UNITED STATES
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

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In re:	Docket Nos. 50-247-LR; 50-286-LR
License Renewal Application Submitted by	ASLBP No. 07-858-03-LR-BD01
Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Operations, Inc.	DPR-26, DPR-64 June 9, 2015
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STATE OF NEW YORK AND RIVERKEEPER, INC. REVISED STATEMENT OF POSITION

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

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PRELIMINARY STATEMENT

In accordance with 10 C.F.R. § 2.1207(a)(1), the Atomic Safety and Licensing Board's ("Board") July 1, 2010 Scheduling Order,[1] the Board's December 9, 2014 Revised Scheduling Order,[2] and the Board's May 27, 2015 Order,[3] the State of New York (the "State") and Riverkeeper, Inc. (collectively, "Intervenors") hereby submit their Revised Statement of Position on Intervenors' admitted Consolidated Contention NYS-26B/RK-TC-1B, concerning the effects of metal fatigue on important reactor components at the Indian Point nuclear facility. Metal fatigue is major age-related phenomenon that must be appropriately managed if the Indian Point facility is to be licensed for an additional 20 years of operation. Failure to properly manage the effects of fatigue – especially as those effects may be enhanced by other aging degradation mechanisms such as embrittlement – could lead to the failure of important reactor components, with a consequent release of radiation and profound safety and environmental consequences for the State and its citizens – as well as Riverkeeper and its members.

Intervenors have previously satisfied the standards set forth in 10 C.F.R. § 2.309 governing contention admissibility – standards that the U.S. Nuclear Regulatory Commission (NRC) and Entergy Nuclear Operations, Inc. ("Entergy" or "the Applicant") have described as "strict by design." Consolidated Contention NYS-26B/RK-TC-1B is supported by numerous

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[1] *Entergy Nuclear Operations, Inc.* (Indian Point, Units 2 and 3), Scheduling Order (July 1, 2010) (unpublished) (ML101820387).

[2] *Entergy Nuclear Operations, Inc.* (Indian Point, Units 2 and 3), Revised Scheduling Order (December 9, 2014) (unpublished) (ML14343A757).

[3] *Entergy Nuclear Operations, Inc.* (Indian Point, Units 2 and 3), Order (Granting New York's Motion for an Eight-Day Extension of the Filing Deadline) (May 27, 2015) (unpublished). The May 27, 2015 Order extended the deadline for the State and Riverkeeper to

documents, including this Revised Statement of Position, the revised prefiled testimony of Dr. Richard T. Lahey, Jr., Edward S. Hood Professor Emeritus of Engineering, Rensselaer Polytechnic Institute (Exh. NYS000530), the revised prefiled testimony and Supplemental Report of Dr. Joram Hopenfeld (Exhs. RIV000142, RIV000144), and numerous supporting technical exhibits. Additionally, this contention is also supported by, among other things, intervenors' initial filings from December 2011, including the Initial Statement of Position (Exh. NYS000343), the December 20, 2011 Report of Dr. Lahey (Exh. NYS000296), the Supplemental Report by Dr. Lahey concerning Entergy's use of the WESTEMS computer code as a tool to analyze the cumulative fatigue condition of important reactor components (Exh. NYS000297), Dr. Lahey's initial prefiled written testimony (Exh. NYS000344), the December 19, 2011 Report of Dr. Joram Hopenfeld (Exh. RIV000035), Dr. Hopenfeld's prefiled written testimony (Exh. RIV000034), and numerous supporting technical documents. Finally, this contention is supported by intervenors' revised testimony from June 2012, including the 2012 Revised Statement of Position (Exh. NYS000439), pre-filed rebuttal testimony from Dr. Lahey (Exh. NYS000440) and Dr. Hopenfeld (Exh. RIV000114), and additional technical exhibits. Intervenors' testimony and exhibits shows that Entergy's license renewal application (LRA) should be denied because Entergy has not established that it has an adequate plan to manage the effects of metal fatigue on key reactor components, in violation of 10 C.F.R. §§ 54.21(a)(3), 54.21(c)(1)(iii) and 54.29(a).

BACKGROUND

A pressurized water nuclear reactor (PWR) is made up of many different systems, components and fittings. In turn, these systems components and fittings are made of many

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file their revised prefiled testimony, affidavits and exhibits from June 1, 2015 to June 9, 2015.

different types of materials. *See* Figure 1. PWRs have water (i.e., the primary coolant) under high pressure flowing through the core in which heat is generated by the fission process. The core is located inside a large steel container known as the reactor pressure vessel (RPV). The heat is absorbed by the coolant and then transferred from the coolant in the primary system to lower pressure water in the secondary system via a large heat exchanger (i.e., a steam generator), which, in turn, produces steam on the secondary side. These steam generator systems are located inside a large containment structure. After leaving the containment building, via main steam piping, the steam drives a turbine, which turns a generator to produce electrical power.

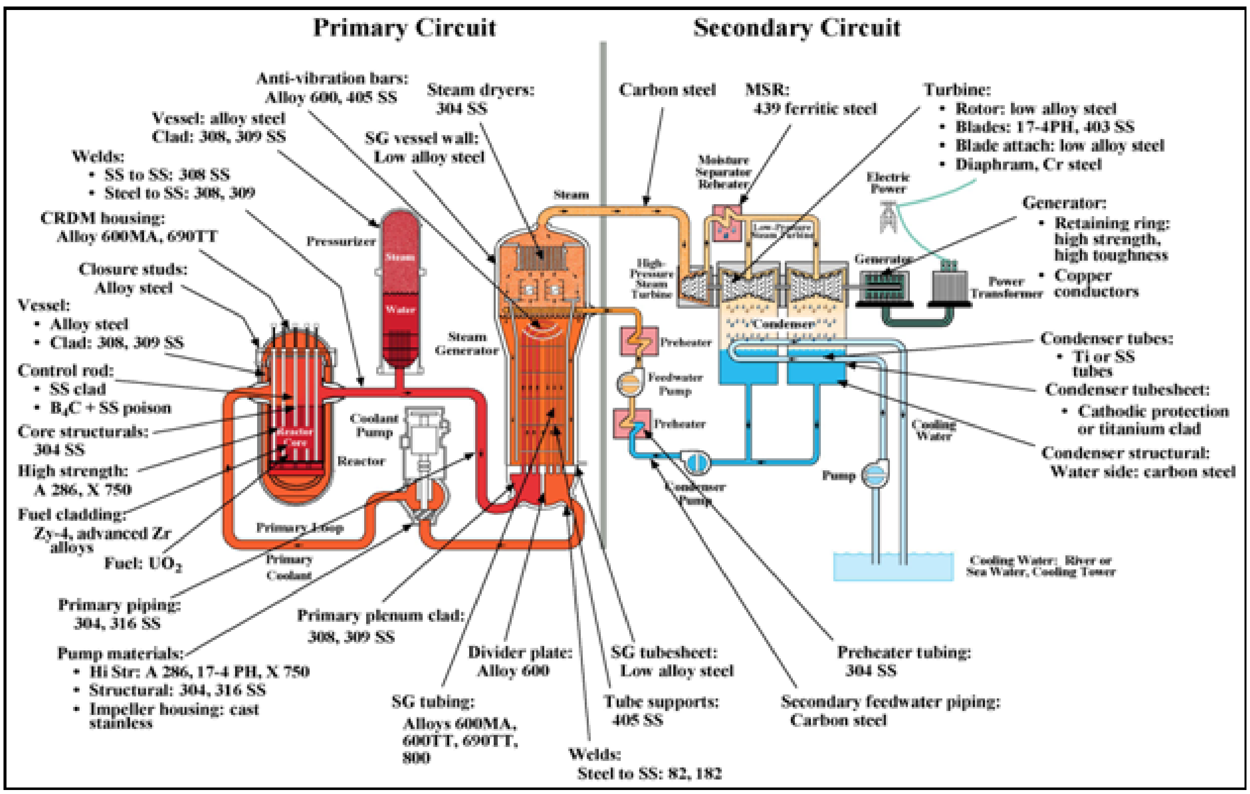

Figure 1. Overview of a PWR reactor, showing the various materials used in the reactor construction. Source: DOE, Light Water Reactor Sustainability Program: Materials Aging and Degradation Technical Program Plan, at 2, Figure 1 (August 2014) (Exh. NYS000485).

Fatigue or "cyclic stress" is a significant aging degradation mechanism that affects metal parts due to repeated stresses during plant operation. Material composition, strain rate,

temperature, and local water chemistry are some of the factors that contribute to fatigue of metal parts. Fatigue can create small cracks that propagate and cause a given component to malfunction, leak, or break entirely. Such failures may occur during steady state or during anticipated or unanticipated transients and may have serious consequences to public health and safety. Other aging effects are known or believed to interact synergistically with fatigue, resulting in greater degradation from the cumulative aging effects than would be anticipated from each degradation effect acting alone. Revised Lahey PFT, at 16-17 (Exh. NYS000530). For example, neutron embrittlement is known to reduce the resistance of fatigued components to crack propagation. *Id.* at 17-19.

A common figure of merit used in the American Society of Mechanical Engineers (ASME) code (Section-III) to appraise the possibility of fatigue failure to appraise the possibility of fatigue failure is the cumulative usage factor (CUF), which is the ratio of the number of cycles experienced by a structure or component divided by the number of allowable cycles for that structure or component. Lahey Report, at 24 (Exh. NYS000296). At a nuclear power plant, the maximum number of cycles that should be experienced by any structure or component should always result in a CUF of less than 1.0 – that is, the number of actual cycles experienced should always be less than the number of allowable cycles. *Id.*; *see* Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Rev. 2 (2010), at X.M1-1 (Exh. NYS000147A-D) ("crack initiation is assumed to have started in a structural component when the fatigue usage factor at the point of the component reaches the value of 1, the design limit on fatigue"). The CUF was developed in laboratory tests using "plain" air that did not take into account the harsh operating environment inside the reactor, which is known to reduce the number of allowable cycles. *See, e.g.*, NUREG/CR-6909 (NYS000357). Therefore, the CUF must be adjusted to reflect

environmental factors. *See, e.g.,* NUREG/CR-5704 (NYS000354); NUREG/CR-6583 (NYS000356). The environmentally adjusted CUF value is known as CUF_{en}.

Entergy operates two Westinghouse-designed PWRs at the Indian Point site in Buchanan, New York, roughly 24 miles north of New York City. The two operating reactors are known as Indian Point Unit 2 (IP2) and Indian Point Unit 3 (IP3).[4] The Indian Point reactors are among the older operating nuclear reactors in the United States. IP2 reached the end of its initial 40-year operating license on September 28, 2013, and IP3 will reach the end of its initial operating license on December 12, 2015.

PROCEDURAL HISTORY

I. Entergy's License Renewal Application and Intervenors' Initial Metal Fatigue Contentions

In 2007, Entergy submitted an LRA seeking permission to operate both Indian Point reactors for an additional 20 years, which would make them among the first nuclear reactors to operate out to 60 years. Entergy's LRA included the results of CUF_{en} calculations on locations identified in NUREG/CR-6260 (Exh. NYS000355) that showed that several components had CUF_{en} values greater than 1. LRA at Tables 4.3-13 and 4.3-14. In particular, the pressurizer surge line and nozzle, and the reactor coolant system charging system nozzle (on the primary side), and the steam generator main feed water nozzles and tube/tube-sheet welds, and the upper joint canopy of the IP-2 control rod drive (CRD) mechanisms, all had unacceptably high CUF_{en} (e.g., $CUF_{en} > 9.0$ for the IP-2 and IP-3 pressurizer surge lines and $CUF_{en} > 15$ for the IP-2 RCS charging system nozzle). *Id*.

[4] A third nuclear reactor, Indian Point Unit 1, is owned by Entergy at the site but does not operate and generate electricity. The license for Unit 1 is DPR-5.

On November 30, 2007, the State submitted a Petition to Intervene (NYS Petition), which included proposed contentions regarding critical deficiencies in Entergy's LRA with respect to public safety, health, and the environment. State of New York Notice of Intention to Participate and Petition to Intervene (Nov. 30, 2007) (ML073400187). Contention NYS-26 alleged that Entergy had failed to include an adequate plan to manage the effects of aging due to metal fatigue. *Id.* at 227. Riverkeeper also submitted a Petition to Intervene, which alleged, in Contention RK-TC-1, that Entergy had failed to demonstrate that aging effects would be managed at the plant, especially with respect to metal fatigue. Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene in the License Renewal Proceeding for the Indian Point Nuclear Power Plant (Nov. 30, 2007), at 7-15 (ML073410093).

On January 22, 2008, Entergy submitted an amendment to its original LRA, denominated "LRA Amendment 2," which included a commitment – "Commitment 33" – to "update" the CUF calculations, including an appropriate environmental factor (F_{en}), for locations identified in LRA tables 4.3-13 and 4.3-14 and to "repair or replace" any "affected locations before exceeding a CUF of 1.0." *See* Letter from Fred Damico, Entergy, to NRC, NL-08-021, Attachment 2, at 15 (Jan. 22 2008) (ML080230637) (Exh. NYS000351). In response, Riverkeeper and the State of New York filed supplemented and amended contentions, denominated RK-TC-1A and NYS 26A, which argued that LRA Amendment 2 and Commitment 33 were inadequate to manage the effects of metal fatigue during the period of extended operations. The Board admitted the State's Contention 26A and Riverkeeper's Contention 1A, over objections from Entergy and NRC Staff, and consolidated the two contentions. *Entergy Nuclear Operations, Inc.* (Indian Point Nuclear Generating Units 2 and 3), LBP-08-13, Memorandum and Order (Ruling on Petitions to Intervene and Requests for Hearing), 68 N.R.C. 43, 217-219 (July 31, 2008).

II. Entergy's Communication NL-10-082 and Intervenors' New and Amended Consolidated Contention NYS-26B/RK-TC-1B

Entergy submitted a communication "regarding completion of Commitment 33" to the Board on August 9, 2010, with updated versions of LRA tables 4.3-13 and 4.3-14 that included "refined" CUF_{en} calculations that had been conducted for Westinghouse for various components that had initially been calculated as more than 1.0. Letter from Kathryn Sutton and Paul Bessette, Counsel for Entergy, to ASLB, NL-10-082 (Aug. 9, 2010) (Exh. NYS000352). The details of these results were reported separately in proprietary-designated documents. *See* Westinghouse, Environmental Fatigue Evaluation for Indian Point Unit 2, WCAP-17199-P, Rev. 0 (June 2010) (Exh. NYS000361); Westinghouse, Environmental Fatigue Evaluation for Indian Point Unit 3, WCAP-17200-P, Rev. 0 (June 2010) (Exh. NYS000362). Rather than performing standard ASME code evaluations, as Entergy had done before, these calculations were done using WESTEMS, a proprietary computer code of Westinghouse. These "refined" CUF_{en} values remained very close to 1.0 for some components, including 0.9434 for the IP2 regenerative heat removal (RHR) system piping and 0.9961 for the IP3 RHR piping. NL-10-082, Attachment 1, at 2-3 (Exh. NYS000352); *see* Figures 2 & 4 (public filing). Nonetheless, Entergy reported that since all calculated CUF_{en} values were now technically below the 1.0 threshold, Commitment 33 had been "resolve[d]" without the need for part repair or replacement. *Id.* at 4.

Table 4.3-13
IP2 Cumulative Usage Factors for NUREG/CR-6260 Limiting Locations

NUREG-6260 Generic Location		IP2 Plant-Specific Location	Material Type	CUF of Record	Per NUREG/CR-6583 or NUREG/CR-5704	
					F_{en}	Environmentally Adjusted CUF
1	Vessel shell and lower head	Bottom head to shell	LAS	0.004	2.45	0.01
2	Vessel inlet and outlet nozzles	Reactor vessel inlet nozzle	LAS	0.05	2.45	0.12
2	Vessel inlet and outlet nozzles	Reactor vessel outlet nozzle	LAS	0.281	2.45	0.69
3	Pressurizer surge line nozzles	Pressurizer surge line nozzle	LAS	~~0.264~~ 0.109	~~2.45~~ 1.74	~~0.646~~ 0.188
3	Pressurizer surge line piping	Surge line piping to safe end weld	SS	~~0.6~~ 0.062	~~15.35~~ 13.26	~~9.21~~ 0.822
4	RCS piping charging system nozzle	Charging system nozzle	SS	~~0.99~~ 0.0323	~~15.35~~ 8.7	~~15.20~~ 0.2809
5	RCS piping safety injection nozzle	NA	SS	~~NA[2]~~ 0.1083	~~15.35~~ 7.8975	~~NA[2]~~ 0.8553
6	RHR Class 1 piping	NA	SS	~~NA[2]~~ 0.0721	~~15.35~~ 13.08	~~NA[2]~~ 0.9434

Figure 2. Updated LRA table 4.3-13 including "refined" CUF_{en} values for IP2 NUREG/CR-6260 locations. Source: Attachment 1 to NL-10-082, at 2 (Exh. NYS000352).

Table 4.3-14
IP3 Cumulative Usage Factors for NUREG/CR-6260 Limiting Locations

	NUREG-6260 Location	IP3 Plant-Specific Location	Material Type	CUF of Record	Per NUREG/CR-6583 or NUREG/CR-5704	
					F_{en}	Environmentally Adjusted CUF
1	Vessel shell and lower head	Bottom head to shell	LAS	0.02	2.45	0.05
2	Vessel inlet and outlet nozzles	Reactor vessel inlet nozzle	LAS	0.049	2.45	0.12
2	Vessel inlet and outlet nozzles	Reactor vessel outlet nozzle	LAS	0.259	2.45	0.64
3	Pressurizer surge line nozzles	Pressurizer surge line nozzle	LAS	~~0.0612~~ 0.0903	~~2.45~~ 1.74	~~2.35~~ 0.157
3	Pressurizer surge line piping	Surge line piping to safe end weld	SS	~~0.6~~ 0.0411	~~15.36~~ 14.45	~~9.21~~ 0.594
4	RCS piping charging system nozzle	NA	SS	~~NA[2]~~ 0.1812	~~15.36~~ 3.98	~~NA[2]~~ 0.722
5	RCS piping safety injection nozzle	NA	SS	~~NA[2]~~ 0.1709	~~15.36~~ 5.0117	~~NA[2]~~ 0.8565
6	RHR Class 1 piping	NA	SS	~~NA[2]~~ 0.1279	~~15.36~~ 7.79	~~NA[2]~~ 0.9961

Figure 3. Updated LRA table 4.3-14, including "refined" CUF_{en} values for IP3 NUREG/CR-6260 locations. Source: Attachment 1 to NL-10-082, at 2 (Exh. NYS000352).

In response to communication NL-10-082, intervenors moved to admit a "New and Amended Contention," denominated Consolidated Contention NYS-26B/RK-TC-1B, that challenged various aspects of the new CUF_{en} calculations and continued to argue that Entergy had not submitted an adequate plan to manage the aging effects of metal fatigue. 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) (ML102670665).

Among other things, intervenors alleged that Entergy should have expanded the range of components for which CUF_{en} values were calculated, to identify potentially more limiting locations in the reactor coolant pressure boundary and to include reactor pressure vessel internals (RVIs). Petitioners State of New York and Riverkeeper, Inc. New and Amended Contention Concerning Metal Fatigue, at ¶¶ 11, 14, 16, 20 (Sept. 9, 2010) (included in ML102670665). The Board, again over the objections of Entergy and NRC Staff, admitted Consolidated Contention NYS-26B/RK-TC-1B and dismissed, as moot, Consolidated Contention NYS-26A/RK-TC-1A. *Entergy Nuclear Operations, Inc.* (Indian Point Nuclear Generating Units 2 and 3), 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), at 2 (Nov 4, 2010) (ML103080987) ("Summary Disposition Order").

In its ruling on the admissibility of NYS-26B/RK-TC-1B, the Board also addressed the applicability of the Commission's decision in *Entergy Nuclear Vermont Yankee, L.L.C., & Entergy Nuclear Operations, Inc.* (Vermont Yankee Nuclear Power Station), CLI-10-17, 72 NRC 1 (July 8, 2010), in which the Commission set forth various standards applicable to contentions related to metal fatigue contentions. The Board held that:

> While it is clear from the Commission's ruling in Vermont Yankee that an Applicant cannot be required to perform CUF_{en} calculations in evaluating TLAAs for meeting Section 54.21(c)(1)(i) and (ii), this is not an issue here because Entergy volunteered to perform CUF_{en} calculations in all instances and has used the results from these calculations in addressing Section 54.21(c)(1)(iii).

Summary Disposition Order at 13, n.58. effects of the coolant environment on component fatigue life.'" (Citation omitted)). In short, the Board held "once an applicant has chosen to include EAFs in its CUF calculations, there is nothing in NRC regulations or in the

Commission's recent decision in Vermont Yankee that prohibits an intervenor from questioning the adequacy, reliability, and breadth of these calculations when applied to Entergy's AMP under Section 54.21(c)(1)(iii), as New York and Riverkeeper have done." Summary Disposition Order at 22-3.

Finally, the Board rejected the assertion that a challenge to the CUF_{en} calculation in this case is prohibited by the Commission ruling in *Vermont Yankee*:

> NYS-26B/RK-TC-1B further differs from the contention that was held inadmissible in Vermont Yankee by challenging whether Entergy's AMP is adequate enough to meet Subsection (iii), and whether it meets the recommendations of the GALL Report. As the Commission noted:
>
>> An applicant may commit to implement an AMP that is consistent with the GALL Report and that will adequately manage aging. But such a commitment does not absolve the applicant from demonstrating, prior to issuance of a renewed license, that its AMP is indeed consistent with the GALL Report. We do not simply take the applicant at its word.
>
> New York and Riverkeeper have provided just such a challenge to the adequacy of Entergy's AMP and specifically to its FMP that has been designated by Entergy to serve as its AMP. Because Entergy calculated CUF analyses as part of its efforts to meet Subsection (iii), the methodology and breadth of these calculations may come under scrutiny.

Summary Disposition Order at 24 (citation and footnote omitted).

III. <u>Initial Briefing on Consolidated Contention NYS-26B/RK-TC-1B</u>

Intervenors filed an Initial Statement of Position on Consolidated Contention NYS-26B/RK-TC-1B, dated December 22, 2011 (Exh. NYS000343), which was supported by Dr. Lahey's December 20, 2011 Report (Exh. NYS000296), Dr. Lahey's the Supplemental Report concerning the WESTEMS computer code (Exh. NYS000297), Dr. Lahey's prefiled written testimony (Exh. NYS000344), the December 19, 2011 Report of Dr. Joram Hopenfeld (Exh. RIV000035), Dr. Hopenfeld's prefiled written testimony (Exh. RIV000034), and numerous

supporting technical documents. Entergy and NRC Staff responded with their own Statements of Position (Exh. ENT000182, Exh. NRC000101) and prefiled testimony (Exh. ENT000183, Exh. NRC000102). In reply, intervenors submitted further support for Consolidated Contention NYS-26B/RK-TC-1B in the form of a Revised Statement of Position, dated June 29, 2012 (Exh. NYS000439), pre-filed rebuttal testimony from Dr. Lahey (Exh. NYS000440) and Dr. Hopenfeld (Exh. RIV000114), and additional technical exhibits. Meanwhile, in February and March 2012, NRC Staff notified the Board and the parties that it could not then prepare a response on a related contention concerning embrittlement (NYS-25), that it also intended to release a Second Supplemental Safety Evaluation Report (SSER2) for Entergy's License Renewal Application (LRA) that would impact issues related to embrittlement and metal fatigue, and requested that a hearing on Consolidated Contention NYS-25 and related issues be delayed until the SSER2 was released. As a result, the proceeding with respect to Contentions NYS-25, NYS-26B/RK-TC-1B, and NYS-38/RK-TC-5 was essentially stayed for almost three years.

IV. Entergy's Commitments 43 and 49

On March 28, 2011, in response to a correspondence from NRC Staff, Entergy committed to conduct a plant-specific evaluation to determine whether the locations on which CUF_{en} calculations had been completed were the most limiting locations for IP2 and IP3. Commitment 43, Letter from Fred Damico to NRC, NL-11-032, Attachment 1, at 26 (March 28, 2011) (Exh. NYS000151). [redacted]

[redacted]

Table 1. Partial List of IP2 Components with CUF_{en} Values Reported Above 1.0 in Westinghouse's 2012 Evaluation

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

[redacted]

Table 2. Partial List of IP3 Components with CUF_{en} Values Reported Above 1.0 in Westinghouse's 2012 Evaluation

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
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[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

[redacted]

In October 2012, Entergy advised Staff that it intended to use the RVI Program to manage the cumulative fatigue aging effects for RVI components that have a time limited aging analysis that determined a CUF. Entergy letter NL-12-140, Attachment 1 (Exh. NYS000499); SSER2 at 3-52 (Exh. NYS000507). Entergy subsequently changed its position and, on May 7, 2013, advised Staff that it would rely instead on its Fatigue Monitoring Program to manage the effects of fatigue on RVIs during the period of extended operation. Entergy Letter NL-13-052, Attachment 1 at 8-9 (Exh. NYS000501). As part of this revision, Entergy also proposed a new Commitment 49, under which the company proposed to perform environmentally assisted fatigue analyses for RVIs by recalculating CUF values to include effects of the reactor coolant environment (CUF_{en}). *Id.*, Attachment 1 at 9; *id.*, Attachment 2 at 20 (proposed LRA Commitment 49). Entergy further indicated that it would undertake corrective action, including further CUF_{en} reanalysis, and/or repair or replacement of the affected component before CUF_{en}

reached 1.0. *Id.* Stated differently, Entergy proposed to take the CUF_{en} analysis used for other locations in the reactor coolant pressure boundary and apply or "import" that type of analysis into the category of reactor vessel internal components.

[redacted]

Notably, none of the details of the "refined" CUF_{en} calculations conducted by Westinghouse on external or RVIs in connection with Commitments 43 and 49 – including the CUF_{en} output values themselves – have been publically released. Instead, the State obtained the results in "calculation notes" submitted by Entergy in the course of mandatory disclosures, but designated as proprietary and thus subject to the Board's Protective Order. Following unsuccessful discussions to obtain the public release of some of the information in the calculation notes – in particular the CUF_{en} output values – the State moved to strike the designation of these documents, in their entirety, as proprietary. *See* State of New York Motion to Withdraw the Proprietary Designation of Various Pressurized Water Reactor Owners' Group

and Westinghouse Documents (April 9, 2015) (ML15099A785 [redacted version]). That motion is currently pending before the Board.

V. SSER2 and Revised Briefing on Consolidated Contention NYS-26B/RK-TC-1B

NRC Staff released the SSER2 in November 2014. NRC Staff acknowledged and found acceptable Entergy's proposal including Entergy's commitment to recalculate CUF_{en} values for RVI components pursuant to proposed Commitment 49. SSER2 at 3-53 (Exh. NYS000507). NRC Staff also certified that Commitment 33 had been completed for both IP2 and IP2, and that Commitments 43 and 49 had been completed for IP2. *Id.* at A-11, A-14 to A-15. Oddly, despite and the certification in the SSER2 of Commitments 43 and 49 as "completed" for IP2, NRC Staff has indicated that it did not possess the Westinghouse calculation notes that included the new CUF_{en} calculations prior to the State's motion for public disclosure, except for copies sent when Entergy disclosed the documents to the State. *See* NRC Staff's Answer to "State of New York Motion to Withdraw the Proprietary Designation of Various Pressurized Water Reactor Owners' Group and Westinghouse Documents" (April 20, 2015), at 5-6.

Following the release of the SSER2 in November 2014, the Board issued a Revised Scheduling Order, which included a provision for intervenors to file revised Statements of Position, Prefiled Testimony, and supporting exhibits to reflect events that have occurred since the first round of briefing in 2011 and 2012. *Entergy Nuclear Operations, Inc.* (Indian Point, Units 2 and 3), Revised Scheduling Order (December 9, 2014) (unpublished) (ML14343A757). Accordingly, Intervenors now submit this Revised Statement of Position, which is supported by revised prefiled testimony from Dr. Lahey (Exh. NYS000530) and Dr. Hopenfeld (Exh. RIV000142), a Supplemental Report from Dr. Hopenfeld (Exh. RIV000144), and additional supporting evidence addressing various recent developments. These documents show that,

despite the passage of time and multiple opportunities to supplement or correct the LRA, Entergy still has not developed an adequate plan to manage the effects of aging due to metal fatigue.

SUMMARY OF ARGUMENT

Consolidated Contention NYS - 26B/RK-TC-1B alleges:

> Entergy's License Renewal Application does not include an adequate plan to monitor and manage the effect of aging due to metal fatigue on key reactor components in violation of 10 C.F.R. § 54.21(c)(1)(iii).[5]

This contention challenges the adequacy of Entergy's reanalyses and its plan to manage the effects of metal fatigue during the requested extended license terms. *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). Under 10 C.F.R. § 54.21(c)(1)(iii), Entergy must establish that "[t]he effects of aging on intended functions(s) will be adequately managed for the period of extended operation." Entergy has failed to demonstrate that its AMP for metal fatigue is legally sufficient because (1) the methodology to determine whether CUF_{en} for any particular component is >1 - *i.e.* the WESTEMs computer program - is technically deficient; (2) the input values chosen by Entergy for its use of WESTEMs are not technically defensible and understate the extent of metal fatigue; and (3) the range of components for which the CUF_{en} calculations are proposed to be conducted is too narrow.

In Applicant's Motion for Summary Disposition of New York State Contentions 26/26A & Riverkeeper Technical Contentions 1/1A (Metal Fatigue of Reactor Components) filed August 10, 2010, Entergy made clear that the legal bases upon which it relies for its metal fatigue AMP

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[5] Entergy has conceded its response to the metal fatigue issue is to seek to demonstrate that it complies with 10 C.F.R. § 54.21(c)(1)(iii) and thus that it must demonstrate that "[t]he effects of aging on the intended function(s) will be adequately managed for the period of extended operation." *See* Summary Disposition Order, at 5.

is that it complies with NUREG-1801, Rev. 1, *Generic Aging Lessons Learned* ("GALL") and NUREG/CR-6260 (Feb. 2005)(NYS000146A-C), *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (Feb. 1995)(NYS000355). However, to prevail in its argument Entergy must demonstrate both that it does in fact comply with those requirements of those two guidance documents and that compliance with them is adequate to meet the requirements of 10 C.F.R. § 54.21(c)(1)(iii). As discussed below, Entergy fails to accomplish either of these goals.

First, Entergy conducted a CUF_{en} analysis of certain components as part of its initial LRA filing and found several components whose CUFen values were >1. LRA at Tables 4.3-13 and 4.3-14. When components are found not to comply with the acceptance criteria (i.e., CUF_{en} >1), "corrective actions" must be taken, which "include a review of additional affected reactor coolant pressure boundary locations." GALL at X M-2 (NYS000146A-C); *see also* MRP-47, Revision 1, Electric Power Research Institute, *Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application*, at 3-4 (2005) ("MRP-47") (NYS000350). Entergy failed to expand its analysis of relevant locations to the extent required by GALL.

Second, by committing to compliance with GALL, Entergy is committed to a metal fatigue AMP that "addresses the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant" and is capable of "[m]aintaining the fatigue usage factor below the design code limit." GALL, Rev. 1, Vol. 2 at X M-1 (NYS000146A-C). However, it is not sufficient for Entergy to merely assert that its AMP methodology for calculating CUF_{en} "addresses the effects of the coolant environment on component fatigue life" and will "[maintain] the fatigue usage factor

below the design code limit. It must demonstrate that the methodology it has accepted will actually achieve those goals." Summary Disposition Order, at 24.

New York and Riverkeeper offer the opinion of Drs. Lahey and Hopenfeld, as well as numerous documents, that demonstrate in several important respects how the AMP methodology that Entergy intends to use for its AMP is deficient and is unable to achieve the GALL goals to which Entergy is committed. In short, Entergy has not established that it has an adequate plan to manage the effects of metal fatigue on key reactor components, in violation of 10 C.F.R. §§ 54.21(a)(3), 54.21(c)(1)(iii) and 54.29(a).

ARGUMENT

Fatigue is a very important age-related safety concern, particularly when a significant plant life extension is being considered. In fact, it is one of the primary things that must be considered when doing a time-limited aging analysis (TLAA) or developing a plant-specific aging management program (AMP) for the extended operation of a nuclear reactor. Significantly, CUF_{en} must be less than 1.0 during extended plant operations. Dr. Lahey and Dr. Hopenfeld have identified a variety of potential sources of error in the calculation of CUF_{en} values relied upon by Entergy and Westinghouse. Considering how close many of the reported CUF_{en} values are to 1.0 – even after conservatisms have been systematically stripped out of the equation – virtually any error in the calculations could mean that some CUF_{en} values exceed the 1.0 threshold. Accordingly, there is a real risk that fatigued components could fail during the period of extended operation. In these circumstances, Entergy has failed to establish that it has an adequate plan to manage the effects of metal fatigue on key reactor components, in violation of 10 C.F.R. §§ 54.21(a)(3), 54.21(c)(1)(iii) and 54.29(a).

[redacted]

[redacted]

[redacted]

I. Reports and Testimony of Dr. Richard T. Lahey

Dr. Lahey has described his ongoing concerns with Entergy's reliance on the WESTEMS computer code and its analytical results to demonstrate that it has a legally adequate AMP for metal fatigue. Revised Lahey PFT, at 62-73 (Exh. NYS000530); Lahey Report, ¶¶24-31 (Exh. NYS000296); Lahey Supp. Report (Exh. NYS000297). First, Dr. Lahey describes how Westinghouse has systematically removed conservatisms from the CUF_{en} cacluation. Second, Dr. Lahey has identified sources of non-conservatism in the calculation of CUF_{en}. Third, Dr. Lahey has noted that without an error analysis, it is inappropriate to rely on CUF_{en} values that are very close to 1.0. Fourth, Dr. Lahey has provided real-world examples of the limitations of CUF_{en} calculations.

A. Westinghouse's "Refined" Calculations Systematically Removed Conservatisms Built Into the CUF_{en} Calculation

Dr. Lahey has described the process by which Westinghouse has systematically removed conservatisms built into the CUF_{en} calculation in order to obtain a result below the 1.0 threshold.

With respect to the 2010 "refined" CUF_{en} results filed with the ASLB by Entergy in NL-10-82, Dr. Lahey observed that the previously most limiting CUF_{en} were reduced by more than an order of magnitude (*e.g.*, the results for the pressurizer surge line piping and the RCS piping charging system nozzle), Attachment 1 to NL-10-082, at 2-3 (Exh. NYS000352), and opined that this was a very significant reduction that significantly reduced the design safety margins implicit in the original ASME code evaluations. Lahey Report, at ¶34 (Exh. NYS000296). Additionally,

Lahey expressed his concern that NL-10-082 reported, for the first time, limiting fatigue analysis results were given for the residual heat removal (RHR) system piping and nozzles, and the results for these components were very close to the unity limit. In particular, for the IP-2 RHR line, CUF_{en} = 0.9434, and for the IP-3 RHR line, CUFen = 0.9961. *Id.* ¶33.

Dr. Lahey has also expressed concerns with respect to Westinghouse's additional round of CUF_{en} calculations in 2012 and 2013 for additional external locations as well as internal locations, using its proprietary WESTEMS computer program. [redacted]

[redacted]

B. Westinghouse Failed to Consider Synergistic Aging Effects in Its CUF_{en} Calculations, Rendering the Resulting CUF_{en} Values Non-conservative

Next, Dr. Lahey notes that the calculated CUF_{en} values are non-conservative, because they do not account for the synergistic effects of embrittlement and other aging degradation mechanisms. Revised Lahey PFT, at 63-64 (Exh. NYS000530). There are a number of degradation mechanisms which impact the integrity of critical components inside the reactor pressure vessel and within the reactor coolant pressure boundary that are not accounted for in the CUF_{en} calculation. Revised Lahey PFT, at 14-15 (Exh. NYS000530). Some of these components, like the reactor vessel internals, are subjected to metal fatigue, embrittlement and stress corrosion cracking. The allowable cycles to failure for components used in the CUF_{en} calculation are determined from small scale experiments using metal test samples which are exposed to simulated reactor coolant environments. *Id.* at 64. However, the fatigue experiments do not use highly embrittled metal test samples, which have a lower expected fatigue life than non-embrittled (ductile) materials. *Id.* The failure to consider the effects of embrittlement and other degradation mechanisms when calculating CUF_{en} values means there is a real possibility that a component could fail, even if its CUF_{en} value is below 1.0. In particular, Dr. Lahey has expressed concern that a highly fatigued, embrittled, or otherwise degraded component, which

does not yet have signs of significant surface cracking, could be exposed to an unexpected seismic event or shock load that could cause it to fail. *Id.* at 68-69. [redacted]

C. <u>Without an Error Analysis, It Is Not Possible to Determine Whether CUF_{en} Values Exceed 1.0</u>

Dr. Lahey has consistently objected to Westinghouse's failure to include an error analysis along with its CUF_{en} calculation results, especially when some of the CUF_{en} values were extremely close to 1.0. Revised Lahey PFT, at 70; Lahey Report, ¶34. Indeed, no error analysis of the WESTEMS results was generated by Westinghouse nor provided to the Board or New York by either Entergy or Westinghouse, nor were any results provided showing that the computational results exhibited nodal convergence, or how they were bench-marked against representative experimental data and/or analytical solutions. Lahey Report, ¶34. Dr. Lahey has testified that one would normally expect to see a detailed 'propagation-of-error' type of analysis, *see, e.g.*, Vardeman & Jobe, "Basic Engineering Data Collection and Analysis," Duxbury, pp. 310-311 (2001) (NYS000347), to determine the overall uncertainty in the CUF_{en} results given by Westingouse. Revised Lahey PFT, at 68. It is well known that all engineering analyses are based on imperfect mathematical models of reality and various code user assumptions which inherently involve some level of error. In addition, as the USNRC Staff confirmed in the SSER, NUREG-1930, Supplement 1 at 4-2 (NYS000326A-F), WESTEMS permits the code user to make assumptions and interventions that can affect the outcome. Lahey Report, ¶34. As a consequence, without a well-documented error analysis, the accuracy of Entergy's new fatigue results are quite uncertain and thus cannot be used to establish the integrity of the structures,

components or fittings being analyzed. *Id.* What is clear is that there are many possible sources of error in the results that Entergy (and Westinghouse) have provided to the ASLB and the parties. Lahey Supplemental Report, ¶5 (Exh. NYS000297).

Dr. Lahey expressed particular concern that the residual heat removal (RHR) systems at IP2 and IP3 have CUF_{en}'s that may exceed 1.0, if error analysis were included. The SMiRT-19 paper, Cranford & Gary-W, 8/07 (NYS000360), notes that a PWR's residual heat removal (RHR) system shares nozzles and piping with the plant's emergency core cooling system (ECCS). In particular, in the fatigue analysis for the ECCS accumulators (which passively inject ECC water into the cold leg of the primary system in the event of a LOCA) and the RHR system (which is normally used during each plant shut-down) one must combine the fatigue usage of both systems for their common components (i.e., the nozzle at the penetration into the cold leg) to obtain the resultant CUF_{en}. This design feature implies that a RHR/accumulator nozzle failure during a LOCA may breach the path for accumulator water injection into the RPVs downcomer region, thus preventing ECC water from reaching the core to mitigate core melting. This is obviously one of the most serious primary system boundary failures that can occur in a PWR. Hence, it is very troubling that one of the largest CUF_{en} calculated for the various components analyzed by Westinghouse were for this critical component. That is, CUF_{en} = 0.9961 for IP-3 RHR, NL-10-082, Table 4.3-14(NYS000352), and CUF_{en} = 0.9434 for IP-2 RHR, *id.*, Table 4.3-13, where both of these values were nearly equal to the CUF_{en} limit of 1.0. [redacted]

[redacted] Other reactor components analyzed using WESTEMS also had disturbingly large values for CUFen. NL-10-082, Tables 4.3-13, 4.3-14 (NYS000352).

Dr. Lahey has noted that there is significant “engineering judgment” implicit in the CUF_{en} results, and, since an error analysis has not been done to bound the uncertainty, and many results are disturbingly close to the CUFen = 1.0 limit, the results produced are not a reliable basis to assure the safety of IP-2 and IP-3 during extended plant operations. Lahey Report, ¶36. Indeed, these results are quite uncertain and this uncertainty should be quantified by doing parametric runs and a detailed error analysis. *Id.* Moreover, because the effect of various shock loads on the failure of these fatigue-weakened components, structures, and fittings has not been considered, the health and safety of the American public is not being adequately protected. *Id.* As previously noted, these results are quite uncertain and this uncertainty needs to be quantified by doing a detailed error analysis. *Id.* Although the USNRC Staff required Entergy to disclose and make clear those user interventions that will be used in future WESTEMS analysis of IP-2 and IP-3, SSER, NUREG-1930, Supplement 1 at 4-2 (NYS000326A-F), this disclosure was not required for the results that Entergy has already submitted to the ASLB for IP-2 and IP-3. This information is needed in order to do a proper review of these important results.

Entergy has objected to intervenors’ request for an “error analysis” of the Westinghouse calculations, claiming that “the actual objection of an EAF analysis is to determine whether or not the CUF_{en} exceeds 1.0, not to calculate a precise CUF_{en} value.” Entergy’s Initial Statement of Position Regarding Contention NYS-26B/RK-TC-1B (Metal Fatigue), at 26 (March 29, 2012)

(footnote omitted) (Exh. ENT000182). However, this argument is non-sensical. A determination of whether or not a certain CUF_{en} exceeds 1.0 requires an understanding of the calculation's degree of accuracy. For example, if the CUF_{en} calculation has an error rate of 0.1 (or ten percent), then any calculated CUF_{en} value above 0.9 might or might not exceed 1.0 – the calculation would not be sufficiently accurate to make that determination.

Entergy has also identified various conservatisms that it alleges have been retained in the CUF_{en} calculations. *Id.* at 38-41. However, neither Entergy nor Westinghouse have quantified the extent of these conservatisms. On the other hand, Dr. Lahey has shown that there are substantial uncertainties and non-conservatisms in the CUF_{en} calculations, the full extent of which has not been evaluated. *See* Revised Lahey PFT, at 63-64 (Exh. NYS000530); Lahey Report, ¶¶34. Furthermore, Westinghouse has systematically stripped conservatisms out of the CUF_{en} calculations in order to bring the results below 1.0. If the error rate or non-conservatisms for the CUF_{en} calculation is greater than the remaining conservatisms, then the results may not be conservative. In short, Entergy has failed to establish that the extent of alleged conservatisms in the CUF_{en} calculation is greater than the non-conservatisms and potential errors.

D. Real World Operating Experience Illustrates the Limitations of CUF_{en} Calculations

Lastly, Dr. Lahey notes that in-core fatigue failures of irradiated baffle-to-former bolts have been observed in operating PWRs. *E.g.*, WCAP-14577, Rev. 1, "License Renewal Evaluation: Aging Renewal Evaluation: Aging Management of Reactor Internals," pg. 2-29 (Oct. 2000) (NYS000324); USNRC Staff Report, "Final Safety Evaluation by the Office of Nuclear Reactor Regulation Concerning Westinghouse Owners Group Report, WCAP-14575, Revision 1, License Renewal Evaluation: Aging Management for Class 1 Piping and Associate Pressure

Boundary Components, Project No. 686," (Nov. 8, 2000) NYS000353), and B&W designed PWRs have had fatigue-induced failures of various in-core components even when CUF < 1.0 (perhaps due to undetected manufacturing flaws). Entergy Email: Esquillo to Stuard et al., Subject: "Section XI - Cracking" (8/30/06) NYS000316). Significantly, the possible effect of fatigue on the failure of RPV internals was apparently well known to Entergy. Entergy Email: Batch to Finnin, Subject: "Need to Evaluate High Cycle Fatigue to IPEC Baffle Bolts?" (12/28/06) (NYS000315). Moreover, unlike postulated nuclear reactor accidents, the fatigue failures of in-core bolts are actual events that have happened and will likely happen again for sufficiently stressed materials. It is not possible to inspect (*e.g.*, using UT techniques) all the bolts within a RPV, and thus the nuclear industry has recommended, EPRI Report, MRP-228; "Materials Reliability Program: Inspection Standard for PWR Internals," (July 2009) (NYS000323), that an analysis be done to support continued operations if bolt failures are found during in-core non-destructive evaluations (NDE). However, it appears that these analyses do not take into account the possibility of various accident-induced pressure and/or thermal shock loads within the RPV, such as those due to a DBA LOCA. Thus, the number of intact bolts which might be adequate for normal operations may be totally inadequate to accommodate shock loads during accidents. Not doing a realistic safety analysis is totally unacceptable since shock-load-induced bolting failures may lead to a blocked or distorted core geometry which, in turn, may not allow the ability to adequately cool the core and can lead to core melting.

More recently, in May 2015, a steam generator feedwater line at IP3 failed and the plant had to be shut down as a result. Event Report 51046 (Exh. NYS000548). The 2007 LRA listed a CUF of 1.0 for the feedwater nozzle for the full period of extended operation (i.e. 60 years). LRA, at 4.3-15, Tbl. 4.3-10. Accordingly, it appears that the feedwater line failed before the

relevant CUF value reached 1.0 – and before the steam generator reached 27 years of operation. Notably, Entergy has previously rejected intervenors' concerns regarding steam generator feedwater lines, noting that "historic issues with failure of feedwater distribution components in steam generators are due to *erosion* – not fatigue . . . – and have been addressed in response to generic industry communications." Entergy's Initial Statement of Position on Consolidated Contention NYS-26B/RK-TC-1B, at 46 (Exh. ENT000182); [redacted] The fact that another aging degradation mechanism may have contributed to the feedwater line failure underscores Dr. Lahey's testimony regarding the shortcomings of a CUF calculation that considers only fatigue. *See* Revised Lahey PFT, at 63-64 (Exh. NYS000530). Moreover, the fact that a feedwater line unexpectedly failed notwithstanding Entergy's belief that the issue had been "addressed" emphasizes Dr. Lahey's testimony regarding the need to maintain safety margins during the period of extend operation. *Id.* at 74-75.

II. <u>Reports and Testimony of Dr. Joram Hopenfeld</u>

Dr. Hopenfeld has previously submitted a Report and Prefiled testimony describing deficiencies in the methods used by Westinghouse in 2010 to calculate "refined" CUF_{en} values for components identified in NUREG/CR-6260, and Entergy's otherwise inadequate program for managing metal fatigue during the proposed PEOs. *See* Hopenfeld Report (Dec. 22, 2011) (Exh. RIV000035); Hopenfeld PFT (Dec. 22, 2011) (Exh. RIV000034). He also opined that Entergy should be required to conduct CUF_{en} calculations for additional locations, to determine if any locations were more limiting than those identified in NUREG/CR-6260. *See* Hopenfeld Report, at 22-25 (Dec. 22, 2011) (Exh. RIV000035).

[redacted]

[redacted] Accordingly, Westinghouse conducted further CUF_{en} analyses that stripped out various conservatisms in order to obtain CUF_{en} values below 1.0. Dr. Hopenfeld has now submitted a Supplemental Report and Revised Prefiled Testimony that describe the continuing deficiencies in the most recent round of CUF_{en} calculations and Entergy's AMP for metal fatigue. Hopenfeld Supplemental Report (Exh. RIV000144); Hopenfeld Revised PFT in Support of Consolidated Contention NYS-26B/RK-TC-1B (Exh. RIV000142). Accordingly, this Revised SOP will first reiterate Dr. Hopenfeld's criticisms of the 2010 CUF_{en} calculations, and then describe his criticisms of the subsequent CUF_{en} calculations.

A. Entergy's and Westinghouse's "Refined" Analyses of Environmental Fatigue Factors in 2010 Were Flawed and Inaccurate, and Failed to Demonstrate that Certain Components Will Not Fail During the Period of Extended Operation.

Dr. Hopenfeld reviewed various Entergy documents that described the "refined" CUF_{en} analysis conducted by Westinghouse for Entergy in 2010.[6] Dr. Hopenfeld found there is a wide margin of error in the calculations, due to Entergy's failure to adequately address many critical underlying assumptions that would be a part of a proper fatigue analysis. Entergy's calculations likely grossly under-predicted the CU_{Fen} values for the components evaluated for the following reasons.

1. The calculations fail to properly adjust laboratory data to account for the actual reactor environment in the calculation of the Fen factors to apply.

Due to significant differences that exist between the laboratory and reactor environment,

[6] Exh. RIV000035, 4-21. Westinghouse, "EnvFat User's Manual Version 1," May 2009, at § 2 (IPECPROP00056785), Exh. RIV000036. Environmental Fatigue Evaluation for Indian Point Unit 2, WCAP-17199-P, Revision 0 (Westinghouse, June 2010), IPECPROP00056486, Exh. NYS000361, Environmental Fatigue Evaluation for Indian Point Unit 3, WCAP-17200-P,

there are numerous uncertainties in applying the Fen equations to actual reactor components. In NUREG/CR-6909, Effect of LWR Coolant Environment on Fatigue Life of Reactor Materials, Argonne National Laboratories ("ANL") identifies numerous such uncertainties, which include such things as material composition, component size and geometry, loading history, strain rate, mean stress, water chemistry, dissolved oxygen levels, temperature, and flow rate. *See* NUREG/CR-6909, at 72 (Exh. NYS000357). Such uncertainties can have a significant effect upon fatigue life and ignoring them will result in underestimated CUF_{en} calculations. For example, variations of temperature when temperature is below 150°C can reduce fatigue life by a factor of two, and increased water conductivity due to the presence of trace anionic impurities in the coolant, which has already been documented to cause stress corrosion cracking at several nuclear plants, may decrease fatigue life of austenitic stainless steels. In addition, surface temperature fluctuations and non-uniform temperature distributions during stratification can increase the potential for crack initiation and growth, thereby reducing fatigue life. So, to appropriately apply the F_{en} equations to actual reactor components, the user must consider all of the relevant uncertainties, and the results must be adjusted to account for the varying parameters. In NUREG/CR-6909, ANL further specifies that appropriate bounding F_{en} values of 12 for stainless steel and 17 for carbon and low alloy steel to account for the numerous uncertainties in using the F_{en} equations. *See* NUREG/CR-6909 at iii, 3 (Exh. NYS000357). These bounding Fen factors are not necessarily conservative, and it is reasonable to expect even higher Fen values in the actual reactor environment, especially for those components that experience stratified flows and thermal striping.

Dr. Hopenfeld's testimony describes how Entergy's "refined" EAF analyses have not

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Revision 0 (Westinghouse, June 2010), IPECPROP00056577, Exh. NYS000362.

adequately evaluated the numerous uncertainties associated with determining acceptable F_{en} values, or, in the alterative, applied bounding F_{en} values to conservatively ensure that such uncertainties are accounted for. Entergy's calculation of fatigue life for selected components would be significantly affected if all relevant uncertainties were actually considered. Entergy's failure to properly account for the numerous uncertainties inherent in determining the appropriate F_{en} value from using equations derived from laboratory tests has resulted in calculations that underestimate the CUF_{en} for the analyzed components. The use of the bounding recommended by ANL, which represent far more realistic values than most of those calculated and used by Entergy, would increase the CUFen values beyond unity for a number of the components analyzed, as Dr. Hopenfeld calculated in his report.

2. The calculations use incorrect values for dissolved oxygen ("DO") levels in the calculation of the Fen factors to apply.

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 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 F_{en} by a factor of 55. The value of F_{en} 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. The difficulty of determining DO levels during transients is well described by EPRI in its guidelines for addressing fatigue in a LRA. EPRI, Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, MRP-47 at 4-19 (Exh. NYS00035). *See also id.* at 4-27. This is particularly true during startup and shutdown transients. 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. *See* John J. Taylor, *R&D Status Report*, Nuclear Power Division, EPRI Journal (Jan/Feb 1983) (Exh. RIV000040). Similar oxygen dependence on temperature can be expected in Pressurized Water Reactors (PWRs).

ANL, the developers of the F_{en} equations, specifically instruct how users should account for transients where temperature varies, as described in the following NRC reports; NUREG/CR-6583 states "the values of temperature and DO may be conservatively taken as the maximum values for the transient." NUREG/CR-6583 at 78 (Exh. NYS000356). 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 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." NUREG/CR-6909 at A-5 (Exh. NYS000357).

Entergy's approach for estimating DO values during transients in calculating Fen is in direct contravention of the specifications provided by ANL and endorsed by NRC, as well as basic laws of physics, and was, therefore, inappropriate and misguided. Entergy has not provided an adequate rational for failing to abide by well-founded recommendations.

3. The calculations do not accurately consider heat transfer coefficients in the calculation of CUF_{en} values.

Heat transfer is a major factor in the determination of CUF_{en} because it controls the cyclic thermal stresses during transients. Thermal stresses arise when there is a change in the local

fluid temperature, such as 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.[7] As of 2007, at least thirteen cases of leakage from thermal fatigue have occurred at nuclear reactors. *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") (Exh. RIV000048). Based on this experience, the International Atomic Energy Agency ("IAEA") has concluded that the frequency of leakage from thermal fatigue will increase with time. *Id.* at 52.

In order to calculate thermal stress and its impact on fatigue life, the temperature distribution of a component during a transient must be determined, or, in other words, the rate at which heat is transferred to the reactor component surface during the transient. Thermal-

[7] 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), Exh. RIV000044; 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), Exh. RIV000045; 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), Exh. RIV000046; 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 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), Exh. RIV000047.

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. Exh. RIV000035 at 14. The coefficient h is an experimental parameter, and has been measured under a range of conditions. It 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.[8] Increase in h increases the corresponding stress and reduces fatigue life. 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. 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. *See* Entergy, Indian Point Unit 3 Flow Accelerated

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[8] 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

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) (Exh. RIV000049). One diagram indicates that in components where flow is not fully developed, component wall thickness can vary by more than 400% at Indian Point. *See id.* (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%).

Dr. Hopenfeld's analysis of Entergy's heat transfer calculations concludes that Entergy employed unrealistically low heat transfer coefficients in the determination of CUF_{en}. A more realistic selection for this key parameter indicates that the CUF_{en} is significantly larger than predicted by Entergy. More realistic heat transfer calculations alone could have increased the CUF_{en} value by as much as a factor of two for several components. Only modest modifications in this value to account for the inherent uncertainties would cause corresponding CUF_{en} for IP 2 to exceed unity.

4. The calculations use an unjustified number of transients in the calculation of CUFen values.

The number of transients used in Entergy's "refined" calculations directly affects the resulting CUF_{en} value. However, the actual number of transients during the proposed extended operation period is unknown. This casts further doubt on the CUF_{en} methodology used by Entergy and, in turn, the accuracy of the new calculations.

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. It is apparent that Entergy has not adequately considered either past or future transients at Indian Point.

To assess the severity of past transients, each transient that has occurred must be

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a concern. *See* NRC Bulletin No. 88-08, *supra*.

described to determine its contribution to the CUF_{en}. 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.

Data regarding events at the Indian Point reactors in the past is incomplete and unreliable. It would appear that Entergy has relied upon some undisclosed model in using other plant data in developing the number of transients for the new calculations. For example, stratification is a plant-specific phenomenon that generally requires 3D modeling; Entergy has failed to show how the number of transients concerning stratification was developed for the pressurized surge line by virtue of using other plant data. Entergy's methodology for calculating the likely future number of transients is also suspect and Entergy fails to provide a credible justification for the extensive engineering judgment upon which it relies for determining future transients.

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" trend. This phenomenon dictates that at the beginning and at the end of life, component failure occurs at a very high frequency. *See* Figure 4. In between these two extremes, the failure is relatively low and constant. For this reason Entergy must justify its use of the straight-line extrapolation for the number of transients it assumed in calculating the CUF_{en} values. Even if Entergy's approach was justifiable, there is doubt they have used the approach correctly in the calculations relied upon.

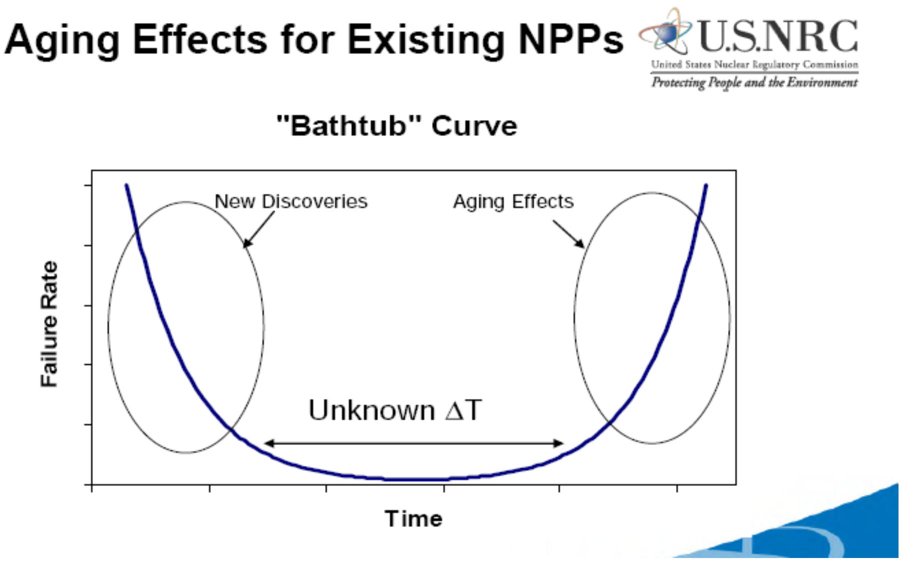

Figure 4. Graph showing "bathtub curve," which predicts that component failure is highest at the beginning and end of nuclear power plant (NPP) operation. Source: USNRC, Slides, "NRC Aging Management Program Including Long Term Operation (LTO)," presented at Workshop on Challenges on the Long Term Operation, 21st SMIRT Meeting, New Delhi, India (Nov. 8-11, 2011), at 7 (ML111801154) (Exh. NYS000305).

B. Entergy's and Westinghouse's Calculation of CUF_{en} Values for Additional Components in Connection with Commitments 43 and 49 Are Flawed and Unreliable.

Dr. Hopenfeld's Supplemental Report (Exh. RIV000144) details the various shortcomings of the additional rounds of CUF_{en} calculations conducted for Entergy by Westinghouse in connection with Commitments 43 and 49. Some of these deficiencies are the same as those described above for the 2010 calculations, reflecting Entergy's and Westinghouse's continued failure to address concerns raised by intervenors in this proceeding. Dr. Hopenfeld has identified four categories of errors in the calculations: (1) they do not properly account for the effects of dissolved oxygen on component fatigue; (2) they do not account for radiation and stress corrosion effects on metal fatigue; (3) they use the 40-year-old CUFs of record in the fatigue analyses; and (4) they have not properly expanded the scope of analysis to bound the most limiting locations. Hopenfeld Supplemental Report, at 6-29 (Exh. RIV000144).

In light of these errors, Dr. Hopenfeld opines that Entergy has not demonstrated that the CUFs of the components at Indian Point will not exceed unity or succumb to metal fatigue during the proposed period of extended operations.

1. Westinghouse and Entergy Have Failed to Properly Account for the Effects of Dissolved Oxygen on Component Fatigue.

Dr. Hopenfeld first describes non-conservatisms in the CUF_{en} calculations resulting from the inadequate consideration of dissolved oxygen (DO) levels. Hopenfeld Supplemental Report, at 6-13 (Exh. RIV000144). Dr. Hopenfeld describes how the formation of oxide films on the surface of stainless steel is believed to play a major role in metal fatigue, but the full effect is not completely understood. *Id.* at 7. In light of this uncertainty, 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. *Id.* Oxygen concentrations in the laboratory tests were uniform in liquid and were conducted at steady state with controlled water chemistry and temperature. *Id.* 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, usually far removed from the component of interest during thermal transients. *Id.*

Dr. Hopenfeld describes several mechanisms that lead to the introduction of DO into the reactor system, and describes guidance regarding how to account for DO levels developed by ANL and EPRI. *Id.* at 7-8. [redacted]

[redacted] Indeed, Westinghouse and Entergy have not provided documentation of plant-specific measurements of DO levels during transients at key locations. *Id.*

[redacted]

2. Westinghouse and Entergy Have Failed to Adequately Consider the Effects of Radiation and Stress Corrosion in Calculating CUF_{en} Values

Dr. Hopenfeld next describes various shortcomings in the CUF_{en} calculations for RVIs relating to Westinghouse's failure to account for the effects of radiation and stress corrosion. Hopenfeld Supplemental Report, at 13-18 (Exh. RIV000144). These concerns mirror Dr. Lahey's testimony regarding the effects of embrittlement and other aging degradation mechanisms on fatigue life. *See* Revised Lahey PFT, at 63-64 (Exh. NYS000530). In particular, Dr. Hopenfeld reiterates that neutron irradiation reduces ductility and increases crack propagation in stainless steels, such as those used in RVI components. Hopenfeld Supplemental Report, at 13-14 (Exh. RIV000144). Dr. Hopenfeld cites reports suggesting that irradiation can reduce fatigue life by one-half for some stainless steels, and can increase crack growth rates by a

factor of 5. *Id.* Dr. Hopenfeld also notes that stress corrosion cracking (SCC) is a separate aging degradation effect from fatigue, and can lead to a further reduction in fatigue life. *Id.* at 13. Dr. Hopenfeld opines that a multiplier of 5 would be an appropriately conservative factor to account for radiation and SCC. *Id.* at 14-15.

[redacted] Dr. Hopenfeld observes that the failure to consider those effects is apparent from their reliance on NUREG/CR-6909 and is confirmed by the SSER2. *Id.* at 15; *see* SSER2, at 3-51 to 3-53, A-14, A-15 (Exh. NYS000507). The apparent justification for not including the synergistic effects of multiple aging effects appears to relate to the assumption that radiation embrittlement will not have a significant effect in the absence of pre-existing flaws or cracks, and that crack initiation is not expected in components with CUF values below 1.0. Hopenfeld Supp. Report, at 15-16. Dr. Hopenfeld demonstrates the flaws in this reasoning, noting that a CUF_{en} of less than one does not necessarily demonstrate that fatigue initiation will not occur during the life of the component. *Id.* at 16-17. Rather, a schematic of crack formation during fatigue life in NUREG/CR-6909 shows that crack or flaws are present from the beginning of the test, throughout the fatigue life of the specimen under cyclic loads. *Id.*; *see* Argonne National Laboratory, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials: Final Report," NUREG/CR-6909, at 7 (Feb. 2007) (Exh. NYS000357). The transition from microscopic cracks to macroscopic cracks occurs at about the time a component reaches its half-life. Hopenfeld Supplemental Report, at 16 (Exh. RIV000144). One study has indicated that when a component experiences a high level of cyclic stress corresponding to a usage factor of less than 1.0, "very small cracks can propagate to sizes that exceed acceptance criteria." G.T.

Yahr, et al., "Case Study of the Propagation of a Small Flaw Under PWR Loading Conditions and Comparison with the ASME Code Design Life," at 2 (RIV000118).

Moreover, Dr. Hopenfeld notes that fatigue in conjunction with other factors may result in synergistic effects on plant components. Hopenfeld Supplemental Report, at 16-17 (Exh. RIV000144). He specifically notes that fatigue and neutron irradiation may act together and accelerate the individual effects. *Id.* Therefore, a CUF_{en} calculation below 1.0 that does not account for radiation effects is inaccurate and unreliable. *Id.* Accordingly, the assumption that synergistic effects need not be considered when the calculated CUF_{en} – which does not include synergistic effects – is below 1.0 "defies logic," according to Dr. Hopenfeld. *Id.* [redacted] Without any analysis of the error rate in the CUF_{en} calculations, it is impossible to determine how non-conservative the "refined" evaluations truly are.

3. The CUF_{en} Calculations Conducted by Westinghouse Are Unreliable [redacted]

Dr. Hopenfeld next criticizes the methodology used to calculate refined CUF_{en} values for relating CUF_{en} to the CUF of record [redacted] Hopenfeld Supplemental Report, at 18-24 (Exh. RIV000144). The CUF of record for Entergy's current licensing basis (CLB) were calculated more than 40 years ago, and were based on calculations that were valid when the plants were initially designed because all of the components were presumably in pristine condition. *Id.* at 18. The F_{en} was determined by comparing fatigue life of similar test specimens in air and water, and applying the F_{en} equation set forth in NURG/CR-

6909, *see id.* at 5, to the plant, with the presumption that the CUF of each component was calculated on the basis that the geometry was well known. *Id.*

However, after 40 years of exposure to a hostile LWR environment, Dr. Hopenfeld notes that most of the components have undergone a change in geometry and surface structure due to erosion/corrosion, stress corrosion, swilling, pitting, and cavitation. *Id.* at 18. The mechanical properties of RVI components also changes due to exposure to radiation. *Id.* Such changes affect the components' fatigue life because they introduce local discontinuities that introduce high local stress concentrations. *Id.* Dr. Hopenfeld notes that the ASME fatigue curves are based on average stresses only, and that more than 100 years of experience has been accumulated to show that sharp surface discontinuities introduce high local stress concentrations where cracks are initiated. *Id.* These changes must be applied to the CUF of record if the F_{en} equation is used to calculate CUF_{en} values. *Id.* at 19. [redacted]

Dr. Hopenfeld describes, in detail, the various factors that have rendered the CUF of record obsolete, including geometry changes, surface finish, heat transfer, strain rate, radiation effects and apparent unexplained changes to the CUF. *Id.* at 19-24. Dr. Hopenfeld notes that the geometry of many components may have changed due to FAC, which causes severe wall thinning and introduces surface discontinuities. *Id.* at 19. Similarly, changes in the surface finish of components may occur due to FAC, pitting, or radiation-induced void swelling, which also may lead to surface discontinuities which high surface stress may occur. *Id.* at 19-20. Additionally, the use of the CLB CUF of record results in a non-conservative CUF_{en}, because it relies on an over-simplified two-dimensional heat transfer model. *Id.* at 20-21. As Dr.

Hopenfeld has consistently testified in this proceeding, this model does not account for thermal stratification or "striping" which has a major effect on the fatigue life for certain components. *Id.* Furthermore, the CUF of record does not adequately account for strain rates or radiation effects, which have a significant effect on carbon and low allow steel. *Id.* at 22. [redacted] Dr. Hopenfeld opines that the refined CUF_{en} values are non-conservative and may exceed 1.0 for multiple components. *Id.* at 23-24.

4. Entergy and Westinghouse Have Failed to Determine the Most Limiting Locations

Dr. Hopenfeld next contends that Entergy and Westinghouse have failed to select the most limiting locations to evaluate CUF_{en} values. Hopenfeld Supplemental Report, at 24-27 (Exh. RIV000144). Initially, he observes that he previously explained the need for Entergy to expand the fatigue analysis beyond the locations identified in NUREG/CR-6260, but that Entergy refused, arguing that its analysis was conservative. *Id.* at 24. However, when Entergy finally did conduct further fatigue analyses at the behest of NRC Staff, it discovered that the initial evaluations [redacted] *Id.* Unfortunately, the expanded analysis remains inadequate and incomplete. *Id.*

Dr. Hopenfeld has previously explained the steps necessary to conduct a properly expanded fatigue analysis. Initial Hopenfeld PFT in Support of Contention NYS-38/RK-TC-5 (June 19, 2012) (Exh. RIV000102). Among the factors to consider, Dr. Hopenfeld observed the

importance of considering components that are vulnerable to thermal stratification and thermal striping. *Id.* at 12. [redacted] Additionally, Hopenfeld observes that Entergy has failed to consider the effects of environment on fatigue for components on the secondary side of the steam generators, and observes that even a minimal correction for environmental effects would cause the CUF_{en} values to exceed 1.0. *Id.* Dr. Hopenfeld specifically identifies various components that Entergy has continued to ignore, but which warrant fatigue review. *Id.* at 25-27.

First, Dr. Hopenfeld notes that the pressurizer surge line, mixing tees, and unisolable branches connected to the RCS piping are particularly susceptible to thermal stratification, and should have been assessed for fatigue life. *Id.* at 25. [redacted] noting that [redacted] an industry document (EPRI's MRP-47) that relies on the environmental correction factor F_{en}. *Id.* However, Dr. Hopenfeld notes several reasons that those F_{en} values are not applicable to components subject to thermal striping: (1) because the F_{en} values were derived from tests where the specimens were not subjected to highly concentrated local stresses from thermal striping, but rather to stress that were distributed over relatively large surface areas; and (2) F_{en} compares conditions in water to conditions in air, but thermal striping effects do not exist in air. *Id.*

[redacted]

[redacted]

Lastly, Dr. Hopenfeld argues that the reactor head penetrations and outlet inlet nozzle safe ends should be evaluated for CUF_{en}. *Id.* at 27. He notes that Alloy 600/82/182 is susceptible to primary water SCC, which can affect metal fatigue. *Id.* [redacted]

In short, Dr. Hopenfeld's evaluation demonstrates that Entergy has not shown that the CUF_{en} calculations it has conducted include the most limiting locations at IP2 and IP3. Accordingly, it remains possible that more limiting locations exist and could fail unexpectedly. 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.

Entergy has not demonstrated 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.

C. Entergy has otherwise not provided sufficient details concerning an Aging Management Program ("AMP") to ensure that the degradation effects of metal fatigue would be adequately handled during the proposed periods of extended operation.

Dr. Hopenfeld further discusses the inadequacy of Entergy's AMP for metal fatigue as a result of its flawed CUF_{en} analyses. *See* Hopenfeld Supplemental Report, at 32 (Exh. RIV000144); *see also* Hopenfeld Report (December 19, 2011), at 21-25 (Exh. RIV000035). 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." NUREG-1801, Rev. 1 § X.M1, 1, p. X M-1 (emphasis added) (NYS000146A-C); *see also* NUREG-1801, Rev. 2, § X.M1, 1 p. X M1-2 (NYS000147A-D) ("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.").

Entergy's plans for managing metal fatigue related degradation depend upon calculating the vulnerability of plant components. Indeed, Entergy intends to rely upon its 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 commitments to inspect, repair, and replace affected locations insufficient to ensure proper management of metal fatigue during the proposed period of extended operation.

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 failed to do this.

CONCLUSION

New York and Riverkeeper have presented a substantial body of technical expertise and documents to demonstrate that the AMP offered by Entergy to demonstrate compliance with 10 C.F.R. § 54.21(c)(iii) fails to achieve the goals set forth in GALL to which Entergy has committed compliance. For all the reasons stated, Entergy’s license renewal application does not comply with the applicable regulatory standards and should be denied.

Respectfully submitted,

__Signed (electronically) by__

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