

[redacted]

### *4.4 Strain Rate*

NUREG/CR-6909 shows that strain rate for carbon and low alloy steel has a significant effect on fatigue life.[73] Reduction of the strain rate from 1 to 0.0001/sec reduces the fatigue life by more than an order of magnitude. [redacted] A look at Equation 1 above shows that when the $F_{en}$ is constant, the $CUF_{en}$ would depend on the strain rate only by virtue of its dependence on the CUF. [redacted] When the original calculations of the CUFs of record were made, it was unknown that the maximum reduction in fatigue life would occurs at low strain rates. Therefore, using the CUF of record without a correction for conservatism with respect to the strain rate would also introduce non-conservatism in the $CUF_{en}$ values.

### *4.5 Radiation Effects*

As discussed above and in my supporting declaration to the amended bases of contention RK-TC-5,[74] [redacted] In addition, [redacted] which represents another way in which the $CUF_{en}$ evaluations are non-conservative and inaccurate.

[redacted] When the CLB CUFs were originally calculated, there was little data on the effects of radiation on mechanical properties, and, therefore, the long-term effects of radiation generally were not included. The stress concentration factors in the CLB CUFs were not based on changes in ductility due to radiation. However, the significant effects of radiation on the stress strain relation has been documented.[75] [redacted]

[73] NUREG/CR-6909, ANL-06-08, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials (2007) at 23 (Figure 12) (Exhibit NYS000357).
[74] Declaration of Dr. Joram Hopenfeld (February 12, 2015) (Exhibit RIV000148).
[75] *See, e.g.*, K L Murty, Materials Ageing and Degradation in Light Water Reactors (Woodhead Publishing Ltd. 2013), at 12 (Exhibit RIV000145).

[redacted]

### *4.6 Unexplained Changes to the CUF*

Table 1 below is a sample of the changes that were made to the CUFs of record from Entergy's screening analysis to Entergy's refined fatigue analysis:

***Table 1:***

| Component | CUF of record from Screening Analysis* | $CUF_{en}$ Values from Screening Analysis* | CUF of record from Refined Analysis** | $CUF_{en}$ Values from Revised Analysis** |
|---|---|---|---|---|
| [redacted] | [redacted] | [redacted] | [redacted] | [redacted] |
| [redacted] | [redacted] | [redacted] | [redacted] | [redacted] |
| [redacted] | [redacted] | [redacted] | [redacted] | [redacted] |
| [redacted] | [redacted] | [redacted] | [redacted] | [redacted] |

* Westinghouse, Indian Point Unit 2 and Unit 2 EAF Screening Evaluations, Calculation Note Number CN-PAFM-12-35 (November 2012), IPECPROP00072778, at Table 5-1, Table 5-2 (Exhibit NYS000510).
** Westinghouse, Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, CN-PAFM-13-32, IPECPROP00078338 (2013), at Table 2-1, Table 2-2 (Exhibit NYS000511).

Based on a review of the various Westinghouse's fatigue calculation documents, and given the uncertainties in the effects of radiation and SCC on fatigue, [redacted]

[redacted]

[76] *See* Westinghouse, Indian Point Unit 2 Reactor Vessel Core Support Pad Fatigue Reevaluation, CN-MRCDA-13-11, IPECPROP00079179, at 8, 9 (Figure 5-1) (Exhibit RIV000156).

[redacted] Furthermore, the stress concentration factors used in the CUF CLB calculations do not account for ductility changes from exposure to radiation.

[redacted] Also, as discussed above, the CUF of record does not include the effects of radiation.

[redacted]

## 5. <u>Inadequate Determination of the Most Limiting Locations</u>

In previously submitted testimony, I have explained the need for Entergy to expand the fatigue analysis for IP2 and IP3 beyond the locations identified in NUREG/CR-6260.[80] Entergy initially failed to undertake such an expanded analysis of additional components (including those in the steam generators), claiming that its analysis was conservative. However, after NRC Staff indicated that it was necessary for Entergy to confirm and/or determine the most limiting, bounding locations at Indian Point, Entergy finally agreed to conduct a screening and potentially expanded analysis.[81]

Notably, it is also beyond clear that Entergy's initial position that its fatigue results were conservative and bounding was wrong, since [redacted]

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[77] Pilkey & Pilkey, Peterson's Stress Concentration Factors, 3rd Edition, ISBN: 978-0-470-04824-5 (2008) (Exhibit RIV000157).

[78] Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, CN-PAFM-13-32, IPECPROP00078338 (2013) at 28-29 (Exhibit NYS000511).

[79] *See id.* at Tables 2-1 to 2-4.

[80] Prefiled Written Testimony of Dr. Joram Hopenfeld Regarding Contention NYS-38/RK-TC-5 (June 19, 2012) (Exhibit RIV000102).

[81] *See* NRC Letter, "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Numbers 2 and 3, License Renewal Application" (February 10, 2011) (Exhibit NYS000199); NUREG-1930, Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Supplement 1 (August 2011) (Exhibit NYS000160) (discussing Commitment 43).

[redacted] as shown by the above discussion.

In any event, Entergy's efforts to expand its analysis to confirm and justify the limiting locations at IP2 and IP3 are inadequate and incomplete. In previous testimony I explained at length the steps necessary to conduct a properly expanded analysis.[83] For example, I stressed the need to consider components that are vulnerable to thermal stratification and thermal striping.[84] [redacted] Thus, Entergy has continued to improperly fail to conduct the necessary expanded analysis.

Importantly, Entergy's analysis did not consider the effects of the environment on fatigue for components on the secondary side of the steam generators. Even a minimal correction for environmental effects would cause the $CUF_{en}$ values of some components on the secondary side to exceed one.

The following sections discuss components that Entergy continued to ignore, but which warrant fatigue review, which renders Entergy's analysis insufficient to demonstrate that fatigue will be adequately managed during the proposed periods of extended operation.

### *5.1 Pressurizer Surge Line, Mixing Tees, Unisolable Branches Connected to RCS Piping*

Entergy did not select the pressurizer surge line for its refined analysis. However, thermal stratification can occur in that component, and as such, fatigue analysis should have been conducted. As discussed above, the 30-year old laboratory tests of the steam generator feed line and [redacted] do not justify the exclusion of the pressurizer surge line from fatigue evaluation.

[redacted]

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[82] Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, CN-PAFM-13-32, IPECPROP00078338 (2013), at Tables 2-1 to 2-4 (Exhibit NYS000511).

[83] Prefiled Written Testimony of Dr. Joram Hopenfeld Regarding Contention NYS-38/RK-TC-5 (June 19, 2012) (Exhibit RIV000102).

[84] *Id.* at 12.

[85] *See* Testimony of Entergy Witnesses Nelson F. Azevedo, Alan B. Cox, Jack R. Strosnider, Robert E. Nickell, and Mark A. Gray Regarding Contention NYS-26B/RK-TC-1B (Metal Fatigue) (March 29, 2012), at 110-111 (citing MRP-47 at 4-23) (Exhibit ENT0000183).

[redacted]

[redacted] he relevant parameters are the high local stress intensities that are created when the surface is subjected to local and repeated temperature changes. High local stress concentrations act as crack initiation sites eventually leading to a network of propagating cracks.

In sum, Entergy did not provide a valid reason for excluding from fatigue analysis components that are vulnerable thermal striping.

### *5.2 Steam Generator Tubes and Steam Generator Secondary Side*

[redacted]

Importantly, the fact that [redacted] Tube fatigue must be based on specific locations. As plants age, more and more crud is accumulated inside the support plates; at some point in time crud accumulation, especially at the upper support plate, would increase the local secondary flow velocity. A sufficiently high velocity in combination with the local geometry could cause fluidelastic instability resulting in high amplitude tube vibration accompanied by wear and fatigue. Tube failures from high cycle fatigue has been discussed and documented in scholarly work.[88] High cycle tube fatigue is an aging phenomenon and must be incorporated into Entergy's Fatigue Monitoring Program. [redacted].

In addition, [redacted] ntergy should have conducted an

[86] Westinghouse, Indian Point Unit 2 and Unit 2 EAF Screening Evaluations, Calculation Note Number CN-PAFM-12-35 (November 2012), IPECPROP00072778, at 41 (Exhibit NYS000510).
[87] *See* Westinghouse, Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, CN-PAFM-13-32, IPECPROP00078338 (2013) at Tables 2-1 to 2-4 (Exhibit NYS000511).
[88] *See* Herve Bodineau & Thierry Sollier, Tube support plate clogging up of French PWR steam generators, Eurosafe, IRSN – Reactor Safety Division, BP17 (2009) (Exhibit RIV000158).
[89] Westinghouse, Indian Point Unit 2 and Unit 2 EAF Screening Evaluations, Calculation Note Number CN-PAFM-12-35 (November 2012), IPECPROP00072778, at Table 5 (Exhibit NYS000510).

expanded refined fatigue analysis for such components. Entergy's ongoing failure to conduct refined analyses for the various components on the secondary side of the steam generators that I previously listed[90] is unacceptable and renders Entergy's program for managing metal fatigue inadequate.

### *5.3 Reactor Head Penetrations, Outlet Inlet Nozzle Safe Ends*

Alloy 600/82/182 are susceptible to primary water SCC which can be expected to affect metal fatigue. Yet, no documentation or discussion by Entergy/Westinghouse is apparent in relation to how the effects of SCC were considered in the determination of limiting locations.

## 6. Summary Assessment and Safety Implications of Entergy's Results

[redacted]

[redacted] various scientific studies show that radiation significantly reduces fatigue life and that, therefore, [redacted]

[redacted]

[90] Prefiled Written Testimony of Dr. Joram Hopenfeld Regarding Contention NYS-38/RK-TC-5 (June 19, 2012), at 14-15 (Exhibit RIV000102).

[91] Westinghouse, Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, CN-PAFM-13-32, IPECPROP00078338 (2013) at Table 2-1 (Exhibit NYS000511).

[92] *See* Westinghouse, Indian Point Unit 2 and Unit 2 EAF Screening Evaluations, Calculation Note Number CN-PAFM-12-35 (November 2012), IPECPROP00072778, at 38 (Exhibit NYS000510).

Entergy's documentation claims that the $CUF_{en}$ results are precise and conservative and that, therefore, as Entergy has maintained throughout the Indian Point license renewal proceeding, an uncertainty discussion and error analysis is not required. This is incredulous given the degree of uncertainty that exists, as described above, [redacted]

[redacted]

Moreover, Entergy's fatigue evaluations remain incomplete and inadequate since Entergy failed to properly expand the scope of its fatigue analysis for IP2 and IP3. Notably, Entergy refused to include components on the secondary side of the steam generators in its fatigue analyses. This directly disregards industry guidelines, which require the inclusion of locations where high usage might be of concern.[94] Steam generator tubes, support plates, feedwater nozzles, and related piping are examples of such locations.

An uncertainty and error analysis would have exposed Entergy's unconventional, unscientific, and unreal theories on dissolved oxygen, radiation effects, the use of CLB CUFs of record, and thermal striping. Even accepting all these clearly misguided theories, a minimal correction of a factor of 3, which is inherent in the experimental, best-fit, $F_{en}$ equation alone,[95] [redacted]

It is clear that the competent analysis called for in Contentions RK-TC-1 and RK-TC-5 continues to be ignored. Instead of using the best available input data, incorrect inputs were selected and used to obtain the desired outcomes. As discussed above, radiation, DO, heat transfer coefficients, strain rates, geometrical changes, stress corrosion, surface roughness, and stress concentration factors were all used incorrectly and/or inadequately considered. For example, [redacted]

[93] *Id.* at Table 2-3.
[94] *See* EPRI, MRP-47, Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, Final Report, Revision 1 (September 2005) (Exhibit NYS000350).
[95] NUREG/CR-6909, ANL-06-08, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials (2007), at 26 (Exhibit NYS000357).
[96] Westinghouse, Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, CN-PAFM-13-32, IPECPROP00078338 (2013) at Tables 2-1 to 2-4 (Exhibit NYS000511).

Entergy believes that its Fatigue Monitoring Program will ensure that the number of plant transients will not exceed the number of cycles that were used in the fatigue analysis, so the $CUF_{en}$ values will remain less than 1.0. The Fatigue Monitoring Program requires that Entergy implement corrective action, including repair or replacement of components where the $CUF_{en}$ exceeds 1.0. However, such a strategy of managing fatigue at Indian Point would only work if the calculated $CUF_{en}$ values bore some measure of certainty and were conservative. The evaluation presented in this report shows that this is not the case. The $CUF_{en}$ values derived by Entergy/Westinghouse are non-conservative, in most cases by at least an order of magnitude. Furthermore, the foregoing assessment demonstrates that it is [redacted] Entergy's defense of their latest calculated $CUF_{en}$ values reflects a profound lack of understanding of the $CUF_{en}$ underlying methodology. The Fatigue Monitoring Program cannot function reliably without a thorough understanding of the uncertainties in the $CUF_{en}$s.

It is Entergy's position that because all the $CUF_{en}$ values are less than one, plant safety will not be compromised by metal fatigue, and that a safety evaluation related to metal fatigue is not required. However, based on the above discussion, Entergy has clearly failed to make this demonstration.

Importantly, NRC safety studies do not include an evaluation of the potential contribution of RVI failures by fatigue to core damage frequency (CDF).[97] [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 CDF per reactor year.[98] The failure of some key safety components during DBAs, such as a Steam Line Break (SLB), would exceed this Commission Guideline.

Contentions RK-TC-1 and RK-TC-5 emphasize that reactor operations with fatigued components—$CUF_{en}$ >1.0—makes the reactor cores at IP2 and IP3 vulnerable to a meltdown followed by large radioactivity releases to the environment both during DBAs and beyond DBAs, such as station blackouts ("SBO") and anticipated transients without scram ("ATWS"). Entergy has continued to improperly fail to adequately discuss how uncertainties in the $CUF_{en}$ calculations can impact plant safety. Apparently, [redacted]

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[97] *See* NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants (1990), *available at*, http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1150/ (Exhibit RIV00146A-RIV000146B).

[98] *See* NUREG/BR-0058, Rev.4, Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission (2004), *available at* http://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0058/br0058r4.pdf (Exhibit RIV000159); U.S. NRC, Commissioner Apostolakis, Application of Risk Assessment and Management to Nuclear Safety (DOE Nuclear Safety Workshop, September 2012), *available at*, http://energy.gov/sites/prod/files/2013/12/f5/Apostolakis.pdf, at 5 (Exhibit RIV000160).

[redacted] his is belied by the above assessment.

## CONCLUSIONS

Figures 4 and 5 below summarize the only conclusions that can be drawn regarding Entergy's latest fatigue evaluations based upon the forgoing discussion on the effects of DO, radiation, and heat transfer on the fatigue life predictions as expressed by the $CUF_{en}$

Figure 4 shows how [redacted] Entergy's approach guarantees that the results are not conservative or accurate:

**Figure 4: Key Parameters that Contributed to Entergy's Non Conservative Predictions of Fatigue Life**

| INPUT | FLAWED VALUE USED | MATERIAL | IMPACT ON $CUF_{en}$ (Conservative: Yes/No) | $CUF_{en}$ REFINED ANALYSIS* |
|---|---|---|---|---|
| DISSOLVED OXYGEN | [redacted] | CARBON STEEL | No | [redacted] |
| | | LOW ALLOY STEEL | No | |
| | | STAINLESS | Yes | |
| RADIATION | | STAINLESS | No | |
| CUF | | STAINLESS | No | |
| SCC | | STAINLESS | No | |
| Surface Roughness | | | No | |

*Westinghouse, Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, CN-PAFM-13-32, IPECPROP00078338 (2013), at Tables 2-1 to Table 2-4

Figure 5 shows that when more realistic assumptions are made related to DO, radiation, and the other parameters discussed in my previous submissions and herein, [redacted]:

[99] Westinghouse, Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations, CN-PAFM-13-32, IPECPROP00078338 (2013) at Tables 2-1 to 2-4 (Exhibit NYS000511).

***Figure 5: Comparison of Entergy's Fatigue Evaluations with Requirements Discussed in RK-TC1 and RK-TC5***

| Requirements/Goals | Entergy Refined Analysis | Findings or Position taken in Contention RK-TC-1 and/or RK-TC-5 | Comments |
|---|---|---|---|
| NUREG -1801: $CUF_{en}$<1.0<br><br>ASME: CUF<1.0 | [redacted] | $CUF_{en}$ > 1.0 for many components analyzed | RK-TC1/RK-TC5 results based on Industry Guidance (NUREG/CR-6909 at A5; EPRI MRP-47 at 4-21; Mohanty (2012)<br><br>[redacted] |
| Core Melt Frequency Goal < $10^{-5}$ R/yr | No safety analysis disclosed or disclosed | potential > $10^{-5}$ R/yr (for example, for Steam Line Break) | Goal based on Commission Guidelines (*See* U.S. NRC, Commissioner Apostolakis, Application of Risk Assessment and Management to Nuclear Safety (DOE Nuclear Safety Workshop, September 2012) |

NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, intended that the $F_{en}$ methodology employed be based on a technical analysis so that the calculated $CUF_{en}$ values could be used reliably to manage fatigue. Entergy has failed to do this. The foregoing evaluation of Energy's fatigue analyses, in conjunction with my previous expert submission related to contentions RK-TC-1 and RK-TC-5, show that the results were not based on a technical considerations. They were fabricated for appearance and regulatory consumption and, therefore, cannot be used to manage fatigue during the proposed periods of extended operation.

The $CUF_{en}$ values represent a creation of numbers that cannot be used to predict fatigue life of reactor components. This renders Entergy's fatigue analyses insufficient to assure that aging effects will be adequately managed at IP2 and IP3.

The accuracy of the calculated $CUF_{en}$s necessarily affects the reliability of the aging management program for metal fatigue. 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/CR-6260. Since the validity and scope of the fatigue monitoring plan depends on the accuracy of the $CUF_{en}$ values, 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 increase the possibility that fatigue susceptible components would remain in service due to inadequately chosen inspection intervals.

Furthermore, because Entergy's screening evaluation was flawed in numerous respects, Entergy has yet to conduct a bounding analysis and has continued to fail to adequately expand the scope of its fatigue analysis as is necessary.

In light of the absence of comprehensive, accurate metal fatigue calculations to properly guide Entergy's aging management efforts, Entergy has failed to define specific criteria to assure that susceptible components are inspected, monitored, repaired, or replaced in a timely manner. Once components with high CUFs have been properly identified, Entergy must describe a fatigue management plan for each such component that should, at a minimum, rank components with respect to their consequences of failure, establish criteria for repair versus defect monitoring, and establish criteria for the frequency of the inspection, and corrective actions.

In light of the foregoing, Entergy has failed to demonstrate that it has an adequate 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*.

June 8, 2015

Rockville, MD

Joram Hopenfeld

Dr. Joram Hopenfeld
1724 Yale Place
Rockville, MD 20850

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