RIV000035
Submitted: December 22, 2011

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

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 Report of Dr. Joram Hopenfeld dated December 20, 2011 as containing Confidential Proprietary Information Subject to Nondisclosure Agreement

REDACTED, PUBLIC VERSION

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REPORT OF

DR. JORAM HOPENFELD

IN SUPPORT OF

CONTENTION RIVERKEEPER TC-1B – METAL FATIGUE

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I. Introduction

I have been retained by Riverkeeper, Inc. ("Riverkeeper") as an expert witness in proceedings concerning the application by Entergy Nuclear Operations, Inc. ("Entergy") for the renewal of two separate operating licenses for the Indian Point nuclear generating facilities, Units 2 and 3, for twenty years beyond their current expiration dates. Indian Point nuclear generating facility is located on the east bank of the Hudson River in the Village of Buchanan, Westchester County, New York. I was asked to review Entergy's Aging Management Program ("AMP") concerning metal fatigue of reactor components, and to assess whether Entergy has demonstrated that this AMP will manage the aging effects of metal fatigue during the proposed period of extended operation.

I am an expert in the field relating to nuclear power plant aging management. I am a mechanical engineer and hold a doctorate in mechanical engineering. I have forty-five years of professional experience in the fields of thermal-hydraulics, material/environment interaction instrumentation, design, project management, and nuclear safety regulation, including 18 years in the employ of the U.S. Nuclear Regulatory Commission ("NRC"). My extensive professional experience has afforded me with knowledge and expertise regarding the material degradation phenomenon of metal fatigue.

I assisted Riverkeeper with the preparation of Contention TC-1, Amended Contention TC-1A, and Amended Contention TC-1B, all of which challenged Entergy's aging management plan for addressing metal fatigue at Indian Point during the extended operating terms. The Atomic Safety and Licensing Board admitted Riverkeeper Contention TC-1, TC-1A, and superseding TC-1B.

I have reviewed the documents identified by Entergy as relevant to Riverkeeper's contentions concerning metal fatigue, as well as NRC guidance documents, technical reports, applicable regulations, and other documents relevant to these contentions. A list of documents I reviewed in reaching my conclusions is included at the end of this report.

Based on my review of these documents and my forty-five years of professional experience, I have concluded that Entergy has, to date, failed to demonstrate that the affects of metal fatigue will be adequately managed at Indian Point during the proposed period of extended operation. The basis for my conclusion is explained forthwith.

II. Background

A. Basic Principles Concerning Metal Fatigue

The aging phenomenon of fatigue refers to when a structure or test specimen is subjected to repeated, "cyclic," loading during plant operation. Under such cyclic loading a crack will be initiated and the structure will fail under stresses that are substantially lower than those that cause failure under static loadings. Material composition, strain rate, temperature and local water chemistry are some of the factors that contribute to fatigue of metal parts. During each loading cycle, a certain fraction of the fatigue life of a component is used up depending on the magnitude of the applied stress. Eventually, after the number of allowable cycles, N, the

structure will use all its fatigue life. The number of cycles actually experienced at any given stress amplitude, n, divided by the corresponding number of allowable cycles, N, is called the usage fatigue factor. The cumulative usage fatigue factor, CUF, is simply a summation of the individual usage factors. The CUF is expressed as,

$$CUF = \sum n_k / N_k$$

The maximum number of cycles that should be experienced by any structure or component should always result in a CUF that does not exceed 1.0, or unity, meaning that the number of actual cycles experienced should always be less than the number of allowable cycles.

The basic equation that describes the crack growth rate for a given stress intensity, includes two empirical constants, C and x. The equation can predict crack growth reliably as long as it is used under the conditions that were used to calibrate C and x.[1] A large database of the empirical constants, C and x, were derived from laboratory tests in air under controlled conditions. In order to account for crack propagation in water (i.e., in the actual reactor environment), which is different than crack growth rates in air, the individual usage factor in air is multiplied by a corresponding environmental correction factor, "$F_{en.}$" F_{en} is simply the ratio of the fatigue life in air at room temperature to the fatigue life in water at the local temperature. The partial environmentally corrected fatigue usage factor during any given stress cycle k, $U_{en \cdot k}$ is given by

$$U_{en,k} = F_{en,k} \cdot U_{,k}$$

The cumulative fatigue usage factor, CUF_{en} that incorporates the effects of the reactor coolant environment throughout the life of the component is then given by,

$$CUF_{en} = \Sigma U_{en,k}$$

F_{en} is derived from laboratory data on strain versus fatigue life, that is, the number of cycles compared to failure. NUREG/CR-6909, *Effect of LWR Coolant Environment on Fatigue Life of Reactor Materials*, Final Report (February 2007) (hereinafter "NUREG/CR-6909"), describes the laboratory tests, which were conducted under controlled conditions at the Argonne National Laboratory ("ANL"), to generate F_{en} factors. Because laboratory tests were not prototypic of the reactor environment, ANL provided a detailed discussion of the required adjustments to be made 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 (shown in a functional form) was derived from tests under controlled and known conditions. This equation specifies the Fen when the test specimen is exposed to specific values of the four variables at any given time. Accordingly, this equation cannot be used

[1] This principle is not unique to engineering and equally applies to other scientific fields such as statistics. The understanding of this principle is very important in assessing how Entergy used laboratory data to calculate fatigue life of selected components at the Indian Point nuclear power plant.

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.

Section III of the American Society Mechanical Engineers ("ASME") Code specifies the procedures for analyzing components for fatigue. The Code provides fatigue curves in air for various materials which specify the allowable number of cycles for a given stress intensity. The Code requires that the CUF at any given location be maintained below one.[2] Since the Code used data from laboratory tests with smooth specimens, it made allowance (2 on stress and 20 on cycles) in recognition that test specimens in air may have a longer fatigue life than actual components due to material variations, surface characteristics and component size.

The present ASME code also provides a simplified set of rules in Subparagraph NB-3600 and a more rigorous rule in Subparagraph NB-3200, which are based on using a finite element analysis. Replacing the simplified analysis by a more detailed analysis has the advantage that it can remove unwanted conservatism from the results of the simplified analysis. Since detailed analysis may require a larger database than the simplified analysis, the user must ascertain that such a database exists. When such information is not available and the user instead makes arbitrary assumptions, the benefit of the detailed analysis is completely negated.

B. Safety Implications of Metal Fatigue

Equipment failures from fatigue may result in small leaks which if not detected in time could result in a pipe rupture. Fatigue may also create small cracks that propagate and cause a given component to malfunction and/or break up and form loose parts which would interfere with the safe operation of the plant. Such failures may occur during steady state or during anticipated or unanticipated transients and may have serious consequences to public health and safety. For example, if one of the feed water distribution nozzles (J tubes) were to fail from fatigue, pieces from the broken nozzle could be lodged between steam generator tubes, causing the tubes to rupture and leading to a potential core melt. Components which are susceptible to fatigue must, therefore, have a planned management program to ensure that the plant functions efficiently and safely.

III. NRC Regulations

NRC regulation 10 CFR § 54.21(c) requires that each license renewal application ("LRA") must include "an evaluation of time-limited aging analyses" ("TLAA") for components covered by the license renewal regulations.[3] If the applicant is unable to demonstrate that TLAAs "remain valid

[2] It is important to understand that the ASME code only requires that the CUFen not exceed one. It makes no difference whether the CUFen is 1.01 or 10 – both are equally unacceptable. Importantly, there is no correlation between the degree to which the CUFen exceeds unity and fatigue life.

[3] TLAAs are defined as: "Those licensee calculations and analyses that: (1) Involve systems, structures, and components within the scope of license renewal, as delineated in § 54.4(a); (2) Consider the effects of aging; (3) Involve time-limited assumptions defined by the current operating term, for example, 40 years; (4) Were determined to be relevant by the licensee in making a safety determination; (5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure and component to perform its intended functions, as delineated in § 54.4(b); and (6) Are contained or incorporated by reference in the CLB [current licensing [basis]." 10 C.F.R. § 54.3.

for the period of extended operation" or that they "have been projected to the end of the period of extended operation," the applicant must demonstrate that "the effects of aging on the intended function(s) will be adequately managed for the period of extended operation."[4]

IV. Entergy's 2010 "Refined" CUF_{en} EAF Analyses

LRA Tables 4.3-13 and 4.3-14 presented the results of Entergy's analysis of the effects of environmental assisted metal fatigue on certain reactor components during the proposed period of extended operation. These results showed that the environmentally corrected cumulative usage factor, CUF_{en} of four risk significant reactor components would exceed unity during the period of extended operation. In an attempt to demonstrate an AMP sufficient to meet applicable regulatory requirements, Entergy committed to performing a refined fatigue analysis in order to lower the predicted CUF_{en} values to less than 1.0.[5] Entergy submitted the results of this "refined" environmentally assisted fatigue ("EAF") analysis in August 2010.[6] Entergy's submission included revised Tables 4.3-13 and 4.3-14, which reported that the CUF_{en} values for all locations were below 1.0. This new analysis only assessed those locations identified in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (1995) (hereinafter "NUREG/CR-6260").[7]

However, the methodology employed to calculate Entergy's new CUF_{en} values is highly suspect and questionable, calling into question the validity of the results. While Entergy describes the general methodology employed to derive the revised calculations, many critical underlying assumptions reveal the potential for a wide margin of error. [redacted] calculations have likely grossly under-predicted the CUFen values for the components evaluated.

A. *Laboratory Data*

[redacted]

[4] 10 C.F.R. § 54.21(c)(1)(i)-(iii).

[5] NL-08-021, Entergy Nuclear Operations, Inc., Indian Point Nuclear Generating Unit Nos. 2 & 3, Docket Nos. 50-247 and 50-286, License Renewal Application Amendment 2 (January 22, 2008), ADAMS Accession No. ML080290659.

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

[7] Riverkeeper and New York State proffered a new contention alleging various deficiencies with this new analysis, including the fact that Entergy did not analyze any additional components beyond those suggested in NUREG/CR-6260, even though such an expanded scope of analysis was warranted given Entergy's original analysis, which showed various CUFens for NUREG/CR-6260 components would exceed unity. *See* Petitioners State of New York and Riverkeeper, Inc. New and Amended Contention Concerning Metal Fatigue (New York State-26-B/Riverkeeper TC-1B (Metal Fatigue) (Sept. 9, 2010).

[redacted][8] As discussed above, these F_{en} equations, developed by ANL, were derived from laboratory tests on the effect of strain and coolant environments on fatigue life. In such equations, F_{en} is expressed in terms of DO, temperature, sulfur content, and strain rate (F_{en} = f(T, DO, S, e)) for several materials of interest.

Because significant differences exist between the laboratory and the reactor environment, there are numerous uncertainties in applying the F_{en} equations to actual reactor components. The Electrical Power Research Institute has explained that "[i]n many respects, the current state of the technology with respect to the F_{en} methodology is incomplete or lacking in detail and specificity."[9] NUREG/CR-6909 identified numerous uncertainties inherent in the determination of CUF_{en}:

> The variables that can affect fatigue life in air and LWR environments can be broadly classified into three groups: (a) Material (i) Composition (ii) Metallurgy: grain size, inclusions, orientation within a forging or plate (iii) Processing: cold work, heat treatment (iv) Size and geometry (v) Surface finish: fabrication surface condition (vi) Surface preparation: surface work hardening (b) Loading (i) Strain rate: rise time (ii) Sequence: linear damage summation or Miner's rule (iii) Mean stress (iv) Biaxial effects: constraints (c) Environment (i) Water chemistry: DO, lithium hydroxide, boric acid concentrations (ii) Temperature (iii) Flow rate.[10]

Thus, identifying the relevant terms of the F_{en} equation is only the beginning of the inquiry. To appropriately *apply* the F_{en} equations to actual reactor components, the user must consider the numerous uncertainties identified in NUREG/CR-6909, and the results must be adjusted to account for the varying parameters. That is, the user must take into account the fact that the equations were not derived on prototypic components in a prototypic environment. The developer of the F_{en} equations, Dr. O. K. Chopra has explained this to the Advisory Committee on Reactor Safeguards ("ACRS"): "To apply those results [i.e., laboratory data] to actual reactor components we need to adjust these results to account for parameters or variables which we

[8] Westinghouse, "EnvFat User's Manual Version 1," May 2009, at § 2 (IPECPROP00056785); Environmental Fatigue Evaluation for Indian Point Unit 2, WCAP-17199-P, Revision 0 (Westinghouse, June 2010), IPECPROP00056486; Environmental Fatigue Evaluation for Indian Point Unit 3, WCAP-17200-P, Revision 0 (Westinghouse, June 2010), IPECPROP00056577.

[9] EPRI's Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, MRP-47, at 4-25 (hereinafter cited as "MRP-47").

[10] NUREG/CR-6909 at 72; *see also id.* at 13, 59 (discussing data scatter); at 14, 34-35 (discussing surface finish); at 62 (discussing size); at 33 (discussing flow rate); at 12, 38-40, 57 (discussing strain rate); at 36 (discussing heat to heat variation); at 62 (discussing loading history, mean stress); at 13 (discussing cyclic strain hardening); at 28 (discussing temperature below 150°C); at A-5 (discussing oxygen); at 30-31, 55 (discussing trace impurities/water conductivity); at 13 (discussing sulfide morphology).

know affect fatigue life, but are not included in this data. And these variables are mean stress, surface finish, size, loading history."[11]

The uncertainties identified in NUREG/CR-6909 can have a significant affect upon fatigue life, and ignoring such uncertainties will result in improperly underestimated CUF_{en} calculations. Several examples illustrate this: (1) Variations in temperature can greatly affect fatigue life. In particular, while the F_{en} equation for carbon predicts a low F_{en} when the temperature value is below 150°C,[12] even when temperature is below 150°C, fatigue life could be reduced by a factor of two,[13] necessitating a larger F_{en}. Thus, when temperature during the transient falls below 150°C, proper procedure dictates that the F_{en} values be increased by a factor of two. In any event, as transients involve variable temperature conditions, NUREG/CR-6909 recommends using an average temperature[14]; (2) increased water conductivity due to the presence of trace anionic impurities in the coolant may decrease fatigue life of austenitic stainless steels.[15] NRC has documented several occurrences at nuclear plants of stress corrosion cracking ("SCC") of stainless steel components due to the presence of trace amounts of chlorides (i.e., anionic impurities) in the water.[16] NRC has explained that "as nuclear plants age, SCC can become an emergent degradation mechanism in PWRs [pressurized water reactors] for environments that contain chlorides and or stagnant flow conditions."[17] Thus, the presence of trace impurities in the coolant must be appropriately accounted for in the determination of CUF_{en}; (3) Variation in sulfide morphology, at a low strain rate, may result in a difference by an order of magnitude in fatigue life[18]; (4) Cyclic strain hardening is an uncertainly, as it is not included in the ASME Code design fatigue curves[19]; and (5) As explained in further detail below, oxygen concentrations during transients can have a considerable impact on CUF_{en} calculations. Additionally, though not discussed in NUREG/CR-6909, surface temperature fluctuations and non-uniform temperature distributions during stratification in components such as the surge line can increase the potential for crack initiation and growth, thereby reducing fatigue life.

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[11] Official Transcript of Proceedings, Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards Subcommittee on Materials, Metallurgy and Reactor Fuels (December 6, 2006), at 22, ADAMS Accession No. ML063550058.

[12] NUREG/CR-6909 at 26-28.

[13] Dr. Chopra, developer of the Fen equations, has indicated this proposition to the ACRS. *See* Official Transcript of Proceedings, Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards Subcommittee on Materials, Metallurgy and Reactor Fuels (December 6, 2006), at 25, ADAMS Accession No. ML063550058.

[14] NUREG/CR-6909 at 40 ("An average temperature for the transient may be used to estimate Fen during a load cycle"), at 56 ("load cycles involving variable temperature conditions may be represented by an average temperature.").

[15] *See* NUREG/CR-6909 at 30-31, 55.

[16] *See* NRC Information Notice 2011-04: Contaminants and Stagnant Conditions Affecting Stress Corrosion Cracking in Stainless Steel Piping in Pressurized Water Reactors (February 23, 2011), ADAMS Accession No. ML103410363 (hereinafter "NRC IN 2011-04").

[17] NRC IN 2011-04 at 4.

[18] *See* NUREG/CR-6909 at 13.

[19] *See* NUREG/CR-6909 at 13.

In recognition of the numerous uncertainties in using the F_{en} equations, NUREG/CR-6909 specifies appropriate bounding F_{en} values of 12 and 17 for stainless steel and carbon and low alloy steel, respectively.[20] These values are based on a review of data from different laboratory tests covering a much wider range of parameters than those considered in NUREG/CR-6583 and NUREG/CR-5704.[21] It is more appropriate to use these bounding values to account for the differences between the laboratory and reactor environments, than to ignore important differences all together. Moreover, these bounding F_{en} factors are not necessarily conservative. Many parameters that control fatigue life in an actual plant are different, including component size and thermal hydraulic conditions. Although F_{en} values of 12 and 17 were bounding in laboratory environments, it is reasonable to expect even higher F_{en} values in the actual reactor environment, especially for those components that experience stratified flows and thermal striping.

[redacted]

Instead, Entergy used unrealistically low F_{en} values[25] [redacted]

[20] NUREG/CR-6909 at iii, 3.

[21] *See* NUREG/CR-6909 at 3.

[22] Westinghouse, "EnvFat User's Manual Version 1," May 2009, IPECPROP00056785; Environmental Fatigue Evaluation for Indian Point Unit 2, WCAP-17199-P, Revision 0 (Westinghouse, June 2010), IPECPROP00056486; Environmental Fatigue Evaluation for Indian Point Unit 3, WCAP-17200-P, Revision 0 (Westinghouse, June 2010), IPECPROP000565772010).

[23] NUREG/CR-6909 at 26-28; Dr. Chopra, developer of the Fen equations, has stated indicated this proposition to the ACRS. *See* Official Transcript of Proceedings, Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards Subcommittee on Materials, Metallurgy and Reactor Fuels (December 6, 2006), at 25, ADAMS Accession No. ML063550058.

[24] *See* NUREG/CR-6909 at 30-31, 55; *see also* NRC IN 2011-04.

[25] *See* NL-10-082, Letter from Fred Dacimo, Entergy, to NRC Document Control Desk, "License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program" (August 9, 2010), (containing revisions to LRA Tables 4.3-13 and 4.3-14, with revised Fen values applied in Entergy "refined" calculation).

[redacted]

The use of the bounding F_{en}s articulated in NUREG/CR-6909, which represent far more realistic values than most of those calculated and used by Entergy, would increase the CUF_{en} values beyond unity for eight, (i.e., half) of the components analyzed, as follows:

Component	**Material**[26]	**F_{en}**	**CUF_{en}**
IP2 Vessel Outlet Nozzle	Low Alloy Steel	17	4.8
IP3 Vessel Outlet Nozzle	Low Alloy Steel	17	4.4
IP2 Surge Line Nozzle	Low Alloy Steel	17	1.4
IP3 Surge Line Nozzle	Low Alloy Steel	17	1.8
IP3 Charging System Nozzles	Stainless Steel	12	2.2
IP2 RCS Piping Injection System Nozzle	Stainless Steel	12	1.3
IP3 RCS Piping Injection System Nozzle	Stainless Steel	12	2.0
IP3 RHR Piping	Stainless Steel	12	1.5

These CUF_{en} values were calculated as follows:

For Low Alloy Steel Components: $\left(\frac{Entergy's revised\ CUfen\ value}{Fen\ factor\ applied\ by\ Entergy}\right) \times 17$

For Stainless Steel Components: $\left(\frac{Entergy's revised\ CUfen\ value}{Fen\ factor\ applied\ by\ Entergy}\right) \times 12$[27]

However, Entergy has previously claimed that using the NUREG/CR-6909 bounding F_{en} values "would actually yield less conservative CUF_{en} values, because the ASME Code design air curves for carbon steel and low-alloy steels contained in air in NUREG/CR-6583 and NUREG/CR-5704 are more conservative than the newer air curves in NUREG/CR-6909."[28] This position is absolutely untenable. Dr. W.E. Cooper, a leading authority on the ASME Code, has repeatedly stressed that the "conservatism" to which Entergy is referring was not intended to provide a safety margin.[29] To be sure, the current ASME code did consider a range of uncertainties and it incorporates a fatigue design margin of 2 on stress and 20 on cycles for carbon and low alloy steels *in air*, which many, though not all, researchers believe is overly conservative.[30] However,

[26] LRA Tables 4.3-13, 4.3-14 and revised Tables 4.3-13, 4.3-14 in NL-10-082, Letter from Fred Dacimo, Entergy, to NRC Document Control Desk, "License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program" (August 9, 2010).

[27] Dividing the CUFen values reported by Entergy by the Fen values applied by Entergy results in the uncorrected CUF value; as CUFen is equal to CUF×Fen, multiplying the uncorrected CUF value by the bounding Fen (17 or 12) is equal to the CUFen values Entergy would have arrived at if it had followed applicable guidance.

[28] 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.

[29] *See* NUREG/CR-6909 at 3. I have personally attended several lectures by Dr. Cooper where this point was stressed frequently.

[30] In contrast, for stainless steel in air, the ASME code is not conservative, as the Code predicts that fatigue life is longer than experimentally observed.

while the conservatisms of the ASME code account for real life variability in material, surface roughness, and component size, such conservatism does not mean that the calculations therefore included a conservative margin of safety. NUREG/CR-6909 specifically explains this: "the factors of 2 and 20 are *not safety margins* but rather adjustment factors that should be applied to the small–specimen data to obtain reasonable estimates of the lives of actual reactor components."[31] The factors considered in the ASME Code were

> [I]ntended to reflect the effects of an industrial atmosphere . . . not the effects of a specific coolant environment. Subsection NB–3121 of Section III of the Code explicitly notes that the data used to develop the fatigue design curves (Figs. I–9.1 through I–9.6 of Appendix I to Section III) did not include tests in the presence of corrosive environments that might accelerate fatigue failure." [32]

Thus, the ASME curves are valid in relation to fatigue life *in air*.

In contrast, NUREG/CR-6909 provides guidance pertaining to appropriate F_{en} factors to account for uncertainties inherent in CUF_{en} due to the reactor, i.e. *water*, environment. The uncertainties discussed in NUREG/CR-6909 are not reflected anywhere in the current ASME code. As NUREG/CR-6909 explains, "[l]aboratory data indicate that under certain reactor operating conditions, *fatigue lives of carbon and low–alloy steels can be a factor of 17 lower in the coolant environment than in air*. Therefore, *the margins in the ASME Code may be less conservative than originally intended*."[33] Thus, while the difference between NUREG/CR-6909 air curve and the ASME code air curve is on the order of a factor of 1.7, this is small in comparison to the many uncertainties in the F_{en}, as discussed above, [redacted] Thus, Entergy's apparent position that its reliance on the ASME code curves in NUREG/CR-6583 and NUREG/CR-5704 automatically results in a more conservative CUF_{en} value [redacted] is *completely* unfounded and simply wrong.

In summary, [redacted] Entergy did not apply the recommended bounding F_{en} factors, [redacted] and Entergy's assessment that no CUF_{en} for the evaluated components exceeds unity remains unsubstantiated.

B. Dissolved Oxygen

One of the largest uncertainties in determining appropriate Fen values is the concentration of dissolved oxygen ("DO") in the water at the surface of each component during the transient. The

[31] NUREG/CR-6909 at 3 (emphasis added).

[32] NUREG/CR-6909 at 3.

[33] NUREG/CR-6909 at 3 (emphasis added).

F_{en} varies exponentially with the DO level. For example, an increase in oxygen concentration by a factor of four, in comparison to steady state values, would increase the Fen by a factor of 55. The value of Fen is, therefore, sensitive to the uncertainties in DO concentrations. For example, the equations for determining F_{en} were experimentally derived under conditions where the temperature and DO at the surface of the specimen were known. In contrast, in a reactor plant, the DO in many cases is unknown. EPRI's Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, MRP-47 ("MRP-47"), discusses in great detail the difficulty in determining DO levels during transients:

> Although DO measurements are normally available through routine chemistry measurements, they are typically very limited with respect to frequency of collection and locations collected in the reactor coolant system (RCS). Therefore, there are several difficulties associated with determining the DO that is appropriate for use in the F_{en} expressions: The DO level is not known at the component location being evaluated. . . . The DO level is not known at all times during a transient (i.e., perhaps DO data is only collected once per day as opposed to continuously during a transient.[34]

This is in particularly true during startup and shutdown transients. During these transients, the oxygen content at the surface of the component varies significantly for two reasons; first, due to oxygen incursions from external sources, and second, due to the fact that DO has a negative solubility coefficient in water, a fundamental law of physics. The level of DO, therefore, increases significantly during shutdown transients. During startup transients, DO will be at a maximum at the beginning of the transient and then decrease towards its steady state value. Oxygen excursions occur during heatups (prior to startups).[35] Data of the Electrical Power Research Institute (EPRI) on actual oxygen concentrations in a Boiling Water Reactor ("BWR") during start up and shutdowns shows that oxygen concentrations vary with the change in temperature by more than an order of magnitude in comparison to oxygen levels during normal operating conditions.[36] Similar oxygen dependence on temperature can be expected in Pressurized Water Reactors (PWRs).

DO levels in operating plants are measured by non-continuous sampling at selected plant locations during normal operations. During transients such measurements would have to be made continuously; however the technology for doing this does not exist. Since DO levels are not measured at the surface of reactor components during transients, the actual DO levels, and resulting F_{en}, are subject to uncertainties. For example, an uncertainty of five in DO levels at the surface of a given component could lead to under-predicting the F_{en} by a factor of five at a minimum. The difficulty presented by this uncertainty can be overcome by using bounding

[34] MRP-47 at 4-19; *see also id.* at 4-27.

[35] In the Matter of Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Vermont Yankee Nuclear Power Station), Docket No. 50-271-LR, Affidavit of Kenneth C. Chang Concerning NEC Contentions 2A & 2B (Metal Fatigue) (May 12, 2008), at 12.

[36] *See* John J. Taylor, *R&D Status Report*, Nuclear Power Division, EPRI Journal (Jan/Feb 1983).

oxygen values during each transient. These values are different for different materials. The bounding oxygen value for stainless steel is the lowest value of oxygen during the transient, while the opposite is true for carbon and low alloy steels.

The developers of the F_{en} equations (ANL) specifically instruct how users should account for transients where temperature varies, as memorialized in two NRC reports: NUREG/CR-6583 states "the values of temperature and DO may be conservatively taken as the maximum values for the transient"[37]; NUREG/CR-6909 is even more specific and quantifies the appropriate level of DO a user should consider:

> The DO value is obtained from each transient constituting the stress cycle. For carbon and low alloy steels, the dissolved oxygen content, DO, associated with a stress cycle is the highest oxygen level in the transient, and for austenitic stainless steels, it is the lowest oxygen level in the transient. A value of 0.4 ppm for carbon and low-alloy steels and 0.05 ppm for austenitic stainless steels can be used for the DO content to perform a conservative evaluation.[38]

EPRI also recommends using the "maximum DO level (for carbon and low alloy steels)" for determining F_{en}.[39]

[redacted]

[redacted]

[37] NUREG/CR-6583 at 78.

[38] NUREG/CR-6909 at A-5. The specifications contained in NUREG/CR-6583 and NUREG/CR-6909 do not differentiate between PWRs and BWRs, even though PWRs run with lower oxygen concentrations than BWRs.

[39] MRP-47 at 4-19.

[40] Entergy, "Table of environmental correction actions for metal fatigue," November 6, 2007, IPEC00012537; Environmental Fatigue Evaluation for Indian Point Unit 2, WCAP-17199-P, Revision 0 (Westinghouse, June 2010), IPECPROP00056486, a pp.5-3, 5-5; Environmental Fatigue Evaluation for Indian Point Unit 3, WCAP-17200-P, Revision 0 (Westinghouse, June 2010), IPECPROP00056577, at p.5-3, 5-5.

[redacted]

[redacted]

[redacted]

[redacted]

[41] Westinghouse, "EnvFat User's Manual Version 1," May 2009, at § 2, IPECPROP00056785.

[42] 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.

[43] NUREG/CR-6909 at 40 ("An average temperature for the transient may be used to estimate Fen during a load cycle"), at 56 ("load cycles involving variable temperature conditions may be represented by an average temperature.").

[44] *See* DC Cook Charging Nozzle License Renewal Environmental Fatigue Evaluations, CN-PAFM-05-114, IPECPROP00061294; Diablo Canyon Unit 1 Insurge/Outsurge and Environmental Fatigue Evaluations, WCAP-17103-P, Rev. 0 (July 2009), IPECPROP00061179, at p.4-5.

[redacted]

[redacted]

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

C. *Heat Transfer*

i. Basic Principles

Heat transfer is a major factor in the determination of CUFen because it controls the cyclic thermal stresses during transients. Thermal stresses arise when there is a change in the local fluid temperature like during heat-ups or cool-downs or due to local mixing of hot and cold fluids. Failures result from either low stress at high cycle or high stress at low cycle. Most of the thermal stresses experienced at Indian Point are of the latter kind. Damage from such stresses is a serious concern. For example, such stress has caused through-the-wall-cracks in pipes at nuclear reactors.[48] As of 2007, at least thirteen cases of leakage from thermal fatigue

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[45] [redacted]

[46] [redacted]

[47] EPRI, MPR 47 at 4-22.

[48] NRC Bulletin No. 88-08: Thermal Stresses in Piping Connected to Reactor Coolant Systems (June 22, 1988) (discussing "circumferential crack extending through the wall of a short, unisolable section of emergency core cooling system (ECCS) piping that is connected to the cold leg of loop B in the RCS" at Farley 2); NRC Bulletin No. 88-08, Supplement 1: Thermal Stresses in Piping Connected to Reactor Coolant Systems (June 24, 1988) (discussing "crack extending through the wall of" a "section of emergency core cooling system (ECCS) piping that is connected to the hot leg of loop 1 of the RCS" at Tihange 1 in Belgium); NRC Bulletin No. 88-08, Supplement 2: Thermal Stresses in Piping Connected to Reactor Coolant Systems (August 4, 1988) (discussing the crack incidents at Farley 2 and Tihange 1); NRC Information Notice 97-46: Unisolable Crack in High-Pressure Injection Piping (July 9, 1997) (discussing "through-wall crack in the weld connecting the MU/HPI pipe and the safe-end of the 2A1

have occurred at nuclear reactors.[49] Based on this experience, the International Atomic Energy Agency ("IAEA") has concluded that the frequency of leakage from thermal fatigue will increase with time.[50]

In order to calculate thermal stress and its impact on fatigue life, one must determine the temperature distribution of a component during a transient, or, in other words, the rate at which heat is transferred to the reactor component surface during the transient. Thermal-hydraulic computer codes together with plant data are used to calculate such temperature distributions. Heat transfer coefficients (h), water temperature, cycling period, and interface motion are all important inputs to this heat transfer analysis, and the consequent determination of the CUF_{en} values. The CUF_{en} value will vary greatly depending on the inputs used to perform the heat transfer analysis.

The heat transfer coefficient h is the most important parameter in this regard. The heat transfer coefficient is commonly expressed in terms of geometry (G), fluid properties (P), flow rates (Q), and temperature difference between the coolant and the surface of the component (ΔT):

$$h = f(G, P, Q, \Delta T)$$

h is an experimental parameter and has been measured and determined for many different geometries, flow rates and rates of temperature change, and is known for well-defined, controlled conditions.

However, the local flow at the surface of many reactor components during transients is not well defined and, therefore, approximations and assumptions are required in calculating the proper h for a given set of conditions. Such approximations lead to uncertainties in the CUF_{en} because uncertainties in h directly impact the errors in the calculated stress. Typical variations in h could increase stress by a factor of 2. Increase in turbulence due to local discontinuities, and increase in the rate of the local temperature change increases h.[51] Increase in h increases the corresponding stress and reduces fatigue life.

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reactor coolant loop (RCL) nozzle" that was "caused by high-cycle fatigue due to a combination of thermal cycling and flow induced vibration" at Oconee Unit 2).

[49] *See* Institute for Energy, Development of a European Procedure for Assessment of High Cycle Thermal Fatigue in Light Water Reactors: Final Report of the NESC-Thermal Fatigue Project, EUR22763, 2007, http://ie.jrc.ec.europa.eu/publications/scientific_publications/2007/EUR22763EN.pdf, at 53 (hereinafter "Assessment of High Cycle Thermal Fatigue, EUR22763").

[50] Assessment of High Cycle Thermal Fatigue, EUR22763 at 52.

[51] Notably, fatigue life reduction due to large temperature differences and temperature fluctuations was not considered during the initial design of PWRs. It was only after many reactors experienced severe cracking in the late 1980s due to stratification that the PWRs became a concern. *See* NRC Bulletin No. 88-08: Thermal Stresses in Piping Connected to Reactor Coolant Systems (June 22, 1988). Following NRC Bulletin 88-11, an extensive effort has been devoted to better understand thermal fatigue with special emphasis on turbulent mixing. The results of these efforts are summarized in an April 2007 EUR22763 report: *See* Assessment of High Cycle Thermal Fatigue, EUR22763.

For example, h along nozzles and bends varies in intensity because of the large variation in turbulence along their surface. This leads to non-uniform heat loads and introduces larger uncertainty in the stress distribution in comparison to simpler flow configurations. Stratified flow in the pressurizer surge line, is another example where non-uniform heat loads exist. Such flows occur when a warmer fluid flows on top of a cooler fluid, with a temperature difference between the two fluids as high as 350°F. Instabilities at the interface between the two fluids are known to produce high frequency temperature fluctuations on the surface of the component, with the potential for accelerating crack initiation and growth.

Another factor that would lead to non-uniform stress distributions is preferential wall wear due to flow accelerate corrosion ("FAC") in low alloy steel components: Entergy ultrasonic examination reports show that wall thickness may vary significantly circumferentially in bends and welds.[52] One diagram indicates that in components where flow is not fully developed, component wall thickness can vary by more than 400% at Indian Point.[53]

ii. Entergy's Heat Transfer Calculations

[redacted] For other components, Entergy has failed to provide sufficient information to justify that heat transfer was appropriately considered, also casting doubt on the accuracy of Entergy's "refined" CUFen calculations.

1. Pressurizer Surge Line

[redacted] The pressurizer surge line is one of the components most vulnerable to fatigue failures. [redacted]

[52] *See* Entergy, Indian Point Unit 3 Flow Accelerated Corrosion, 3RF13 Outage, 2005 (Ultrasonic Examination Report of Main Steam/FAC-05-TD-03 (January 6, 2005); Ultrasonic Examination Report of HD/FAC-05-VCD-08(01-02) (March 23, 2005)).

[53] *See* Entergy, Indian Point Unit 3 Flow Accelerated Corrosion, 3RF13 Outage, 2005 (Ultrasonic Examination Report of Main Steam/FAC-05-TD-03 (January 6, 2005) (schematic indicating that in one section of piping, wall thickness varied from 0.059" to 0.257", which represents a difference of a factor of 4, or 400%).

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

[redacted]

[redacted] There is extensive data that has been generated in the last decade throughout the world to resolve fatigue issues in PWRs during stratification.[56] [redacted][57]

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

[redacted]

[55] *See* Assessment of High Cycle Thermal Fatigue, EUR22763.

[56] *See* Assessment of High Cycle Thermal Fatigue, EUR22763.

[57] 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

[redacted]. In particular, Entergy's "refined" CUF_{en} values for the surge lines in Indian Point Unit 2 and Indian Point Unit 3 are 0.822 and 0.594 respectively. [redacted] An uncertainty of only 30% in the heat transfer coefficients would cause the corresponding CUF_{en} for Indian Point Unit 2 to exceed unity.

2. Inlet and Outlet Reactor Vessel Nozzles

Entergy also provided documentation related to the heat transfer analysis for the inlet and outlet reactor vessel nozzles.[58] The temperature distributions for these components were based on over 40-year old Combustion Engineering calculations. These calculations did not use a finite element analysis and were based on a simplified 2-D model where the heat transfer coefficient was assumed constant along the flow and thermal properties were taken as independent of the temperature. The calculations were based on "as installed" nozzle dimensions. Due to nozzle geometry, the heat transfer is not uniform along the nozzle and can vary by 20%-30% depending on flow velocity location along the nozzle and flow direction. Such variation would result in axial thermal stress, which is not included in the 2-D analysis.

Additionally, it is apparent that this analysis related to the inlet and outlet reactor vessel nozzles neglected to account for the observed fact that low alloy steels are subjected to wall thinning due to FAC.[59] The reduction in wall thickness after 60 years of operation is expected to reduce fatigue life. Entergy has failed to justify, and therefore, cannot assume, that the nozzles would maintain their original dimensions after 60 years of operation.

3. Other Table 4.3-13 and Table 4.3-14 Components

In relation to the other components evaluated in Entergy's "refined" CUF_{en} analysis, i.e., the RCS charging system, RCS injection nozzles, and RHR class 1 piping, Entergy has not provided sufficient information to allow for meaningful comment on the heat transfer calculations. In particular, Entergy's documentation does not provide the actual equations employed to determine the heat transfer coefficients. [redacted]

[58] 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).

[59] *See* Entergy, Indian Point Unit 3 Flow Accelerated Corrosion, 3RF13 Outage, 2005 (Ultrasonic Examination Report of Main Steam/FAC-05-TD-03 (January 6, 2005); Ultrasonic Examination Report of HD/FAC-05-VCD-08(01-02) (March 23, 2005)).

[redacted] Notably, the WESTEMS program, which may include the details of the equations used, was never provided.

However, Entergy has purported to provide sufficient information concerning the calculation of heat transfer coefficients for the evaluated components.[61] Thus, based on the information relied upon by Entergy to support the "refined" evaluation, it is simply impossible to conclude that the new CUF_{en} values for these other components are accurate. The analyses and documents Entergy relies upon do not specify the heat transfer coefficients used, or how h was determined, for the relevant components. To assess the uncertainty of h, it is imperative to know the component geometry, the piping geometry upstream of the component, the flow velocities, and the corresponding expressions for h, none of which are specified by Entergy for the three components at issue. [redacted]

Given the uncertainties associated with determining the heat transfer coefficient, h, as discussed here as well as above, it is imperative to ensure that the methodology and assumptions employed were adequate to account for such uncertainties. Entergy has failed to do this. Without an understanding of the values of h and the assumptions used to arrive at such values, the methodology employed by Entergy to re-calculate CUF_{en} for the three relevant components remains questionable. [redacted]

D. Number of Transients

The number of transients used in Entergy's "refined" calculations directly affects the resulting CUF_{en} value. [redacted]

To evaluate the remaining fatigue life of a given component, it is necessary to consider the past as well as future loading during all transients. In other words the fatigue status of the component

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[60] *See* Environmental Fatigue Evaluation for Indian Point Unit 2, WCAP-17199-P, Revision 0 (Westinghouse, June 2010), IPECPROP00056486 [redacted]

[61] *See* Letter from P. Bessette, K. Sutton (Entergy) to D. Brancato (Riverkeeper), Re: Entergy Nuclear Operations, Inc., (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-247-LR and 50-286-LR (Oct. 18, 2011).

must be known from the time it was installed to the time it is removed from service. An accurate number of past and anticipated future transients is essential to the calculation of the CUF_{en} values. It is apparent that Entergy has not adequately considered either past or future transients at Indian Point, as follows.

i. Past Transients

To assess the severity of past transients, each transient that has occurred must be described to determine its contribution to the CUF_{en}. Yet considering past transients can be difficult because of the fact that degradation of some components was not included in the fatigue analysis when nuclear power plants were originally built. For example, the pressurizer surge line falls into this category. Historical records in such cases are incomplete and insufficient to provide adequate inputs for the number of heat-up and cool-down transients, stratification frequency, and system ΔT. These parameters are required for thermal hydraulic and stress calculations.

[redacted]

ii. Future Transients

[redacted]

[62] *See* Westinghouse, Indian Point Unit 3 Pressurizer Insurge/Outsurge Transient Development, CN-PAFM-09-64 (June 2010), IPECPROP00058077, at 44 (IPECPROP00058120).

[63] *See* Westinghouse, Indian Point Unit 3 Pressurizer Insurge/Outsurge Transient Development, CN-PAFM-09-64 (June 2010), IPECPROP00058077, at 44 (IPECPROP00058120).

[64] *See* Environmental Fatigue Evaluation for Indian Point Unit 2, WCAP-17199-P, Revision 0 (Westinghouse, June 2010), IPECPROP00056486, at 3-1; Environmental Fatigue Evaluation for Indian Point Unit 3, WCAP-17200-P, Revision 0 (Westinghouse, June 2010), IPECPROP00056577, [redacted] LRA at p. 4.3-2 ("Rate per day was calculated for each event and that rate was projected to the end of the extended operations"; this is a linear extrapolation).

[redacted]

Justification of an appropriate number of future transients is critical in light of the fact that Indian Point will be entering an *extended* period of operation. It is commonly accepted that the useful life of most engineering components and structures follow a " bathtub curve" behavior. A "bathtub curve" is defined as "the phenomenon that the fraction of products failing in a given timespan is usually high early in the lifecycle, low in the middle, and rising strongly towards the end. When plotted as a curve, this looks like the profile of a bathtub."[66] This phenomenon dictates that at the beginning and at the end of life, component failure occurs at a very high frequency. In between these two extremes, the failure is relatively low and constant. [redacted]

[redacted]

E. Summary and Conclusions Concerning Entergy's "Refined" CUFen EAF Analysis

While many of the parameters discussed above can best be determined experimentally in the plant during transients, for most components this is not practical and it is necessary to resort to scientific publications, analysis, and assumptions. It, thus, becomes necessary to understand all of the underlying assumptions employed by Entergy in order to determine the validity of the "refined" CUFen calculations. It is a common engineering practice to clearly state all assumptions especially when the equations used deviate from the conditions for which they were originally derived. Such informed judgment is required in order to determine whether Entergy has adequately accounted for the significant degree of uncertainty associated with calculating CUF_{en}.[68]

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[65] *See* Systems Standard Design Criteria, Nuclear Steam Supply System Design Transients, Jan. 11, 1971, IPECPROP00059695, at 4 (IPECPROP00059698).

[66] WordIQ.com, Bathtub curve – Definition, http://www.wordiq.com/definition/Bathtub_curve (last visited Dec. 21, 2011).

[67] *See* Westinghouse, Indian Point Unit 3 Pressurizer Insurge/Outsurge Transient Development, CN-PAFM-09-64 (June 2010), IPECPROP00058077, at 44.

[68] NRC has acknowledged the need to account for such uncertainties. *See* Safety Evaluation by the Office of Nuclear Reactor Regulations Related to Amendment No. 229 to Facility Operating License No. DPR-28 Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Vermont Power Station, Docket No. 50-571), at § 2.8.7.1, p. 190, ADAMS Accession No. ML060050028 ("Application of NRC-approved Analytical Methods and Codes. . . . In general, the analytical methods and codes are assessed and benchmarked against measurement data, comparisons to actual nuclear plant test data and research reactor measurement data. The validation and benchmarking process provides the means to establish the associated biases and uncertainties. The uncertainties

[redacted]

Entergy's failure to sufficiently account for all relevant parameters has resulted in predictions that are non-conservative. Given the large uncertainties in the input parameters and other assumptions used to generate the revised metal fatigue calculations, the methodology employed by Entergy suggests the likelihood of a wide margin of error, and the detrimental effects of the environment on fatigue strength, and resulting predicted fatigue life, of the components evaluated are likely grossly underestimated.

In fact, many of the revised CUF_{en} values remain very close to unity (for example, 0.9434). The fact that Entergy's "refined" CUF_{en} values are reported to the fifth significance figure (i.e. to a ten-thousandth of a decimal point), with several just a hair below unity clearly shows that Entergy does not appreciate the uncertainties inherent in calculating the CUF_{en} values. With a margin of error to account for varying input data and other undisclosed assumptions, such numbers could be considerably higher than the 1.0 regulatory threshold. In any event, without an error analysis, the claimed high degree of accuracy of the results remains questionable at best. Entergy repeatedly applies the term "bounding" to its analyses and results, implying that such results are conservative, and that no error analysis is necessary. However, there is patently deficient explanation to show that the results are in any respect bounding or conservative. Indeed, the foregoing lengthy discussions point to numerous areas that plainly show that the CUF_{en} calculations were *not* conservative.

Based on the foregoing, Entergy's "refined" CUF_{en} calculations cannot be used as the basis for concluding that the aging effects of metal fatigue will be adequately managed at Indian Point during the PEO.

V. Inadequacy of Entergy's Aging Management Plan ("AMP") for Metal Fatigue.

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

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associated with the predicted parameters and the correlations modeling the physical phenomena are accounted for in the analyses.").

A. Improper Reliance on Entergy's "Refined" Fatigue Evaluations

In response to CUF_{en} values in excess of regulatory limits, Entergy opted to conduct additional analyses, and update its calculations. As explained above, despite Entergy's assertions that the recalculated usage factors are within proper limits, there is paltry evidence to suggest that the recalculated CUF_{en} values are accurate to the degree Entergy now claims. This fails to comply with the AMP articulated in the *GALL Report*, which specifies that acceptable corrective action includes "a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded during the extended period of operation."[69] Based on the discussion above, Entergy's August 2010 metal fatigue calculations fail to make such a demonstration. These new calculations do not demonstrate that Entergy's program for managing metal fatigue during the PEO is adequate.

B. Improper Failure to Identify Additional Components for Review

In order for Entergy to have an effective AMP to monitor for metal fatigue, it must expand the scope of the fatigue analysis beyond simply representative components, to identify other components whose CUF_{en} may be greater than 1.0. This is necessary at Indian Point for several reasons. First, Entergy's initial findings presented in Tables 4.3-13 and 4.3-14 of April the 2007 LRA indicated that the CUF_{en} values for various components exceeded the regulatory threshold of 1.0. Under these circumstances, regulatory and industry guidance requires Entergy to identify additional reactor locations for potential high susceptibility to metal fatigue in order to ensure that appropriate aging management measures are taken in a timely fashion. In particular, EPRI's MRP-47 indicates that when "plant-unique evaluations . . . show that some of the NUREG/CR-6260 locations do not remain within allowable limits for 60 years of plant operation when environmental effects are considered . . . plant specific evaluations should *expand* the sampling of locations accordingly to include other locations where high usage factors might be a concern."[70] NUREG-1801, *GALL*, Revision 1 also contemplates such an expanded review when CUF_{en} values have been found to exceed unity, stating that "[f]or programs that monitor a sample of high fatigue usage locations, corrective actions include a review of *additional* affected reactor coolant pressure boundary locations," and that sample locations identified in NUREG/CR-6260 are simply the "minimum" set of components to analyze.[71]

Entergy's "refined" calculations, which purport to show that for all components evaluated, the CUF_{en} is now less than 1.0, does not alleviate Entergy of its obligation to expand the number of components reviewed. To begin with, the fact that Entergy's initial findings showed CUF_{en} values over 1.0, by itself, triggered the responsibility to perform an expanded review, notwithstanding whether any additional calculations later show all CUF_{en} values within regulatory acceptance criteria. Moreover, discussed at length above, the accuracy of the

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[69] NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, Rev. 1 § X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary, ¶ 7, pp. X M-1 to X M-2 (hereinafter cited as "NUREG-1801, Rev. 1"); NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, Rev. 2 § X.M1, Fatigue Monitoring, ¶ 7, p. X M1-2 (hereinafter cited as "NUREG-1801, Rev. 2").

[70] MRP-47 3-4 (2005) (emphasis added).

[71] NUREG-1801, Rev. 1, § X.M1, ¶¶ 5, 7, pp. X M-1 to X M-2 (emphasis added).

"refined" evaluation is highly suspect, and, in actuality, the CUF_{en} values for the evaluated components indeed may exceed unity during the proposed period of extended operation. Thus, an expanded review must still be conducted.[72]

Moreover, the most recent version of the *GALL Report*, Revision 2, specifies that the sample set for fatigue calculations that consider the effects of the reactor water environment "should include the locations identified in NUREG/CR-6260 *and additional plant-specific component locations* in the reactor coolant pressure boundary *if they may be more limiting than those considered in NUREG/CR-6260.*"[73] Entergy has, to date, not provided any analysis that would support a conclusion that that the CUF_{en} values in LRA Tables 4.3-13 and 4.3-14 (which represent the same locations as those identified in NUREG/CR-6260) bound all other components at the plant. To the contrary, Entergy's initial findings indicated that the CUF_{en} value for several components would exceed unity, demonstrating that such components were not necessarily the most limiting. Furthermore, the extensive discussion above about Entergy's "refined" calculations shows that the revised CUF_{en} values for the components evaluated are likely under-predicted and that, if properly evaluated, many could exceed unity. This discussion squarely demonstrates that the locations analyzed may not be bounding for the entire plant.

In fact, NRC Staff has now conceded that there may be more limiting components, and that the CUF_{en} values in LRA Tables 4.3-13 and 4.3-14 may not be bounding. Specifically, on February 10, 2011, NRC Staff issued a Request for Additional Information ("RAI") to Entergy, asking that Entergy "[c]onfirm and justify that the plant-specific locations listed in LRA Tables 4.3-13 and 4.3-14 are bounding for the generic NUREG/CR-6260 components" and "[c]onfirm and justify that the locations selected for environmentally-assisted fatigue analyses in LRA Tables 4.3-13 and 4.3-14 consist of the most limiting locations for the plant (beyond the generic components identified in the NUREG/CR-6260 guidance)."[74]

In response, Entergy only indicated that the locations in LRA Tables 4.3-13 and 4.3-14 are the same as those locations provided in NUREG/CR-6260, and, in Commitment 43, promised to determine at some future point in time, though before entering the period of extended operation, whether the NUREG/CR-6260 locations evaluated are the limiting locations for the Indian Point plant configurations. Entergy's apparent reason for requiring additional time was the need to review "design basis ASME Code Class 1 fatigue evaluations."[75] In August 2011, NRC Staff issued Supplement 1 to the Safety Evaluation Report pertaining to the Indian Point license

[72] *See* ASLB Memorandum and Order (Ruling on Motion for Summary Disposition of NYS-26/26A/Riverkeeper TC-1/1A (Metal Fatigue of Reactor Components) and Motion for Leave to File New Contention NYS-26B/Riverkeeper TC-1B) (November 4, 2010), at 20.

[73] NUREG-1801, Rev. 2, § X.M1, p. X M1-2 (emphasis added).

[74] Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Numbers 2 and 3, License Renewal Application (Feb. 10, 2011), at 13, ADAMS Accession No. ML110190809.

[75] NL-11-32, Response to Request for Additional Information (RAI), Aging Management Programs, Indian Point Nuclear Generating Unit Nos. 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64 (March 28, 2011), at 26, ADAMS Accession No. ML110960360.

renewal proceeding, memorializing this exchange, and accepting Entergy's responses and new commitment.[76]

However, NRC Staff's acceptance of a vague commitment to perform necessary metal fatigue investigation and analyses in the future, and Entergy's failure to actually "confirm and justify" bounding and limiting locations for Indian Point renders Entergy's AMP insufficient to comply with 10 C.F.R. § 54.21(c), or the regulatory guidance of NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, as it does not demonstrate that metal fatigue will be appropriately monitored, managed and corrected during the PEO. No party now disputes that Entergy's LRA Tables 4.3-13 and 4.3-14, which lists the same components as those identified in NUREG/CR-6260, may not represent limiting locations for the entire plant. As such, according to the guidance of the *GALL Report*, in order to have an adequate aging management program, Entergy must identify the locations that may be more limiting, and which will be the subject of CUF_{en} calculations, *now,* and not just articulate a plan to determine such locations later.

Instead of conducting the analysis that would be required to confirm and justify that the components in LRA Tables 4.3-13 and 4.3-14 are bounding, Entergy only agreed to perform such an assessment in the future. Notably, such an assessment is necessarily complex and not a clearly defined calculation. In fact, there are numerous considerations and factors for determining the most limiting locations. In particular, such an analysis would require an assessment of experience at Indian Point as well as other pressurized water reactor (PWR) plants; and identification and ranking of all components that are susceptible to thermal fatigue (including but not limited to nozzles, reducers, mixing tees and bends in feed water lines, surge lines, spray lines, and volume control system lines), in terms of numerous key parameters that are known to effect fatigue life (including the ratios of the local heat transfer coefficient and the local material conductivity, wall thickness, fluid temperature, ΔT, dissolved oxygen levels, flow velocities, number of transients, magnitude and cycling frequency of surface temperatures and loads, and surface discontinuities, and flow discontinuities in each component). Moreover, thermal striping during stratification should be generally considered as these effect fatigue life.[77] Entergy has demonstrably failed to provide any information about how the analysis to determine the most limiting locations as Indian Point will be performed, such that Intervenors could meaningfully comment upon the adequacy of the analysis.

An analysis to determine the most limiting locations for which Entergy will have to perform CUF_{en} evaluations must be performed *before* a determination is made about license renewal. Accordingly, NRC Staff's acceptance of Entergy's commitment in SER Supplement 1 to determine whether the NUREG/CR-6260 locations evaluated are the limiting locations for the Indian Point plant configurations at some time in the future is not warranted or acceptable.

In light of Entergy's failure to expand the scope of review and identify further components to be subject to CUF_{en} analyses, Entergy has not demonstrated that it has a program to monitor,

[76] Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Supplement 1 (August 2011), *available at*, http://pbadupws nrc.gov/docs/ML1124/ML11242A215.pdf.

[77] NUREG-1801 requires that environmental effects be included in the calculations and it does not exclude thermal striping from such requirements.

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*.

C. Failure to Define Specific Criteria Concerning Component Inspection, Monitoring, Repair, or Replacement

The lack of a reliable, transparent, complete assessment of CUF_{en} values for susceptible plant components at Indian Point fails to comply with the "Scope of Program" articulated in the *GALL Report*, which specifies that a program for managing metal fatigue must include adequate "*preventative measures* to mitigate fatigue cracking of metal components of the reactor coolant pressure boundary caused by anticipated cyclic strains in the material."[78]

However, Entergy's plans for correcting metal fatigue related degradation depend initially upon calculating the vulnerability of plant components. Indeed, Entergy intends to rely upon future CUF_{en} calculations throughout the period of extended operation to manage metal fatigue. Entergy's calculations are meant to signify when components require inspection, monitoring, repair, or replacement, and, according to Entergy, will determine when such actions are taken. Accordingly, the validity of Entergy's monitoring program depends upon the accuracy of the calculations of the CUF_{en}. When a fatigue monitoring program is entirely based on a predictive analysis and not on actual measurements, and the analysis is flawed, the monitoring program is invalid. Thus, Entergy's flawed methodology for calculating CUF_{en}, as discussed above, which Entergy ostensibly intends to employ throughout the period of extended operation, as well as Entergy's failure to expand the scope of components to be assessed, renders Entergy's vague commitments to inspect, repair, and replace affected locations insufficient to ensure proper management of metal fatigue during the proposed PEO.

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 CUF_{en} values 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 (considering, for example defect size changes and uncertainties in the stress analysis and instrumentation), and allow for independent and impartial reviews of scope and frequency of inspection. Entergy has not done this.

VI. Conclusion

The evidence discussed above clearly demonstrates the following:

[78] NUREG-1801, Rev. 1 § X.M1, ¶ 1, p. X M-1 (emphasis added); *see also* NUREG-1801, Rev. 2, § X.M1, ¶ 1 p. X M1-2 ("The program ensures the fatigue usage remaining within the allowable limit, thus minimizing fatigue cracking of metal components caused by anticipated cyclic strains in the material.").

- [redacted]

- As a result of these deficiencies, Entergy's calculations have a wide margin of error and have likely underestimated the CUF_{en} values. As many of the values derived by Entergy's new evaluation are very close to unity, the regulatory threshold, it is highly likely that if Entergy had accurately considered all relevant factors, many of the CUF_{en} values would, in actuality, exceed 1.0;

- Entergy has improperly failed to confirm that the components evaluated represent the most limiting locations providing a bounding analysis, and failed to expand the scope of components to be subject to a CUF_{en} analysis accordingly;

- In light of the insufficiency of Entergy's CUF_{en} analyses, Entergy has otherwise failed to provide any details concerning the inspection, monitoring, repair, and replacement to ensure that the degradation effects of metal fatigue would be sufficiently handled.

Accordingly, Entergy has failed to demonstrate that the aging effects of metal fatigue will be adequately managed for the period of extended operation, and has, thus, failed to comply with 10 C.F.R. § 54.21(c) or regulatory guidance, including the *GALL Report*.

December 19, 2011

Rockville, MD

Dr. Joram Hopenfeld
1724 Yale Place
Rockville, MD 20850

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