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In the Matter of:	Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3)		
ASLBP #:	07-858-03-LR-BD01		
Docket #:	05000247 | 05000286		
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NRCR20101
Submitted: August 10, 2015

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket Nos. 50-247/286-LR
	)	
(Indian Point Nuclear Generating Units 2 and 3)	)	

NRC STAFF'S REVISED STATEMENT OF POSITION REGARDING NYS 26B/RK-TC-1B

INTRODUCTION

Pursuant to 10 C.F.R. § 2.1207(a)(2) and the Atomic Safety and Licensing Board's orders,[1] the Staff of the U.S. Nuclear Regulatory Commission ("Staff") submits its revised statement of position (NRCR20101) on the State of New York ("NYS" or the "State") and Riverkeeper, Inc.'s ("Riverkeeper" or "RK") (collectively, "Intervenors") Consolidated Contention NYS-26B/RK-TC-1B (Metal Fatigue of Reactor Components). The Staff's position is supported by the testimony of Dr. Allen Hiser, Dr. Ching Ng, Mr. Gary Stevens, P.E., and Mr. On Yee,[2] and is directed to the June 9, 2015 revised[3] statement of position of Intervenors[4] and their witnesses

[1] Licensing Board Scheduling Order (July 1, 2010) (unpublished) ("Scheduling Order"); Licensing Board Amended Scheduling Order (June 7, 2011) (unpublished); Licensing Board Revised Scheduling Order, at 2 (December 9, 2014) (unpublished) (permitting for the filing of new or revised written statements of position, written testimony with supporting affidavits, and exhibits); Licensing Board Order (Granting New York's Motion for an Eight-Day Extension of Filing Deadline) (May 27, 2015) (unpublished).

[2] NRC Staff Testimony of Dr. Allen Hiser, Dr. Ching Ng, Mr. Gary Stevens, P.E., and Mr. On Yee, Concerning Contentions NYS-26B/RK-TC-1B and NYS-38/RK-TC-5 (Ex. NRC000168) ("NRC Fatigue Test.").

[3] Per the Scheduling Order, as amended, by late 2011 the Intervenors timely filed their initial written statements of position, written testimony, and exhibits. *Entergy Nuclear Operations, Inc.* (Indian Point, Units 2 and 3), LBP-13-13, 78 NRC 246, 276-77 (2013), *aff'd in part and rev'd in part* CLI-15-6, 81 NRC 340 (2015). The NRC Staff and Entergy filed their statements of position, testimony, and exhibits in March 2012, and the Intervenors subsequently filed with rebuttal testimony and exhibits. *Id.* The Board

Dr. Richard Lahey, Jr.,[5] and Dr. Joram Hopenfeld.[6] For the reasons set forth below and in the testimony filed herewith, the Staff submits that a careful evaluation of the evidence demonstrates that Intervenors' challenge to aspects of metal fatigue issues in Entergy Nuclear Operations, Inc. ("Entergy" or "Applicant") application for renewal of the Indian Point Nuclear Generating Units 2 and 3 operating licenses cannot be sustained.

BACKGROUND

On April 23, 2007, Entergy filed its license renewal application ("LRA") (ENT00015A-B) to renew the operating licenses for Indian Point Nuclear Generating Units 2 and 3 ("IP2" and "IP3"), for an additional period of 20 years.

On November 30, 2007, petitions for leave to intervene were filed by various petitioners, including the State of New York[7] and Riverkeeper.[8] Both the State and Riverkeeper filed contentions challenging the LRA's provisions for aging management related to metal fatigue, as set forth in New York Contention 26 and Riverkeeper Contention TC-1.[9] On January 22, 2008, Entergy submitted Amendment 2 to its LRA,[10] which directly affected Riverkeeper's and New York's aging management contentions. Riverkeeper subsequently filed its Amended Contention

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split the hearing into "Track 1" and "Track 2" contentions, and, at request of the Staff, placed NYS-26/RK-TC-1B on "Track 2." *Id.* at 278-79.

[4] State of New York and Riverkeeper, Inc. Revised Statement of Position on Consolidated Contention NYS-26B/RK-TC-1B (June 9, 2015)(Ex. NYS000529).

[5] Revised Pre-Filed Witten Testimony of Richard T. Lahey, Jr. in support of Contention NYS-26B/RK-TC-1B (June 2, 2015) (Ex. NYS000530).

[6] Revised Prefiled Testimony and Supplemental Report of Dr. Joram Hopenfeld (Exhs. RIV000142, RIV000144).

[7] *See* "New York State Notice of Intention to Participate and Petition to Intervene," filed November 30, 2007 ("New York Petition").

[8] *See* "Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene in the License Renewal Proceeding for the Indian Point Nuclear Power Plant," filed November 30, 2007 ("Riverkeeper Petition").

[9] *See* New York Contention 26 ("Entergy's [LRA] Does Not Include an Adequate Plan to Monitor and Manage the Effects of Aging Due to Metal Fatigue on Key Reactor Components"), New York Petition at 227-33; Riverkeeper Contention TC1 ("Inadequate Time Limited Aging Analyses and Failure to Demonstrate That Aging Will Be Managed Safely"), Riverkeeper Petition at 7-15.

[10] Entergy Letter NL-08-021, from Fred R. Dacimo, Vice President, Entergy, to NRC Docket Control Desk, dated January 22, 2008 ("NL-08-021") (Ex. NYS000351).

TC-1A on March 5, 2008, and New York filed its Supplemental Contention 26-A on April 7, 2008.

On July 31, 2008, the Board issued its ruling on standing and the admissibility of contentions, finding, *inter alia*, that Contention New York 26/26A and Riverkeeper TC-1/1A were admissible as contentions of omission, since Entergy's LRA did not include fatigue evaluations using environmentally-assisted cumulative usage factors ("CUF_{en}") for all of its components, and a description of the specific corrective actions it would take to manage the aging effects of metal fatigue on key reactor components.[11] The Board directed Riverkeeper and New York to submit a consolidated contention,[12] which they did on August 21, 2008.[13]

In August 2010, Entergy (through its contractor, Westinghouse Electric Co., LLC) completed its refined fatigue analyses to determine the CUF_{en} for relevant locations at the facility, and transmitted the results of its calculations to the Board and parties in this proceeding.[14] On August 25, 2010, Entergy filed a motion for summary disposition of the consolidated metal fatigue contention,[15] on the grounds, *inter alia,* that (a) an applicant is not required to submit refined CUF_{en} calculations prior to issuance of a renewed license, as held in

[11] LBP-08-13, 68 NRC at 138, 140, and 172.

[12] *Entergy Nuclear Operations, Inc.* (Indian Point, Units 2 and 3), LBP-08-13, 68 NRC 43, 131-140, 166-72, 218-19 (July 31, 2008).

[13] "Consolidated Contention of Petitioners State of New York (No. 26/26-A) and Riverkeeper, Inc. (TC-1/TC1-A) - Metal Fatigue and Designation of the State of New York as Lead Litigator for this Consolidated Contention" (Aug. 21, 2008).

[14] *See* Letter from Paul Bessette, Esq. to the Licensing Board, dated August 10, 2010, transmitting Entergy Letter NL-10-082, from Fred R. Dacimo, Vice President, Entergy, to NRC Docket Control Desk, "[LRA] – Completion of Commitment # 33 Regarding the Fatigue Monitoring Program, dated August 9, 2010 (Ex. NYS000352).

[15] "Applicant's Motion for Summary Disposition of New York State Contentions 26/26A & Riverkeeper Technical Contentions 1/1A (Metal Fatigue of Reactor Components)" (Aug. 25, 2010). Accompanying Entergy's Motion were 16 attachments, including (1) a "Statement of Material Facts," dated August 25, 2010 ("Material Facts") and (2) the "Declaration of Nelson F. Azevedo in Support of Applicant's Motion for Summary Disposition of Contentions NYS-26/26A and Riverkeeper TC-1/1A," dated August 20, 2010 ("Azevedo Decl."). Entergy included its calculations in (proprietary) attachments to its motion. *See* Attachment 15, "Westinghouse Electric Co., WCAP-17199-P, Rev. 0, *Environmental Fatigue Evaluation for Indian Point Unit 2"* (June 2010) (Ex. NYS000361), and Attachment 16 "Westinghouse Electric Co., WCAP-17200-P, Rev. 0, *Environmental Fatigue Evaluation for Indian Point Unit 3" (*June 2010) (NYS000362).

the Commission's 2010 *Vermont Yankee* decision,[16] and (b) in any event, Entergy had completed those calculations and had shown that the refined CUF_{en} values were less than 1.0. Responses to Entergy's motion were filed by Intervenors[17] and the Staff[18] on September 14, 2010.

On September 9, 2010, based upon the information filed by Entergy in support of its motion for summary disposition, Intervenors submitted a motion for leave to file a new and amended contention[19] along with New York 26-B/Riverkeeper TC-1B (Metal Fatigue) ("NYS-26B/RK-TC-1B"), which alleged:

> Entergy's License Renewal Application does not Include an Adequate Plan to Monitor and Manage the Effects of Aging Due to Metal Fatigue on Key Reactor Components In Violation of 10 C.F.R. § 54.21(c)(1)(iii).[20]

On November 4, 2010, Board admitted NYS-26B/RK-TC-1B, denied Entergy's Motion for Summary Disposition of NYS-26/26A/RK-TC-1/1A as moot, and dismissed NYS-26/26A/RK-TC-1/1A as moot finding that both the previously-admitted consolidated contention and the Motion for Summary Disposition had been superseded by Entergy's August 2010 reanalyses and the challenges to those reanalyses found in NYS-26B/RK-TC-1B.[21] The Board stated that Entergy and the NRC Staff say that there is sufficient detail in the Applicant's AMP, while New York's and Riverkeeper's expert witnesses raise doubt as to the adequacy of Entergy's AMP, and the

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[16] *Entergy Vermont Yankee, L.L.C*. (Vermont Yankee Nuclear Power Station), CLI-10-17, 72 NRC 1 (2010).

[17] State of New York and Riverkeeper, Inc. Combined Response to Entergy Motion for Summary Disposition of Combined Contentions NYS 26/26A and RK TC-1/TC1-A (Metal Fatigue) (Sept. 14, 2010).

[18] NRC Staff's Answer to Applicant's Motion for Summary Disposition of New York Contention 26/26A and Riverkeeper Contention TC-1/1A -- Metal Fatigue (Sept. 14, 2010).

[19] 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);

[20] Petitioners State of New York and Riverkeeper, Inc. New and Amended Contention Concerning Metal Fatigue (Sept. 9, 2010).

[21] 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) (Nov. 4, 2010) (unpublished) (ADAMS Accession No. ML103080987) ("MSD Order"), at 2.

Board proposed to sort out these differing opinions at hearing because it is apparent to the Board that such a review of Fatigue Monitoring Program (FMP) details is needed to determine whether Entergy's AMP is consistent with the Generic Aging Lessons Learned (GALL) Report[22] and meets the requirements of Sections 54.21(a) and 54.21(c)(1)(iii).[23] Concerning interpretation of the Commission's decision metal fatigue *Vermont Yankee*, CLI-10-17, the Board stated, "Because Entergy calculated CUF analyses as part of its efforts to meet [Section 54.21(c)(1)(iii)], the methodology and breadth of these calculations may come under scrutiny."[24]

DISCUSSION

I. Applicable Regulations and Guidance

A. Standards for Issuance of a Renewed License

Underlying the Commission's renewal regulations is the principle that each nuclear power plant has a plant-specific current licensing basis ("CLB")[25] that must be maintained during the renewal term in the same manner and to the same extent as during the original licensing term. *Entergy Nuclear Generation Co. and Entergy Nuclear Operations, Inc.* (Pilgrim Nuclear Power Station), CLI-10-14, 71 NRC 449, 453 (*citing* Final Rule, Nuclear Power Plant License Renewal; Revisions, 60 Fed. Reg. 22,461, 22,464 (May 8, 1995) ("License Renewal Rule")). In accordance with 10 C.F.R. Part 54, *inter alia*, the Commission may issue a renewed license if:

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[22] NUREG-1801, Rev. 1, *Generic Aging Lessons Learned (GALL) Report*, (Sept. 2005), Vol. 1 (ADAMS Accession No. ML052770419) & Vol. 2 (ADAMS Accession No. ML052110006) (Ex. NYS00146A-C) ("GALL Report Rev. 1"); NUREG-1801, Rev. 2, *Generic Aging Lessons Learned (GALL) Report – Final Report*, (Dec. 2010) (ADAMS Accession No. ML103490041) ("GALL Report Rev. 2") (Ex. NYS000147A-D).

[23] MSD Order at 15.

[24] *Id.* at 24.

[25] The CLB is "the set of NRC requirements (including regulations, orders, technical specifications, and license conditions) applicable to a specific plant, and includes the licensee's written, docketed commitments for ensuring compliance with applicable NRC requirements and the plant-specific design basis." *Pilgrim*, CLI-10-14, 71 NRC at 453-54 (*citing* 10 C.F.R. § 54.3). Both during the original license term and continuing through the renewal term, the NRC continually assesses the both the adequacy of the CLB, as well as the licensee's compliance with its CLB, through the NRC regulatory oversight process, generic and plant-specific reviews, plant inspections, and enforcement actions. *Id.*

> (a) Actions have been identified and have been or will be taken with respect to the matters identified in paragraphs (a)(1) and (a)(2) of this section, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB, and that any changes made to the plant's CLB in order to comply with this paragraph are in accord with the Act and the Commission's regulations. These matters are:
>
> > (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under § 54.21(a)(1); and
> > (2) time-limited aging analyses that have been identified to require review under § 54.21(c).

10 C.F.R. § 54.29(a). The regulations in 10 C.F.R. § 54.29(a) define the findings the Commission must make in order to issue a renewed license to a nuclear power plant, and define the scope of any hearing on the renewal application. *See* 60 Fed. Reg. at 22,481 (describing changes to 10 C.F.R. §§ 54.29 and 54.30 to "minimize any possibility of misinterpreting the scope of the renewal review and finding"). In turn, the scope of Commission review determines the scope of admissible contentions in a license renewal hearing. 10 C.F.R. § 2.309(f)(iii). The regulations in 10 C.F.R. Part 54 set forth the safety standards for license renewal.

B. <u>The Scope of a Contested Renewal Hearing is Limited</u>

As the Board in this proceeding has recognized, the scope of license renewal proceedings is significantly circumscribed from the scope of the original licensing proceedings. The Board stated:

> [The] Commission determined that the safety issues relevant to reactor relicensing are significantly different from, and defined more narrowly than, those relevant during the original licensing proceedings that authorize facility construction and operation.

Entergy Nuclear Operations Inc. (Indian Point, Units 2 and 3), LPB-08-13, 68 NRC 43, 67 (2008) (discussing technical review for reactor licensing); *compare* 10 C.F.R. § 54.29 (findings needed to issue a renewed license) with 10 C.F.R. § 50.57 (findings needed to issue operating licenses). Moreover, the Board recognized that certain safety issues that were reviewed for the initial license are already closely monitored and inspected by the NRC, and do not need to be re-reviewed in the context of a license renewal application. *Indian Point,* LBP-08-13, 68 NRC at 67 (*citing* Nuclear Power Plant License Renewal, Final Rule, 56 Fed. Reg. 64,943, 64,946 (Dec. 13, 1991); *Florida Power & Light Co.* (Turkey Point Nuclear Generating Plant, Units 3 and 4), CLI-01-17, 54 NRC 3, 7 (2001)).

Issues related to a licensee's compliance with its current licensing basis are not within the scope of a license renewal proceeding. 10 C.F.R. §54.30. Issues relating to a plant's CLB are ordinarily beyond the scope of a license renewal review because those issues already are monitored, reviewed, and commonly resolved as needed by ongoing regulatory oversight. *Turkey Point*, CLI-01-17, 54 NRC at 8. Because they are not subject to physical aging effects, programmatic aspects of the current licensing basis such as radiation protection, physical protection (security), and quality assurance are beyond the scope of a license renewal review. *See* Nuclear Power Plant License Renewal; Revisions, 60 Fed. Reg. 22,461, 22,474-75 (May 8, 1995).

C. <u>The Scope of a Contention Is Limited By Its Bases</u>

As the Commission has held, "[T]he scope of a contention is limited to issues of law and fact pled with particularity in the intervention petition, including its stated bases, unless the contention is satisfactorily amended in accordance with our rules." *Southern Nuclear Operating Co.* (Early Site Permit for Vogtle ESP Site), CLI-10-5, 71 NRC 90, 100 (2010) (footnotes omitted). The Commission recently emphasized that the "reach of a contention necessarily hinges upon its terms *coupled with* its stated bases." *NextEra Energy Seabrook, LLC* (Seabrook Station, Unit 1), CLI-12-5, 75 NRC 301, 310 n.50 (2012) (quoting *Entergy Nuclear Generation Co.* (Pilgrim Nuclear Power Station), CLI-10-11, 71 NRC 287, 309 & n.103 (2010) (emphasis in original; footnote and internal quotation marks omitted)).

D. Burden of Proof

In an NRC licensing proceeding, the applicant has the burden of proof:

> [t]he ultimate burden of proof on the question of whether the permit or the license should be issued is … upon the applicant. But where … one of the other parties contends that, for a specific reason … the permit or license should be denied, that party has the *burden of going forward* with evidence to buttress that contention. Once [the party] has introduced sufficient evidence to establish a *prima facie* case, the burden then shifts to the applicant who, as part of [its] overall burden of proof, must provide sufficient rebuttal to satisfy the Board that it should reject the contention as a basis for denial of the permit or license.

AmerGen Energy Co., LLC (Oyster Creek Nuclear Generating Station), CLI-09-7, 69 NRC 235, 269 (quoting *Louisiana Power and Light Co.* (Waterford Steam Electric Station, Unit 3), ALAB-732, 17 NRC 1076, 1093 (1983), quoting *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-123, 6 AEC 331, 345 (1973) (emphasis in original)).

Thus, insofar as the LRA is contested, Entergy must demonstrate how its programs will be effective in managing the effects of aging during the proposed period of extended operation ("PEO"). *Turkey Point,* CLI-01-17, 54 NRC at 8 (citing 10 C.F.R. § 54.21(a)). Further, Entergy must identify actions that will need to be taken to manage adequately the detrimental effects of aging. *See id.* (citing "Nuclear Power Plant License Renewal; Revisions," 60 Fed. Reg. 22,461, 22,463 (May 8, 1995)).

The regulations at 10 C.F.R. §§ 54.21 and 54.29 require Entergy to establish aging management programs that provide "reasonable assurance" that the structures and components will continue to perform their intended functions consistently with the CLB during the period of extended operation. *AmerGen Energy Co., LLC* (Oyster Creek Nuclear Generating Station), CLI-09-7, 69 NRC 235, 263 (2009) (*affirming* LBP-07-17, 66 NRC 327 (2007)). In this regard, "'[r]easonable assurance' ... is not susceptible to formalistic quantification or mechanistic application." *Oyster Creek*, LBP-07-17, 66 NRC at 339.

“Reasonable assurance” is based on sound technical judgment and on compliance with the Commission's regulations. *Oyster Creek*, CLI-09-7, 69 NRC at 263.

Entergy must show that its LRA meets the regulatory requirements of 10 C.F.R. § 54.21 by a “preponderance of the evidence.” *Amergen Energy Co., LLC* (Oyster Creek Nuclear Generating Station), CLI-09-7, 69 NRC 235, 263 (2009).

E. License Renewal Guidance

In reviewing the safety aspects of license renewal applications, the NRC Staff is guided primarily by two documents—the GALL Report, as revised, and the License Renewal Standard Review Plan ("SRP-LR").[26] *See Amergen Energy Co., LLC* (Oyster Creek Nuclear Generating Station), CLI-08-23, 68 NRC 461, 466 (2008).

The GALL Report identifies generic aging management programs that the Staff has determined to be acceptable, based on the experiences and analyses of existing programs at operating plants during the initial license period. *Oyster Creek*, CLI-08-23, 68 NRC at 467. The GALL Report recognized that the Staff’s reviews of the first sets of license renewal applications found that many of the programs that the licensees rely on to manage aging effects during the renewal period were already in place during the initial license period. *Id.* at 467 n.15. The report describes acceptable aging management programs with respect to the ten program elements defined in the SRP-LR. *Id.* at 467.

The SRP-LR assigns review responsibilities among Staff technical organizations and describes methods for identifying the systems, structures, and components (SSCs) that are subject to aging effects within the scope of license renewal review. *Id* at 467. The SRP-LR defines ten aging management program elements—(1) scope of program, (2) preventive actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring

[26] NUREG-1801, Rev. 1, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, (Sept. 2005) (Ex. NYS000195); NUREG-1801, Rev. 2, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, NUREG-1800, Rev. 2 (Dec. 2010) (Ex. NYS000161) ("SRP-LR Rev. 2"). Throughout this statement of position, where no revision is provided in the text, the statement provides equally to both revisions of SRP-LR.

and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience -- which are essential to an effective aging management program. *Id.* & SRP-LR Rev. 2 at A.1-3 through A.1-9.

As described in the SRP-LR, an applicant is required to demonstrate that the effects of aging on structures and components subject to an Aging Management Review (AMR) will be managed adequately to ensure that their intended functions will be maintained consistent with the CLB of the facility for the period of extended operation. SRP-LR at A.2-1.Therefore, those aspects of the AMR process that affect quality of safety-related structures, systems, and components are subject to the quality assurance (QA) requirements of Appendix B to 10 CFR Part 50. *Id.* For nonsafety-related structures and components subject to an AMR, the existing 10 CFR Part 50 Appendix B QA program may be used by the applicant to address the elements (7) corrective actions, (8) confirmation process, and (9) administrative controls. *Id.*

As the Commission explained in *Oyster Creek*, the SRP-LR provides that a license renewal application may rely on an AMP that is consistent with the GALL Report, or may use a plant-specific AMP. *Oyster Creek*, CLI-08-23, 68 NRC at 467. Using an AMP identified in the GALL Report (i.e. when an applicant ensures and certifies[27] that its programs correspond to those reviewed in the GALL report) constitutes reasonable assurance that the AMP will manage the targeted aging effect during the renewal period. *Id.* The Commission has recently re-iterated this principle, stating:

> If the NRC concludes that an aging management program (AMP) is consistent with the GALL Report, then it accepts the applicant's commitment to implement that AMP, finding <u>the commitment itself to be an adequate demonstration of reasonable assurance under section 54.29(a).</u>

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[27] The Commission has emphasized that it is neither possible nor necessary for the Staff to verify each and every factual assertion in LRAs, but the Commission's regulations require that an LRA be complete and accurate in all material respects, and submitted under oath. *Oyster Creek*, CLI-08-23, 68 NRC at 480-481. Nevertheless, and as demonstrated by the attached testimony, the NRC does not simply take the applicant at its word. When an applicant claims consistency with the GALL Report, the Staff draws its own independent conclusion as to whether the applicant's programs are, in fact, consistent with the GALL Report. *See Vermont Yankee*, CLI-10-17, 72 NRC at 37.

Seabrook, CLI-12-05, 75 NRC at 304.

F. Commission Evaluation of Metal Fatigue

The Commission issued a definitive ruling discussing at-length the technical aspects and legal aspects of metal fatigue as an aging effect for license renewal reviews. *See* generally *Vermont Yankee*, CLI-10-17, 72 NRC 1 (2010).[28] Regarding the technical aspects, the Commission wrote:

> Metal fatigue can be defined as the weakening of a metal due to mechanical and thermal stresses, which are variously referred to as load cycles, stress cycles, and cyclical loading. Metal components experience these stresses during "transients" such as significant temperature changes during plant startup and shutdown. An excessive number of load cycles or transients may result in a fracture or a significant reduction in the strength of a component. These fractures or significant reductions are called "fatigue failure." For any material, there is a characteristic number of stress cycles that it "can withstand at a particular applied stress level before fatigue failure occurs." The period during which this number of load cycles occurs for *all* types of stress is called the material's "fatigue life."

Id. at 14 (footnotes omitted).

The fatigue that a metal component experiences is quantified by the "Cumulative Usage Factor" or "Cumulative Use Factor" ("CUF"). *Id.* at 5 n.9 (*citing AmerGen Energy Co., LLC* (Oyster Creek Nuclear Generating Station), CLI-08-28, 68 NRC 658, 663 (2008)).

The Commission described the calculation of fatigue, and the differences between nuclear and non-nuclear calculations and curves in the American Society of Mechanical Engineers Boiler & Pressure Vessel Code ("ASME Code"):

> Determining the stress acting on the component during a transient is a complicated inquiry, requiring detailed knowledge of material

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[28] In the *Vermont Yankee* decision the Commission, *inter alia*, (i) *denied* as moot NEC's motion to stay the proceeding, (ii) *granted*, in part, the Staff's petition for review of LBP-08-25, (iii) *reversed* the Board's rulings in LBP-08-25 regarding NEC's Contentions 2A and 2B insofar as those rulings relate to the calculation of the CUFen, for the core spray and reactor recirculation outlet nozzle, (iv) *granted*, in part, NEC's petition for review of LBP-09-9, and (vii) *remanded* the proceeding for the limited purpose of giving NEC the opportunity to submit a revised Contention 2. *Vermont Yankee*, CLI-10-17, 72 NRC at 54-55.

> properties, component design, and the temperature profile of the transient, among other parameters. A detailed stress analysis uses the methodology from the ASME Code to consider six different stress inputs.
>
> The ASME Code contains fatigue design curves for various materials, such as low alloy steel and stainless steel used in nuclear power plants. These curves indicate the allowed number of stress cycles at any applied stress. In addition, ASME took actual laboratory fatigue data, derived from tests performed at room temperature in the air, and then adjusted the laboratory data by reducing the stress - where stress is expressed as the number of cycles - to account for the difference in a material's behavior in a controlled laboratory environment as compared with a real-world non-nuclear industrial setting where the component could be used. From these adjusted data, an applicant can calculate the CUF for a component at a particular location on that same component, i.e., the applicant can quantify "the fatigue that a particular [location on a] metal component experiences during … operation" of a non-nuclear industrial facility.

Id. at 15 (footnotes omitted).

The Commission described the reason why an environmental adjustment factor is considered:

> [T]he correction factors applied by ASME were not intended to account for the potentially corrosive environment present in a light water reactor - an environment that may accelerate fatigue failure. The effects of the reactor environment can be significant under certain circumstances. To take the reactor environment into account, a license renewal applicant may apply a concept called the "environmental fatigue correction factor," or Fen, which yields the environmentally adjusted CUF, i.e., the CUFen....

Id. at 15-16 (footnotes omitted).

The regulations in 10 C.F.R. Part 50 include provisions requiring compliance with the ASME Code. In particular, § 50.55a(c)(1) generally requires components which are part of the reactor coolant pressure boundary to meet the requirements for Class 1 components in Section III of the ASME Code or the applicable Code Edition which was in effect at the time of issuance of the plant's construction permit. *Id.* at 16-17. The ASME Code provides the methodology for calculating the CUFs for nuclear power plant components, and specifies a design limit of 1.0 for the CUF of any given component, including any additional stress cycles that may occur during the period of extended operation. *Id.* at 17 (*citing* ASME Code, Section III, Division 1, Subsection NB, Paragraph NB-3222.4. *See also* NUREG/CR-6909, *Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials*, at 1 (Feb. 2007) (Ex. NYS000357); SRP-LR Rev. 1 § 4.3.1.1, at p. 4.3-1 (Ex. NYS000195).

If a given component has a CUF analysis which is part of the licensee's CLB (*i.e.* the licensing basis that exists *before* any granting of a renewed license, *see* 10 C.F.R. § 54.3(a)), the CUF analyses are treated as time-limited aging analyses ("TLAA")[29]. *See Vermont Yankee*, CLI-10-17, 72 NRC at 34.

Pursuant to 10 C.F.R. § 54.21(c)(1)(i)-(iii), an applicant may address, for license renewal purposes, an existing TLAA by: (1) showing the analyses remain valid for the period of extended operation ("PEO"), (2) projecting (reanalyzing) the TLAA to the end of the PEO, or (3) demonstrating that the effects of aging will be adequately managed. *See Id.* at 35.

Under § 54.21(c)(1)(iii), the applicant may manage metal fatigue using an AMP. As the Commission described in *Vermont Yankee*:

> The SRP permits an applicant who chooses to implement an AMP under 10 C.F.R. § 54.21(c)(1)(iii) to reference Chapter X of the GALL Report:
>
> > [NRC] staff has evaluated a program for monitoring and tracking the number of critical thermal and pressure transients for the selected reactor coolant system components. The staff has determined that this program is an acceptable aging management program to address metal fatigue of the reactor coolant system components according to 10 CFR 54.21(c)(1)(iii). The GALL [R]eport may be

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[29] *Time-limited aging analyses*, for the purposes 10 C.F.R. Part 54 "Requirements For Renewal of Operating Licenses for Nuclear Power Plants," are 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.

10 C.F.R. 54.3(a)

> referenced in a license renewal application and should be treated in the same manner as an approved topical report. In referencing the GALL [R]eport, the applicant should indicate that the material referenced is applicable to the specific plant involved and should provide the information necessary to adopt the finding of program acceptability as described and evaluated in the report.

Vermont Yankee, CLI-10-17, 72 NRC at 19 (quoting SRP-LR , § 4.3.2.1.1.3, at p. 4.3-4). The Commission also repeated the holding of *Oyster Creek*, CLI-08-23: "the license renewal applicant's use of an aging management program identified in the GALL Report constitutes reasonable assurance that it will manage the targeted aging effect during the renewal period"). *Id.* at 19 n.85 (*citing Oyster Creek*, CLI-08-23, 68 NRC at 468).

Further, the Commission explained its requirements for what is needed to demonstrate under § 54.21(c)(1)(iii) the effects of aging will be adequately managed: "One way to do this is to reference the Metal Fatigue AMP that is approved in the GALL Report." *Vermont Yankee*, CLI-10-17, 72 NRC at 20 (*citing* GALL Report at § X.M1, at pp. X M-1 to X M-2 (description of the "Metal Fatigue of Reactor Coolant Pressure Boundary" AMP). Regarding when calculations performed as part of an AMP must be done, the Commission has stated

> Our regulations contain no requirement that an applicant complete a subsection (iii) fatigue analysis prior to the issuance of a renewed license, and an applicant need not do so unless the analysis is needed to support a demonstration that the tracking AMP will satisfy our regulatory requirements - [in *Vermont Yankee*], such an analysis would be used to demonstrate that the AMP is consistent with the GALL Report.

Vermont Yankee, CLI-10-17, 72 NRC at 36 (footnotes omitted).

ARGUMENT

II. Entergy's Fatigue Monitoring Program Meets 10 CFR Part 54

In this proceeding, the Board, while admitting Contention NYS-26B/Riverkeeper TC-1B and denying Entergy's motion for summary disposition, held:

> Because Entergy is using CUF calculations to demonstrate the adequacy of its AMP, these calculations are subject to review in this proceeding. New York's and Riverkeeper's challenge to the "refined" CUF analysis is addressed specifically to the use of the results of these analyses in meeting Section 54.21(c)(1)(iii), which is exactly the kind of challenge that the Commission has permitted through its Vermont Yankee decision.

MSD Order at 24. The Board indicated that "a review of [the fatigue monitoring program] details is needed to determine whether Entergy's AMP is consistent with the GALL Report and meets the requirements of Sections 54.21(a) and 54.21(c)(1)(iii)." *Id.* at 15.

As explained below, the Staff has (1) performed a review of the LRA and documented the acceptability of the application in accordance with the Staff's normal practices, and (2) as set forth in prepared expert testimony, provided additional explanation of the Staff's views as to why Entergy's LRA is acceptable, and rebutted assertions made by Intervenors' witnesses Drs. Lahey and Hopenfeld.

A. <u>NRC Staff Expert Witnesses</u>

The NRC Staff Testimony of Dr. Allen Hiser, Dr. Ching Ng, Mr. Gary Stevens, P.E., and Mr. On Yee, Concerning Contentions NYS-26B/RK-TC-1B and NYS-38/RK-TC-5 (Ex. NRC000168) presents the expert opinions of four highly-qualified NRC Staff witnesses who explain why the information provided by the Applicant on the aging management of metal fatigue provides an adequate demonstration of reasonable assurance under 10 C.F.R. § 54.29(a), and further explain why the additional information sought by NYS and Riverkeeper is not necessary for a determination of adequacy.

1. <u>Dr. Allen Hiser, Jr.</u>

Dr. Allen Hiser, Jr. has worked at the NRC for 25 years in the Office of Nuclear Regulatory Research ("RES") and the Office of Nuclear Reactor Regulation ("NRR"). Dr. Hiser is employed as the Senior Technical Advisor for License Renewal Aging Management in the Division of License Renewal, NRR, in Rockville, MD. NRC Fatigue Test. at A1. He received

Bachelor of Science and Master of Science degrees in Mechanical Engineering from the University of Maryland at College Park. *Id.* He also received a Ph.D. degree in Materials Science and Engineering from Johns Hopkins University. *Id.* He has been a participant in ASME Working Groups on Flaw Evaluation and Pipe Flaw Evaluation dating back to the early 1980s. *Id.* For some of this time, he was the voting member and the NRC representative on these working groups. *Id.* Currently, he is a member of the ASME Special Working Group on Nuclear Plant Aging Management. *Id.* He is chairman of the Steering Committee of the International Atomic Energy Agency (IAEA) program to develop aging management standards for international use and the International Generic Aging Lessons Learned program. *Id.* In addition, he has been a team member on several IAEA missions to evaluate the aging management for international plants pursuing license renewal, and he has been a trainer in international workshops for regulators and plants on aging management for license renewal. *Id.* Dr. Hiser's responsibilities include providing technical advice and assistance to the Division of License Renewal on a variety of technical, regulatory and policy issues related to aging management of nuclear power plant systems, structures, and components. *Id.* at A2. His statement of his professional qualifications was previously submitted (Ex. NRCR00103).

For the Indian Point review, Dr. Hiser's Branch was responsible for the review of several portions of the Indian Point LRA. *Id.* at A3...He also assisted and guided the Staff in its review of information submitted by Entergy on environmentally-assisted fatigue analyses, which was used to develop SER Supp. 1. *Id.* at A3.

2. Mr. On Yee

Mr. On Yee has been working at the NRC for approximately ten years. *Id.* at A1. He is currently employed as a Reactor Systems Engineer in the Containment & Balance of Plant Branch, Japan Lessons-Learned Division, NRR, in Rockville, MD. *Id.* He was employed as a Mechanical Engineer in the Aging Management of Reactor Systems Branch, Division of License Renewal, NRR, NRC, in Rockville, MD. *Id.* He received a Bachelor of Science degree in

Mechanical Engineering from Polytechnic University, in Brooklyn, NY. *Id.* A revised statement of his professional qualifications is submitted herewith (Ex. NRCR000104).

For Indian Point, Mr. Yee assisted in the review of the existing FMP, metal fatigue time limited aging analyses (TLAAs) and environmentally-assisted fatigue analyses associated with the IP2 and IP3 LRA. *Id.* at A3. As part of those activities, he assisted in the review of Entergy's on-site technical documentation that described its existing FMP, which will be used as its aging management program for license renewal. *Id.* He also assisted in the review of Entergy's existing metal fatigue analyses, which are time-limited aging analyses as defined in 10 C.F.R. 54.3, and Entergy's environmentally-assisted fatigue analyses, which are not TLAAs as defined in 10 C.F.R. 54.3. *Id.* He worked with the principal reviewer in the preparation of the review of these issues, the results of which are documented in NUREG-1930, *Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3*, (November 2009) (Ex. NYS00326A-F) ("SER"). *Id.*

3. Dr. Ching Ng

Dr. Ching Ng has been working at the NRC for more than eight years. *Id.* at A1. He is currently employed as a Reliability and Risk Analyst in the Probabilistic Risk Assessment Operations and Human Factors Branch, Division of Risk Assessment, NRR, in Rockville, MD. Previously he was employed as a Mechanical Engineer in the Aging Management of Reactor Systems Branch, Division of License Renewal, NRR, in Rockville, MD. *Id.* He received Bachelor of Science, Master of Science, and Ph.D. degrees in Mechanical Engineering from the University of California, Berkeley. *Id.* A revised statement of his professional qualifications is submitted herewith (Ex. NRCR00105).

From June 2010 to January 2012, Dr. Ng served as a reviewer for the environmentally-assisted fatigue analyses associated with the IP2 and IP3 LRA. *Id.* at A3. As part of his duties, he developed the updated environmentally-assisted fatigue section in Section 4.3.3 of the Staff's SER, Supp. 1. *Id.* at A3.

4. Mr. Gary L. Stevens

Mr. Gary L. Stevens has been employed by the NRC for more than five years. He is currently employed as a Senior Materials Engineer in the Vessel and Internals Integration Branch in the Division of Engineering, NRR, in Rockville, MD. *Id.* at A1. Prior to March 2015, he was employed as a Senior Materials Engineer in the Component Integrity Branch, Division of Engineering, Office of Nuclear Regulatory Research (RES), in Rockville, MD. *Id.* He received a Bachelor of Science degree in Mechanical Engineering from California Polytechnic State University in San Luis Obispo, CA, and a Master of Science degree in Mechanical Engineering from San Jose State University. *Id.* He has been a participating member in American Society of Mechanical Engineers (ASME) Code, Section XI Committees for more than 25 years. *Id.* His statement of qualifications is submitted herewith (Ex. NRC000227).

Mr. Stevens is the NRC's current subject matter expert on environmentally assisted fatigue (EAF), and is leading the NRC's research activities to update and revise Regulatory Guide (RG) 1.207, *Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors*, (March 2007) (ADAMS Accession No. ML070380586) (Ex. NRC000179) ("RG 1.207") and the associated update and revision of the supporting technical basis document, NUREG/CR-6909, *Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials*, (February 2007) (Ex. NYS000357) ("NUREG/CR-6909"). *Id.* at A2. He has led these research activities for the NRC since 2010. *Id.*

B. Review of the Fatigue Monitoring Program.

The NRC Staff reviewed the Indian Point Nuclear Generating Unit Nos. 2 and 3 license renewal application for compliance with the requirements of 10 C.F.R. Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and documented its findings in its

NUREG-1930, *Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3* (Nov. 2009) (Ex. NYS00326A-F), as supplemented.

In NUREG-1930, the Staff stated its overall conclusion on Entergy's application:

> The staff of the U.S. Nuclear Regulatory Commission (NRC) (the staff) reviewed the license renewal application (LRA) for Indian Point Nuclear Generating Unit Nos. 2 and 3, in accordance with NRC regulations and NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," dated September 2005. Title 10, Section 54.29, of the *Code of Federal Regulations* (10 CFR 54.29) sets the standards for issuance of a renewed license. Pursuant to 10 CFR 54.29(a), the Commission may issue a renewed license if it finds that actions have been identified and have been or will be taken, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis (CLB).
>
> On the basis of its review of the LRA, the staff determines that the requirements of 10 CFR 54.29(a) have been met.

NUREG-1930, Vol. 2 (Ex. NYS00326E) at 6-1.

Section 4.3 of NUREG-1930 addressed metal fatigue, which, as stated above, is age-related degradation caused by cyclic stressing of a component by either mechanical or thermal stresses. *Id.* at 4-18. Entergy's FMP tracks and evaluates design transients and requires corrective actions if the number of analyzed transients are approached, to keep the number of transient cycles experienced by the plant within the analyzed numbers of cycles, thus keeping the component cumulative usage factors (CUFs) below the values calculated in the design-basis fatigue evaluations. *Id.* at 4-19. Appendix B to the LRA provides further details on the FMP. In Section 4.3.3 of NUREG-1930, the Staff documented its review of Entergy's evaluation of the effects of the reactor coolant system environment on fatigue life of piping and components. *Id.* at 4-40 to 4-46. As noted therein, the fatigue data for the ASME Code, Section III,[30] fatigue curves result from tests performed in air at room temperature and constant strain

[30] ASME Code Section III "Rules for Construction of Nuclear Facility Components" provides general requirements addressing the material, design, fabrication, examination, testing and overpressure protection of the items specified within each respective Subsection, assuring their structural integrity.

rate. NUREG-1930, Vol. 2, (Ex. NYS00326E) at 4-40. Research and studies over the potential effect of elevated temperature, reactor coolant chemistry environments, and different strain rates are documented in NUREG/CR-5999, "Interim Fatigue Design Curves for Carbon, Low-Alloy, and Austenitic Stainless Steels in LWR Environments" and NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." NUREG-1930, Vol. 2 at 4-40 (Ex. NYS00326E).

Based on NUREG/CR-6260 and the IP2 and IP3 plant design, the following component locations were shown to be the most sensitive to reactor water environmental effects: (1) reactor vessel (RV) shell and lower head; (2) RV inlet and outlet nozzles; (3) pressurizer surge line (including hot leg and pressurizer nozzles); (4) reactor coolant system (RCS) piping charging system nozzle; (5) RCS piping safety injection nozzle; and (6) residual heat removal (RHR) Class 1 piping. *Id.* Therefore, Entergy evaluated the limiting locations using the guidelines of the GALL Report, Volume 2, Section X.M1, which calls for following the guidance (formulas) of (1) NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," issued April 1999, for austenitic stainless steel; and (2) NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," issued February 1998, for carbon steel and low-alloy steel to calculate environmentally assisted fatigue correction factors (Fen). LRA Tables 4.3-13 (IP2) and 4.3-14 (IP3) list the environmentally adjusted CUF values for the applicant's NUREG/CR-6260 limiting locations. *Id.* at 4-40 to 4-41.

The Staff considered whether or not the applicant: (1) included the critical components selected in NUREG/CR-6260; (2) addressed usage of environmental correction factors to the ASME Code fatigue analyses for the sample of critical components; and (3) addressed formulas for calculating the environmental life correction factors as provided in NUREG/CR-6583 for carbon and low-alloy steels and those in NUREG/CR-5704 for austenitic stainless steels or approved technical equivalents. *Id.* at 4-41. The Staff is satisfied that Entergy's metal fatigue program meets the Commission's regulations. *Id.* at 4-19 to 4-38 (Class 1 fatigue), 4-39 to 4-40 (non-class 1 fatigue), 4-40 to 4-46 (effects of reactor water environment on fatigue life).

The Staff separately determined in its review of Entergy's QA program that all of Entergy's AMPs were consistent with elements 7 (Corrective Actions), 8 (Confirmation Process), and through 9 (Administrative Controls) of GALL AMPs. *See* NUREG-1930 (Ex. NYS00326C) at 3-220 to 3-22.

On August 30, 2011, the Staff issued Supplement 1 to NUREG-1930 (Ex. NYS000160). This supplement documents the NRC staff's review of supplemental information provided by the applicant since the issuance of NUREG-1930, including annual updates required by 10 CFR 54.21(b), and updated information and commitments in response to NRC staff requests for additional information. In SER Supplement 1, the Staff concluded that the additional information provided by Entergy does not alter the conclusions stated in the SER and that the requirements of 10 CFR 54.29(a) have been met. NUREG-1930, Supp. 1 (Ex. NYS000161) at 6-1.

On July 7, 2015, the NRC staff issued Supplement 2 to NUREG-1930 (Ex. NYS000507). This supplement documents the NRC staff's review of supplemental information provided by the Applicant since the issuance of Supplement 1, Entergy's Commitment No. 30 (pertaining to reactor vessel internals) and Commitment No. 49 (pertaining to using the FMP for RVI), information required by 10 CFR 54.21(b), updated information and commitments, as well as information provided in response to NRC staff requests for additional information. In SER Supplement 2, the staff concluded that the additional information provided by Entergy does not alter the conclusions stated in the SER and that the requirements of 10 CFR 54.29(a) have been met. NUREG-1930, Supp. 2 (Ex. NYS000507) at 6-1.

The Staff compared Entergy's FMP with the ten aging management program elements of the Staff's guidance, i.e. (1) scope of program, (2) preventive actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience. NRC Fatigue Test. at A101-A102. The AMP Audit Report (Ex. NRC000108) describes how, based on the Staff's review of Entergy's onsite documents, review of Entergy's responses to the Staff's questions, and interviews with Entergy personnel, the Staff determined that elements (1) scope of program, (2) preventive actions, (5) monitoring and trending and (6) acceptance criteria were consistent with the GALL Report. *Id.* at A101. The Staff's SER documented the conclusions concerning elements (3) parameters monitored or inspected, (4) detection of aging effects, and (10) operating experience. *Id.* at A102. Last, the Staff also was satisfied with how Entergy addressed the quality assurance elements (7) corrective actions, (8) confirmation process, and (9) administrative controls. *Id.* Thus, the Staff considered all ten program elements of the aging management program and found all of them to be acceptable. *Id.*

C. <u>ASME Code, Quality Assurance, Iterative Calculations, and Metal Fatigue</u>

In NYS-26B/RK-TC-1B, the Intervenors essentially claim that Entergy has made, or might make in the future, a mistake while performing a calculation under an aging management program. But, because the calculations are subject to the licensee's quality assurance program, Intervenors' claim of errors in the calculations is tantamount to a challenge to the licensee's compliance with its quality assurance program. Thus, the claims in NYS-26B/RK-TC-1B of mistakes and errors during calculations are challenges to the CLB (i.e. compliance with quality assurance) and are beyond the scope of a license renewal review. *See* Nuclear Power Plant License Renewal; Revisions, 60 Fed. Reg. 22,461, 22,474-75 (May 8, 1995) (listing radiation protection, security, and quality assurance as examples of programs not subject to aging effects). Nonetheless, to address the Intervenors' arguments, the Staff's testimony provides affirmative discussion of: ASME code requirements, CUFs, environmentally-assisted CUFs, iterative reanalysis of CUFs, iterative removal of conservative assumptions or "conservatisms", repair or replacement vs. reanalysis, and quality assurance. Further discussion of these items is provided in the Staff's rebuttal of the witnesses.

1. ASME Code Requirements

In their testimony, the Staff's witnesses explain that 10 C.F.R. §§ 54.33 and 54.35 require that a license renewal applicant comply with the requirements in 10 CFR Part 50, including the provisions requiring compliance with the ASME Code Section III during the period of extended operation. NRC Fatigue Test. at A14. Further, the ASME Code, developed by industry experts, is updated to reflect operating experience and ongoing research and development activities in materials science and analytical areas. *Id.* In addition, the Staff's witnesses explain that 10 C.F.R. § 50.55a(c)(1) requires that reactor coolant pressure boundary components meet the metal-fatigue requirements for Class 1 components in Section III of the ASME Code. *Id.*

2. Cumulative Usage Factors

In particular, the Staff's witnesses observe that the ASME Code provides the methodology for calculating the cumulative usage factors ("CUF") for nuclear power plant components, and specifies a design limit of 1.0 for the CUF of any given component, including any additional stress cycles that may occur during the period of extended operation. NRC Fatigue Test. at A14

The Staff's witnesses explain that an ASME Code Class 1 fatigue evaluation is a calculation that was performed by an applicant in accordance with Section III; and is part of an applicant's CLB for the plant. *Id.* at A23. An ASME Code Class 1 fatigue analysis is one part of a larger stress evaluation that is required by ASME to show compliance with Section III for ASME certification of the component for service. *Id.* The fatigue evaluation part of the stress evaluation is a measure that identifies the likelihood of a component initiating a fatigue crack caused by cyclic loading. *Id.*

The Staff's witnesses discuss the "cumulative usage factor" or "CUF," and what the CUF means in relationship to a crack forming, or failure of a component, if a CUF is less than or equal to 1.0. *Id.* at A23 to A31. As explained therein, the Staff is not aware of any scientific evidence to indicate that a fatigue crack has formed when CUF value exceeds 1.0, but the Staff conservatively assumes that a fatigue crack may have formed and is growing when a CUF value is greater than 1.0. *Id.* at A31.

3. Environmentally-Assisted Cumulative Usage Factor

The Staff's testimony explains that the environment (e.g. hot water) to which metal is exposed can affect its cumulative usage factor. The Staff describes how an environmentally-assisted fatigue usage factor (CUF_{en}) is calculated by (1) calculating the CUF using the methodology from ASME Section III; (2) calculating the environmental adjustment factor (F_{en}) by using the guidance recommended in the GALL Report; and (3) calculating the CUF_{en} as the product of the CUF and the F_{en} factor. *Id.* at A35. The Staff further describes the extensive research and literature, some of which is documented in the SRP-LR, on environmentally-assisted fatigue analyses. *Id.* at A40-41.

4. Iterative Reanalysis of CUFs

The Staff's witnesses discuss the reanalysis of CUFs . NRC Fatigue Test. at A50-53, 208. The analyst's goal when performing a Section III fatigue analysis is to ensure the component will continue to meet the component fatigue limit. The re-analyses accomplish this goal by using more precise and refined methods to calculate a CUF value. They explain there is sufficient safety margin inherent in the Section III CUF limit of 1.0; a limit that indicates a likelihood of initiating a fatigue crack, not failing the structure or component; accordingly a reanalysis of the CUF that demonstrates compliance with the limit does not correspond to a reduction in safety margin. Iterative calculations are routine for fatigue evaluations. *Id.* at A50-54, 208.

5. <u>Iterative Removal of Conservative Assumptions or “Conservatisms”</u>

Next, the Staff explains how conservatism is removed through refinements; these refinements are considered acceptable because they more accurately reflect the actual operating conditions of the component compared to the conservative assumptions used during the design of the component. NRC Fatigue Test. at A54:

Design fatigue calculations for components generally use conservative assumptions in the calculation of the CUF, typically to make performing the calculation more simple. *Id.* These conservative assumptions can relate to the severity of the transients (where less severe transients are grouped with and treated as higher severity transients), the number of transients (where the number of transients expected to occur over the service life of the component is intentionally increased for use in the calculation), and the stresses on the component generated by the transient (where simplifying assumptions are made to provide conservative estimates of the component stresses). *Id.* If the calculated CUF values with these various conservative assumptions are less than or equal to the limit of 1.0, then the analysis is considered acceptable because it has met the acceptance limit defined by the ASME Code and the analyst stops work on the analysis. *Id.* Although the analyst could have reduced conservatism in the calculation to achieve a more accurate value of CUF, this isn’t typical because the objective of acceptability was achieved by demonstrating that the CUF is less than 1.0. *Id.*

When a value of CUF_{en} is calculated for that same component, that value may exceed the allowable value of 1.0 because the F_{en} value, which is multiplied by the CUF value to calculate the CUF_{en}, can be larger than 1.0. *Id.* Since the limit for CUF_{en} is also 1.0, the analyst can perform a refined analysis for the component to reduce the CUF to an acceptable level by removing some of the conservatisms that were used in the original calculation. *Id.* In other words, the analyst continues the calculation originally developed for the component with the addition of an extra multiplier. *Id.* The iterative process that results from the addition of the F_{en} multiplier often involves the use of fewer conservative assumptions than originally applied so that a more refined value of CUF_{en} can be calculated. *Id.* Since these iterative calculations involve the use of fewer conservative assumptions and methods, the refined value of CUF_{en} can be lower than the original value. *Id.*

Importantly, the actual plant transients that occur are typically less severe than the design transients, which are defined on a generic basis by the vendor for the component design. The use of actual severity of the transients experienced by the plant would typically result in a CUF_{en} value that is lower than that from the original design calculation. In addition, transients may occur less frequently than specified by the original design, which would lead to lower calculated CUF and CUF_{en} values for the component, or the design calculations may have grouped less severe transients with more severe transients because it simplified the CUF calculation. *Id.* In the iterative analyses, these transients may be ungrouped and individually analyzed as separate transients. *Id.*

The witnesses note that Entergy uses the FMP to ensure that the assumptions (e.g., transient severity, number of transients) made in the underlying CUF or CUF_{en} evaluations remain valid based on the actual operation of the plant. *Id.* The FMP for both IP2 and IP3, as described in Entergy's implementing procedures, require corrective actions when a single transient type (e.g., heat-up transient or cool-down transient) approaches the respective number of transients used in the fatigue evaluation for that transient type. *Id.* Since a design fatigue calculation or environmentally-assisted fatigue evaluation for a component typically include multiple transients (e.g., not just a single transient), this approach provides additional margin against the CUF or CUF_{en} exceeding its limit. *Id.* at A56:

6. <u>Repair or Replacement vs. Reanalysis</u>

The ASME Code does not require the preemptive repair or replacement of components that have calculated CUF values less than or equal to 1.0, or even for components where the calculated CUF is greater than 1.0. NRC Fatigue Test. at A51. A commitment to repair or replace components when the CUF approaches unity is not required or necessary. *Id.*

The Staff's witnesses explain that Entergy will have taken corrective actions, in accordance with its FMP if the monitoring of the plant transients indicates the potential for a condition outside those analyzed in the underlying fatigue evaluation and prior to the calculated CUF, CUF_{en} and fatigue limit of 1.0 being approached. *Id.* at A108.

7. Quality Assurance

As with all licensees, Entergy is required by Appendix B to 10 C.F.R. Part 50 to implement a QA Program that takes into account the need for special controls, processes, and skills to attain the required quality, and the need for verification of quality when performing an analysis or calculation. NRC Fatigue Test. at A59. Accordingly, an individual performing any evaluation must have specialized experience and be specifically trained so that a quality analysis or evaluation is produced. *Id.* In addition, Entergy's QA Program requires training of personnel performing activities affecting quality, as necessary, to assure that suitable proficiency is achieved and maintained. *Id.* Entergy is also required by its QA Program to ensure that all analyses have sufficient records and there is adequate record maintenance and retention for these records to properly document all activities affecting quality. *Id.* Furthermore, Entergy's QA Program requires that design analyses and calculations are sufficiently detailed such that a person technically qualified in the subject area can review and understand the analyses, and reproduce and verify the adequacy of the results without recourse to the originator. *Id.* Finally, Entergy's QA Program provides measures for verifying or checking the adequacy of design, such as by the performance of design reviews and independent design verifications. *Id.*

8. Overall Conclusion Concerning Entergy's Fatigue Monitoring Program

The Staff's witnesses explain where to find the details of the Staff's determination that Entergy's FMP, including the environmentally-assisted fatigue analyses for IP2 and IP3 are acceptable, and fulfill the applicable regulatory criteria in Part 54 of the Commission's regulations. NRC Fatigue Test. at A47. Specifically, the conclusions and bases for the FMP are in Section 3.0.3.2.6 of the Staff's SER. SER at 3-76 through 3-79 (Ex. NYS00326B). The conclusions and bases for Entergy's metal fatigue TLAAs are documented in SER Section 4.3. SER at 4-18 through 4-41 (Ex. NYS00326E). Finally, the conclusions and bases for Entergy's environmentally-assisted fatigue analyses are documented in Section 4.3.3 of the Staff's SER and Section 4.3.3 of the Staff's SER Supp. 1. SER at 4-41 through 4-46 (Ex. NYS00326E) and SER Supp. 1 at 4-1 through 4-3 (Ex. NYS000160). In addition, the Staff's aging management program audit report[31] and the Staff's scoping and screening audit report[32] provide additional information that supports the Staff's conclusions in the Staff's SER and SER Supp. 1. Furthermore, details about the results of the IP71002 inspection performed for IP2 and IP3 regarding the FMP are contained in the inspection report. IP71002 Report at 4 (Ex. NRC000107).

D. Rebuttal of Intervenors' Witnesses and Position

On June 9, 2015, the Intervenors submitted, *inter alia*, their Revised Statement of Position (Ex. NYS000529), the revised prefiled testimony of Dr. Richard T. Lahey, Jr., (Ex. NYS000530), the revised prefiled testimony of Dr. Joram Hopenfeld (Ex. RIV000142), and the Supplemental Report of Dr. Joram Hopenfeld dated June 8, 2015 (Ex. RIV000144). Although the Intervenors submitted revised testimony and a revised statement of position, the Intervenors state that their contention is supported by their previous filings. Revised Statement of Position at 2 (Ex. NYS000529). They do not explain, however, which portions of their previous filings need to be examined beyond the matters presented in their more recent testimony. As expressed in their Revised Statement of Position (Ex. NYS000529),

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[31] Audit Report for Plant Aging Management Programs and Reviews, Indian Point Nuclear Generating Unit Nos. 2 and 3, (January 13, 2009) (Ex. NRC000108) ("AMP Audit Report"),

[32] Scoping and Screening Methodology Audit Trip Report for Indian Point, Units 2 and 3, (January 13, 2009) (Ex. NRC000124) ("Scoping Audit Report")

the Intervenors believe that Entergy's FMP is deficient for the reasons proffered by Drs. Lahey and Hopenfeld. The Staff addresses the testimony of each witness below.

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

Dr. Lahey testified, in part, on metal fatigue (Revised Lahey PFT, at 62-73 (Ex. NYS000530)) and provided several reports (Lahey Report, ¶¶24-31 (Ex. NYS000296); Lahey Supp. Report (Ex. NYS000297)). Dr. Lahey's reports and testimony make four assertions: (1) that Westinghouse removed conservatism from its CUFen calculations; (2) that some non-conservatisms were ignored; (3) that a lack of an error analysis makes it inappropriate to rely on CUFen values that are close to unity; and (4) that there are real-world examples of the limitations of CUFen calculations.

The Staff's witnesses, having carefully reviewed the reports and testimony of Dr. Lahey, conclude that Dr. Lahey's concerns are without merit.

a. Iterative CUFen Calculations

Dr. Lahey expresses his concern over the iterative process used by Westinghouse whereby "safety margin" is removed until the CUFen is at or below 1.0. Revised Lahey PFT, at 67-68 (Ex. NYS000530); State of New York and Riverkeeper, Inc. Revised Statement of Position on Consolidated Contention NYS-26B/RK-TC-1B at 21-22 (NYS000529).

Dr. Lahey's concerns about removing conservatisms and the calculated value being closer to one do not demonstrate that Entergy's FMP fails to satisfy 10 C.F.R. Part 54. This is because, as the Staff's witnesses explain, the practice of refining calculations and removing conservatisms is an appropriate practice and is an expected part of any realistic analysis. NRC Fatigue Test. at A209. In fact, it would be unusual for an experienced analyst to perform a fatigue calculation without some degree of iterative removal of conservatism. *Id.*.

Design fatigue calculations for components generally use conservative assumptions in the calculation of the CUF, typically to make performing the calculation more simple. *Id.* These conservative assumptions can relate to the severity of the transients (where less severe transients are binned with and treated as higher severity transients), the number of transients (where the number of transients expected to occur over the service life of the component is intentionally increased for use in the calculation), and the stresses on the component generated by the transient (where simplifying assumptions are made to provide conservative estimates of the component stresses). *Id.* The analyst can perform a refined analysis for the component to reduce the CUF by removing some of the conservatisms that were used in the original calculation. *Id.* Since these iterative calculations involve the use of fewer conservative assumptions and methods, the refined value of CUFen can be lower than the original value. NRC Fatigue Test. at A54. This iteration would continue as appropriate until the CUFen was calculated to be less than or equal to 1.0. *Id.* at A209.

In fact, iteration during design has been quite common in ASME Code fatigue calculations for more than 50 years. *Id.* Nothing in the ASME Code prohibits the iterative process performed by Westinghouse. *Id.* The iterative process that results from the addition of the Fen multiplier often involves the use of fewer conservative assumptions than originally applied so that a more refined value of CUFen can be calculated. *Id.* at A54. Further calculations can be performed to reduce the existing stress values by using more realistic loadings or more detailed analysis models, which would result in lower CUFs. *Id.* at 209..

In sum, instead of identifying a deficiency in Entergy's FMP, Dr. Lahey simply highlights the long-standing, and acceptable, iterative methodology used by analysts.

b. <u>Synergistic Aging Effects</u>

Dr. Lahey asserts that 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 (Ex. NYS000530). Further, Dr. Lahey alleges that degraded components could be exposed to an unexpected seismic event or shock load that could cause failure. Revised Lahey PFT 68-69 (Ex. NYS000530).

Dr. Lahey's concerns about unaddressed synergistic aging effects, and about unexpected seismic or shock loads, do not demonstrate that Entergy's FMP fails to satisfy 10 C.F.R. Part 54. Furthermore, seismic and other accident loads are considered as part of the design calculations, and Dr. Lahey's argument is irrelevant because the important failure mode for these severe loads is gross structural overload and deformation, not fatigue crack initiation. NRC Fatigue Test. at A145, A213. Is also beyond the scope because the license renewal purpose of the FMP is to maintain the CLB in accordance with 10 C.F.R. Part 54, meaning a proper aging management program leaves unchanged the plant's response to the hypothesized seismic event or shock load. *Id.* at A14.

With respect to managing age-related degradation, the FMP is consistent with the state of current knowledge and understanding. NRC Fatigue Test. at 201. Entergy's aging management programs are consistent with NRC's current understanding of age-related degradation, which is documented in the GALL Report. *Id.*

Dr. Lahey does not provide a meaningful supporting basis regarding any aging affect that is unaddressed that presents a new seismic concern. See *id.* at A211 to A214.

c. CUFen Values May Be Used Without an Error Analysis

Dr. Lahey alleges that it is an error for Westinghouse not to include an error analysis along with its CUF_{en} calculation results. Revised Lahey PFT at 70 (Ex. NYS000530); Lahey Report, ¶34 (Ex. NYS000296).

Dr. Lahey's concerns about the lack of an error analysis do not demonstrate that Entergy's FMP fails to satisfy 10 C.F.R. Part 54.

An error analysis involves the investigation of errors, or mistakes, made in an analysis, and is addressed under Appendix B to 10 C.F.R. Part 50, which provides measures for verifying or checking the adequacy of design, such as by the performance of design reviews, and which ensures that design analyses and calculations are sufficiently detailed such that a person technically qualified in the subject area can review and understand the analyses, and independently verify the adequacy of the results. NRC Fatigue Test. at A171. Appendix B to 10 C.F.R. Part 50 does not address the possible variation, or uncertainty, of inputs used in a calculation as Dr. Lahey discusses in his testimony. *Id.*

Fatigue calculations performed in accordance with ASME Code Section IIII, such as those done by Entergy (i.e., CUF or CUF_{en}) are deterministic. *Id.* at A171. The characteristics of a deterministic calculation are to use conservative, bounding input values and required safety factors to produce results that are conservative as compared to what is actually expected. *Id.*. There is no requirement, either in the ASME Code or in any NRC regulations, to perform uncertainty analyses for these types of deterministic fatigue calculations. *Id.*

An uncertainty analysis involves the investigation of the variation, or uncertainty, which is possible with inputs used in a calculation. *Id.* However, such analyses are only performed in probabilistic calculations that estimate the probabilities that certain outcomes will occur. *Id.* The fatigue calculations performed by Entergy (i.e., CUF or CUF_{en}) are not probabilistic calculations, so an uncertainty analysis is not necessary. *Id.* Accordingly, the example Dr. Lahey cites where an uncertainty analysis was performed for a Monte Carlo criticality analysis is inapposite. *Id.* at 172. Indeed, the report cited by Dr. Lahey shows that uncertainty analysis is not performed for deterministic analyses and does not support his contention that an error analysis should be performed for the IP CUF_{en} analyses. *Id.*

Within the ambit of error analysis, the Intervenors express concerns regarding Westinghouse's computer code WESTEMS. Revised SOP at 25-26. But, in the end, Dr. Lahey has not supported his concerns, regarding the WESTEMS thermal-hydraulic models and framework, in that he does not identify any instance where the models and framework employed by WESTEMS™ for IP2 and IP3 invalidated the Applicant's environmentally-assisted fatigue calculations. NRC Fatigue Test. at A173.

d. Real World Operating Experience

Lastly, Dr. Lahey notes that in-core fatigue failures of irradiated baffle-to-former bolts have been observed in operating PWRs . *See* NRC Fatigue Test. at A215. Dr. Lahey's observations do not demonstrate that Entergy's FMP is insufficient to satisfy 10 C.F.R. Part 54.

In their Statement of Position, the Intervenors assert that in-core fatigue failures of irradiated baffle-to-former bolts have been observed in operating PWRs. Revised Statement of Position on Consolidated Contention NYS-26B/RK-TC-1B at 27-28 (Ex. NYS000529). The Revised SOP cites several supporting documents, including NYS-000324, NYS000353, NYS000316, and NYS000315. In most cases, however, the contents of the cited document have either been taken without the context provided or needed context has not been provided. NRC Fatigue Test. at A214. For example, NYS000324 states that "Possible causes of baffle/former bolt cracking are irradiation-assisted SCC (IASCC), irradiation embrittlement, stress relaxation, and fatigue, or some combination of these." NYS000324 at 2-30 (emphasis added); NRC Fatigue Test. at A216. In fact, MRP-51, *Hot Cell Testing of Baffle/Former Bolts Removed from Two Lead PWR Plants*, (November 2001) (Ex. NRC000176) ("MRP-51), analysis of failed bolts removed from Point Beach Unit 2 determined that the "fractures appear to be caused by irradiation assisted stress corrosion cracking (IASCC)." NRC Fatigue Test. at A217-218.

Contrary to Dr. Lahey's apparent belief, fatigue usage factors alone are not sufficient to manage all forms of cracking in all cases, such as the cited case where IASCC is the predominant mechanism governing degradation of these bolts. *Id.* at A223. That is exactly the reason why aging management of the baffle-to-former and barrel-to-former bolts addressed by the RVI Inspection Program uses volumetric examination of the bolts to identify cracks without reliance on a fatigue usage calculation. *Id.* at A223.

2. Reports and Testimony of Dr. Joram Hopenfeld

The Staff's witnesses, having carefully reviewed the reports and testimony of Dr. Joram Hopenfeld, conclude that Dr. Hopenfeld's concerns are without merit. Dr. Hopenfeld's reports and testimony make multiple related assertions that essentially boil down to three claims: (1) Entergy incorrectly performed calculations in 2010; (2) Entergy incorrectly performed calculations associated with Commitment Nos. 43 and 49; and (3) Entergy failed to provide sufficient details in its FMP.

As explained below, the issues Dr. Hopenfeld describes do not show any inadequacy in Entergy's FMP.

a. Refined Analyses of Environmental Fatigue Factors

First, regarding the 2010 calculations, the Intervenors assert that Dr. Hopenfeld found that there is a wide margin of error in Entergy's calculations, and, consequently, Entergy likely grossly under predicted CUFen values for four reasons. Revised SOP at 30. The four claimed errors are: (1) failure to adjust laboratory data to account for the actual reactor environment; (2) use of incorrect values for dissolved oxygen (DO) levels; (3) use of inaccurate heat transfer coefficients; and (4) use of an unjustified number of transients. Revised SOP at 30-37. As explained below, the Staff's witnesses disagree with Dr. Hopenfeld, and conclude that his assertions regarding the 2010 calculations do not demonstrate that Entergy's FMP is deficient.

i. Correctly Accounting for Actual Reactor Environment

Concerning the claims that the calculations failed to adjust laboratory data to account for the actual reactor environment, the Intervenors want Entergy to use bounding values recommended by Argonne National Laboratory. Revised SOP at 31-32. The Intervenors believe these bounding values are "far more realistic." Revised SOP at 32.

The Staff disagrees with Dr. Hopenfeld's testimony and the cited inadequacies in Entergy's analyses. Fatigue Test. at A119. Dr. Hopenfeld provides no substantiation that the values calculated and used by Entergy are not incorrect. *See Id.* Instead, Dr. Hopenfeld seems to misinterpret NUREG/CR-6909 to indicate that the bounding values identified in the ANL report are "realistic" rather than what they actually are – i.e. bounding values that are among the highest values identified by ANL in the report. *Id.*

ii. <u>Correctly Selecting Values for Dissolved Oxygen</u>

Intervenors' asserted position with respect to dissolved oxygen is that Entergy's approach directly contravenes basic laws of physics and specifications provided by ANL and endorsed by the NRC. Revised SOP at 32-33.

The Staff disagrees. *See* NRC Fatigue Test. at A120 to A123. Westinghouse provided an explanation regarding the treatment of DO levels in the refined CUF analyses. *Id.* at A122. Indeed, Entergy's Water Chemistry Control - Primary and Secondary Program, which the Staff found to be acceptable, maintains DO at IP2 and IP3 consistent with the DO values used by Entergy in WCAP-17199-P and WCAP-17200-P, negating the need to use the bounding values that Dr. Hopenfeld suggests. NRC Fatigue Test. at A120. Westinghouse's approach was correct. *Id.* at A121.

iii. <u>Correctly Selecting Heat Transfer Coefficients</u>

Intervenors assert that Entergy used unrealistically low heat transfer coefficients. Revised SOP at 36. As a consequence, they claim the CUFen is incorrect. Revised SOP at 36.

In all cases, for the IP2 and IP3 pressurizer surge lines and the IP2 and IP3 reactor inlet and outlet nozzles, Dr. Hopenfeld questions Entergy's analyses that are part of its CLB. NRC Fatigue Test. at A124. Specifically, the heat transfer coefficient, which Dr. Hopenfeld is questioning, is used to calculate the stress values, which in turn are then used as an input to the

CUF calculation; again, these are CUF calculations that are part of the CLB and are not part of the review for license renewal. *Id.* Dr. Hopenfeld's assertions should therefore be disregarded.

The Staff also noted that, through Entergy's and its vendors' QA Programs in accordance with Appendix B to 10 C.F.R. Part 50, the review and verification processes would ensure that reasonable and appropriate inputs, such as the heat transfer coefficient, are used in these calculations. *Id.*

iv. Correctly Selecting the Number of Transients

Dr. Hopenfeld states that Entergy's fatigue evaluations use an unjustified number of transients and do not adequately consider either past or future transients at IP2 and IP3. See Hopenfeld at 17 (Ex. RIV000034).

The Staff disagrees. NRC Fatigue Test. at 125. As a threshold matter, Dr. Hopenfeld does not provide any data to support his assertions that Entergy's records are in any way incomplete or insufficient. *Id.*

Second, Entergy's existing license already requires, pursuant to TS 5.5.5., Entergy to track the transients at the plant. *Id.*

As part of license renewal, the Staff determined that Entergy tabulated the appropriate number of past transients, and responded to questions about future predictions. *Id.* Moreover, Entergy's FMP includes an enhancement to Entergy's current IP3 FMP to revise appropriate procedures to include all the transients identified, to assure all fatigue analysis transients are included with the lowest limiting numbers, and to update the number of design transients accumulated to date. *Id.* The data for past transients at IP2 and IP3 will be incorporated into each existing FMP. *Id*.

Dr. Hopenfeld's assertion that Entergy has not justified a straight line extrapolation to determine the remaining number of transients lacks merit, because Entergy is using its FMP to verify the extrapolated number of transients and take corrective action in the event those extrapolations are found to be non-conservative. *Id.* This aging management approach for the CUF_{en} calculations assures that the fatigue limit of 1.0 will not be exceeded. *Id.* Entergy's procedures for IP2 and IP3 specifically prohibit the actual number of transient cycles from exceeding the number of transient cycles assumed in the fatigue calculations without corrective action. *Id.*

Finally, the "past transients" for IP2 and IP3, referred to by Dr. Hopenfeld, will be incorporated in Entergy's FMP prior to entering the period of extended operation, and Entergy's use of this program does not rely on the prediction of "future" transient cycles to justify continued operation. *Id.* Such projections only give an indication that components continue to meet the fatigue limit of 1.0 during the period of extended operation consistent with their fatigue analyses. *Id.* Should Entergy find that the actual number of occurrences for any transient exceeds the projected number assumed in the fatigue calculations, it would implement corrective actions consistent with its program to ensure that the CUF and CUF_{en} calculations remain valid. *Id.*

b. Commitment Nos. 43 and 49

Dr. Hopenfeld's Supplemental Report (Ex. RIV000144) details the various shortcomings of the additional rounds of CUFen calculations conducted for Entergy by Westinghouse in connection with Commitment Nos. 43 and 49. He further asserts that 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 the Intervenors in this proceeding. Dr. Hopenfeld identifies 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 (Ex. RIV000144).

The NRC Staff disagrees with Dr. Hopenfeld's assertions. The Staff's review of Commitment No. 43 is documented in the Section 4.3.3.2 of SER Supp. 1. SER Supp. 1 at 4-1 through 4-3 (Ex. NYS000160); NRC Fatigue Test. at A84. In Commitment No. 43, Entergy stated that it will review its ASME Code Class 1 fatigue evaluations to confirm that the NUREG/CR-6260 locations, which were evaluated for the effects of the reactor water environment, are the limiting locations for IP2 and IP3. NRC Fatigue Test. at A68. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor water environment and added to the FMP. *Id.* Further, Entergy will use the NUREG/CR-6909 methodology, which currently represents the best methodology for nickel alloys, in the evaluation of any nickel alloy limiting locations. *Id.*

In addition, Entergy is required by Appendix B to 10 C.F.R. Part 50 to implement a QA Program that takes into account the need for special controls, processes, and skills to attain the required quality, and the need for verification of quality when performing an analysis or calculation. *Id.* at A59. This means that any fatigue analyses that may be performed in the future, as well as the evaluation that was completed as part of Commitment No. 43, are governed by Entergy's QA Program. *Id.*

In Commitment No. 49, Entergy stated that it will recalculate each of the limiting CUFs provided in Section 4.3 of the LRA for the reactor vessel internals to include the reactor coolant environment effects (Fen) as provided in the IPEC FMP using NUREG/CR-5704 or NUREG/CR-6909. In accordance with the corrective actions specified in the FMP, corrective actions include further CUF re-analysis, and/or repair or replacement of the affected components prior to the CUFen reaching 1.0. NRC Fatigue Test. at A68. Regarding any errors in fatigue calculations associated with Commitment No. 49, Entergy is required by Appendix B to 10 C.F.R. Part 50 to implement a QA Program that takes into account the need for special controls, processes, and skills to attain the required quality, and the need for verification of quality when performing an analysis or calculation. *Id.* at A92.

The assumption that Entergy is not following its QA program correctly constitutes a challenge to the current operating license and is beyond the scope of a license renewal proceeding.

i. Dissolved Oxygen

Dr. Hopenfeld makes several assertions regarding the usage of DO in Entergy's calculations. *See* e.g. Hopenfeld Supplemental Report at 9 (Ex. RIV000144). Dr. Hopenfeld asserts that non-conservatisms in the CUFen calculations result from the inadequate consideration of DO levels. Hopenfeld Supplemental Report, at 6-13 (Ex. RIV000144). Dr. Hopenfeld would require usage of generic DO values in place of measured values, asserting measured values are not well known during transients. *Id.* at 7.

The Staff disagrees with Dr. Hopenfeld's claims. Water chemistry is tightly controlled by one of Entergy's aging management programs, the Water Chemistry Control - Primary and Secondary Program. NRC Fatigue Test. at A120. This program, which is described in LRA Section B.1.41, relies on monitoring and control of reactor water chemistry based on the EPRI water chemistry guidelines in TR-1002884 Rev. 5. LRA at B-137 (Ex. ENT00015A-B); NRC Fatigue Test. at A120. The available plant measurements of DO are the best indicator of DO levels in the plant, so they are appropriate for use in the Fen calculations. NRC Fatigue Test. at A152. In this regard, the Staff considers it to be reasonable to take into account the normal variation in the measurements to yield a conservative Fen calculation. *Id.* In addition, the guidance in MRP-47 uses a time-averaged value of DO from plant records as a best-estimate of DO for use in the Fen relationships. *Id.* Such an approach is acceptable for use when actual plant DO measurements are used in the Fen expressions. *Id.*[33] The NRC finds this to be a reasonable approach that is within the accuracy of the Fen method. *Id.*

ii. <u>Effects of Radiation and Stress Corrosion</u>

Dr. Hopenfeld asserts that there are various shortcomings in the CUFen calculations for RVIs due to Westinghouse's failure to account for the effects of radiation and stress corrosion. Hopenfeld Supplemental Report, at 13-18 (Ex. 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 (Ex. NYS000530). Dr. Hopenfeld states that any analysis of the effects of the LWR environment on fatigue must consider the synergistic effects of radiation, SCC and thermal embrittlement, and that a first step toward this end would be to incorporate the effects of radiation into the Fen equation. Hopenfeld Supplemental Report at 15 (Ex. RIV000144).

As a threshold matter, Dr. Hopenfeld does not offer any specific research data or evidence to support his assertion that treating these mechanisms separately is inadequate, nor does he provide any synergistic models, methods, or evaluations to support his assertion. NRC Fatigue Test. at A153.

The effects of SCC and thermal embrittlement are addressed separately from fatigue. The NRC and consensus standards such as the ASME currently treat these effects separately, and it is intended that the separate evaluation approach for these mechanisms is conservative. The NRC continues to find such an approach acceptable and conservative, and is not aware of any specific data to indicate otherwise. *Id.* Because Entergy's FMP ensures CUF and CUFen

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[33] Entergy presented such an approach for using measured DO values for *Vermont Yankee* using a mean plus one standard deviation value during ASLB Hearings in 2008. Entergy Nuclear Vermont Yankee, LLC (Vermont Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 807-809 (2008)(addressing a claim supported by Dr. Hopenfeld, and opposed by, *inter alia*, Gary Stevens, that Entergy's CUFen calculations do not adequately account for the DO chemistry of the reactor water), (reversed in part on other grounds CLI-10-17, 72 NRC 1 (2010)). The Board stated in part: "The use of actual DO data from the feedwater system, as well as the use of industry guidance DO values in other systems, was reasonable and appropriate." *Id.* at 809.

calculations remain valid and that the fatigue limit of 1.0 is not exceeded, there is no reason to assume the presence of cracks caused by fatigue that could lead to potentially significant irradiation effects on the structural integrity of the components. *Id.*

Generally speaking, radiation increases the mechanical properties of materials (i.e., yield and ultimate tensile strengths) which, in turn, improves fatigue life (increases the number of applied cycles before the onset of fatigue crack initiation). However, the total amount of irradiated research test data available is insufficient to perform a detailed, comprehensive, statistical evaluation of data in the same manner that was done for unirradiated data to develop the Fen expressions. *Id.* Nonetheless, the Staff believes application of the current Fen expressions and fatigue curves is conservative for application to irradiated components. *Id.* at A154. To this end, as discussed by the Staff's witnesses, the NRC and Argonne National Laboratory, concluded that it is still appropriate to apply the F_{en} method to irradiated components. *Id.* at A153.

iii. <u>CUF of Record Usage</u>

Dr. Hopenfeld asserts that the methodology used to calculate refined CUFen values based on the CUF of record are unreliable. Hopenfeld Supplemental Report, at 18-24 (Ex. RIV000144). He asserts that ASME fatigue curves cannot be used due to various changes in the components due to aging effects (e.g., erosion/corrosion). Hopenfeld Supplemental Report, at 18-19 (Ex. RIV000144). His view is that the CUF of record is obsolete for various reasons including consideration of strain rates or radiation effects. Hopenfeld Supplemental Report, at 19-24 (Ex. RIV000144).

The Staff disagrees with Dr. Hopenfeld's claims. NRC Fatigue Test. at A158. Dr. Hopenfeld implies that the effects of the LWR environment were not considered in the design of the Indian Point reactor vessels, which is incorrect. In fact, it is a requirement of ASME Code, Section III, to which the Indian Point reactor vessels were designed, to address the environment to which the component will be exposed. *Id.* For other, less prominent aging mechanisms, Section XI requires inspections to look for degradation and, if found, to evaluate the degradation or repair it. Dr. Hopenfeld does not provide any supporting inspection data or other evidence to support his claims that the IP units are experiencing any type of significant degradation from other effects such as swelling, pitting, or cavitation, which would invalidate the governing CUF calculations. *Id.* Finally, using the CUF of record, with possible adjustments necessitated by considerations from actual plant operation, is consistent with Section XI guidance. *Id.*

Concerning Dr. Hopenfeld's claims about the need to re-visit the CUFs of record, to consider changes in material surfaces, Dr. Hopenfeld does not offer any inspection evidence to support his claim that the surfaces of any components in the Indian Point reactors have undergone changes that lead to overwhelming effects on fatigue life, nor is the Staff aware of any field experience that supports this assertion. NRC Fatigue Test. at A159: As described in NUREG/CR-6909 at 71 (Ex. NYS000357), factors such as material variability and data scatter, size effects, surface finish effects, and load sequence effects were specifically developed to account for the observed variation in test results and the differences in key parameters between laboratory test specimens and actual components. *Id.* at A147.

Dr. Hopenfeld's other assertions concerning why the CUFs of record are now flawed do not withstand scrutiny. Concerning, for example, his claim about strain rates, the Staff is not aware of any "errors" in the CUF of record associated with strain rates, and Dr. Hopenfeld doesn't provide an example. *Id.* A162. In fact, it is well-recognized by component design experts in the nuclear industry that the CUF of record has been shown to be consistently conservative with respect to actual plant operation. *Id.* A source of conservatism in the CUF of record is a common assumption made in those analyses, that rapid step-changes in temperature occur thereby maximizing hypothetical thermal stress in components, whereas real world temperature changes are a slower function of the physical properties of the component and its environment. *Id.* The Staff's evaluations of the competing effects of stress vs. strain rate support the conclusion that use of the CUF of record is generally conservative and appropriate for use when performing CUF_{en} calculations. *Id.* Dr. Hopenfeld's testimony does not offer any specific examples or evidence to the contrary. *Id.*.

iv. Determining the Most Limiting Locations

Dr. Hopenfeld contends that Entergy and Westinghouse have failed to select the most limiting locations to evaluate CUFen values. Hopenfeld Supplemental Report, at 25 (Ex. RIV000144). He alleges that various other locations should be evaluated for fatigue. Hopenfeld Supplemental Report, at 25-28 (Ex. RIV000144).

The Staff disagrees that more action is needed in this regard. In Commitment No. 43, Entergy stated that it will review its ASME Code Class 1 fatigue evaluations to confirm that the NUREG/CR-6260 locations that were evaluated for the effects of the reactor water environment are the limiting locations for IP2 and IP3. NRC Fatigue Test. at A57. The purpose of Commitment No. 43 is for Entergy to confirm that the representative sample of components that were selected for an older vintage Westinghouse plant in NUREG/CR-6260 is sufficient for IP2 and IP3, in that the limiting CUF locations for IP2 and IP3 are included by considering the NUREG/CR-6260 locations. *Id.* at A62. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor water environment and added to the FMP. *Id.* at A57. Further, Entergy will use the NUREG/CR-6909 methodology, which currently represents the best methodology for nickel alloys, in the evaluation of any nickel alloy limiting locations. *Id.* The Commission limited the scope of the license renewal review to the effects of age-related degradation related to the license renewal operating period, stating that the on-going regulatory process provides reasonable assurance that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety for operation during any renewal period. *Id.* at A58. Limiting this commitment to ASME Code Class 1 fatigue evaluations is consistent with Commission policy, since these evaluations are the only CLB fatigue evaluations. *Id.*

Moreover, Entergy's review of additional locations is subject to the requirements of Appendix B to 10 C.F.R. Part 50, which requires Entergy to implement a QA Program that takes into account the need for special controls, processes, and skills to attain the required quality, and the need for verification of quality when performing an analysis or calculation. *Id.* at A59.

c. <u>Sufficiency of Details</u>

Dr. Hopenfeld contends that Entergy failed to follow the intent of NUREG-1801, Generic Aging Lessons Learned (GALL) Report Hopenfeld Supplemental Report, at 25-28 (Ex. RIV000144). Dr. Hopenfeld states that the results of Entergy's fatigue analyses were "fabricated for appearance and regulatory consumption and, therefore, cannot be used to manage fatigue during the proposed periods of extended operation." *Id.* at 32. Dr. Hopenfeld states that Entergy has failed to define specific criteria to assure that susceptible components are inspected, monitored, repaired, or replaced in a timely manner, and that it should establish criteria for repair versus defect monitoring, and establish criteria for the frequency of the inspection, and corrective actions. *Id*.

First, with respect to meeting the Staff's guidance document, the ten program elements of Entergy's FMP were reviewed by the Staff during the on-site Aging Management Program audit, the Scoping and Screening audit, in-office reviews, and the IP71002 inspection, as a result of which the Staff found the program to be acceptable. NRC Fatigue Test. at A146.

Second, contrary to Dr. Hopenfeld's view, the program is not insufficiently-defined. For example, the FMP includes corrective actions that are initiated if the monitoring of the plant transients indicates the potential for a condition outside those analyzed in the underlying fatigue evaluation. *Id.* at A53. Corrective actions include performing a more rigorous analysis to remedy the condition, or repair or replacement of affected components, before the CUF or

CUFen exceeds 1.0. *Id.* These corrective actions are consistent with the recommendation in the GALL Report for AMP X.M1, which is an acceptable option for managing metal fatigue for the reactor coolant pressure boundary, considering environmental effects. *Id.*

Based on its reviews, the Staff has concluded that the metal fatigue TLAAs, the environmentally-assisted fatigue analyses, and the FMP for license renewal of IP2 and IP3 are acceptable, and there is no merit in the contention's assertion that Entergy's LRA does not include an adequate plan to monitor and manage the effects of aging due to metal fatigue on key reactor components, in violation of 10 C.F.R. § 54.21(c)(1)(iii). *Id.*

CONCLUSION

Based upon the lack of specifics in the testimony of Drs. Lahey and Hopenfeld, as described by the Staff's witnesses, the Board should accord little weight to the Intervenors' testimony and report. In the end, their testimony amounts to speculation that Entergy will make a mistake in the future. For the reasons stated above, the LRA for IP2 and IP3 is adequate with respect to the AMP for metal fatigue. Accordingly, NYS-26B/RK-TC-1B should be resolved in favor of the Applicant.

Respectfully submitted,

/Signed (electronically) by/

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