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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	Docket Nos. 50-247-LR and 50-286-LR
ENTERGY NUCLEAR OPERATIONS, INC.	ASLBP No. 07-858-03-LR-BD01
(Indian Point Nuclear Generating Units 2 and 3)	

APPLICANT'S ANSWER TO NEW AND AMENDED CONTENTION NEW YORK STATE 26B/RIVERKEEPER TC-1B (METAL FATIGUE)

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October 4, 2010

TEMPLATE = SECY-036

DS-03

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NUCLEAR REGULATORY COMMISSION

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In the Matter of	)	Docket Nos. 50-247-LR and 50-286-LR
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ENTERGY NUCLEAR OPERATIONS, INC.	)	
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(Indian Point Nuclear Generating Units 2 and 3)	)	
	)	October 4, 2010

APPLICANT'S ANSWER TO NEW AND AMENDED CONTENTION
NEW YORK STATE 26B/RIVERKEEPER TC-1B (METAL FATIGUE)

Pursuant to 10 C.F.R. § 2.309(h)(1) and the Atomic Safety and Licensing Board's ("Board") July 1, 2010, Scheduling Order, Entergy Nuclear Operations, Inc. ("Entergy") submits this Answer to the Amended Contention jointly filed by Petitioners New York State ("NYS") and Riverkeeper, Inc. ("Riverkeeper") on September 9, 2010.[1] This proceeding concerns Entergy's license renewal application ("LRA") for Indian Point Nuclear Generating Units 2 and 3 ("IP2" and "IP3"), also referred to as the Indian Point Energy Center ("IPEC").

The Amended Contention seeks to amend two previously-admitted and consolidated contentions that challenge the manner in which the IPEC Fatigue Monitoring Program will manage the effects of environmentally-assisted fatigue ("EAF") on key reactor components during the period of extended operation. As set forth below, the Amended Contention is not admissible because it raises issues beyond the scope of this proceeding, lacks adequate factual and legal support, and fails to raise a genuine dispute on a material issue of law or fact, as required by 10 C.F.R. § 2.309(f)(1)(iii)-(vi). The Amended Contention also is untimely in at

[1] *See* State of New York's and Riverkeeper's Motion for Leave to File a New and Amended Contention Concerning the August 9, 2010 Entergy Reanalysis of Metal Fatigue (Sept. 9, 2010) ("Motion for Leave"); Petitioners State of New York and Riverkeeper, Inc. New and Amended Contention Concerning Metal Fatigue (Sept. 9, 2010) ("Amended Contention"). NYS also filed the Declaration of Richard T. Lahey, Jr., dated September 8, 2010 ("Lahey Decl.") and the Declaration of Dr. Joram Hopenfeld, dated September 9, 2010 ("Hopenfeld Decl.").

least one respect, insofar as Petitioners belatedly argue, without good cause, that Entergy must consider reactor pressure vessel ("RPV") "in-core" structures and certain accident loads as part of its fatigue analyses. Apart from lacking a legal basis, these arguments could have been presented based on the original LRA.

I. PRELIMINARY STATEMENT

Certain metal components in a nuclear power reactor have a distinctive number of stress cycles that the material can withstand at a particular applied stress level before fatigue crack initiation occurs. 10 C.F.R. § 50.55a(c)(1) requires that metal components that are part of the reactor coolant system ("RCS") pressure boundary meet the metal fatigue requirements for Class 1 components in Section III of the American Society of Mechanical Engineers ("ASME") Boiler and Pressure Vessel Code ("ASME Code"). The ASME Code provides the methodology for calculating the cumulative usage factors ("CUF") for covered nuclear plant components. The CUF represents the fraction of the total allowable fatigue cycles that the component is projected to incur during its operation and, under the ASME Code, must not exceed 1.0.

The Nuclear Regulatory Commission ("NRC") Staff has directed license renewal applicants to address the effects of the reactor coolant environment on component fatigue life (*i.e.*, EAF) when formulating aging management programs ("AMPs") in support of license renewal.[2] NRC guidance states that an applicant may assess EAF effects on the six critical RCS pressure boundary component locations identified in NUREG/CR-6260.[3] An applicant may evaluate these six locations by applying environmental correction factors ("F_{en}") to CUFs determined in ASME Code fatigue analyses, to obtain environmentally-adjusted cumulative

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[2] *See* NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants* at 4.3-3 (Rev. 1, Sept. 2005) ("SRP-LR") (Attach. 1); NUREG-1801, Vol. 2, *Generic Aging Lessons Learned Report* at X M-1 to X M-2 (Rev. 1, Sept. 2005) ("GALL Report") (Attach. 2).

[3] *See* SRP-LR at 4.3-5; GALL Report at X M-1; LRA at 4.3-21; NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* at 4-1, 5-62 (Feb. 1995) (Attach. 3).

usage factors ("CUF_{en}").[4] Section 4.3.3 of the April 2007 LRA included a screening evaluation in which Entergy applied conservatively-calculated F_{en} values to existing CUFs for the NUREG/CR-6260 locations at IP2 and IP3, to determine preliminary, bounding CUF_{en} values for the NUREG/CR-6260 locations at IP2 and IP3, some of which exceeded 1.0.

The original ASME Code fatigue evaluations for plants such as IPEC contain substantial conservatisms that are due to the methods and assumptions used in the analyses; *e.g.*, the use of bounding heat transfer and stress analysis techniques or bounding design transients that are more severe than those experienced in actual reactor service.[5] The ASME Code permits the use of new and improved fatigue evaluation methods, such as finite element analysis, that provide more accurate CUF values by decreasing conservatisms typically included in prior analyses.[6] The Commission recently recognized the use of new and improved fatigue evaluation methods in CLI-10-17—a key decision that Petitioners do not cite once in their Amended Contention.[7]

Accordingly, in June 2010, consistent with Commitment 33 and the IPEC Fatigue Monitoring Program, Westinghouse completed new EAF analyses for IPEC at Entergy's direction. Those analyses demonstrate that the new, more accurate CUF_{en} values for *all* of the relevant IPEC RCS pressure boundary components are below 1.0 when projected to 60 years. Entergy promptly disclosed the results of the new EAF analyses to the NRC Staff, the Board, and the Petitioners.[8]

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4 SRP-LR at 4.3-7; GALL Report at X M-1; LRA at 4.3-21.

5 *See, e.g.*, NUREG/CR-6909, *Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials*, at 71 (Feb. 2007) (Attach. 4) (stating that current ASME Code fatigue evaluations rely on fatigue evaluation procedures and/or fatigue design curves that are "quite conservative").

6 *See id.* (stating that the ASME Code permits new and improved approaches to fatigue evaluations that can "significantly decrease the conservatism in the current fatigue evaluation procedures").

7 *See Entergy Vt. Yankee, L.L.C.* (Vermont Yankee Nuclear Power Station), CLI-10-17, slip op. at 42-50 (July 8, 2010) (holding that a license renewal applicant may determine new, more accurate CUF_{en} values to address EAF effects during the period of extended operation, but that such analyses are not a prerequisite to license renewal).

8 *See* Letter from Paul M. Bessette, Morgan, Lewis & Bockius, LLP, to Administrative Judges, "Notification of Entergy's Submittal Regarding Completion of Commitment 33 for Indian Point Unit 2 and 3" (Aug. 10, 2010).

Petitioners subsequently filed the Amended Contention, which, as shown below, is inadmissible for numerous reasons. Petitioners erroneously assume that, because certain (but now superseded) CUF_{en} values in the original LRA exceeded 1.0, the Fatigue Monitoring Program and updated EAF analyses are somehow inadequate. In particular, they allege that Entergy's new EAF analyses merely "make it appear" that the CUF_{en} values are all less than 1.0.[9] Petitioners also suggest that Entergy is bound to the preliminary, screening-level CUF_{en} values that it provided in its original LRA to determine whether updated EAF analyses or other corrective actions are necessary for the period of extended operation.

In addition, Petitioners seek to impose requirements on IPEC that are not contained in NRC regulations or the ASME Code; *e.g.*, the alleged need to perform a propagation of error analysis and EAF analyses for in-core reactor vessel components that are not part of the RCS pressure boundary, as specifically governed by Section 50.55a(c)(1).[10] Thus contrary to 10 C.F.R. §§ 2.335(a) and 2.309(f)(1)(iii), Petitioners' arguments run counter to NRC and ASME code requirements that have governed nuclear plant design and operation for decades.

In claiming that Entergy's updated EAF analyses rely on incorrect or undisclosed parameters, Petitioners also overlook directly-relevant and readily-available information contained in the LRA and the new EAF analyses. Similarly, in alleging that IPEC lacks a detailed and prescriptive AMP, Petitioners do not account for the detailed acceptance criteria of ASME Code Section XI, as implemented through the Fatigue Monitoring Program.

Finally, Petitioners neither substantiate the alleged errors and uncertainties in Entergy's new CUF_{en} analyses, nor show that the alleged errors and uncertainties will result in a violation

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9 Motion for Leave at 1-2.

10 *Id.* at 2 n.2, 6.

of an applicable statutory or regulatory requirement. Instead, Petitioners insist that they need more information, despite the sufficient level of detail contained in the new CUF_{en} analyses."[11]

II. PROCEDURAL HISTORY

On July 31, 2008, the Board admitted NYS-26/26A:

> to the limited extent that it asserts that the LRA is incomplete without the calculations of the CUFs as *threshold* values necessary to assess the need for an AMP, that Entergy's AMP is inadequate *for lack of the final values*, and that the LRA must specify actions to be carried out by the Applicant during extended operations to manage the aging of key reactor components susceptible to metal fatigue.[12]

The Board considered it a requirement for Entergy to include CUF_{en} calculations as part of its LRA to comply with the time-limited aging analysis ("TLAA") regulations (10 C.F.R. § 54.21(a)(3)), notwithstanding Entergy's stated reliance on an AMP pursuant to Section 54.21(c)(1)(iii).[13] In other words, the Board did not accept Entergy's commitment (in Commitment 33) to perform the updated CUF_{en} analyses within two years of the period of extended operation in accordance with its NRC-accepted Fatigue Monitoring Program.[14] The Board also found that Commitment 33 did not describe in sufficient detail Entergy's methodologies for recalculating and verifying CUF_{en} values, or summarize the CUF_{en} values for each location.[15] The Board also admitted Riverkeeper TC-1/1A and consolidated it with NYS-26/26A.[16]

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[11] The Commission has emphasized that petitioners are not entitled to discovery for the purpose of framing a proposed contention. *See AmerGen Energy Co. LLC* (Oyster Creek Nuclear Generating Station), CLI-08-28, 68 NRC 658, 676 (2008) (affirming Board decision, in *Amergen Energy Co.* (License Renewal for Oyster Creek Nuclear Generating Station), LBP-08-12, 68 NRC 5, 27 n.23 (2008), denying intervenors' request in that proceeding that the applicant disclose information "underlying [the confirmatory analyses] and any documents that were referenced by the analyses to support the assumptions made").

[12] *Entergy Nuclear Operations, Inc.* (Indian Point, Units 2 & 3), LBP-08-13, 68 NRC 43, 140 (2008) (emphasis added).

[13] *Indian Point*, LBP-08-13, 68 NRC at 137, 140.

[14] *See id.* at 138-39.

[15] *See id.* at 138.

[16] *Id.* at 166-72. The Board directed NYS and Riverkeeper to confer and submit a draft of the Consolidated Contention for the Board's consideration. *Id.* at 172, 219-20. NYS and Riverkeeper submitted the Consolidated Contention on August 21, 2008, and identified NYS as the lead party. *See* Consolidated Contention of Petitioners State of New

Thus, consistent with Commitment 33, Entergy retained Westinghouse in 2008 to determine new, more accurate CUF_{en} values for the critical RCS pressure boundary components identified in LRA Tables 4.3-13 and 4.3-14 and NUREG/CR-6260. Entergy provided the results of the new CUF_{en} analyses to the NRC Staff, Board, and Petitioners in August 2010.[17]

Westinghouse's analyses determined that the CUF_{en} values for the critical RCS pressure boundary components at IPEC are all less than 1.0 when projected to 60 years. Those analyses are described in two Westinghouse proprietary reports, which Entergy identified in its August 2, 2010 mandatory disclosures.[18] On August 4, 2010, NYS requested copies of the reports, which Entergy produced on August 11, 2010, subject to the Board's Protective Order.[19]

On August 25, 2010, after extended consultations with NYS and Riverkeeper,[20] Entergy moved for summary disposition of NYS-26/26A and Riverkeeper TC-1/1A.[21] Based on the

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

17 *See* NL-10-082, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program," attach. 1, at 2-4 (Aug. 9, 2010) ("NL-10-082") (Attach. 5) (including revised LRA Tables 4.3-13 and 4.3-14), *available at* ADAMS Accession No. ML102300504. As noted above, the Board and Parties also were provided with a copy of NL-10-082 on Aug. 10, 2010.

18 Westinghouse Electric Co., WCAP-17199-P, *Environmental Fatigue Evaluation for Indian Point Unit 2* (Rev. 0, June 2010) (proprietary); Westinghouse Electric Co., WCAP-17200-P, *Environmental Fatigue Evaluation for Indian Point Unit 3* (Rev. 0, June 2010) (proprietary). Copies of these proprietary reports were provided to the Board and Petitioners as Attachments 15 and 16 to Entergy's August 25, 2010 summary disposition motion. *See* note 21, *infra*.

19 *See* Board Protective Order (Sept. 4, 2009). Riverkeeper did not request a copy of the proprietary Westinghouse EAF reports until August 19, 2010. Entergy counsel provided copies of the reports to Riverkeeper counsel the next day (August 20, 2010).

20 During the parties' consultations, NYS requested (1) a copy of the user's manual for the proprietary WESTEMS™ computer program used by Westinghouse to perform the EAF analyses and (2) Westinghouse's error analysis. Entergy maintained that the validity of the new CUF_{en} values is not the subject of any currently-admitted contention (and, therefore, any related discovery). Nonetheless, Entergy counsel agreed to request a copy of the WESTEMS™ user's manual from Westinghouse. On September 3, 2010, after reviewing the Board's July 31, 2008 admissibility ruling, the admitted contention, and the Protective Order, Westinghouse provided excerpts from two proprietary documents: (1) the *EnvFat 1.0 User's Manual 1.0* (May 2009) and (2) the *WESTEMS™ User's Manual,* Version 4.5, Vol. 2, Rev. 0, "Design Analysis (2007), as relevant to the fatigue analyses. Entergy forwarded those documents to NYS and Riverkeeper the same day. Based on communications with Westinghouse and Entergy, counsel for Entergy understands that the postulated "error analysis" sought by Petitioners and described in their Amended Contention was not prepared because it is neither required by NRC/ASME Code requirements nor customarily prepared for an ASME Code-based EAF analysis. *See* Section IV, *infra*.

21 *See* Applicant's Motion for Summary Disposition of New York State Contentions 26/26A & Riverkeeper Technical Contention 1/1A (Metal Fatigue of Reactor Components) (Aug. 25, 2010).

Commission's recent ruling in CLI-10-17, Entergy asserted that its commitment to conduct its new EAF analyses before the period of extended operation is sufficient to meet the applicable Part 54 requirements.[22] Entergy further asserted that it fully addressed all issues raised in NYS-26/26A and Riverkeeper TC-1/1A by showing that new CUF_{en} values all are less than 1.0.[23]

On September 9, 2010, Petitioners filed the Amended Contention. Petitioners allege that Entergy has: (1) inappropriately limited the number of component locations for which EAF analyses must be performed; (2) failed to provide a propagation of error analysis; (3) improperly excluded reactor pressure vessel ("RPV") "in-core" structures and fittings from the scope of the EAF analyses; (4) not disclosed sufficient information about Westinghouse's thermal hydraulic analysis; (5) relied on incorrect or undisclosed assumptions regarding F_{en} factors, dissolved oxygen levels, and numbers of transients; and (6) failed to provide a "detailed, reliable, and prescriptive" AMP.[24] As shown below, each of these allegations is unfounded.

III. LEGAL STANDARDS FOR THE ADMISSION OF AMENDED CONTENTIONS

An intervenor may file new or amended safety contentions only with leave of the presiding officer upon a showing that the new or amended contention is based on information that was not previously available and is materially different than information previously available.[25] Thus, a new contention "is not an occasion to raise additional arguments that could have been raised previously."[26] If an intervenor cannot satisfy the criteria of Section 2.309(f)(2),

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22 *See Vt. Yankee,* CL-10-17, slip op. at 42-50 (holding that because CUF_{en} analyses are not contained within an applicant's current licensing basis ("CLB"), they are not TLAAs, and NRC regulations do not require a license renewal applicant to calculate CUF_{en} values prior to issuance of a renewed license).

23 In Answers filed on September 14, 2010, NYS/Riverkeeper and the NRC Staff opposed and supported Entergy's summary disposition motion, respectively. *See* State of New York and Riverkeeper, Inc. Combined Response to Entergy Motion for Summary Disposition of Combined Contentions NYS/26A and RK TC-1/TC-1A (Metal Fatigue) (Sept. 14, 2010); NRC Staff's Answer to Applicant's Motion for Summary Disposition of New York Contention 26/26A and Riverkeeper Contention TC1/1A – Metal Fatigue (Sept. 14, 2010). The Board's ruling is pending.

24 *See* Amended Contention at 6-13.

25 10 C.F.R. § 2.309(f)(2)(i)-(iii) (emphasis added).

26 *Duke Energy Corp.* (McGuire Nuclear Station, Units 1 & 2; Catawba Nuclear Station, Units 1 & 2), CLI-02-28, 56 NRC 373, 385-86 (2002).

then a contention is considered "nontimely," and the intervenor must successfully address the late-filing criteria in 10 C.F.R. § 2.309(c)(1)(i)-(viii). Absent good cause for a failure to timely raise the issue, a petitioner's demonstration on the other factors must be particularly strong.[27]

A proposed contention also must satisfy, "without exception," each of the admissibility criteria in 10 C.F.R. § 2.309(f)(1)(i)-(vi).[28] Several principles bear emphasis here. First, a contention that contravenes applicable statutory requirements or the basic structure of the NRC regulatory process must be rejected as outside the scope of the proceeding.[29] Second, "the Board is not to accept uncritically the assertion that a document or other factual information or an expert opinion supplies the basis for a contention."[30] Absent a reasoned basis or explanation, an expert declaration is insufficient to support admission of a contention.[31] Third, petitioners must establish that a genuine dispute exists with the applicant on a *material* issue of law or fact.[32] Finally, discovery is not permitted to assist a petitioner in framing contentions.[33]

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27 *Tex. Utils. Elec. Co.* (Comanche Peak Steam Elec. Station, Units 1 & 2), CLI-92-12, 36 NRC 62, 73 (1992) (*quoting Duke Power Co.* (Perkins Nuclear Station, Units 1, 2 & 3), ALAB-431, 6 NRC 460, 462 (1977)).

28 *S.C. Elec. & Gas Co.* (Virgil C. Summer Nuclear Station, Units 2 & 3), LBP-10-06, slip op. at 3 (Mar. 17, 2010). *See also* Final Rule, Changes to Adjudicatory Process, 69 Fed. Reg. at 2189-90, 2221 (Jan. 14, 2004) (hearings should "cover only genuine and pertinent issues of concern" to be "effective and focused on real, concrete issues").

29 *Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant Units 1), LBP-07-11, 66 NRC 41, 57-58 (*citing Phila. Elec. Co.* (Peach Bottom Atomic Power Station, Units 2 & 3), ALAB-216, 8 AEC 13, 20 (1974)).

30 *Private Fuel Storage* (Indep. Spent Fuel Storage Installation), LBP-98-7, 47 NRC 142, 181, *aff'd on other grounds*, CLI-98-13, 48 NRC 26 (1998). Thus, mere reference to articles or documents without "explanation or analysis" does not supply an adequate basis for admitting a contention. *See USEC, Inc.* (Am. Centrifuge Plant), CLI-06-10, 63 NRC 451, 472 (2006). Nor can a petitioner's imprecise reading of a document be the basis for a litigable contention. *See Ga. Inst. of Tech.* (Ga. Tech Research Reactor, Atlanta, Ga.), LBP-95-6, 41 NRC 281, 300 (1995).

31 *See Fansteel, Inc.* (Muskogee, Okla. Site), CLI-03-13, 58 NRC 195, 203-05 (2003) (holding that a petitioner also must explain the significance of any factual information upon which it relies); *Dominion Nuclear Conn., Inc.*, CLI-08-17, 68 NRC 231, 240 (Aug. 13, 2008) (noting that an expert must provide more than speculation).

32 10 C.F.R. § 2.309(f)(1)(iv) & (vi) (emphasis added). In this regard, the Commission has emphasized that there is a difference between contentions that allege an "omission" of information and those that challenge substantively and specifically how particular information has been discussed in a license application. *See McGuire/Catawba*, CLI-02-28, 56 NRC at 382-83. Where a contention alleges the omission of particular information or an issue from an application, and the information is later supplied by the applicant, the contention is moot. *Id.*

33 *Consumers Energy Co.* (Palisades Nuclear Plant), CLI-07-18, 65 NRC 399, 416 (2007) ("We have long precluded petitioners from using discovery as a device to uncover additional information supporting the admissibility of contentions.").

IV. THE AMENDED CONTENTION IS INADMISSIBLE UNDER 10 C.F.R. § 2.309

A. Entergy Has Not "Inappropriately Limited" the Number of Component Locations For Which EAF Analyses Must Be Performed

Petitioners allege that the scope of Entergy's new EAF analyses (which they originally contended should be included as part of the April 2007 IPEC LRA) "is incorrect and improperly narrow."[34] They contend that because LRA Tables 4.3-13 and 4.3-14 *originally* contained certain CUF_{en} values greater than 1.0, the Fatigue Monitoring Program—despite the acceptable results of the new EAF analyses—still should have included other locations where high usage factors might be a concern.[35] This argument lacks a legal or regulatory basis and fails to establish a genuine material dispute, as required by 10 C.F.R. § 2.309(f)(1)(v) and (vi).

At its core, the Amended Contention misapprehends the Commission-approved approach by which Entergy addressed EAF in its April 2007 LRA. Entergy addressed EAF effects on those IPEC components corresponding to the six critical locations identified in NUREG/CR-6260 by either projecting the analyses to the end of the period of extended operation, per Section 54.21(c)(1)(ii), or demonstrating that aging effects will be adequately managed, per Section 54.21(c)(1)(iii).[36] The original LRA included a screening evaluation to ascertain whether an AMP was needed. The evaluation applied conservative F_{en} values to the existing CUFs to determine preliminary, bounding projected CUF_{en} values.[37]

Consistent with Commission-approved practice, an applicant may perform a more accurate fatigue analysis by evaluating, for example, the *actual* plant transient cycles rather than

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34 Amended Contention at 8.

35 *Id.* at 5 (*quoting* EPRI, MRP-47, *Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* at 3-4 (Rev. 1, Sept. 2005) (Attach. 6)).

36 *See* LRA at 4.3-20 to 4.3-25, *available at* ADAMS Accession No. ML071210517.

37 *Id.* at 4.3-21.

using assumed *design* cycles.[38] Actual plant transients generally are less severe and occur less frequently than the design transients, which are defined on a generic basis for similar plants for the design of the component. This established practice yields a new CUF value (to which the F_{en} is then applied) that is more accurate (lower in this case) than that of the original design calculation.[39] If a CUF_{en} still exceeds 1.0 *after* more detailed fatigue analysis using actual plant transients, then NRC guidance recommends reviewing additional RCS pressure boundary locations (beyond the NUREG/CR-6260 locations) where high usage factors might be a concern.[40]

The component locations in LRA Tables 4.3-13 and 4.3-14 initially projected to have CUF_{en} values greater than 1.0 included the pressurizer surge line piping for IP2 and IP3, the RCS piping charging system nozzles for IP2 and IP3, and the pressurizer surge line nozzles for IP3.[41] To address these locations, Entergy committed to take one of the following actions: (1) update the fatigue analyses, at least two years before entering the period of extended operation, to determine valid IPEC-specific CUF_{en} values less than 1.0; (2) manage the effects of aging due to fatigue at the affected locations by an NRC-approved inspection program during the period of extended operation; or (3) repair or replace the affected locations before exceeding a CUF of 1.0 during the period of extended operation (collectively referred to as "Commitment 33").[42]

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38 *Vt. Yankee*, CLI-10-17, slip op. at 23 n.99 (citation omitted). *See also* MRP-47 at 3-7 ("Possible reasons for updating the fatigue analysis could include . . . [e]xcess conservatism in original fatigue analysis with respect to modeling, transient definition, transient grouping and/or use of an early edition of the ASME Code.").

39 *See* MRP-47, at 4-4 (stating that techniques for removing excess conservatisms from the input (stress) values of CUF calculations are "generally well understood by engineers performing these assessments throughout the industry"). Calculation of CUF and F_{en} are two separate mathematical processes. *Vt. Yankee*, CLI-10-17, slip op. at 26 n.110.

40 NUREG-1801, Vol. 2 at X M-2; MRP-47 at 3-4 to 3-5.

41 LRA tbls. 4.3.13 & 4.3-14. The preliminary CUF_{en} values for the three NUREG/CR-6260 *reactor vessel* locations were *less than 1.0* for both IP2 and IP3 and, therefore were conservatively projected to the end of the period of extended operation (in the LRA) in accordance with 10 C.F.R. § 54.21(c)(1)(ii). *Id.* at 4.3-22.

42 *Id.* at 4.3-22 to 4.3-23. A 60-year projected CUF greater than 1.0 does not indicate that fatigue cracking necessarily will occur. Rather, it indicates that there is a *potential* for cracking to *initiate* at the affected location at some point during the period of extended operation. This is not necessarily failure of the component. *Id.* at 4.3-22.

Subsequently, in January 2008, as a part of LRA Amendment 2, Entergy amended the Fatigue Monitoring Program to provide information on cycle counting and the methodology used to determine stresses and fatigue usage, including environmental effects, in accordance with the NRC-endorsed ASME Code.[43] Entergy also amended the LRA to place Commitment 33 within the scope of the Fatigue Monitoring Program, by stating that it will use that AMP to manage the effects of reactor water environment on fatigue life, in accordance with Section 54.21(c)(1)(iii).[44] Pursuant to revised Commitment 33, if Entergy does not demonstrate valid CUF_{en} values below 1.0 (Option 1) by conducting more detailed CUF_{en} analyses, then it must implement Option 2 of the commitment. Option 2 requires Entergy to repair or replace the affected locations before their CUF_{en} values exceed 1.0, consistent with the Fatigue Monitoring Program.

Westinghouse completed plant-specific EAF analyses to determine more accurate CUF_{en} values for IPEC in June 2010. These EAF analyses (the results of which are reflected in revised LRA Tables 4.3-13 and 4.3-14), show that the updated CUF_{en} values for *all* of the NUREG/CR-6260 locations at IPEC are less than 1.0 for 60 years.[45] The new CUF_{en} values were calculated specifically for the IPEC locations, and they are more accurate than, and *supersede*, the values contained in the April 2007 LRA.[46]

Petitioners cite no regulation to support their claim that Entergy must "broaden" its EAF analysis beyond the representative components identified in NUREG/CR-6260 and LRA Tables 4.3-13 and 4.3-14. Instead, they argue that such an action is somehow required by the GALL Report and EPRI's MRP-47, presumably because certain (now-superseded) CUF_{en} values in the

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43 *See* NL-08-021, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application Amendment 2," (Jan. 22, 2008) ("LRA Amendment 2") (Attach. 7), *available at* ADAMS Accession No. ML080290659.

44 *Id.*

45 *See* NL-10-082, attach. 1, at 2-4.

46 The Commission has explicitly confirmed that performing new, more accurate EAF analyses for the locations specified in NUREG/CR-6260 locations is one acceptable option under GALL Report Section X.M1. *Vt. Yankee*, CLI-10-17, slip op. at 53 n.236 (*quoting* NUREG-1801, Vol. 2 at X M-1 to X M-2).

April 2007 LRA exceeded 1.0.[47] However, Entergy's Fatigue Monitoring Program is fully consistent with all ten program elements in GALL Report Section X.M1 and the guidance in MRP-47. Consistent with the GALL Report, the new EAF analyses prepared in accordance with Entergy's Fatigue Monitoring Program have demonstrated that the 60-year-projected CUF_{en} values for the critical NUREG/CR-6260 component locations at IPEC are all less than 1.0. Under the Fatigue Monitoring Program, Entergy will monitor the actual IPEC cycles incurred to ensure that they do not exceed the analyzed numbers of cycles, such that the CUF_{en} analyses remain valid throughout the period of extended operation, and take any necessary corrective actions (*e.g.*, repair or replacement) for the affected locations *before* exceeding a CUF_{en} of 1.0.[48] Furthermore, consistent with the GALL Report, MRP-47 states that because the CUF_{en} results for all NUREG/CR-6260 locations are less than or equal to 1.0 for the 60-year operating life, "additional evaluations or locations *need not be considered*."[49] In short, Petitioners' contention that Entergy must rely on *preliminary* (but now superseded) CUF_{en} values, instead of the *new* EAF analyses is not supported by NRC regulations, guidance, or case law.[50]

B. <u>Entergy Has Not Improperly Failed to Perform or Disclose an "Error Analysis"</u>

Petitioners allege that Entergy should have prepared and disclosed an error analysis for its new EAF analyses, especially for CUF_{en} values that are "very close" to 1.0.[51] While Dr. Lahey vaguely posits that there are "many possible sources of error" in the new EAF analyses

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47 *See Vt. Yankee*, CLI-10-17, slip op. at 48 ("But as guidance documents, they cannot impose . . . a requirement.").

48 NUREG-1930, Vol. 2, *Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3*, at 4-44 to 4-45 (Nov. 2009) ("SER").

49 MRP-47 a 3-4 (emphasis added).

50 *See Yankee Atomic Elec. Co.*, LBP-96-2, 43 NRC 61, 90 (1996), *rev'd in part on other grounds & remanded*, CLI-96-7, 43 NRC 235 (1996) (stating that any supporting material provided by a petitioner, including those portions thereof not relied upon, is subject to Board scrutiny, "both for what it does and does not show").

51 Amended Contention at 8, 10, 12, 17. *See also* Lahey Decl., ¶¶ 10-11.

that "*could* lead to a violation" of the 1.0 limit,"[52] he fails to substantiate or quantify the postulated errors or uncertainties. He points to no NRC regulation or ASME Code provision mandating an error analysis, and to Entergy's knowledge, ASME Code-based analyses for both current and proposed nuclear power reactors do not include such an error analysis. Petitioners' argument lacks a legal or factual basis, as required by 10 C.F.R. § 2.309(f)(1)(v).

This basis also fails to establish a genuine material dispute with Entergy, as required by 10 C.F.R. § 2.309(f)(1)(vi). Specifically, Petitioners fail to show that the new EAF analyses violate any NRC or ASME Code requirement; *i.e.*, a new CUF_{en} value exceeds 1.0. In fact, Petitioners do not acknowledge the well-documented, extensive conservatisms associated with the ASME Section III fatigue evaluation procedures and design fatigue curves (from which CUF values are derived). This failure violates their "iron-clad obligation" to examine the relevant, publicly-available documentary material with sufficient care in formulating contentions.[53]

NUREG/CR-6909, a document on which Petitioners and their experts explicitly rely, explains that various entities, including the ASME, NRC, U.S. Department of Energy national laboratories, and EPRI, have evaluated the conservatisms in the relevant ASME Code calculation inputs, analytical methods, and ASME Code fatigue design curve margins.[54] One report issued by Sandia and EPRI, and discussed in NUREG/CR-6909, states:

> After review of numerous Class 1 stress reports, it is apparent that there is a *substantial amount of conservatism* present in many existing component fatigue evaluations. . . . It was concluded that the potential increase in predicted fatigue usage due to *environmental effects* should be more than offset by decreases in predicted fatigue usage if re-

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52 Lahey Decl., ¶ 10 (emphasis added). *See also* Hopenfeld Decl. ¶ 4 (stating that "the CUF_{en} values now cited by Entergy *may* underestimate the detrimental effects of the environment on fatigue strength" and "such numbers *could* be considerably higher than the 1.0 regulatory threshold") (emphasis added).

53 *N. States Power. Co.* (Prairie Island Nuclear Generating Plant, Units 1 and 2), CLI-10-27, slip op. at 18 (Sept. 30, 2010) (citations omitted). *See also Consumers Energy Co.* (Palisades Nuclear Power Plant) CLI-07-18, 65 NRC 399, 414 n.46 (2007) (stating that a petitioner is "obligated to put forward and support contentions when seeking intervention, based on the application and information available").

54 *See* NUREG/CR-6909, Sec. 7 (Margins in ASME Code Fatigue Design Curves).

> analysis were conducted to reduce the conservatisms that are present in existing component fatigue evaluations.[55]

ASME Code Section III, Class 1, component fatigue analyses are conservative in two principal ways. First, the Code rules contain conservatisms that were intentionally included to compensate for a variety of uncertainties. For example, the original Code developers applied a margin of 2 on strain or a margin of 20 on cycles, whichever is greater, to account for variations in materials, surface finish, data scatter, and environmental effects.[56] Second, when applying the Code rules, component designers and analysts use a number of conservative assumptions, which include, or relate to, the use of design transients considerably more severe than those experienced in actual reactor service, grouping of transients, bounding heat transfer and stress analysis, simplified elastic–plastic analysis, material property selection, and Code edition.[57] By overlooking these well-known conservatisms, Petitioners fail to meet their settled "obligation to conduct [their] own due diligence" in formulating new contentions.[58]

Argonne National Laboratory ("Argonne") has performed numerous studies on fatigue of carbon and low-alloy steels and wrought and cast austenitic stainless steels ("SSs") in simulated light water reactor ("LWR") coolant environments. These studies, which also are summarized and referenced in NUREG/CR-6909, include extensive reviews of literature data to evaluate the conservatisms in the existing ASME Code fatigue evaluations. NUREG/CR-6909 concludes that, for all materials, "the current Code requirement of a factor of 20 on cycles, to account for

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55 SAND94-0187, *Evaluation of Conservatisms and Environmental Effects in ASME Code, Section III, Class 1 Fatigue Analysis,* at iii (Aug. 1994) (Attach. 8) (emphasis added).

56 *See* NUREG/CR-6909, at 72, 80-81. *See also* MRP-47 at 3-7, 4-4 (discussing sources of conservatism in existing ASME Code fatigue analyses).

57 SAND94-0187, at 7-1 to 7-2.

58 *Prairie Island,* CLI-10-27, slip op. at 18.

the effects of material variability and data scatter, specimen size, surface finish, and loading history, is conservative by at least a factor of 1.7."[59]

In fact, on this basis, the Board ultimately rejected a similar fatigue-related challenge by an intervenor in the *Vermont Yankee* license renewal proceeding. There, the intervenor and Dr. Hopenfeld argued that Entergy should have validated its CUF_{en} analyses by performing an "error analysis" to show the admissible range for each variable.[60] The Board found that the lack of an error analysis for each variable in the CUF_{en} analyses does *not* render the analyses inadequate.[61] The Board based its finding, in part, on the well-known conservatisms included in the ASME design fatigue curves for carbon steel/SS in the LWR coolant environment that apply equally to IPEC. The Board relied on Staff testimony citing two Argonne studies and noting that the curves "have been adjusted for uncertainties that are associated with material and loading conditions."[62]

In contending that a detailed error analysis is necessary, Petitioners also argue that Entergy has failed to show that results obtained from WESTEMS™ have been "bench-marked against representative experimental data and/or analytical solutions."[63] However, the EAF analyses prepared by Westinghouse explicitly state that WESTEMS™ has been accepted as a tool to perform CUF evaluations by the NRC in the SER for the Shearon Harris Nuclear Power Plant, Unit 1 LRA.[64] That SER, in turn, states that *benchmarking verification results* submitted

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59 NUREG/CR-6909, at 81.

60 *Entergy Vt. Yankee, LLC.* (Vt. Yankee Nuclear Generating Station) LBP-08-25, 68 NRC 763, 804, 814 (2008), *rev'd & remanded on other grounds*, CLI-10-17, slip op. (July 8, 2010).

61 *See id.* at 814.

62 *Id.* The *Vermont Yankee* Board also cited Entergy testimony to the effect that "error analysis" is further unnecessary because refined CUF_{en} analyses apply very conservative to bounding input parameters to maximize stresses. *Id.*

63 Lahey Decl., ¶ 11.

64 WCAP-17199-P, at 1-1; WCAP-17200-P, at 1-1 (*both citing* NUREG-1916, Vol. 2, *Safety Evaluation Report Related to the License Renewal of Shearon Harris Nuclear Power Plant, Unit 1* (Nov. 2008) (Attach. 9)). WESTEMS™ uses the transfer function method ("TFM") to calculate six components of stresses due to time-varying mechanical and thermal loads. The resulting through wall stress components are processed and categorized according ASME Section III, Division 1, Subarticle NB-3200 criteria. The Staff has found this method to be acceptable. *See* NUREG-1916, Vol. 2 at 4-27.

by the applicant and Westinghouse on the Shearon Harris docket show that "stress evaluation by [the WESTEMS™] fatigue analysis software is acceptable."[65] Petitioners fail to challenge or even acknowledge these benchmarking results, which apply to the use of WESTEMS™ at other Westinghouse PWRs, including IP2 and IP3.[66]

C. RPV "In-Core" Structures and Fittings Do Not Require Metal Fatigue Analyses Under Parts 50 and 54 Because They Are Not Part of the RCS Pressure Boundary

Petitioners also argue that the LRA is deficient because it does not present a fatigue evaluation of reactor vessel internals, including, but not limited to, bolting.[67] First, they allege that a fatigue evaluation is necessary for license renewal because "there are many other in-core structures and fittings that will be both highly irradiated (and embrittled) and fatigued-weakened."[68] Second, they allege that Entergy did not evaluate the potential failure of highly fatigued structures and fittings (both internal and external to the RPV) due to design-basis accident ("DBA") loss of coolant accident ("LOCA"), secondary-side LOCA, and anticipated transient without scram ("ATWS") loads.[69] This basis fails to raise a material issue and establish a genuine material dispute, as required by 10 C.F.R. § 2.309(f)(1)(iv), (v), and (vi). It also is

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65 NUREG-1916, Vol. 2 at 4-28. *See* Letter from Thomas J. Natale, Harris Nuclear Plant, to NRC Document Control Desk, "Shearon Harris Nuclear Power Plant, Unit No. 1, Docket No. 50-400/License No. NPF-63, [LRA] Amendment 2: Changes Resulting from Responses to Site audit Questions Regarding Time-Limited Aging Analyses," encl. 3, at 67-93 (Aug. 31, 2007) (Harris Nuclear Plant License Renewal Audit Question and Response Database) (Attach. 10), *available at* ADAMS Accession No. ML072540804 (providing benchmarking verification results for WESTEMS™). *See also* Memorandum from Peter Wen, Sr., ACRS Staff Engineer, to ACRS Members, "Certification of the Minutes of the ACRS Plant License Renewal Subcommittee Meeting Regarding Shearon Harris Nuclear Power Plant on May 7, 2008 – Rockville Maryland," attach. at 9 (July 1, 2008) (Attach. 11), *available at* ADAMS Accession No. ML081830260 (ACRS meeting minutes discussing WESTEMS™ benchmarking results).

66 For example, Westinghouse provided the same benchmarking validation results in support of the LRA for Beaver Valley Power Station, Units 1 and 2, for which the NRC issued renewed operating licenses in November 2009. *See* Westinghouse FENOC-08-109, Letter from K. Blanchard to C. Custer, FENOC, "FirstEnergy Nuclear Operating Company, Beaver Valley Unit 1 and 2, Responses to NRC RAIs Regarding Pressurizer Surge Line Environmental Fatigue" (Rev. 1, June 25, 2008) (enclosed as Enclosure B to Attachment 1, to the Letter from Mark A. Manoleras, FENOC, to NRC Document Control Desk, "Reply to Request for Additional Information for the Review of the Beaver Valley Power Station, Units I and 2, License Renewal Application, and License Renewal Application Amendment No. 15 (July 11, 2008)) (Attach. 12), *available at* ADAMS Accession No. ML081970436.

67 Amended Contention at 9.

68 *Id.* at 9-10.

69 *Id.* at 10.

untimely under the criteria set forth in 10 C.F.R. §§ 2.309(f)(2)(i) and 2.309(c)(1), because Petitioners fail to establish good cause for belatedly presenting an issue that they plainly could have raised at the outset of this proceeding based on Entergy's LRA and Amendment 2.

Petitioners' first argument fundamentally fails under Section 2.309(f)(1)(iv) and (vi) because the in-core structures and fittings simply are *not* part of the reactor coolant pressure boundary.[70] Section 50.55a(c)(1) requires that components that are part of the *reactor coolant pressure boundary* meet the metal-fatigue requirements for Class 1 components in ASME Code, Section III, Subarticle NB-3200, which provides the CUF calculation methodology and specifies a design limit of 1.0.[71] Consistent with Section 50.55a(c)(1), GALL Report Section X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary," directs applicants "to mitigate *fatigue cracking* of metal components of the *reactor coolant pressure boundary* caused by anticipated cyclic strains in the material," considering the effects of the reactor water environment.[72]

Petitioners' second argument is similarly flawed because ASME Code Section III, NB-3200 requires that cyclic loads associated with *normal service conditions* be maintained within the approved design limits. That is, NB-3200 limits fatigue analysis to normal operation and transient conditions (Level A, Level B, and Testing Limits).[73] It does not require consideration of postulated accident and post-accident conditions (Level C and Level D Limits). Petitioners contend that the IPEC fatigue analyses must consider LOCA and ATWS loads. LOCA and

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70 *See* 10 C.F.R. § 50.2 (defining reactor coolant pressure boundary); NUREG-1801, Vol. 2 at IV.A2-1 (describing specific reactor components that are a part of the PWR vessel pressure boundary).

71 *Vt. Yankee*, CLI-10-17. slip op. at 18 (citing 10 C.F.R. § 50.55a(c)(1)). *See* ASME Code, Section III , Rules for Construction of Nuclear Facility Components, Division 1, Subarticle NB-3200 (Design by Analysis) (1998 Edition) (Attach. 13).

72 Reactor vessel internals are addressed under a separate AMP—the Reactor Vessel Internals Program, which manages the effects of aging on reactor vessel internals using the guidance from EPRI Materials Reliability Program ("MRP"). Entergy's aging management of IPEC reactor vessel internals is the subject of a separate admitted contention (NYS-25). On September 15, 2010, NYS filed additional bases in support of NYS-25. Entergy's and the NRC Staff's answers are due the week of October 11, 2010.

73 ASME Code, Section III, Subsections NB-3222.4(b) & NB-3222.4(d) n.9 (stating that peak stress intensity is derived from normal service conditions and defining "normal service conditions" to include Class A, Class B, and Testing Limits).

ATWS loads are not normal service conditions as defined in NB-3222.4 and, therefore, are not included in an NB-3200 fatigue analysis of RCS Class 1 components.[74] Furthermore, because the updated CUF_{en} values of the relevant IPEC components are less than 1.0 (*i.e.*, crack initiation is not predicted), there is no basis for Petitioners' suggestion that that such components are more susceptible to failure when subjected to accident loads.

In summary, Petitioners' assertions that Entergy's EAF analyses must include reactor vessel "in-core" components and consider LOCA and ATWS loads are inconsistent with Part 50 and ASME Code requirements and beyond the scope of this proceeding.[75]

D. Petitioners' Criticisms of Westinghouse's Thermal Hydraulic Analysis Are Unduly Vague, Inadequately Supported, and Insufficient to Establish a Material Dispute

Dr. Lahey accuses Westinghouse of using a "crude analytical approach" that purportedly leads to unspecified errors in the transient thermal stresses and the resultant fatigue analysis.[76] He contends that detailed 3-D computational fluid dynamic ("CFD") models are needed to accurately evaluate the transient developing boundary layers, lest the computed CUF_{en} values will be "too small."[77] Dr. Hopenfeld claims that the EAF evaluations do not specify the heat transfer coefficients used for each component, and that CUF_{en} values will "vary greatly" depending on the heat transfer coefficient.[78] He further alleges that Westinghouse's "highly questionable" methodology leads to unquantified uncertainties in the new CUF_{en} values.[79]

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74 *See* 10 C.F.R. Part 50, App. A (defining LOCA); 10 C.F.R. § 50.62(b) (defining ATWS).

75 *See* 10 C.F.R. § 2.335(a) & § 2.309(f)(1)(iii). *See also Duke Energy Corp.* (McGuire Nuclear Station, Units 1 & 2; Catawba Nuclear Station, Units 1 & 2), LBP-02-4, 55 NRC 49, 77-78 (2002) ("[T]o the extent that [petitioners] challenge NRC regulations relating to the ASME standards, they are inadmissible under 10 C.F.R. § [2.335], and no request for a waiver of the rule has been made, either explicitly or implicitly.").

76 Lahey Decl., ¶ 11(i).

77 *Id.*, ¶ 11(ii).

78 Hopenfeld Decl., ¶ 12.

79 *Id.*

None of these criticisms of Westinghouse's evaluation of thermal-hydraulic conditions at IPEC is sufficiently supported *or* material to warrant admission of the Amended Contention and the consequent expenditure of additional hearing resources. Dr. Lahey merely advocates the use of different modeling techniques or assumptions. But as this Board has aptly noted, seeking "an alternative analysis is, without more, insufficient to support a contention alleging that the original analysis failed to meet applicable requirements."[80] Thus, a proposed contention cannot stand merely on an expert's rejection of the applicant's analyses or representations, in Dr. Lahey's words, as "incomplete, inadequate and unacceptable."[81]

Section 5.3.1.1 of WCAP-17199-P and WCAP-17200-P describes the thermal-hydraulic model used by Westinghouse, the WESTEMS™ PZR Global-to-Local model, which is used along with the transient characteristics developed from the plant-specific data to determine thermal stratification loads and temperature loads, as applicable to the component being analyzed. In questioning Westinghouse's thermal-hydraulic methods, Dr. Lahey reiterates Intervenors' vague and recurring assertion that the EAF analyses are "quite uncertain and this uncertainty must be quantified with a detailed error analysis."[82] As discussed above, an error analysis is not required, necessary, or customarily prepared when performing ASME Code fatigue evaluations for current or new reactors. Moreover, as this Board has specifically noted, more than speculation is required to support the admission of a contention.[83]

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80 *Indian Point*, LBP-08-13, 68 NRC at 187. *See also Duke Energy Corp.* (McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station, Units 1 and 2), LBP-03-17, 58 NRC 221, 238 (2003), *aff'd on other grounds*, CLI-03-17, 58 NRC 419 (2003) (rejecting a petitioner's contention that the applicant had "understated" accident consequences by relying upon "unreasonable and unsupported" assumptions).

81 Lahey Decl., ¶ 5. *See USEC*, CLI-06-10, 63 NRC at 472 (*quoting Private Fuel Storage*, LBP-98-7, 47 NRC at 181) ("[A]n expert opinion that merely states a conclusion (*e.g.*, the application is 'deficient,' 'inadequate,' or 'wrong') without providing a reasoned basis or explanation for that conclusion is inadequate because it deprives the Board of the ability to make the necessary, reflective assessment of the opinion" alleged to provide a basis for the contention).

82 Lahey Decl., ¶ 11.

83 *See Indian Point*, LBP-08-13, 68 NRC at 110 (stating that is improper to "assume[] that, because [an intervenor] cannot check all analysis details, the [applicant's] analysis is incomplete or incorrect. This is speculation . . . [which]

E. Petitioners' Criticisms of the Fen Factors, Dissolved Oxygen Values, and Plant Transient Numbers Used by Westinghouse Lack a Factual Basis

Petitioners' other criticisms of Westinghouse's EAF analyses also lack adequate support and fail to establish a genuine material dispute, as required by 10 C.F.R. § 2.309(f)(1)(v) and (vi). First, they argue that Westinghouse's calculation of F_{en} values is "highly suspect and questionable" because it does not use "appropriate bounding F_{en} values of 12 and 17 for stainless steel and carbon, respectively," as specified in NUREG/CR-6909.[84] The new EAF analyses, however, make clear that Westinghouse scrupulously followed SRP-LR and GALL Report guidance for determining F_{en} factors for license renewal applications.

Specifically, in determining the new CUF_{en} values, Westinghouse used F_{en} factors, calculated as described in NUREG/CR-5704, for the stainless steels in the pressurizer surge line, the reactor coolant piping charging and safety injection system nozzles, and the residual heat removal system Class 1 piping.[85] In addition, Westinghouse applied F_{en} factors, calculated as described in NUREG/CR-6583, for the carbon steel associated with the pressurizer surge nozzle.[86] Westinghouse calculated the F_{en} factors using the detailed inputs described in the applicable NUREG and applied them directly to the updated ASME Code fatigue result.[87] This well-established methodology is acceptable under 10 C.F.R. Part 54.[88]

This, in fact, was the conclusion reached by the Board in *Vermont Yankee* license renewal proceeding. There, Dr. Hopenfeld similarly alleged Entergy used "outdated" statistical equations

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is insufficient to support the admissibility of [a] contention"). *Cf. McGuire/Catawba*, LBP-03-17, 58 NRC at 236 ("The Board here finds that there is no NRC requirement for uncertainty analyses in the situation before us.").

84 Hopenfeld Decl., ¶ 10.

85 *See* WCAP-17199-P, at 5-2 to 5-4; WCAP-17200-P, at 5-2 to 5-4; NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* (Apr. 1999) (Attach. 14).

86 *See* WCAP-17199-P, at 5-4 to 5-5; WCAP-17200-P, at 5-4 to 5-5; NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low–Alloy Steels* (Mar. 1998) (Attach. 15).

87 WCAP-17199-P, at 1-2; WCAP-17200-P, at 1-2.

88 *See* NUREG-1801, Vol. 2 at X M-1.

in NUREG/CR-5704 and -6583, instead of the more recent equations in NUREG/CR-6909, to calculate the F_{en} factors. The Board disagreed, finding that Entergy's use of NUREG/CR-5704 and -6583 in the determination of the CUF_{en} values is sufficient to provide the reasonable assurance required by 10 C.F.R. § 54.29(a) and produces CUF_{en} values "that are more conservative than those produced by the calculation method espoused by Dr. Hopenfeld."[89] Nonetheless, Dr. Hopenfeld again urges the use of worst-case F_{en} values (17 for carbon steel and 12 for stainless steel) purportedly derived from NUREG/CR-6909 without adequate technical justification.[90]

Dr. Hopenfeld also contends that Westinghouse used inappropriately low dissolved oxygen ("DO") values in its EAF analyses, and did not specify DO values used in the calculations of F_{en} for each component during the transients.[91] The Westinghouse EAF analyses state that, for all cases, DO is assumed to be less than 0.05 parts per million ("ppm"); *i.e.*, the bounding default value specified NUREG/CR-5704 and NUREG/CR-6583.[92] Westinghouse explained that this is an appropriate assumption for Westinghouse PWRs, for which DO concentrations are "extremely low" for most operating conditions.[93] Specifically, Entergy maintains a DO content of less than or equal to 0.005 ppm for IPEC power operation, in

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89 *Vt. Yankee*, LBP-08-25, 68 NRC at 805-06.

90 This is somewhat surprising given the *Vermont Yankee* Board's unequivocal rejection of Dr. Hopenfeld's F_{en} argument as "unsound." *Vt. Yankee*, LBP-08-25, 68 NRC at 823 (". . . Dr. Hopenfeld's recalculations predict that the regulatory requirement (*i.e.*, unity) would have been exceeded within 4.63 years after the VYNPS commenced operations, and it is obvious to the Board that this did not occur."). As that Board further noted, the Staff applies NUREG/CR-6909 only to new reactor construction permits or operating license applications. *See id.* at 799 (*citing* Regulatory Guide 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* at 2-3 (Mar. 2007)).

91 Hopenfeld Decl., ¶ 11.

92 *See, e.g.*, WCAP-17199-P, at 5-3, 5-5 & 5-24.

93 *Id.* at 5-24. Dr. Hopenfeld cites an "R&D Status Report" from January/February 1983 issue of the EPRI Journal (*see* Attach. 16), claiming that EPRI data on actual "oxygen concentrations vary by more than an order of magnitude with the change in temperature" during plant tart ups and shut downs. Hopenfeld Decl., ¶ 11. He neglects to mention, however, that the cited report and data relate to *BWR* water chemistry and do not apply to PWRs such as IP2 and IP3.

accordance with plant operating procedures.[94] This value is an order of magnitude lower than the DO concentration that Westinghouse used in calculating F_{en} factors on fatigue usage. Dr. Hopenfeld ignores these critical facts, and merely speculates that Entergy's "apparently flawed" method "strongly suggests" that at least some of the new CUF_{en} values would exceed 1.0.[95]

Finally, in alleging that Entergy made certain unknown assumptions in obtaining the number of transients used in the calculations, Dr. Hopenfeld inexplicably ignores the contents of the Westinghouse EAF analyses. Section 3 (Transient Development) of WCAP-17199-P and WCAP-17200-P lists the transient numbers used for the detailed EAF evaluations of each critical component analyzed by Westinghouse.[96] As discussed in LRA Section 4.3.1, the 60-year cycle projections that Entergy used to develop transient numbers were projected based on actual numbers of cycles accrued to date.[97] Generally, Westinghouse conservatively used CLB cycles that were higher than the numbers of transients projected based on actual plant operation.[98] In a limited number of cases, Westinghouse used the 60-year projected cycles if: (1) the use of 60-year cycles was necessary to reduce extra conservatism, or (2) the 60-year projected cycles were higher than the CLB cycles.[99] Petitioners fail to account for any of this directly-relevant information, again contravening their duty to carefully review the information available to them.

F. <u>The IPEC Fatigue Monitoring Program Complies with Section 54.21(c)(1)(iii)</u>

Finally, Petitioners contend that (1) the *alleged* flaws in Entergy's CUF_{en} reanalysis methodology render its Fatigue Monitoring Program inadequate, and (2) that Entergy lacks a

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94 *See* SER at 4-43.

95 Hopenfeld Decl., ¶ 11.

96 *See* WCAP-17999-P, at 3-1 to 3-22; WCAP-17200-P, at 3-1 to 3-22.

97 *See* LRA at 4.3-2 & 4.3-3.

98 *See* WCAP-17199-P, at 5-1 ("Each location was evaluated with the goal of demonstrating acceptable fatigue usage using the CLB cycles justified for 60-year operation in the License Renewal Application. If this was not feasible, then 60-year projected cycles were used for controlling transients.").

99 *See id.* For example, see WCAP-17999-P, at 3-13 to 3-14, Table 3-7 (Final Transient Set Evaluated for Pressurizer Surge Nozzle) (including note 5).

"tangible, detailed, reliable, and prescriptive" AMP.[100] As a threshold legal matter, the Commission has expressly stated that there is nothing in NRC regulations "to suggest that 'baseline' CUF_{en} calculations are prerequisites to establish the 'parameters' of the AMP."[101]

Petitioners' allegations, in any case, lack adequate factual support. Petitioners erroneously assume that certain CUF_{en} values exceed 1.0. Again, the *preliminary*, bounding CUF_{en} values contained in the April 2007 LRA have been superseded by the *new* CUF_{en} values contained in the revised LRA—and the new CUF_{en} values are all less than 1.0. As described above, Westinghouse performed the EAF analyses fully in accordance with the requirements of the ASME Code, Section III, as required by 10 C.F.R. § 50.55a(c), and in accordance with its NRC-approved Quality Assurance ("QA") Program, as required by 10 C.F.R. Part 50, App. B. Contrary to the conclusory statements of Petitioners' experts, Westinghouse's EAF analyses are not "highly suspect" or "unacceptable."[102]

Furthermore, the IPEC Fatigue Monitoring Program complies fully with NRC regulations and GALL Report recommendations, and provides the level of detail necessary for an AMP.[103] Entergy has committed to managing the effects of fatigue throughout the period of extended operation by monitoring cycles incurred and ensuring they do not exceed the analyzed numbers of cycles, such that the CUF_{en} analyses remain valid.[104] Under this AMP, Entergy will track the numbers of actual plant transients and evaluate those numbers against the design transient

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100 Amended Contention at 6-7.

101 *Vt. Yankee*, CLI-10-17, slip op. at 50 (rejecting Vermont's allegation that the parameters of the AMP monitoring cannot be determined if any CUF_{en} value exceeds 1.0).

102 Hopenfeld Decl., ¶ 10; Lahey Decl., ¶ 5.

103 *See* SER at 3-78 to 3-81 (finding the IPEC Fatigue Monitoring Program to be consistent with the GALL Report AMP).

104 *See* SER at 4-44 to 4-45; NL-08-084, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex" attach. 1, at 3-4 (May 16, 2008) ("NL-08-084") (Attach. 17), *available at* ADAMS Accession No. ML081490317.

numbers.[105] The plant transient counts will be updated at least once each operating cycle, which is an acceptable frequency since the evaluation during each update determines if the number of design transients could be exceeded prior to the next update.[106]

There also is no ambiguity or uncertainty about the timing or scope of repair and replacement activities under the Fatigue Monitoring Program. The program requires that corrective action be implemented *before* the plant exceeds the analyzed number of transient cycles.[107] IPEC procedures contain specific "alert levels" that trigger the initiation of corrective actions under the Fatigue Monitoring Program.[108] Any necessary future analysis updates would be governed by Entergy's QA program, as discussed above.[109] Repair or replacement of a component, *if necessary*, would be done in accordance with established plant procedures that are governed by Entergy's Repair and Replacement and QA programs. These are the same programs that govern current plant operations and have been approved by the NRC under the CLB. As required by 10 C.F.R. § 50.55a, repair and replacement will be in accordance with the applicable requirements of ASME Code Section XI, "Inservice Inspection of Nuclear Power Plant Components," which contains the "concrete and verifiable"[110] details sought by Petitioners (*e.g.*, acceptance standards for examination evaluations, repair procedures, inservice test requirements, and replacements for ASME Class 1 components.).[111] Petitioners' allegations of vagueness thus are entirely unfounded.

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105 NL-08-084, at 4; SER at 4-45

106 NL-08-084, at 4; SER at 3-79, 4-44.

107 NL-08-084, at 4; SER at 4-44 to 4-45.

108 SER at 4-44.

109 NL-08-084, at 4.

110 Amended Contention at 6.

111 *See* NL-08-084, at 4; SER at 3-173 to 3-189 (discussing ASME Code Section XI inservice inspection requirements and their implementation at IPEC during the period of extended operation).

V. CONCLUSION

For the reasons set forth above, the Amended Contention should be rejected as inadmissible under the requirements of 10 C.F.R. § 2.309.

CERTIFICATION OF COUNSEL UNDER 10 C.F.R. § 2.323(b)

I certify that I have made a sincere effort to make myself available to listen and respond to the moving parties, and to resolve the factual and legal issues raised in the motion, and that my efforts to resolve the issues have been unsuccessful.

Respectfully submitted,

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Dated in Washington, D.C.
this 4th day of October 2010

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and
	)	50-286-LR
ENTERGY NUCLEAR OPERATIONS, INC.	)	
	)	
(Indian Point Nuclear Generating Units 2 and 3)	)	
	)	October 4, 2010

CERTIFICATE OF SERVICE

I hereby certify that copies of the "Applicant's Answer to New and Amended Contention New York State 26B and Riverkeeper TC-1B (Metal Fatigue)" and the Supporting Attachments, dated October 4 2010, were served this 4th day of October, 2010 upon the persons listed below, by first class mail and e-mail as shown below.

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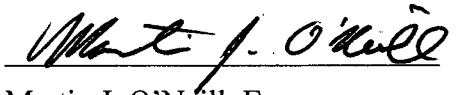

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