NRCR00147
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 ON STATE OF NEW YORK AND RIVERKEEPER'S JOINT CONTENTION NYS-38/RK-TC-5

INTRODUCTION

Pursuant to 10 C.F.R. § 2.1207(a)(2) and the Atomic Safety and Licensing Board's ("Board") orders,[1] the Staff of the U.S. Nuclear Regulatory Commission ("Staff") submits its revised written statement of position and written testimony with supporting affidavits on the State of New York ("NYS") and Riverkeeper, Inc. ("RK") (together, "Intervenors") Joint Contention NYS-38/RK-TC-5. For the reasons set forth in (1) 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); (2) Revised NRC Staff Testimony of Dr. Allen L. Hiser and Mr. Kenneth J. Karwoski Concerning Portions of Joint Contention NYS-38/ RK-TC5 (Ex. NRCR000161); and (3) NRC Staff Testimony of Dr. Allen Hiser, Mr. Jeffrey Poehler, and Mr. Gary Stevens (Ex. NRC000197) ("Hiser/Poehler/Stevens Testimony"), and associated Staff Exhibits, the Staff submits that a careful evaluation of this evidence demonstrates that the contention's challenge to Entergy Nuclear Operations, Inc.'s ("Entergy" or

[1] Licensing Board Scheduling Order (July 1, 2010) (unpublished); Licensing Board Amended Scheduling Order (June 7, 2011) (unpublished); Licensing Board Revised Scheduling Order, at 2 (Dec. 9, 2014) (unpublished); Licensing Board Order (Granting New York's Motion for an Eight-Day Extension of Filing Deadline) (May 27, 2015) (unpublished).

"Applicant") application for renewal of the Indian Point Nuclear Generating Units 2 and 3 ("IP2" and "IP3") operating licenses is without merit.

BACKGROUND

On April 23, 2007, Entergy filed a license renewal application ("LRA"), seeking to renew the operating licenses for IP2 and IP3, for an additional period of 20 years beyond their initial expiration dates of September 28, 2013 and December 12, 2015, for IP2 and IP3, respectively. The Staff reviewed the LRA for compliance with the safety requirements of 10 C.F.R. Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." On August 11, 2009, the Staff issued its "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3" ("SER"), which it published as NUREG-1930, Vols. 1 and 2 (Ex. NYS000326A-F).

On August 30, 2011, the Staff issued Supplement 1 to NUREG-1930.[2] 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.[3]

On July 7, 2015, the NRC staff issued Supplement 2 to NUREG-1930.[4] 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 fatigue monitoring program for reactor vessel internals (RVI)), information required by 10 CFR 54.21(b), updated information

[2] NUREG-1930, Supplement 1 (Ex. NYS000160).

[3] *Id.* at 6-1.

[4] NUREG-1930, Supplement 2 (NYS000507).

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.[5]

Following the issuance of SER Supplement 1, on September 30, 2011, New York and Riverkeeper filed Contention NYS-38/RK-TC-5[6] alleging that:

> Entergy is not in compliance with the requirements of 10 C.F.R. §§ 54.21(a)(3) and (c)(1)(iii) and the requirements of 42 U.S.C. §§ 2133(b) and (d) and 2232(a) because Entergy does not demonstrate that it has a program that will manage the affects [sic] of aging of several critical components or systems and thus NRC does not have a record and a rational basis upon which it can determine whether to grant a renewed license to Entergy as required by the Administrative Procedure Act[.]

Contention at 1 (capitalization omitted). Specifically, Intervenors claim that Entergy's license renewal application is deficient for these reasons:

> Basis 1: Entergy has deferred defining the methods used for determining the most limiting locations for metal fatigue calculations and the selection of those locations;
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> Basis 2: Entergy has not specified the criteria it will use and assumptions upon which it will rely for modifying the WESTEMS computer model for cumulative usage factors (CUF) which have been adjusted for the reactor environment (CUFen)[7] calculations;

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[5] NUREG-1930, Supplement 2, at 6-1 (Ex. NYS000160).

[6] The filing comprises a transmittal letter dated September 30, 2011 (ADAMS Accession No. ML11273A193), along with (1) "State Of New York And Riverkeeper's Joint Motion For Leave To File A New Contention Concerning Entergy's Failure To Demonstrate That It Has All Programs That Are Required To Effectively Manage The Effects Of Aging Of Critical Components Or Systems" ("Motion") (ADAMS Accession No. ML11273A195); (2) "State Of New York And Riverkeeper's New Joint Contention NYS-38/RK-TC-5" ("Contention") with Attachments (ADAMS Accession No. ML11273A196), (3) Declaration of Dr. Richard T. Lahey, Jr. (ADAMS Accession No. ML11273A192), (4) Declaration of Dr. Joram Hopenfeld with Attachment (ADAMS Accession No. ML11273A194), and (5) Certificate of Service (ADAMS Accession No. ML11273A191).

[7] "CUF" is a means of quantifying the fatigue that a metal component experiences during plant operation; "CUFen" is a CUF which has been modified by an environmental adjustment factor (Fen) to account for the environmental conditions experienced by the metal inside the reactor. *Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc.* (Vermont Yankee Nuclear Power Station), CLI-10-17, 72 NRC 1, 30 n.168 (2010). Metal fatigue is further discussed in Section II, *infra*.

Basis 3: Entergy has not adequately defined how it will manage primary water stress corrosion cracking (PWSCC), and

Basis 4: Entergy does not adequately describe the contents of its Aging Management Program (AMP)[8] for reactor vessel internals which is based on a revised version of the Materials Reliability Program 227 (MRP-227) guidance document.

Contention NYS-38/RK-TC-5 at 1-3 (Sep. 30, 2011) (ADAMS Accession No. ML11273A196).

The Staff opposed admission of Contention NYS-38/RK-TC-5 because, *inter alia*, the proffered contention was impermissibly late, was not based upon new, materially different information, and failed to demonstrate a genuine dispute with the application.[9] The Applicant also opposed admission of NYS-38/RK-TC-5, and noted that the contention erroneously alleges that Entergy relies on commitments to define, in the future, the AMPs and activities required by Part 54, when, in fact, Entergy had <u>already</u> defined the requisite AMPs and aging management activities.[10] The State of New York and Riverkeeper filed timely replies.[11]

On November 10, 2011, the Board admitted the contention.[12] On December 6, 2011, the Board granted, in part, Entergy's motion for clarification of its ruling.[13]

On April 23, 2012, the Board confirmed that those aspects of NYS-38/RK-TC-5 that pertain to RVIs (i.e. the fourth basis for NYS-38/RK-TC-5 which concerns the use of a modified

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[8] An AMP is a program intended to manage the effects of aging on a particular component. *Vermont Yankee*, CLI-10-17, 72 NRC at 12 n.44.

[9] *See* NRC Staff's Answer to State of New York and Riverkeeper's Joint Motion to File a New Contention, and New Joint Contention NYS-38/RK-TC-5, at 1 (Oct. 25, 2011) (ADAMS Accession No. ML11298A379).

[10] *See* Applicant's Opposition to New York State's and Riverkeeper's Joint Motion to Admit New Contention NYS-38/RK-TC-5, at 2 (Oct. 25, 2011) (ADAMS Accession No. ML11298A380).

[11] *See* State of New York and Riverkeeper's Joint Reply in Support of Admission of Proposed Contention NYS-38/RK/TC-5 (Nov. 1, 2011) (ADAMS Accession No. ML11305A269) & Declaration of Dr. Richard T. Lahey (Nov. 1, 2011) (ADAMS Accession No. ML11305A266) (responding to Entergy NL-11-107).

[12] Memorandum and Order (Admitting New Contention NYS-38/RK-TC-5) (Nov. 10, 2011), at 10-11 n.47 (unpublished) (ADAMS Accession No. ML11314A211).

[13] *See* Order (Granting Entergy's Motion for Clarification of Licensing Board Memorandum and Order Admitting Contention NYS-38/RK-TC-5), (Dec. 6, 2011) (unpublished) (ADAMS Accession No. ML11340A088).

inspection plan for RVIs, to which Entergy has committed) would be addressed separately and on the same hearing schedule as Contention NYS-25.[14]

On June 19, 2012, NYS submitted the Intervenors' initial evidentiary materials on this contention, comprising (a) Intervenors Initial Statement of Position in support of Contention NYS-38/RK-TC-5 (Ex. NYS000371) (ADAMS Accession No. ML12172A018) ("SOP"), (b) the Pre-filed Testimony of Dr. David J. Duquette (Ex. NYS000372) ("Duquette PFT"), (c) the Report of Dr. David J. Duquette (Ex. NYS000373) ("Duquette Rpt."), (d) the Pre-filed Testimony of Dr. Richard T. Lahey (Ex. NYS000374) ("Lahey PFT"), and (e) Exs. NYS000375 through NYS000397. In addition, RK filed the Pre-filed Testimony of Joram Hopenfeld in Support of RK-TC-5 (Ex. RIV000102) ("Hopenfeld PFT"), along with exhibits RIV000103 to RIV000106.[15]

On July 6, 2012, the NRC Staff filed an unopposed motion for an extension of time to file its initial and rebuttal statement of position, testimony, and exhibits,[16] which the Board granted.[17] Also, on July 6, 2012, Entergy moved to strike portions of Intervenors' filing;[18] on July 16, 2012, the Intervenors opposed Entergy's motion.[19] In its Order issued on August 16, 2012, the Board agreed with New York and Riverkeeper that "NYS-38 is a broad challenge to Entergy's

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[14] Order (Denying NRC Staff's Motion for Partial Reconsideration and State of New York/Riverkeeper's Cross-Motion to NRC Staff's Motion for Reconsideration), (Apr. 23, 2012) (unpublished) (ADAMS Accession No. ML12114A248).

[15] As noted by NYS and Riverkeeper, the procedural background of NYS-38/RK-TC-5 is complex and includes a now-resolved discovery dispute. *See* SOP at 14-24.

[16] *See* NRC Staff's Unopposed Motion for Extension of Time for the Filing of Testimony, Exhibits and Statements of Position on Contention NYS-38/RK-TC-5 (July 6, 2012) (ADAMS Accession No. ML12188A745) (requesting Aug. 20 filing date).

[17] Order (Memorializing Items Discussed During the July 9, 2012, Status Conference), at 2 (July 12, 2012) (unpublished) (ADAMS Accession No. ML12194A538).

[18] Entergy's Motion In Limine To Exclude Portions Of Intervenors' Prefiled Direct Testimony, Expert Report, Statement Of Position, And Exhibits For Contention NYS-38/RK-TC-5 (Safety Commitments) (July 6, 2012) (ADAMS Accession No. ML12188A747).

[19] State Of New York And Riverkeeper's Joint Answer To Entergy's Motion In Limine To Exclude Portions Of Intervenors' Prefiled Direct Testimony, Expert Report, Statement Of Position, And [Exhibits] for Contention NYS-38/RK-TC-5 (July 16, 2012) (ADAMS Accession No. ML12198A548).

Commitments – <u>delineated in the SSER</u> – and thus is not limited [to] specific commitments."[20] Further, the Board cited its ruling of December 6, 2011, in which it clarified that Contention NYS-38/RK-TC-5 is not limited "solely" to Commitment No. 41, stating: "Rather, in finding [NYS-38] admissible, we admitted the Intervenors' 'broad' contention, which relied on 'multiple bases' including the 'claim that there is insufficient information in Entergy's recent commitments' that were addressed in the SSER."[21] In summarizing its ruling, the Board observed as follows:

> [W]e reiterate that NYS-38 is a broad contention, the scope of which is not limited to Entergy Commitments 30, 41, 43, and 44. Instead it broadly encompasses the claim that there is insufficient information in Entergy's commitments as addressed in the SSER.[22]

The Staff's SSER, to which the Board referred, delineated a total of <u>six</u> commitments that had not been delineated in the Staff's original SER (Commitments 41, 42, 43, 44, 45, and 46).[23] The Board's rulings indicate that these are the commitments at issue in Contention NYS-38/ RK-TC-5.[24]

In November 2014, the Staff issued Supplement 2 to NUREG-1930, "Safety Evaluation Report Related to the License Renewal of Indian Point Generating Unit Nos. 2 and 3" (SSER2) (Ex. NYS000507).

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[20] Order (Denying Entergy's Motion in Limine Seeking to Exclude Portions of Intervenors' Direct Evidence Addressing Contention NYS-38/RK-TC-5), at 3 (Aug. 16, 2012) (unpublished) (emphasis added).

[21] *Id.* at 3 (quoting Order (Granting Entergy's Motion for Clarification of Licensing Board Memorandum and Order Admitting Contention NYS-38/RK-TC-5), at 3 (Dec. 6, 2011) (unpublished).

[22] *Id.* at 3-4 (emphasis added).

[23] *Compare* SSER, App, A, at A-22 – A-25, *with* SER, Vol. 2, App. A, at A-25.

[24] The SSER addresses Commitment 41, at 3-19; Commitment 42, at 3-22 – 3-23; Commitment 43, at 4-1 – 4-2; Commitment 44, at 4-2; Commitment 45, at 4-2 – 4-3; and Commitment 46, at 3-15. In addition, the SSER further addresses four commitments (Commitments 15, 25, 30, and 40) that were addressed in the original SER*, i.e.,* Commitment 15, at 3-8; Commitment 25, at 3-14; Commitment 30, at 3-20; and Commitment 40, at 3-6 and 3-8; these four commitments were not raised in this contention.

In February 2015, the Intervenors moved[25] to supplement previously admitted contention NYS-25[26] and joint contention NYS-38/RK-TC-5. The Intervenors argued that Entergy had substantially modified and replaced its previous approach relating to RVIs.[27] Further, they asserted that SSER2 revealed for the first time that NRC Staff would accept Entergy's current proposal to manage the combined and synergistic effects of various aging mechanisms on RVIs through periodic inspections, rather than preventative actions, and that the Staff would accept Entergy's continued reliance on NUREG/CR-5704 and NUREG/CR-6909 (Rev. 0) to calculate CUF_{en} for various components.[28]

The Staff and Entergy opposed Intervenors' proposed the change to NYS-38/RK-TC-5, because, *inter alia*¸ the Intervenors were effectively changing the contention from a challenge to the use of commitments to a challenge to the adequacy of Entergy's AMP for RVI.[29] On March 31, 2015, the Board found that New York and Riverkeeper's proposed supplement and additional bases for Contention NYS-38/RK-TC-5 are admissible.[30] In doing so, the Board noted:

> Although the Board admitted NYS-38/RK-TC-5 as a contention challenging the sufficiency of Entergy's reliance on commitments to develop an aging management program that satisfies NRC regulatory requirements, the contention continues to be admissible as it alleges that Entergy failed to comply with 10 C.F.R. §§

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[25] State of New York and Riverkeeper's Joint Motion for Leave to Supplement Previously-Admitted Joint Contention NYS-38/RK-TC-5 (Feb. 13, 2015) (ML15044A500); New York State February 2015 Supplement To Previously-Admitted Contention NYS-25 (Feb. 13, 2015) (ML15044A507).

[26] NYS-25 alleges that Entergy's LRA "does not include an adequate plan to monitor and manage the effects of aging due to embrittlement of the reactor pressure vessels ("RPVs") and the associated internals." LBP-08-13, 68 NRC 43, 129, 131 (2008).

[27] State of New York and Riverkeeper's Joint Motion for Leave to Supplement Previously-Admitted Joint Contention NYS-38/RK-TC-5 (Feb. 13, 2015) at 5

[28] State of New York and Riverkeeper's Joint Motion for Leave to Supplement Previously-Admitted Joint Contention NYS-38/RK-TC-5 (Feb. 13, 2015) At 5, 6.

[29] NRC Staff's Answer To (1) State Of New York's Motion To Supplement Contention NYS-25, And (2) State Of New York And Riverkeeper Inc.'s Joint Motion To Supplement Contention NYS-38/Rk-Tc-5 (March 10, 2015) (Ml15069a590)

[30] Memorandum and Order (Granting Motions for Leave To File Amendments To Contentions NYS-25 And NYS-38/RK-TC-5) at 15. (March 31, 2015) (unpublished).

> 54.21(a)(3) because it did not "demonstrate that it has a program" to manage aging effects on certain components.[31]

On June 9, 2015, the Intervenors submitted their Revised Statement of Position on Joint Contention NYS-38/RK-TC-5 (NYS000531), asserting that the LRA should be denied because it does not contain (1) sufficient information, (2) adequate programs, and (3) enforceable, binding commitments. Revised Statement of Position on Joint Contention NYS-38/RK-TC-5 (NYS000531) at 1. Further, blurring the line between Contention NYS-26B/TC-1B (metal fatigue) and NYS-38/RK-TC-5, the Intervenors raise the following five deficiencies: (1) Deferred identification of limiting locations; (2) deferred disclosure of environmentally assisted fatigue analyses using WESTEMS; (3) deferred inspections of steam generator divider plate assemblies for PWSCC and development of plans for management of tube-to-tubesheet welds; (4) deferred baffle former bolt inspections; and (5) deferred replacement or other appropriate corrective action for highly embrittled components. Revised Statement of Position on Joint Contention NYS-38/RK-TC-5 (NYS000531) at 3-4.

The Intervenors' revised Statement of Position is supported by: the testimony of Dr. David J. Duquette ("Duquette Revised PFT" (Ex. NYS00532), submitted June 9, 2015, as corrected on June 12, 2015)), the revised testimony of Dr. Richard T. Lahey, Jr. ("Lahey Revised PFT" (Ex. NYS000562)); the revised testimony and supplemental report of Dr. Joram Hopenfeld ("Hopenfeld Revised PFT" (Ex. RIV000143), "Hopenfeld Supplemental Report" (Ex. RIV000144)), and the exhibits cited therein; and the June and November 2012 testimony and report of Dr. Duquette; the June and November 2012 testimony of Dr. Lahey; the June and November 2012 testimony of Dr. Hopenfeld; and all documents referenced in the witnesses revised testimony. Revised Statement of Position on Joint Contention NYS-38/RK-TC-5 (NYS000531) at 5.

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[31] *Id.* at 14 (footnotes omitted).

On June 23, 2015, the Intervenors, Entergy, and the Staff submitted a joint stipulation of issues not in dispute with respect to NYS-38/RK-TC-5.[32] Specifically, with respect to the testimony of the State's expert, Dr. Duquette (NYS000532) on issues related to vibration-induced wear of steam generator tubes, steam generator tube plugging, and steam generator foreign objects, the Intervenors affirm that the testimony is offered solely for the purpose of presenting Dr. Duquette's opinions on the adequacy of Entergy Commitment Nos. 41 and 42 and is not offered as a general challenge to Entergy's Steam Generator Integrity Aging Management Program.[33]

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")[34] 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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[32] State of New York, Riverkeeper, Inc., Nuclear Regulatory Commission Staff, and Entergy Nuclear Operations, Inc., Joint Stipulation Regarding State of New York Pre-Filed Testimony for Contention NYS-38/RK-TC-5 (safety commitments) (June 23, 2015) (ADAMS Accession No. ML15174A081).

[33] *Id.* at 1.

[34] 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:
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> > (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).

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. The Scope of a Contention Is Limited By Its Bases

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)).

Thus, the bases define the scope of a contention, and no testimony may be entertained on any contention, such as NYS-38/RK-TC-5, that strays beyond the bases used to support admission of the contention.

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").[35] *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

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[35] 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[36] 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:

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[36] 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.

> 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>[37]

F. <u>Commission Evaluation of Metal Fatigue</u>

Bases 1 and 2 of NYS-38/RK-TC-5 address the issue of metal fatigue, which is also the subject of Contention NYS-26B/RK-TC-1B. Contention at 1-3. The Commission has described the issue of metal fatigue as follows:

> 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."

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Vermont Yankee Nuclear Power Station), CLI-10-17, 72 NRC 1, 14 (2010) (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)). When the corrosive environment in a reactor is considered, it is reflected by an "Environmental Adjustment Factor" or "F_{en}" which is used to modify the CUF into a "Cumulative Use [or usage] Factor Environmentally Adjusted" or "CUF_{en}". *Id.* An applicant may manage metal fatigue by using an AMP[38] under 10 C.F.R. § 54.21(c)(1)(iii). *Id.* at 19-22. As the Commission stated, "One way to do this [i.e., to satisfy

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[37] *Seabrook*, CLI-12-05, 75 NRC at 304.

[38] Entergy has committed to implement an AMP (the "Fatigue Monitoring Program") during the period of extended operation. LRA Amendment 2 at 15.

10 C.F.R. § 54.21(c)(1)(iii)] is to reference the Metal Fatigue AMP that is approved in the GALL Report." *Id.* at 20 (referring to NUREG-1801, "Generic Aging Lessons Learned (GALL) Report" (Sept. 2005) (Ex. NYS00146A-C) ("GALL Rev. 1")).[39]

Finally, the Commission observed that NRC regulations contain no requirement that an applicant using an AMP to manage metal fatigue must complete its CUF_{en} analyses prior to the issuance of a renewed license. *Id.* at 34. Rather, an applicant is only required to complete such an analysis prior to license renewal if "the analysis is needed to support a demonstration that the tracking AMP will satisfy [NRC] regulatory requirements," for example, to demonstrate that a proposed "AMP is consistent with the GALL Report." *Id.* at 36.

ARGUMENT

II. Entergy's Commitments Are Legally Acceptable For Demonstrating Reasonable Assurance That It Will Adequately Manage Aging During The PEO

In its binding *Vermont Yankee* holding, the Commission stated:

> We reiterate here that a commitment to implement an AMP that the NRC finds is consistent with the GALL Report constitutes one acceptable method for compliance with 10 C.F.R. §54.21(c)(1)(iii).

Vermont Yankee, CLI-10-17, 72 NRC at 36 (emphasis added). Thus, when it comes to the adequacy of an AMP, future implementation of an action which an applicant commits to perform is clearly permissible as part of a "reasonable assurance" finding. The Commission's regulations in 10 C.F.R. § 54.29(a) contemplate a finding that "[a]ctions have been identified and have been or will be taken" with respect to managing the effects of aging. As such, a fair reading of 10 C.F.R. § 54.29(a) leads to the conclusion that an applicant can commit to take an action *in the future*, and thereby satisfy the regulation.

A license renewal application does not need to include detailed results, but instead may meet the requirements of 10 C.F.R. § 54.29 by describing the measures that *will be taken* to

[39] The GALL Report was subsequently revised in December 2010 (Ex. NYS00147A-D)("GALL Rev. 2").

manage aging during the period of extended operations. *Vermont Yankee*, CLI-10-17, 72 NRC at 36.[40] In other words, an applicant's use of a GALL-consistent program is a sufficient demonstration that it will manage the targeted aging effect during the period of extended operations.[41]

Further, the Commission has recognized that 10 C.F.R. § 54.29(a) speaks to both past and *future* actions, referring to actions that "have been or *will be*" taken to manage aging. *Id.* In *Oyster Creek*, CLI-08-23, 68 NRC at 468, the Commission expressly interpreted 10 C.F.R. § 54.21(c)(1) to permit a demonstration *after* the issuance of a renewed license, stating that an 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. *Vermont Yankee*, 72 NRC at 36 (discussing *Oyster Creek*, CLI-08-23, 68 NRC at 468).[42]

In *Vermont Yankee,* the applicant made a "commitment to implement the Fatigue Monitoring Program during the period of extended operation." *Vermont Yankee*, CLI-10-17, 72 NRC at 43 n.236. The Commission then viewed subsequent steps taken by the applicant (e.g. CUFen calculations) as <u>part of the AMP</u>. *Id.* The Commission went on to distinguish the <u>commitment</u> to implement an AMP from the <u>execution</u> of an AMP stating:

> The [latest CUFen] calculations constitute "corrective actions" in the form of "a more rigorous analysis of the component to

[40] Permissible AMPs may be found where the AMP included "state-of-the-art" tests, about which no further details were provided. *Seabrook*, CLI-12-05, 75 NRC at 311.

[41] During its review of an LRA, the Staff validates the claims of consistency made therein. As the Commission stated in *Vermont Yankee*:

> An applicant may commit to implement an AMP that is consistent with the GALL Report and that will adequately manage aging. But such a commitment does not absolve the applicant from demonstrating, prior to issuance of a renewed license, that its AMP is indeed consistent with the GALL Report. We do not simply take the applicant at its word. When an applicant makes such a statement, the Staff will draw its own independent conclusion as to whether the applicant's programs are in fact consistent with the GALL Report.

Vermont Yankee, CLI-10-17, 72 NRC at 37.

[42] "If the applicant uses a different method for managing the effects of aging for particular SSCs at its plant, then the applicant should demonstrate to the Staff reviewers that its program includes the ten elements cited in the GALL Report and will likewise be effective." *Oyster Creek*, CLI-08-23, 68 NRC at 468.

> demonstrate that the design code limit will not be exceeded during the extended period of operation" pursuant to the GALL Report § X.M1, at pp. X M-1 to X M-2.

Id. Thus the commitment to implement an AMP is distinct from the actual implementation of the AMP. *See id.* In *Vermont Yankee* the Commission rejected as "lack[ing] legal support" a demand to essentially execute the corrective actions element of an AMP as a prerequisite to designing the AMP. *See id.* at 40-41, *but see id.* at 36 (stating that an applicant need not do so unless the analysis is needed to support a demonstration that the tracking AMP will satisfy regulatory requirements).[43] The commitment is sufficient, but the Staff will draw its own independent conclusion as to whether the applicant's programs are in fact consistent with the GALL Report. *See id.* at 37.

In sum, the Commission's regulations allow applicants to commit to take future actions as part of their implementation of AMPs. As demonstrated by the cases discussed above, the Commission finds no fault with this practice. The Commission has held that an LRA is not deficient for using commitments to identify future actions. Moreover, as the Commission has held, an LRA does not lack sufficient specificity where the LRA states it will implement a GALL-compliant AMP.

III. The Intervenors' Legal Arguments Do Not Withstand Scrutiny

The Intervenors argue that (1) "Entergy Has Not Carried Its Burden Of Demonstrating That It Will Effectively Manage Aging Degradation Of Certain Components Of The Reactor Coolant Pressure Boundaries At Indian Point Units 2 And 3" (Revised SOP at 45-51) and (2) "Many Of Entergy's Proposed Approaches For Aging Components Are Not Enforceable By The Federal Government, States, Or Citizens And, Therefore, Cannot Support The Required Regulatory And Statutory Findings" (Revised SOP at 51-57). Under the second topic, the

[43] The Staff does not believe that completion of various reanalyses are needed to demonstrate that the AMP is sufficient. 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 at A104-A105 (Ex. NRC000168).

Intervenors further allege (A) "Entergy and NRC Staff Offer Differing Accounts of What Is, and Is Not, Enforceable " (Revised SOP at 51-54) and (B) "The State's Concern is Supported by a 2011 NRC Inspector General Report Finding that NRC Staff Routinely Fails to Monitor Licensee Commitments" (Revised SOP at 55-57). The Intervenors' arguments are without merit.

More specifically, the Intervenors' arguments are unavailing because (a) the LRA demonstrates that the specified components of the reactor coolant pressure boundary are being appropriately managed to address the effects of aging, and (b) the Intervenors' experts do not show any link between the adequacy of an AMP and methods of enforcement.[44] In addition, as discussed in section C.2, *infra*, the Applican't' commitments will be incorporated into the plants' FSARs and licensing bases, and will be subject to regulatory controls governing how changes to those commitments may be made.

A. <u>The LRA Does Not Preclude a Public Review of Safety Questions</u>

The Intervenors assert that information is missing from the LRA, and that the public, the Board, and the State are being excluded from the review of the missing information. Revised SOP at 48 (alleging "extra-hearing resolution of important safety questions"). These claims lack merit. There are no important safety questions that are being left for post-hearing resolution. Indeed, to issue a renewed license, the Commission must make the safety and environmental findings specified in 10 C.F.R. § 54.29. The Staff's SER, as supplemented, and the commitments listed therein, demonstrate the resolution of these safety issues in accordance with established Commission practice -- which includes addressing aging management through commitments to take future actions. *Vermont Yankee*, CLI-10-17, 72 NRC at 36.

The Intervenors argue that the public can add value to identification and resolution of important safety and environmental concerns. Revised SOP at 48. Far from excluding the

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[44] Often, the Intervenors' arguments proceed from a false assumption that their disagreements over enforcement and with the structures of the NRC rules and regulations are litigable issues. *See, e.g.*, Revised SOP at 54 (Intervenors object to use of portion of the Commission's rules addressing exemptions).

public from this role, the Commission's practices and regulations provide ample opportunity for public participation through, for example, the current hearing on the Indian Point LRA. Testimony presented by the Intervenors has not revealed any reason which would preclude the Applicant's use of its commitments to identify actions that will be taken to manage the effects of aging, pursuant to 10 C.F.R. § 54.29(a).

B. The Intervenors Misapply the Requirements of the Atomic Energy Act

The Intervenors argue that LRA falls short of the requirements of the Atomic Energy Act and the Administrative Procedure Act. Revised SOP at 49-51. Specifically, for "the three AMPs identified in this contention" the Intervenors assert there is insufficient information to develop a record of the Staff's determinations about whether the AMPs meets the GALL Report. *Id.* at 50. The Intervenors object to the view that a commitment to develop plans and programs is sufficient. *Id.* at 51. However, as discussed above, in *Vermont Yankee* the Commission explicitly found that a commitment to implement a GALL Report AMP to be sufficient. Furthermore, as documented throughout the Staff's SER, as supplemented, inspection reports, and audit reports, the Staff found sufficient information to draw its own independent conclusion as to whether the Applicant's programs are in fact consistent with the GALL Report, as required under V*ermont Yankee* at 37.

C. The Intervenors' Assertions Regarding Enforceability are Incorrect and Raise Issues that Are Beyond the Scope of the Proceeding

The Intervenors assert that they have "substantial concerns" over whether Entergy's proposals memorialized in its 2011 regulatory communications with NRC and in NRC Staff's August 2011 Supplemental Safety Evaluation Report are enforceable in an NRC administrative enforcement proceeding or in a federal court action. Revised SOP at 51. They argue that the statements made by Entergy or NRC Staff and relied upon by the Board must be enforceable in the future by NRC Staff and by the public through operation of 10 C.F.R. §§ 50.100, 2.206, or other means available under NRC regulations or applicable statutes. Revised SOP at 55. The

Intervenors base their concerns on two items. First, they cite a letter written by an NRC Staff official to an official of the State of Vermont Department of Public Safety ("VT DPS"),[45] which responded to questions from VT DPS concerning commitments. Second, the Intervenors cite an NRC Office of the Inspector General ("OIG") audit report concerning the NRC Staff's overview of licensee commitments.[46] *See* SOP at Revised SOP at 51-54, 55-57. However, as explained below, neither the Vermont Yankee Letter nor the OIG Report supports the Intervenors' position and concerns.

1. <u>Existing Regulations Address Enforceability of the LRA Commitments</u>

Regarding the Intervenors' concern over the reliability and enforceability of statements made by Entergy or the Staff (e.g. Revised SOP at 51-52), it is important to recognize that any information provided by the Applicant is required by regulation to be "complete and accurate in all material respects." 10 C.F.R. § 54.13(a). The provision of materially false information is subject to NRC enforcement action.

As to the Staff, its performance and official conduct is to be accorded a presumption of legitimacy,[47] and is not subject to challenge in an NRC licensing proceeding. Further, 10 C.F.R. § 50.100 provides the Commission the ability to take "for cause" action based on misinformation in the application, or new information, to revoke, suspend, or modify a license; in addition 10 C.F.R. § 2.206 allows the public to request the NRC to initiate such a proceeding "for any other action as may be proper." Thus, the regulations provide appropriate enforcement tools which could be used to address any inappropriate changes to a commitment that may occur after a renewed operating license has been issued. *See* 10 C.F.R. §§ 2.206, 50.100 and 54.13; *see*

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[45] Letter from Christopher G. Miller, Director, Division of Reactor Safety, Region I, NRC, to Sarah Hofmann, Deputy Commissioner, Vermont Department of Public Service (March 20, 2012) (ADAMS Accession No. ML12103A158) (Exhibit NYS000396) ("Vermont Yankee Letter").

[46] OIG Audit of NRC's Management of Licensee Commitments, OAG-A-17 (Sept. 19, 2011), (ADAMS Accession No. ML112620529) (Exhibit NYS000171) ("OIG report").

[47] *All Operating Boiling Water Reactor Licensees with Mark I and Mark II Containments: Order Modifying Licenses With Regard to Reliable Hardened Containment Vents (Effective Immediately)*, LBP-12-14, 76 NRC 1, 8 n.36 (2012) (*quoting United States Dep't of State v. Ray*, 502 U.S. 164, 179 (1991)).

also 10 C.F.R. § 54.37(a) (requiring records to show compliance with 10 C.F.R. Part 54) and 10 C.F.R. § 54.37(b) (requiring inclusion of newly-identified systems into license renewal).

2. A License Condition Will Be Imposed To Address Commitments

Significantly, facts presented by the Intervenors do not address the adequacy of any AMP in the LRA. Rather, the Intervenors argue that "commitments" preceding the Staff's SSER 1 are not binding, enforceable, or tracked, and therefore no reasonable assurance may be found. Revised SOP at 55-57.

In fact, the NRC's regulations and conditions imposed upon renewed licenses already address these concerns. Specifically, pursuant to 10 C.F.R. § 54.21(d), a license renewal application is required to include a supplement to the plant's Final Safety Evaluation Report ("FSAR"). The FSAR supplement for the facility must contain a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses for the period of extended operation. 10 C.F.R. § 54.21(d). The Commission's established practice is to insert a license condition in the renewed license to indicate that the FSAR supplement, as supplemented by the appropriate active commitments made by the applicant and documented in the Staff's safety evaluation report, are henceforth part of the FSAR, are subject to the recordkeeping and reporting requirements in 10 CFR § 50.71(e), and can only be changed through the proper application of 10 C.F.R. § 50.59. In addition, the Commission imposes a license condition into the renewed license specifying that the licensee is to inform the Commission of the completion of commitments which were made during the LRA review.[48]

The Intervenors have not identified any reason to believe that these regulations and regulatory practices do not provide a sound basis for finding reasonable assurance that aging effects will be adequately managed following license renewal.

[48] Written commitments made in docketed licensing correspondence, the NRC's safety evaluation report, and the licensee's FSAR are part of the CLB and subject to enforcement if changed in a manner contrary to regulations.

a. The Vermont Yankee Letter Does Not Support Intervenors' Concerns

The Intervenors state that a letter to the State of Vermont (Ex. NYS000396) explaining "Regulatory Commitments" shows that such commitments might not become legally binding. Revised SOP at 53. But, as discussed in section II.C.2., *supra*, the Intervenors' concerns over enforceability are readily resolved through imposition of a binding license condition. Thus, the Vermont Yankee Letter fails to support their position.

b. The NRC's Regulations Control Changes to Commitments

The Intervenors note that the Vermont Yankee Letter discusses NEI 99-04, "Managing NRC Commitment Changes," July 1999 (ADAMS Accession No. ML003680088), and the uncontested fact that not all changes need prior NRC approval. Revised SOP at 53. The Intervenors further assert that the Staff and the public are unaware of changes that may be made. *Id.*.

The concerns are without merit. As described in section II.C.2, *supra*, the commitments associated with the renewed license become incorporated into the FSAR and are subject to the Commission's established change process described in 10 C.F.R. § 50.59. Thus, whenever a holder of a renewed license seeks to change a commitment that is incorporated in the FSAR, it is required to screen the change against the regulatory requirements of 10 C.F.R. § 50.59 to determine whether it must request NRC prior approval of the change. For some changes, the licensee would need to submit to the NRC a license amendment request ("LAR"), which would result in an opportunity for a hearing. *See* 10 C.F.R. § 50.59(c)(2).

c. The Regulations Control Recordkeeping

The Intervenors express concern that the NRC may not be informed when changes to commitments are made which do not involve a LAR. Revised SOP at 53. However, the Intervenors' concerns are resolved by 10 C.F.R. § 50.59(d)(1)-(3), under which a licensee is required both to maintain records of changes *and* to submit to the NRC, at least every 24 months, reports of the changes made. These are existing rules and requirements which are beyond the scope of NYS-38/RK-TC-5, and the Intervenors' concerns with these processes are impermissible topics for a license renewal proceeding.

d. The Intervenors Impermissibly Attack Commission Regulations

The Intervenors would require that all future changes to "statements relied upon by the Board" must be made via the license amendment process, and they assert that Staff cannot use 10 C.F.R. § 50.12 to grant exemptions from the Commission's regulations. *See* Revised SOP at 54. These assertions constitute an impermissible challenge to the regulations, and are irrelevant to a determination as to the sufficiency of a license renewal application and the topics addressed in NYS-38/RK-TC-5. *See* 10 C.F.R. § 2.335(a) (no rule is subject to attack by argument in an adjudicatory proceeding). In the future, if the Intervenors believe that the licensee has incorrectly changed "statements relied upon by the Board" via an exemption request under 10 C.F.R. § 50.12, or via execution of 10 C.F.R. § 50.59, they may file a petition under 10 C.F.R. § 2.206. *See Yankee Atomic Elec. Co.* (Yankee Nuclear Power Station), CLI-94-3, 39 NRC 95, 101 n.7 (1994) (holding that a "member of the public may challenge an action taken under 10 C.F.R. § 50.59 only by means of a petition under 10 C.F.R. § 2.206"). They may not, however, improperly preclude the licensee from properly applying an exemption or using the Commission's regulation in 10 C.F.R. § 50.59 to evaluate proposed changes to its license.

3. The OIG Audit Report Does Not Support NYS-38/RK-TC-5.

The Intervenors argue that the OIG report shows that the process used by the NRC and its licensees for creating, tracking and handling commitments does not produce the evidence needed for a "reasonable assurance" finding. Revised SOP at 57. This claim is without merit.

For license renewal, the NRC inspects completion of commitments using Inspection Procedure (IP) 71003. See IP71003 at 1 (Ex. ENT000251). For the case of IP2, the Staff performed an on-site inspection regarding the completion of commitments in accordance with TI 2516/001 in 2013. *See* Fatigue Testimony for NYS-26B/RK-TC-1B and NYS-38/RK-TC-5 at A88. In the case of IP3, the Staff will perform an on-site inspection regarding the completion of commitments in accordance with IP71013 in the Fall of 2015. *See* Fatigue Testimony for NYS-26B/RK-TC-1B and NYS-38/RK-TC-5 at A94. This inspection verifies the Applicant's actions implementing its commitments for license renewal and will ensure that selected aging management programs are implemented in accordance with the license renewal regulations. *See* Fatigue Testimony for NYS-26B/RK-TC-1B and NYS-38/RK-TC-5 at A19. Thus, the NRC Staff inspects and audits the commitments made in connection with license renewal. Further, as discussed in section II.C.2, *supra*, the Commission imposes a licensing condition, thereby controlled by the change process established in § 50.59. In short, the OIG report is not relevant to and does not invalidate the Commission's reliance on commitments as part of the license renewal process.

D. <u>Reasonable Assurance Does Not Depend Upon Public Participation</u>

Finally, the Intervenors assert that there can be no "reasonable assurance" if changes to commitments can be made without an opportunity for public participation. Revised SOP at 57. This claim, like others discussed above, is without merit. First, the content of an LRA is specified by the regulations in 10 C.F.R. Part 54, and the content does not change based on public participation. Second, the demonstration the applicant must make to satisfy the standards for issuance of a renewed license are specified in 10 C.F.R. Part 54 and do not change based upon public participation. Third, there *is* public participation by the Intervenors.. Moreover, in the event a renewed license is issued, additional hearing opportunities will be made available for those changes found to require prior NRC approval, under the § 50.59 change process.

Thus, the Commission's regulations address when and how public participation is available. *See e.g.* 10 C.F.R. §§ 2.206, 50.59, and 50.90. An opportunity for a hearing under section 189a of the Atomic Energy Act exists for those actions that are identified in section 189a as licensing proceedings. *Yankee*, CLI-94-3, 39 NRC at 101. With respect to a change to a commitment made unilaterally by the licensee through 10 C.F.R. § 50.59, the Commission holds that a member of the public may challenge such an action taken under only by means of a petition under 10 C.F.R. § 2.206. *Id.* at 101 n.7.

III. Contention NYS-38/RK-TC-5 Lacks Technical Merit

The Staff's expert testimony shows that the concerns raised in NYS-38/RK-TC-5 lack merit and should be resolved in Entergy's favor. In NYS-38/RK-TC-5, the Intervenors raise a multitude of technical issues, including (a) deferred identification of the most limiting locations for metal fatigue, (b) deferred disclosure of CUFen analyses using WESTEMS, (c) deferred inspections of steam generator divider plate assemblies for primary water stress corrosion cracking (PWSCC), (d) deferred baffle former bolt inspections, and (e) deferred replacement or other corrective action for "highly embrittled" components. Revised SOP at 3-4. To address portions of NYS-38/RK-TC-5 related to metal fatigue,[49] the Staff presents 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) ("NRC Fatigue Testimony"). To address the RVI portion of NYS-38/RK-TC-5, the Staff presents the expert testimony of Ex. NRC000197, Hiser/Poehler/Stevens Testimony. Last, to address issues related to steam generators, the Staff presents the revised testimony and certifications of Dr. Allen Hiser and Mr. Kenneth Karwoski (Ex. NRC000161R) ("Hiser/Karwoski Testimony"), along with Staff's Ex. NRC000189 through NRC000194. Previously, the Staff submitted exhibits NRC000158 through NRC000160 in support of its initial statement of position and testimony regarding steam generators; those exhibits are also relied upon by the Staff in its revised testimony and this Statement of Position.

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[49] Concerning NYS-38/RK-TC-5, the Metal Fatigue testimony addresses Basis (1) and Basis (2), which relate to determining the most limiting locations for CUFen calculations, and criteria and assumptions for modifying the WESTEMSTM computer model. Basis (1) is related to Entergy's Commitment No. 43 and Basis (2) is related to Entergy's Commitment No. 44. In addition, the Metal Fatigue Testimony addresses Basis (4), only in part, as it relates to calculation of cumulative usage factors adjusted for environmental effects for various reactor vessel internals. Metal Fatigue Test. at A11.

1. Dr. Allen Hiser, Jr.

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 Fatigue Monitoring Program, 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 Fatigue Monitoring Program, 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.*

5. Mr. Kenneth Karwoski

Mr. Kenneth Karwoski has 24 years of experience at the NRC and another 6 years with the U.S. Navy. Ex. NRC000157; *See* Hiser/Karwoski Testimony at A1b-A2b. His current responsibilities include providing technical advice and guidance to senior NRC management related to steam generator integrity and related issues. *Id.* Mr. Karwoski holds Bachelor of Science degrees in Chemistry and Computer Science from Wayne State University and University of Maryland University College, respectively. A statement of his professional qualifications was previously submitted (Ex. NRC000157). Mr. Karwoski's testimony addresses the NRC Staff's analysis of the Applicant's proposed aging management for the steam generator divider plate assemblies ("SGDPs") and tube-to-tubesheet welds ("TTSW"), and the Staff's views with respect to Contention NYS-38/RK-TC-5. *Id.* at A4b.

6. Mr. Jeffrey Poehler

Mr. Jeffrey Poehler is a Professional Engineer with Bachelor of Science and Master of Science degrees in Material Science and Engineering from Johns Hopkins University. His current responsibilities include serving as the NRC Staff's lead technical reviewer for reactor vessel internals programs, including AMPs and other related issues for reactor vessels and reactor vessel internals.

Mr. Poehler has over 22 years of experience developing and reviewing materials programs for a variety of power plant systems including nuclear and conventional power plant systems. A statement of his professional qualifications is submitted herewith (Ex. NRC000226).

D. Metal Fatigue Technical Background

Contention NYS-38/RK-TC-5 deals with, *inter alia,* metal fatigue.[50] 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).[51] 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 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.

[50] In the NRC Staff’s Statement of Position Regarding NYS 26B/RK-TC-1B (Ex. NRCR20101), which is also supported by 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), the Staff presents greater detail about metal fatigue.

[51] 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.

> 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).

E. Staff Review of Fatigue Monitoring Program and Commitments

The NRC Staff reviewed the IP2 and IP3 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).

With respect to meeting the Staff’s guidance document, the ten program elements of Entergy’s Fatigue Monitoring Program were reviewed by the Staff during a combination of the on-site Aging Management Program audit, the Scoping and Screening audit, in-office reviews, and the IP71002 inspection. NRC Fatigue Test. at A102. On the basis of its review, the Staff found the program to be acceptable. *Id.* Entergy is managing metal fatigue and EAF with its Fatigue Monitoring program that (1) tracks actual plant transients, (2) evaluates these actual transients against design transient definitions to ensure the actual severity is not greater than the design severity, and (3) ensures that the numbers of cycles experienced by the plant remain within the analyzed numbers of cycles in the fatigue evaluations. *Id.* at A105. This ensures that the accumulated fatigue usage, including environmental effects when applicable, will not exceed the ASME Code design limit of 1.0, during both the initial 40-year operating license term and the period of extended operation. *Id.* at A105.

As will be elaborated below, the Staff’s witnesses explain that Entergy does not need to provide the results of, *inter alia*, Commitment Nos. 43 and 44, to demonstrate that the aging effects for metal fatigue and EAF will be managed. NRC Fatigue Test. at A105. The completion of these commitments is not necessary to demonstrate that the aging effects of metal fatigue will be managed. *Id.*

IV. The Bases Statements for Contention NYS-38/RK-TC-5 Lack Merit

A. Basis (1): Deferral of Defining Methods Used To Determine Most-Limiting Locations For Metal Fatigue Calculations And The Selection Of Those Locations

Basis (1) of NYS-38/RK-TC-5 asserts that the LRA is deficient because Entergy has deferred defining the process to be used to determine the most limiting locations for environmentally-assisted metal fatigue calculations (CUF_{en} calculations) and selection of those locations. NRC Fatigue Test. at A10. Basis (1) of Contention NYS-38/RK-TC-5 is related to Entergy's Commitment No. 43, which reads:

> Entergy will review the governing IP ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 locations that were evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting locations for the IP2 and IP3 configurations. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor coolant environment on fatigue usage. Entergy will use the NUREG/CR-6909 methodology in the evaluation of any limiting locations fabricated from nickel alloy material.

NRC Fatigue Test. at A68 (quoting SER Supp. 1 at A-24 (Ex. NYS000160)). The purpose of Commitment No. 43 is for Entergy to confirm that the representative sample of components, i.e., those that were selected for an older vintage Westinghouse plant in NUREG/CR-6260, is sufficient for IP2 and IP3. *Id.* A62.

In connection with Commitment No. 43, Entergy's vendor Westinghouse conducted a screening analysis aimed to determine the most limiting locations for fatigue evaluation purposes, as well as subsequent refined fatigue evaluations. Hopenfeld PFT at 2 (Ex. RIV000143). Dr. Hopenfeld prepared a report which, *inter alia*, assesses the adequacy of the new analyses. *Id.* at 4 (referring to Supplemental Report of Dr. Joram Hopenfeld In Support of Contention NYS-26/RK-TC-1B and Amended Contention NYS-38/RK-TC-5, dated June 8, 2015 ("Hopenfeld Supplemental Report") (Ex. RIV000144)). With respect to Commitment No. 43, Dr.

Hopenfeld asserts that the refined fatigue evaluations (1) failed to consider dissolved oxygen, (2) failed to consider synergistic aging effects of radiation, thermal embrittlement, and stress corrosion cracking, and (3) improperly relied upon CUF values of record without accounting for changes in geometry, surface finish, heat transfer, strain rate, and radiation. *Id.* at 4. With respect to Commitment No. 43, Dr. Hopenfeld also asserts that the analysis, opinions, and conclusions from his earlier written testimonies remain valid and applicable. *Id.*at 5.

1. Entergy Has Not Deferred Defining Commitment No. 43 Methods

As a threshold matter, with respect to the Intervenors' assertion that Entergy deferred defining the methods used to determine the most limiting locations for metal fatigue calculations and selection of those locations as related to Commitment No. 43, Entergy has not deferred defining those methods. NRC Fatigue Test. at A109. To the contrary, Commitment No. 43 has been completed by Entergy for IP2, and, for IP3, Commitment No. 43 will be completed by Entergy before its entrance into the extended period of operation in December 2015. *Id.* The Staff provide extensive discussion of the methodology. E.g., *id.* at A63.

2. The Results Of Commitment No. 43 Are Not Required In the LRA Because Entergy Will Manage Limiting Locations Identified Through Commitment No. 43

Since the locations evaluated in NUREG/CR-6260 represent a generic evaluation, the commitment will cause Entergy to consider the plant-specific configurations and CUF analyses of record at IP2 and IP3 to ensure that Entergy's evaluation covers the locations that are most susceptible to fatigue when considering environmental effects. NRC Fatigue Test. at A66. Entergy's evaluation in Commitment No. 43 will confirm that the representative sample of components that were selected for the older vintage Westinghouse plant in NUREG/CR-6260 are sufficient for IP2 and IP3. *Id.* If Entergy's evaluation identifies additional locations that should be managed, then it will also manage these additional locations with its Fatigue Monitoring Program. *Id.*

Contrary to Dr. Hopenfeld's view, the results related to Commitment No. 43 need not be provided before a licensing decision is reached. *Id.* at 134. The LRA is complete and there is no missing information. *Id.* In any event, the licensee confirmed that license renewal Commitment No. 43 for IP2 is complete, and Commitment No.43 for IP3 will be complete prior to December 12, 2015. *Id.*

3. <u>Sufficiency of Method to Fulfill Commitment No. 43.</u>

With respect to Commitment No. 43, the issues raised by the Intervenors under NYS-38/RK-TC-5 go beyond simply making Commitment No. 43; Intervenors include challenges to the methods used to fulfill Commitment No. 43, and allege various errors in those methods, and the results. The Staff in its NRC Fatigue Testimony fully addresses and disputes the technical issues that the Intervenors' witnesses associated with Commitment No. 43, consistent with the Staff's Statement of Position on NYS-26B/RK-TC-1B. Below the Staff addresses the three issues that Dr. Hopenfeld said were applicable to NYS-38/RK-TC-5 in his most-recent testimony. *See* Hopenfeld PFT at 4. (Ex. RIV000143).

a. <u>Consideration of Dissolved Oxygen</u>

Dr. Hopenfeld makes several assertions concerning treatment of dissolved oxygen, stating in part that Westinghouse made an incorrect assumption. NRC Fatigue Test. at A149. The Staff disagrees with Dr. Hopenfeld's assessment of Entergy's consideration of dissolved oxygen. NRC Fatigue Test. at A149-A152. Instead of simply assuming extreme bounding values for dissolved oxygen, the Staff believes it is appropriate to use real-world measurements. *Id.* at A152. It is true that an analyst could make simplifying, conservative assumptions, but such assumptions are not necessary, but are acceptable. *Id.* at A148, A151. Available plant measurements of DO are acceptable for determining Fen factors for carbon steel, low-alloy steel and stainless steel components, even if these measurements may be taken during steady state operation. *Id.* Also, in the real world, the reactor water chemistry controls will limit DO levels to very low levels at temperatures of significance for metal fatigue. *Id.* at A149.

b. Synergistic Effects

Dr. Hopenfeld states that any analysis of the effects of the LWR environment on fatigue must consider the synergistic effects of radiation, stress corrosion cracking and thermal embrittlement and that a first step towards this end would be to incorporate the effects of radiation into the Fen equation. Hopenfeld Supplemental Report at 15 (Ex. RIV000144).

The NRC and consensus standards such as the ASME Code currently treat the effects of stress corrosion cracking and thermal embrittlement separately from fatigue, and it is intended that the separate evaluation approach for these mechanisms is conservative. NRC Fatigue Test. at A153. Dr. Hopenfeld does not offer any specific research data or evidence to support his contention that treating these mechanisms separately is inadequate, nor does he provide any synergistic models, methods, or evaluations to support his concern. *Id.* Because Entergy's Fatigue Monitoring Program ensures CUF and CUF_{en} 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.* With respect to the effects of radiation on fatigue, the F_{en} methods are considered appropriate for application to materials exposed to significant levels of irradiation, including austenitic stainless steel RVI components, when mandated by regulation or required by the CLB, as discussed in Section 1.3.2 of the draft of NUREG/CR-6909, Rev. 1, *Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials*, (March 2014) (Ex. NYS00490A). In fact, research shows that the fatigue life of irradiated stainless steel was longer than that of un-irradiated stainless steel, thus it is still appropriate to apply the F_{en} method to irradiated components without consideration of an unspecified synergistic effect. *See* NRC Fatigue Test. at A153.

c. CUF of Record

Dr. Hopenfeld states that the methodology employed by Entergy erroneously uses the CUF of record without correcting for changes in the plant that occurred during operation and with the passage of time. Hopenfeld Supplemental Report at 19-20 (Ex. RIV000144). The Staff explain that both at the time of design, and during operation, the ASME Code addresses Dr. Hopenfeld's concern though design specifications and inspection requirements. NRC Fatigue Test. at A158. 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 measured degradation from other effects, such as swelling, pitting, cavitation, or surface topography which would obviate the governing CUF calculations. *Id.* at A158-160.

B. Basis (2): Prescriptive Documentation of WESTEMS User Intervention

Basis (2) of NYS-38/RK-TC-5 is that the LRA is deficient because it has not specified the criteria and assumptions for modifying the WESTEMS computer model for environmentally adjusted CUFen calculations (i.e., ASME Code Section III design stress and fatigue analyses). NRC Fatigue Test. at A10. Basis (2) concerns Entergy's Commitment No. 44, which is: "Entergy will include a written explanation and justification of any user intervention in future evaluations using the WESTEMSTM 'Design CUF' module." NRC Fatigue Test. at A63 (quoting SER Supp. 1 at A-25 (Ex. NYS000160)). The concern over "user intervention" was documented by the Staff in NRC Regulatory Issue Summary (RIS)-2011-14, *Metal Fatigue Analysis Performed By Computer Software*, (December 2011) (Ex. NRC000112) ("RIS 2011-14"). NRC Fatigue Test. at A231-A232.

Concerning Dr. Hopenfeld's support of this claim, as related to Commitment No. 44, he is also not clear about the "criteria and assumptions" that he is referencing because the use of assumptions and engineering judgment is inherent in any fatigue analysis regardless if it is performed with or without computer software. *Id.* at 140. The Staff's concern with "user intervention" was associated only with sufficient documentation of modifications of stress peaks and valleys by properly trained analysts using the WESTEMS™ software, and not with the engineering judgment exercised by the analyst or the results of the analyses. *Id.*

Similarly, Dr. Lahey's characterization of "user intervention" is not consistent with the concerns identified with the use of WESTEMS™. *Id*. at A230. Dr. Lahey never specifies the effects that the "user intervention," as described in the RIS, has, if any, on the results from these fatigue calculations. *Id.*

Relative to Commitment No. 44, the issue of concern is documentation of, selection of, and modification of, stress peaks and valleys. *Id.* at A233. Such documentation should be sufficient so a person technically qualified in the subject area can review, understand, and verify the adequacy of the results without needing to contact the analyst who was running the software and making the documentation. *Id.* Entergy's documentation for any analysis, not just fatigue analyses, must be performed in accordance with their Quality Assurance program that is required in accordance with Appendix B to 10 CFR Part 50. *Id.*

C. Basis (3): The Potential For PWSCC in the Steam Generator Divider Plate and Tube-to-Tubesheet Welds

1. Indian Point's Approach for Verifying the Adequacy of the Water-Chemistry Program for Managing PWSCC in the Steam Generator Divider Plate and Tube-to-Tubesheet Welds Does not Frustrate the Rights of New York, Riverkeeper, or the Pubic

The Intervenors state that NYS-38's central concern is that the Staff "allow[s] certain safety issues to be resolved … *after* the NRC has issued an operating license and *outside* the framework of an Atomic Energy Act § 189 proceeding." Revised SOP at 1-2 (emphasis in original). The Intervenors' concern is misplaced, as the Applicant has proposed an adequate aging management program in combination with a one-time verification test that meets the statutory and regulatory requirements of license renewal. In sum, the Applicant's program identifies "the actions that have been or will be taken" with respect to "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)." 10 C.F.R. § 54.29(a)(1).

Turning to Intervenors' specific concerns regarding PWSCC in the steam generator's divider plates and in the tube-to-tubesheet welds, the contention states:

> Entergy has acknowledged a problem with primary water stress corrosion cracking ("PWSCC") for the nickel alloy or nickel-alloy clad steam generator ("SG") divider plates exposed to reactor coolant, a problem which could impact components directly relevant to plant safety. SSER at 3-18 to 3-19. Entergy originally proposed, and the SER approved, the water chemistry program for managing that problem, but it now concedes that it is not sufficiently effective to meet the aging management program objectives and requirements. *Id.* Entergy has not yet determined how to address that problem; rather, it intends to rely on the EPRI Steam Generator Management Program ("SGMP") Engineering and Regulatory Technical Advisory Group report, which is not expected to be available for a number of years and, in the meantime, to institute an unspecified inspection program to ascertain, after commencement of the license renewal period, whether stress corrosion cracking is actually occurring in the divider plates of the steam generators. *Id.* Entergy and NRC Staff have also acknowledged a concern with the steam generator tubesheet cladding and the propagation of primary water stress corrosion cracking to the tube-to-tubesheet welds. SSER at 3-20 to 3-23. Entergy proposes to "develop a plan" to address this issue but the plan lacks detail and will not be developed until well into the period of extended operations. SSER at 3-22 to 3-23.

Revised SOP at 3-4. The Intervenors fault Entergy for seeking to rely upon EPRI's future development of a steam generator management program and to thereby gain the benefit of that program, while, in the meantime, conducting inspections to verify that PWSCC is not occurring in the SGDPs. In doing so, the Intervenors overlook or dismiss the actions Entergy has proposed to manage PWSCC in the steam generator divider plates and tube-to-tubesheet welds, simply because Entergy also seeks to gain the benefit of continuing industry research on new or better ways to accomplish the same goals in the future.

As Dr. Hiser and Mr. Karwoski explain, Entergy has proposed a combination of specific programs and actions, including the existing water chemistry program and one-time inspections that are well-defined as to the timing and acceptance criteria. Hiser/Karwoski Testimony at A25-A26, A56-A59, A92-A96. With respect to the one-time inspections of the steam generator divider plate and tube-to-tubesheet welds, the precise method of testing and the resulting corrective actions (if any are required) are left for the Applicant to determine at the time of the inspection. Neither of these matters need to be identified prior to the inspection, inasmuch as the inspection and follow-up corrective actions must satisfy the applicable acceptance criteria. Accordingly, leaving those items to be determined at the time of the inspection does not preclude the Commission from reaching a conclusion as to the adequacy of Entergy's AMP and does not prevent the Intervenors from fully participating in these proceedings.

a. Entergy's Proposed Actions for PWSCC in the Steam Generator Divider Plate and Tube-to-Tubesheet Welds Are Well and Appropriately Defined

The Intervenors argue that "Entergy's proposed plan for the steam generator divider plates assemblies, tubesheets, and welds contain several unknowns." Revised SOP at 15. Specifically, the Intervenors seem to want more details on the "inspection methods or technique …, acceptance criteria, monitoring and trending protocols, or corrective actions responses." *Id.* The Intervenors argue that Entergy's plan is "wholly deficient" and that it is a mere ruse until a plan is developed by EPRI. *Id.* at 16.

First, the Intervenors raise concerns that the monitoring and trending protocols are deficient. *Id.* at 16-17. This concern is unfounded, in that it appears to arise from a misunderstanding of the acceptance criteria, discussed below, and the schedule for examining the steam generator divider plates and tube-to-tubesheet welds. The current program proposed by Entergy is the existing water chemistry program applied in conjunction with a one-time inspection of the steam generator divider plate and tube-to-tubesheet welds. Hiser/Karwoski Testimony at A25-A26, A98, A104. The proposed one-time inspection program does not have a monitoring or trending protocol because the proposed inspection will only occur once. *See Id.* at A95-A96. As such, monitoring or trending need not occur. Rather, the one-time inspection in combination with the very conservative acceptance criteria provides a clear point where Entergy would be required to take additional actions to repair, replace, or further justify the suitability of the steam generator divider plates and tube-to-tubesheet welds. *Id.* at A17-A20.

Second, the Intervenors argue that Entergy's acceptance criteria are not sufficiently well-defined. Revised SOP at 15-16, 21. As Dr. Hiser and Mr. Karwoski explain, the acceptance criteria for the one-time inspections are simple, clear, and easily identified. *See* Hiser/Karwoski Testimony at A25-A26, A56-A59, A92-A96. For example, the acceptance criterion is that there be "no detectable" PWSCC in either the steam generator divider plates or the tube-to-tubesheet welds. *Id.* at A25-A26, A56-A59, A92-A96. Even though the Intervenors seem to want Indian Point's program to identify length, depth, and orientation, the actual criterion (no detectable PWSCC) is clearer and more absolute than any criteria that could have been used to define the acceptable combinations of crack depth, location, and length. *See id.* The "no detectable" PWSSC criterion is easily translated into the Intervenors' paradigm, as PWSCC cracks may not exceed lengths or depths of "0 inches" and may not occur on either of the components. *See id.*

Third, the Intervenors assert that Indian Point has not identified the inspection methods or techniques that will be used to detect cracking. Revised SOP at 15-16. But, the Intervenors recognize that Entergy will be performing an inspection for cracking that includes "visual, volumetric, or surface" examinations. *Id.* As the Staff's experts explain, the selection of the appropriate examination will depend on a number of factors including the conditions found in the steam generators, the as-found geometry, and worker safety, among others. *See* Hiser/Karwoski Testimony at A77-A78. As the Staff's experts state, selecting the precise test now to be used in a number of years in the future would preclude applicants from making use of the best available technology at the time of the inspection. *Id.* Further, the Applicant's choice of testing techniques is not unlimited. *Id.* The technique(s) chosen by the Applicant must be able to detect PWSCC, which certain current inspection techniques have been shown to be acceptable as evidenced by the French experience in which cracks were identified in some SGDPs. *Id*.

Fourth, the Intervenors raise concerns that Entergy's actions do not specifically define the appropriate corrective actions. Revised SOP at 15. Here, again, it would be improper to prescribe the appropriate corrective actions, since corrective measures will necessarily depend on the results of the inspection and other conditions in the steam generator, which might impact any proposed action. *See* Hiser/Karwoski Testimony at A57-A59, A92-A96. The Staff's experts explain that "appropriate corrective action[s] will depend on the actual inspection findings, but could include more detailed evaluation of the inspection results, repair of the affected component(s), or replacement of the affected component(s)." *Id.* at A58.

b. The Inspections of the Steam Generator Divider Plates and Tube-to-Tubesheet Welds are Appropriately Timed Based on the Operating History at Indian Point

Dr. Lahey complains that "the anticipated resolution of these issues [PWSCC in the SGDP and TTSW] appear to be beyond the time frame for submission of testimony and the evidentiary hearings … and thus will not allow for a testing of the adequacy of the proposed resolution of these issues in this proceeding." Revised SOP at 8. This concern is premised on two fundamental misunderstandings related to the license renewal process and the existing conditions at the two Indian Point plants.

First, Dr. Lahey and Intervenors' other experts want to move up all of the inspections regarding PWSCC and any potential responses to the inspections to a time prior to the hearing on Indian Point's proposal. *Id.* at 2, 8. This conflates the adequacy of Indian Point's eventual performance in implementing the application's proposals with a pre-license renewal determination of the adequacy of the program to manage the effects of aging such that the intended functions will be maintained. *See* 10 C.F.R. § 54.29(a)(1).

Second, moving the scheduled inspections to a time prior to the hearing date would only serve to diminish their usefulness. The Staff's experts clarify that the steam generators at IP2 and IP3 have already been replaced; they are not the originally-installed steam generators. *See* Hiser/Karwoski Testimony at A52-A54, A71. IP2's steam generators were replaced in 2000 and IP3's steam generators were replaced in 1989. *Id.* at A53. The Staff's experts explain that the steam generators need to operate over a period of time in order for PWSCC to potentially initiate and become detectable to available methods. *Id.* at A52-A54, A71. As such, inspecting the steam generator divider plate and tube-to-tubesheet welds prior to the period of extended operations may not be conclusive with respect to whether PWSCC is occurring and may not serve to confirm the effectiveness of the water chemistry program. *See Id.* at A54, A71.

c. Entergy's License Renewal Commitments <u>Are Enforceable</u>

The Intervenors assert that Entergy's license renewal commitments regarding its AMPs are not enforceable, and that Entergy will be able to change these commitments at will without limitation. Revised SOP at 51-57. The Intervenors base this assertion on their interpretation of an NRC letter sent to the State of Vermont regarding Vermont Yankee's change of commitments prior to the renewed license being issued. Revised SOP at 51-52. The Intervenors' arguments are without basis. The letter sent to the State of Vermont clearly indicated that commitments can become requirements through a number of different actions. Ex. NYS000396, Enclosure 1 at 1-2. Most pertinent to this license renewal proceeding, is the fact that license renewal commitments are made into mandatory requirements through the use of a license condition. *See* Hiser/Karwoski Testimony at A153-A158; Ex. NYS00326A, SER at 1-22. IP2 and IP3 will have license conditions imposed as part any renewed license that would require Entergy's license renewal commitments to be placed into the licensee's FSAR. *See* Hiser/Karwoski Testimony at A153-A158; Ex. NYS00326A, SER at 1-22. Although the precise wording of this license condition does vary from licensee-to-licensee based on the particular commitments and licensee, the license condition generally contains wording similar to the following (Callaway) license condition:

> (17) <u>License Renewal License Conditions</u>
>
> (a) The information in the Final Safety Analysis Report (FSAR) supplement, submitted pursuant to 10 CFR 54.21(d), is henceforth part of the FSAR which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs and activities described in the FSAR supplement, without prior Commission approval, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

> (b) The licensee's FSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and as revised in accordance with license condition 2.C.(17)(a), describes certain programs to be implemented and activities to be completed prior to the period of extended operation.
>
> 1. [Licensee] shall implement those new programs and enhancements to existing programs no later than April 18, 2024 [6 months prior to the commencement of the PEO].
>
> 2. [Licensee] shall complete those designated inspection and testing activities, as noted in Appendix A of the [Safety Evaluation Report for License Renewal], no later than April 18, 2024 [6 months prior to the commencement of the PEO]., or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.
>
> 3. [Licensee] shall notify the NRC in writing within 30 days after having accomplished item (b)1 above and include the status of those activities that have been or remain to be completed in item (b)2 above.

See Hiser/Karwoski Testimony at A156.

This license condition imposed on Callaway or a similar one will be imposed on IP2 and IP3. *See* Hiser/Karwoski Testimony at A156-A157; Ex. NYS00326A, SER at 1-22. As such, IP2 and IP3 will have an enforceable requirement placed on their license renewal commitments, including the commitments related to the steam generator divider plate inspections and the fatigue monitoring program, among many others. *See* Hiser/Karwoski Testimony at A153-A158. The condition will require that Entergy put these license renewal commitments (including inspection schedules and AMPs) into its FSAR for IP2 and IP3. *Id.* Once these commitments are placed into the FSAR for each plant, they become an enforceable part of the licensing basis through NRC inspections or public requests under 10 C.F.R. § 2.206. *See* Hiser/Karwoski Testimony at A157.

The license condition requires that the programs and inspections are implemented and completed by certain dates. *See* Hiser/Karwoski Testimony at A156-A157. Changes to the FSAR are controlled under 10 C.F.R. § 50.59. *See* Hiser/Karwoski Testimony at A157.

Although some changes to the FSAR are allowed to be made by the licensee without NRC approval, the documentation supporting those changes must be kept for audit and inspections. *See* Hiser/Karwoski Testimony at A157. Other changes under 10 C.F.R. 50.59, as explained in that regulation, require NRC approval through a license amendment submitted under 10 C.F.R. 50.90. *See* Hiser/Karwoski Testimony at A156. Regardless of whether the licensee makes FSAR changes without NRC approval (using the 10 C.F.R. 50.59 process) or seeks NRC approval (under 10 C.F.R. 50.90), the changes are controlled under the Commission's regulations. As such, New York's concerns about the transparence or enforceability of the IP2 and the IP3 commitments is not material to whether the Applicant's aging management programs are adequate.

D. Basis (4) Deferred Inspections, and Replacements for the RVI AMP Deferred Baffle Former Bolt Inspection

The Intervenors state that NYS-38's central concern is that the Staff "allow[s] certain safety issues to be resolved … *after* the NRC has issued an operating license and *outside* the framework of an Atomic Energy Act § 189 proceeding." Revised SOP at 1-2 (emphasis in original). With respect to the RVI AMP, New York raises two concerns: (1) deferral of the baffle-former bolt inspections and (2) deferral of replacement or other corrective action for other "highly embrittled" components. Revised SOP at 4. As discussed in the NRC Staff's testimony and Statement of Position on Contention NYS-25 (concerning the RVI AMP), the Intervenors' concern is misplaced, inasmuch as (a) the Applicant has proposed to conduct the baffle-former bolts inspection in a timely manner with an appropriate inspection technique and acceptance criteria, and (b) other components are being appropriately monitored by the RVI AMP through regularly scheduled inspections that are likely to identify aging such that appropriate corrective actions can be taken to analyze, repair, replace components prior to the loss of intended functions. As such, the IP2 and IP3 RVI AMP is capable of "managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified … under §54.21(a)(1) or (c)." 10 C.F.R. § 54.29(a)(1) and (2).

1. Indian Point's Approach for Inspecting the Baffle Former Bolts Is Timely and Well-Defined by the RVI AMP

Turning to the Intervenors' specific concern regarding the baffle-former bolts, the contention states:

> Entergy and NRC acknowledge that cracking of the baffle former bolts is a potential problem aging management issue, yet Entergy has proposed to delay inspections of these bolts until well into the period of extended operations. Moreover, Entergy has yet to develop acceptance criteria for these components.

Revised SOP at 4. Again, the Intervenors' concerns appear to be misplaced. The inspections of the baffle-former bolts were established based on the operating experience for IP2 and IP3. As Dr. Hiser and Mr. Poehler explain, the baffle-former bolts initial inspections under the RVI AMP are scheduled to be conducted upon reaching 25 to 35 effective full power years ("EFPY"). Ex. NRC000197, Hiser/Poeheler/Stevens Testimony at Q80. Thus, these inspections are scheduled to occur between 15 to 5 years prior to completing 40 years of equivalent full power operations. *Id.* This schedule takes into account the operating history of the plants including extended shut-down periods, to ensure that sufficient operation would have occurred to make the resulting inspections useful in evaluating the aging effects on the baffle-former bolts *Id.* Once the initial inspection is completed, the baffle-former bolts would then be inspected every 10 years. *Id.*

The Intervenors assert that acceptance criteria have not been developed for the baffle-former bolts. Revised SOP at 4. In this aspect, the Intervenors are mistaken. The baffle-former bolts have explicit and conservative acceptance criteria. Ex. NRC000197, Hiser/Poehler/Stevens Testimony at Q277-Q280. Baffle-former bolts would fail to meet their acceptance criteria if any flaw is detected. *Id.* The Staff's guidance and Entergy proposed program assume, for the purpose of acceptance criteria, that bolts with any detectable flaws are non-functional. *Id*. This finding applies to an individual bolt and not necessarily to the remaining bolts. *Id*. If a flaw is discovered, the applicant would then have to determine what additional testing is required, based on the actual detection of a flaw in a baffle–former bolt. *Id*.

At IP2 and IP3, there are approximately 1000-plus baffle-former bolts installed in each plant. *Id.* at Q222, Q243, Q319. The industry and the Staff have examined the impact of baffle-former bolt failures on reactor safety and performance. *Id*. The research on baffle-former bolt failures has demonstrated that approximately 1.5% of baffle-former bolts fail. *Id*. In order to maintain the intended function, only about 20-30% of the baffle-former bolts need to remain intact. *Id*. The existence of numerous installed baffle-former bolts demonstrates a high degree of redundancy, such that a widespread replacement of all baffle-former bolts is not required merely because an aging effect may be occurring that does not challenge the safe operation of the plant during normal or accident conditions. *Id*.

2. Indian Point's Approach for Inspecting Other RVI Components Will Timely Identify and Replace <u>Components Prior to the Loss of Intended Function</u>

Turning to the Intervenors' concern regarding deferred replacement or other appropriate corrective actions, the contention states:

> Reactor vessel internals typically experience, by the end of life, neutron irradiation at levels well beyond those known to cause significant reduction in ductility. This loss of fracture toughness may not be detectable using currently available non-destructive inspection techniques. Since Entergy's aging management plan for reactor vessel internals is an inspection-based plan, highly embrittled components subject to brittle fracture may not be identified for timely replacement or corrective action until well after the period of extended operations.

Revised SOP at 4. In raising these issues, the Intervenors appear to assert that essentially all components managed by the RVI AMP should be preemptively replaced regardless of whether the component is showing any aging effect or indicates that it may not be able to perform its intended function until the next scheduled inspection. The Intervenors' concerns are without merit.

Dr. Lahey argues that the clevis insert bolts should be pre-emptively replaced consistent with Entergy's plan to replace the split pins. EX. NYS000562, Lahey Testimony at p. 59 ln. 5 – p. 60 ln. 3. This claim is without merit. First, the split pins are being replaced because there is operating history with the split pins failing, and the process for inspecting the split pins requires that they be removed from the reactor vessel. Ex. NRC000197, Hiser/Poeheler/Stevens Testimony at Q155-Q156, Q240-Q242. Once removed, even if no aging or degradation was detected, the split pins may be re-inserted and re-used or, instead, replaced with materials that do not experience the same aging effects. *Id.* In its RVI AMP, the Applicant conservatively determined that replacing the split pins would be more effective than having to inspect the component repeatedly using a technique that requires its removal from the reactor vessel. *Id.*

Another example of components that are not being preemptively replaced is the lower core support columns (column caps).[52] *Id.* at Q161-164, Q172, Q211-Q213, Q220, Q317. Entergy addressed thermal and irradiation embrittlement as one of the action items that was specifically identified in the Staff's safety evaluation concerning the topical report for MRP-227-A. *Id.* Action item 7 requires an applicant or licensee to perform a plant-specific analysis of cast austenitic stainless steel ("CASS") RVI components to demonstrate the components will remain capable of performing their intended functions through the period of extended operation. *Id.* The analyses must account for the potential loss of fracture toughness of the components due to both thermal embrittlement and irradiation embrittlement, as applicable. *Id*. The requirement to perform the functionality analysis is focused on components that require aging management according to the guidance of MRP-227-A. *Id*. (MRP-227, Rev. 0 SE at pp. 34 Exhibit ENT000230).

[52] The lower core support columns are at times referred to as lower support columns and core support columns. The column caps consist of the top portion of the lower support column. The top portion of the lower core support column receives more fluence than other portions of the lower support column.

The Staff determined during its review of the IP2/IP3 LRA that: (1) Entergy had evaluated the risk-significant CASS components of the RVI (i.e., the lower core support columns (column caps)); (2) Entergy screened the column caps for thermal embrittlement and irradiation embrittlement using plant-specific materials data and determined that the column caps are not susceptible to thermal embrittlement[53]; (3) Entergy provided information on fabrication non-destructive examinations ("NDE"), demonstrating that pre-existing flaws are unlikely to exist in the column caps; (4) Entergy provided information on the expected stresses and neutron fluence for the column caps that demonstrated that service-induced cracking due to irradiation-assisted stress corrosion cracking ("IASCC") is unlikely; and (5) Entergy modified its RVI Inspection Program to include a link to a lead component that is an appropriate predictor of IASCC and irradiation embrittlement for the column caps, with an appropriate schedule for performing the "Expansion" inspection if necessary. *Id*. Therefore, the Staff found that the information provided by Entergy provides reasonable assurance that the functionality of the lower core support columns (column caps) will be maintained during the period of extended operation for IP2 and IP3. (*Id*.; SSER2 at 3-47, Exhibit NYS000507).

In sum, in accordance with the RVI AMP, Entergy will conduct appropriate inspections in a timely manner to identify potential degradation, including thermal embrittlement and irradiation embrittlement, such that appropriate corrective actions will be taken prior to loss of intended function for these components.

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[53] The staff independently confirmed this screening using its own screening criteria.

CONCLUSION

For the reasons stated above, Intervenors' challenge to Entergy's LRA, as presented in NYS-38/RK-TC-5, cannot be sustained, and the issue should be resolved in Entergy's favor.

Respectfully Submitted,

/Signed (electronically) by/

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