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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

APPLICANT'S ANSWER TO AMENDED CONTENTION NEW YORK STATE 25 CONCERNING AGING MANAGEMENT OF EMBRITTLEMENT OF REACTOR PRESSURE VESSEL INTERNALS

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I. INTRODUCTION

Pursuant to 10 C.F.R. § 2.309(h)(1) and the Atomic Safety and Licensing Board's ("Board') July 1, 2010, Scheduling Order, Entergy Nuclear Operations, Inc. ("Entergy") submits this Answer to the Amended Contention filed by New York State ("NYS") on September 15, 2010.[1] This proceeding concerns Entergy's license renewal application ("LRA") for Indian Point Units 2 and 3 ("IP2" and "IP3"), also known as the Indian Point Energy Center ("IPEC").

NYS seeks to amend admitted contention NYS-25, which challenges the manner in which IPEC will manage the effects of aging due to embrittlement of the reactor pressure vessels ("RPV") and RPV internals during the period of extended operation. As set forth below, the Amended Contention is inadmissible because it raises issues beyond the scope of this proceeding, lacks adequate factual and legal support, and fails to raise a genuine dispute on a material issue of law or fact, contrary to the requirements of 10 C.F.R. § 2.309(f)(1)(iii) to (vi).

[1] *See* State of New York's Motion for Leave to File New Additional Bases for Previously-Admitted Contention NYS-25 in Response to Entergy's July 14, 2010 Proposed Aging Management Program for Reactor Pressure Vessels and Internal Components (Sept. 15 , 2010) ("Motion for Leave"); Petitioner State of New York's Additional Bases for Previously-Admitted Contention NYS-25 (Embrittlement of Reactor Pressure Vessels and Associated Internals) (Sept. 15, 2010) ("Amended Contention"); NYS also filed the Declaration of Richard T. Lahey, Jr., dated Sept. 15, 2010 ("Lahey Decl.").

It also is untimely, in part, under 10 C.F.R. §§ 2.309(f)(2) and 2.309(c)(1), insofar as Petitioners belatedly argue, without good cause, that Entergy must consider the "combined synergistic aging effects" of embrittlement *and* fatigue on reactor vessel internal components.[2]

To the extent that NYS relies on expert opinion, the source of that opinion, Dr. Richard Lahey, is a self-described "international authority in *multiphase flow and heat transfer* technology"—not a metallurgist.[3] While Entergy does not dispute Dr. Lahey's evident expertise in the field of thermal-hydraulics, it is readily apparent from his declaration and resume that such expertise does not encompass the *metal* aging-degradation mechanisms (*e.g.*, embrittlement and metal fatigue) discussed in his declaration. This fact alone casts serious doubt on the validity of the opinions and conclusions contained in his declaration.

Notably, NYS and Dr. Lahey refer to RPV components and various metallurgical phenomena that have *no* relevance to the Reactor Vessel Internals ("RVI") Aging Management Program ("AMP") submitted by Entergy to the Nuclear Regulatory Commission ("NRC") in July 2010—the purported source of the Amended Contention. In fact, the effects of aging on the RPV (as opposed to RPV internals) are managed by a different AMP altogether. This misstep underscores their confusion relative to reactor vessel and internals embrittlement issues.

The crux of Dr. Lahey's declaration is that metal fatigue and neutron embrittlement can and will act in concert (*i.e.*, synergistically) at IPEC to make RPV and RPV internals more susceptible to failure when exposed to "accident-induced shock loads."[4] However, Dr. Lahey provides no support to demonstrate that such a phenomenon even exists. NYS and Dr. Lahey

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2 Motion for Leave at 5; Amended Contention at 1. *See also* Lahey Decl. ¶¶ 13-15.

3 *See* http://www.rpi.edu/~laheyr/ (emphasis added). *See also* Curricula Vitae of Richard T. Lahey, Jr., Ph.D. (attached to Lahey Decl.) (providing no indication that Dr. Lahey has expertise in solid metal degradation mechanisms).

4 Lahey Decl. ¶¶ 14, 15.

also fail to demonstrate that NRC regulations require any further analysis or management of the aging effects of embrittlement and metal fatigue.

Furthermore, NYS fails to state why it could not have raised this issue at the outset of the proceeding, especially given its initial identification of embrittlement *and* fatigue issues in admitted contentions NYS-25 and NYS-26/26A, respectively, nearly three years ago. Similarly, NYS fails to explain how Entergy's recent submittal of the RVI Program could possibly be the sole catalyst for its concerns about synergistic aging effects, which NYS suggests have "profound safety consequences."[5]

Finally, Dr. Lahey focuses mainly on perceived deficiencies in NRC rules and guidance rather than on the adequacy of the program elements described in the IPEC RVI Program and the detailed Electric Power Research Institute ("EPRI") inspection and evaluation guidelines on which that program is explicitly based. In this regard, his allegations are beyond the scope of this proceeding. Moreover, NYS does not challenge directly-relevant information in the RVI Program and related EPRI guidelines that addresses the very omissions it incorrectly alleges exist in the RVI Program relative to inspection timing, inspection methods, and corrective actions. NYS thus fails to satisfy the contention admissibility criteria of 10 C.F.R. § 2.309. For all of these reasons, the Board should deny admission of the Amended Contention.

II. BACKGROUND

A. Technical Background

IP2 and IP3 are Westinghouse pressurized water reactors ("PWRs").[6] PWRs contain water (*i.e.*, the primary coolant) under high pressure flowing through the core in which heat is generated by the fission process.[7] The reactor coolant system ("RCS") provides the boundary for

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5 Amended Contention at 1 & 2.

6 LRA at 1-6.

7 *See id.* at 2.3-2.

containing the pressurized primary coolant.[8] The RCS has various mechanical components, including the RPV and its internals.[9]

The RPV houses the nuclear fuel and serves as a key part of the reactor coolant pressure boundary.[10] For IPEC, the RPV components include, for example, the shell, top and bottom heads, closure head stud assembly, primary nozzles and safe ends, and control rod drive mechanism ("CRDM") housing penetrations.[11]

The role of the *RPV internals* is to direct the coolant flow, support the reactor core, and guide the control rods.[12] The three major parts of the IP2 and IP3 RPV internals are the lower core support structure, the upper core support structure, and the incore instrumentation support structure.[13] The lower core support structure comprises numerous components, including core baffle/former assembly bolts and plates, and a one-piece thermal shield.[14]

Importantly, the RPV and RPV internals are made of different materials. The RPV materials are fabricated primarily from ferritic materials (*e.g.*, carbon and low-alloy steels), whereas reactor vessel internals are made of austenitic stainless steel.[15] The types of steel that comprise the RPV and RPV Internals have different unirradiated material properties, and also vary in the manner in which their material properties change as a result of irradiation.[16] With different material compositions and different functions, the RPV and RPV internals are managed

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8 *Id.*

9 *Id.*

10 *See id.* at 2.3-3.

11 *See id.* at 2.3-15.

12 *See id.* at 2.3-3.

13 *Id.*

14 *See* LRA Tbls. 3.1.2-2-IP2 & 3.1.2-2-IP3.

15 *See id.* at 3.1-2 to 3.1-3.

16 *See id. See also* LRA Tbls. 3.1.2-1-IP2 through 3.1.2-2-IP3.

by separate AMPs.[17] Time-limited aging analyses ("TLAAs") demonstrate that irradiation embrittlement of the types of steels comprising the RPV is managed primarily through 10 C.F.R. Part 50 regulations for pressure-temperature ("P-T") limits, pressurized thermal shock ("PTS"), and Charpy upper shelf energy ("USE"). The Reactor Vessel Surveillance Program ensures that the reactor vessel TLAAs, which project fracture toughness based on projected neutron fluence, remain valid through the period of extended operation.[18] RPV internals are managed by a separate AMP—the Reactor Vessel Internals Program—which, as discussed below, Entergy submitted to the NRC in July 2010.

B. <u>10 C.F.R. Part 54 Requirements and Related Guidance</u>

Section 54.21(a)(3) requires an applicant to demonstrate in its LRA that the effects of aging on structures and components subject to an AMR *will* be adequately managed, so that there is "reasonable assurance" that their intended functions will be maintained consistent with the current licensing basis ("CLB") for the period of extended operation.[19] The NRC Staff reviews LRAs in accordance with the *Standard Review Plan* for license renewal applications, or "SRP-LR."[20] The NRC's *Generic Aging Lessons Learned Report* ("GALL Report") provides guidance to applicants for license renewal.[21] The SRP-LR provides that for each of the structures and components identified, the applicant may credit an AMP that is consistent with the GALL Report, or may choose to use a plant-specific AMP.[22] The GALL Report describes generic

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17 *See* LRA at 3.1-2, 3.1-3.

18 *See id.* at 3.1-2 to 3.1-3, 4.2-1 to 4.2-11; *id.*, App. B at B-111. *See also* NUREG-1930, Vol. 2, *Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3*, at 3-112 to 3-115 (Nov. 2009) ("SER").

19 *See* 10 C.F.R. §§ 54.21(a)(3), 54.29(a). *See also Entergy Nuclear Vt. Yankee, L.L.C.* (Vt. Yankee Nuclear Power Station), CLI-10-17, slip op. at 19, 37 (July 8, 2010).

20 *See* NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants* (Rev. 1, Sept. 2005) ("SRP-LR") (excerpt attached as Attach. 1).

21 *See* NUREG-1801, Vol. 2, *Generic Aging Lessons Learned Report* (Rev. 1, Sept. 2005) ("GALL Report") (excerpts attached as Attach. 2).

22 *See* SRP-LR at 3.0-2.

AMPs that the Staff has found acceptable for meeting the requirements of Part 54, based on its evaluations of aging management programs.[23]

Section IV.B2 of the GALL Report addresses aging management of Westinghouse RPV internals. For the aging effects related to loss of fracture toughness and neutron irradiation embrittlement, the GALL Report concludes that no further aging management review is necessary, if the applicant specifically commits to: (1) participate in the industry programs for investigating and managing aging effects on RPV internals; (2) evaluate and implement the results of industry programs as applicable to the RPV internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for RPV internals to the NRC for review and approval.[24]

III. PROCEDURAL HISTORY

In its April 2007 LRA, Entergy committed to take the three actions specified in GALL Report Section IV.B2, as listed above, to manage the loss of fracture toughness of RPV internals components due to neutron irradiation embrittlement.[25]

On November 30, 2007, NYS submitted NYS-25 in response to Entergy's April 2007 LRA.[26] As proffered, NYS-25 contended that the LRA fails to include an adequate plan to monitor and manage the effects of aging due to *embrittlement* of the RPV and the associated internals as required by 10 C.F.R. § 54.21(a), and does not include an evaluation of TLAA as

[23] *See id.* at 3.0-1. According to the Commission, "the GALL Report [is] a guidance document that was prepared at our behest and that we have cited with approval." *Vt. Yankee*, CLI-10-17, slip op. at 45.

[24] GALL Report at IV B2-4.

[25] *See* LRA at 3.1-7.

[26] *See* New York State Notice of Intention to Participate and Petition to Intervene at 223-27 (Nov. 30, 2007) ("NYS Petition").

required by 10 C.F.R. § 54.21(c).[27] Entergy and the NRC Staff opposed the admission of NYS-25 on the grounds that it did not fully meet the criteria in 10 C.F.R. § 2.309(f)(1).[28]

On July 31, 2008, the Board admitted NYS-25, finding that the asserted need for an AMP to manage the effects of embrittlement of the RPVs and associated internals is within the scope of this proceeding.[29] The Board also found that a genuine material dispute existed because NYS's expert, Dr. Richard Lahey, had alleged deficiencies in "specific portions" of Entergy's LRA based on his professional judgment.[30]

On July 14, 2010, Entergy filed LRA Amendment 9, which revised the LRA to provide details on the RVI Program.[31] The RVI Program is a new, plant-specific program that will manage aging effects of RPV internals using guidance developed from nearly a decade of extensive industry research and contained in EPRI Materials Reliability Program ("MRP") documents MRP-227 and MRP-228.[32] MRP-227 provides comprehensive inspection and

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27 *See id.* at 223.

28 *See* Answer of Entergy Nuclear Operations, Inc. Opposing New York State Notice of Intention to Participate and Petition to Intervene at 135-41 (Jan. 22, 2008); NRC Staff's Response to Petitions for Leave to Intervene Filed by [the State of New York] at 75-77 (Jan. 22, 2008).

29 *See Entergy Nuclear Operations, Inc.* (Indian Point Nuclear Generating Units 2 & 3), LBP-08-13, 68 NRC 43, 131, (2008).

30 *Id.* As shown below, in his September 15, 2010 declaration, Dr. Lahey does not raise particularized or supported challenges to specific portions of the IPEC RVI Program.

31 *See* NL-10-063, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Amendment 9 to License Renewal Application (LRA) – Reactor Vessel Internals Program," attach. 1 (July 14, 2010) ("LRA Amendment 9") (excerpts attached as Attach. 3), *available at* ADAMS Accession No. ML102010102. Entergy informed the Board and other parties of the submittal of the RVI Program on July 15, 2010. Letter from Paul Bessette, Counsel for Entergy, to Administrative Judges, "Notification of Entergy's Submittal of the Reactor Vessels Internals Program for Indian Point Units 2 and 3" (July 15, 2010), *available at* ADAMS Accession No. ML102030120. Entergy disclosed MRP-227 to NYS on November 30, 2009, and identified MRP-228, an EPRI-proprietary report, in its July 1, 2010 mandatory disclosures. On July 6, 2010, NYS requested a copy of the report, which Entergy produced on August 6, 2010, pursuant to the terms of the Board's September 4, 2009 Protective Order. Four days later, NYS sought an additional month in which to file amended or new contentions based on LRA Amendment 9. *See* State of New York's Motion to Extend Time in Which to File New or Supplemental Contentions concerning Entergy's Ninth Amendment to the License Renewal Application (Aug. 10, 2010). The Board granted the extension. *See* Licensing Board Order (Granting New York's Motion to Extend Deadline for Filing New Contentions) (unpublished) (Aug. 12, 2010).

32 *See* NL-10-063, at 82-84. *See also* EPRI, MRP-227, *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines* (Rev. 0, Dec. 2008) (attached as Attach. 4); EPRI, MRP-228, *Materials Reliability Program: Inspection Standard for PWR Internals* (July 2009) (proprietary).

evaluation guidelines for managing the effects of aging on PWR vessel internals.[33] MRP-228 contains inspection requirements specific to the inspection methods delineated in MRP-227, as well as requirements for qualification of the nondestructive examination ("NDE") systems used to perform those inspections.[34] In January 2009, EPRI submitted MRP-227 to the NRC for formal Staff review and approval. That review is ongoing.

On September 15, 2010, NYS filed the Amended Contention and the accompanying new declaration of Dr. Lahey. The Amended Contention alleges, in principal part, that Entergy's RVI Program is deficient because it purportedly does not: (1) consider the "synergistic" effects of embrittlement and metal fatigue on the RPV and its internals; (2) provide sufficient details about when IPEC will conduct and complete baseline inspections; (3) include adequate inspection techniques to identify embrittlement issues for certain RPV internals; and (4) provide sufficiently specific details or commitments regarding when and how IPEC will implement corrective actions to address any future embrittlement-related issues.[35] As set forth below, each of these allegations is unfounded and fails to support the admission of the Amended Contention, contrary to the requirements of 10 C.F.R. § 2.309(f).

IV. <u>LEGAL STANDARDS FOR THE ADMISSION OF AMENDED CONTENTIONS</u>

A. <u>Timeliness Requirements</u>

An intervenor may file new or amended safety contentions only with leave of the Board upon a showing that the new or amended contention is based on information that was not previously available and is materially different than information previously available.[36] Thus, a new contention is not admissible to the extent that it raises additional arguments that could have

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[33] NL-10-063, at 84.

[34] *Id.*

[35] Motion for Leave at 6. *See also* Amended Contention at 1-5.

[36] 10 C.F.R. § 2.309(f)(2)(i)-(iii).

been raised previously.[37] In a decision issued less than two weeks ago, the Commission reemphasized the importance of complying with its timeliness rules:

> [NRC] contention admissibility and timeliness rules require a high level of discipline and preparation by petitioners, who must examine the publicly available material and set forth their claims and the support for their claims at the outset. There simply would be no end to NRC licensing proceedings if petitioners could disregard our timeliness requirements and add new contentions at their convenience during the course of a proceeding based on information *that could have formed the basis for a timely contention at the outset of the proceeding*. Our expanding adjudicatory docket makes it critically important that parties comply with our pleading requirements and that the Board enforce those requirements.[38]

On this point, the Commission further noted that a petitioner's obligation to review the available documentary materials and to conduct its own due diligence is "iron-clad."[39]

If a petitioner cannot satisfy the criteria of Section 2.309(f)(2), then a contention is considered "nontimely," and the petitioner must successfully address the late-filing criteria in 10 C.F.R. § 2.309(c)(1)(i) to (viii).[40] The first factor identified in that regulation, whether good cause exists for the failure to file on time, is entitled to the most weight.[41] Without good cause, a petitioner's demonstration on the other factors must be particularly strong.[42]

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

38 *N. States Power Co.* (Prairie Island Nuclear Generating Plant, Units 1 & 2), CLI-10-27, slip op. at 18 (Sept. 30, 2010) (emphasis added).

39 *Id.*

40 *See* 10 C.F.R. § 2.309(c)(2) ("The requestor/petitioner *shall* address the factors in paragraphs (c)(1)(i) through (c)(1)(viii) of this section in its nontimely filing.") (emphasis added). *See also* Licensing Board Scheduling Order at 5-6 (July 1, 2010) (unpublished).

41 *See New Jersey* (Dep't of Law & Pub. Safety's Requests Dated Oct. 8, 1993), CLI-93-25, 38 NRC 289, 296 (1993).

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

B. Substantive Admissibility Requirements

A proposed contention also must satisfy each of the admissibility criteria in 10 C.F.R. § 2.309(f)(1)(i) to (vi).[43] These criteria are intended to ensure that hearings cover only genuine and pertinent issues of concern and are effective and focused on real, concrete issues.[44] Given the nature of NYS's arguments, several principles bear emphasis here.

First, a contention that challenges applicable statutory requirements or regulations must be rejected as outside the scope of the proceeding.[45] Second, a contention that simply states the petitioner's views about what regulatory policy ought to be does not present a litigable issue.[46] Third, the Board may not accept uncritically the assertion that a document or other factual information or an expert opinion supplies the basis for a contention.[47] Absent a reasoned basis or explanation, an expert declaration is insufficient to support the admission of a contention.[48] Finally, a petitioner must establish that a genuine dispute exists with the applicant on a material issue of law or fact.[49] A petitioner's oversight or failure to perform its own due diligence cannot provide the foundation for an admissible contention.[50] Further, an allegation that some aspect of

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43 *S.C. Elec. & Gas Co.* (Virgil C. Summer Nuclear Station, Units 2 & 3), LBP-10-06, slip op. at 3 (Mar. 17, 2010).

44 Final Rule, Changes to Adjudicatory Process, 69 Fed. Reg. 2182, 2189-90 (Jan. 14, 2004).

45 *See* 10 C.F.R. § 2.335(a) (absent a waiver, "no rule or regulation of the Commission . . . is subject to attack by way of discovery, proof, argument, or other means in any adjudicatory proceeding"). *See also Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant Units 1), LBP-07-11, 66 NRC 41, 57-58 (*citing Phila. Elec. Co.* (Peach Bottom Atomic Power Station, Units 2 & 3), ALAB-216, 8 AEC 13, 20 (1974)).

46 *See Peach Bottom*, ALAB-216, 8 AEC at 20-21.

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

48 *See Fansteel, Inc.* (Muskogee, Okla. Site), CLI-03-13, 58 NRC 195, 203-05 (2003) (holding that a petitioner also must explain the significance of any factual information upon which it relies); *Dominion Nuclear Conn., Inc.* (Millstone Nuclear Power Station, Unit 3), CLI-08-17, 68 NRC 231, 240 (Aug. 13, 2008) (noting that an expert must provide more than speculation).

49 10 C.F.R. § 2.309(f)(1)(iv) & (vi) (emphasis added).

50 *See Dominion Nuclear Conn., Inc.* (Millstone Nuclear Power Station, Units 2 & 3), LBP-04-15, 60 NRC 81, 95-96 *aff'd*, CLI-04-36, 60 NRC 631 (2004). For example, if a petitioner alleges an omission, and the purportedly missing information is in the relevant licensing submittal, then the contention does not raise a genuine material issue.

a license application is inadequate or unacceptable does not give rise to a genuine dispute unless it is supported by facts and a reasoned statement of *why the application* is unacceptable in some *material* respect.[51]

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

A. NYS's Allegation That Entergy Must Evaluate the "Synergistic Effects" of Embrittlement and Metal Fatigue Lacks Adequate Support and Is Untimely

NYS contends that the RVI Program should address the "synergistic" aging effects of embrittlement and metal fatigue on the RPV and RPV internals.[52] This argument fails to support the admission of the Amended Contention because it: (1) lacks an adequate legal or factual foundation, (2) raises issues beyond the scope of this proceeding, and (3) fails to establish a genuine material dispute with Entergy, in contravention of 10 C.F.R. § 2.309(f)(1)(iii) to (vi). It also is inexcusably late under the timeliness criteria of 10 C.F.R. §§ 2.309(f)(2) and 2.309(c)(1).

First, Dr. Lahey's statements regarding "synergistic" aging effects are rooted in conjecture rather than accepted science. For example, Dr. Lahey states that "[h]ow the rather complex metal degradation mechanisms associated with fatigue and irradiation interact is still an area of active research."[53] Dr. Lahey, however, provides no information about the sponsors, status, or results of this research. Moreover, he cites no peer-reviewed and/or NRC-sponsored studies documenting the purported synergistic effects of embrittlement and metal fatigue on RPV

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[51] *See Fla. Power & Light Co.* (Turkey Point Nuclear Generating Plant, Units 3 & 4), LBP-90-16, 31 NRC 509, 521, 521 n.12 (1990). This principle applies equally assertions made by a proffered expert. *See USEC*; CLI-06-10, 63 NRC at 472 (*quoting Private Fuel Storage*, LBP-98-7, 47 NRC at 181) ("[A]n expert opinion that merely states a conclusion (*e.g.*, the application is 'deficient,' 'inadequate,' or 'wrong') without providing a reasoned basis or explanation for that conclusion is inadequate because it deprives the Board of the ability to make the necessary, reflective assessment of the opinion" alleged to provide a basis for the contention).

[52] Motion for Leave at 5; Amended Contention at 1. *See also* Lahey Decl. ¶¶ 13-15. Specifically, NYS and Dr. Lahey identify the following structures and components: the core baffle, intermediate shells, former plates and bolts (particularly the re-entrant corners), and including the baffle-to-baffle bolt locations, the core barrel-to-former bolt locations, and baffle-to-former bolt locations, core barrel (and its welds), lower core plate and support structures, clevis bolts, fuel alignment pins, thermal shield, the lower support column and mixer, and the control rods and their associated guide tubes, plates, and welds. Amended Contention at 1; Lahey Decl. ¶ 11.

[53] Lahey Decl. ¶ 10.

internals. Dr. Lahey also provides no technical or factual basis for his *assumption* that IPEC RPV and vessel internals components will be fatigued to such an extent that they will be more vulnerable to other aging effects or failure mechanisms.[54] Instead, he merely hypothesizes that "thermal shock *may* cause *highly* embrittled and fatigued incore components to fail, *perhaps* leading to an uncoolable core geometry and core melt."[55] Speculation, even if by a proffered expert, does not provide adequate support for the admission of a contention.[56]

Second, NYS and Dr. Lahey refer to numerous components and degradation mechanisms that are not germane to the RVI Program (or even subject to AMR).[57] For example, the intermediate shell and stub tubes referenced by Dr. Lahey are not PWR vessel *internals* components.[58] Therefore, they are not included in the RVI Program or MRP-227. In addition, the IPEC control rods are not subject to AMR because they perform their intended function with moving parts or a change in configuration.[59] Similarly, Dr. Lahey refers to certain aging effects/components (*e.g.*, thermally-aged cast stainless steel in-core components, boric acid corrosion of the upper RPV head) that are not identified in NYS-25 and are not the subject of the new RVI Program.[60] Accordingly, these arguments do not controvert the RVI Program, as

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54 In actuality, the LRA indicates that the metal fatigue analyses for IP2 and IP3 reactor vessel internals will remain valid for the period of extended operation, per 10 C.F.R. § 54.21(c)(1)(i). LRA at 4.3-11 to 4.3-12.

55 Lahey Decl. ¶ 14 (emphasis added).

56 *See Indian Point*, LBP-08-13, 68 NRC at 63.

57 NYS repeatedly loses sight of the fact that the AMP in question relates solely to the RPV internals. For example, NYS refers to "[t]he *RPV* and internal components that are the subject of the recently proposed [RVI] AMP"; "Entergy's now-modified attempt to provide an adequate AMP for *RPV* and internals"; "inadequate management of the effects of embrittlement on *RPV* and internals" (Motion for Leave at 4 (emphasis added)); and "aggressive corrosion and wasting of the unclad outer surface of the *upper head of the RPV*." Lahey Decl. ¶ 18 (emphasis added).

58 Lahey Decl. ¶¶ 11, 16. The intermediate shell refers to the plates that make up the reactor vessel. At IPEC, these plates are addressed by other AMPs. These other AMPs include the Water Chemistry Control—Primary and Secondary Inservice Inspection, Reactor Vessel Surveillance Program, and Inservice Inspection Program.

59 *See* 10 C.F.R. § 54.21(a)(1)(i); LRA at 2.3-14 ("The control rods are active components and are not subject to aging management review.").

60 Lahey Decl. ¶¶ 9, 10, 17.

required by Section 2.309(f)(1)(vi), and seek impermissibly to expand the scope of the original contention.[61]

Third, NYS cites no NRC regulation or guidance document that requires or otherwise directs applicants to evaluate the so-called "synergistic" aging effects of neutron embrittlement and metal fatigue for those components that are properly included in the scope of the RVI Program.[62] As noted above, Dr. Lahey simply claims that this postulated metallurgical phenomenon is the subject of "active research" and constitutes a "significant safety issue" but provides no reasoned basis or explanation for this conclusion.[63] When a petitioner alleges an omission (*i.e.*, analysis of supposed "synergistic" effects), the petitioner is required to show that the missing information is required by law.[64] NYS has not done so here.

Fourth, NYS's contention that the RVI Program does not adequately manage the effects of both embrittlement and metal fatigue on affected components lacks a factual basis. The RVI Program is based on MRP-227, which includes screening criteria that incorporate susceptibility levels for eight postulated aging mechanisms relevant to reactor vessel internals, including *fatigue and irradiation embrittlement*.[65] Thus, programs based on MRP-227 monitor the effects of the eight aging degradation mechanisms on the intended function of PWR internals through one-time, periodic, and conditional examinations, taking into account the relative susceptibility

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61 *See McGuire/Catawba*, CLI-02-28, 56 NRC at 386 ("An intervenor may not freely change the focus of an admitted contention at will as litigation progresses, but is bound by the terms of the contention.") (internal quotation marks and citations omitted).

62 In contrast, GALL Report Section XI.M13 (Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)) specifically directs license renewal applicants to consider the "synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement." GALL Report at XI M-52.

63 Lahey Decl. ¶¶ 10, 15.

64 *See* 10 C.F.R. § 2.309(f)(1)(vi).

65 *See* MRP-227 at 1-1, 2-1, 3-12 to 3-16, 3-23 to 3-24 (Tbl. 3-3, Final disposition of category B and C Westinghouse internals).

of PWR internals to the eight postulated aging mechanisms.[66] NYS neither acknowledges nor challenges any of the relevant discussion in MRP-227.

Fifth, NYS fails to present any specific, material challenges to Entergy's RVI Program and the detailed EPRI guidelines on which that program is explicitly based. Apart from a few perfunctory and passing references to the RVI Program and MRP-227, Dr. Lahey does not meaningfully address the specifics of the RVI Program.[67] Notably, he does not challenge the adequacy of the specific inspection methods or acceptance criteria described in MRP-227. For example, MRP-227 provides detailed information concerning EPRI's initial component and screening categorization process, functionality assessment of degradation for components and assemblies of components, recommended component inspection or examination techniques, examination acceptance and expansion criteria, and evaluation methodologies.[68] Dr. Lahey fails to mention, much less challenge, any of this information.

Indeed, much of Dr. Lahey's declaration is devoted to broad, generic criticisms of current NRC rules and guidance, which clearly are not grounds for an admissible contention in this proceeding. For instance, Dr. Lahey states that the SRP-LR and GALL Report suffer from "serious omissions" and overlook a "significant safety issue."[69] In particular, he posits that those documents do not consider how highly-embrittled and fatigued internal RPV structures and fittings will respond to severe transient decompression shock loads, such as those associated with a design-basis accident ("DBA") loss of coolant accident ("LOCA").[70] He further contends that

66 *See id.* at 2-1.

67 Dr. Lahey states only that the IPEC RVI Program "does not call for an analysis of the synergistic impacts of fatigue and embrittlement" (Lahey Decl. ¶ 13) and that industry programs that Entergy has proposed to follow "are mute on the serious age-related safety concern of the coolability of PWR cores subsequent to an accident-induced failure of highly embrittled and fatigued RPV internals." *Id.* ¶ 14. This presupposes, of course, that such an analysis is required by regulation or recommended by the GALL Report—which it is not.

68 *See generally*, MRP-227, Secs. 3-6.

69 *See* Lahey Decl. ¶¶ 13, 14, 15.

70 *See id.* ¶¶ 14-15.

the NRC's leak-before-break rule improperly excludes in-vessel DBA LOCA decompression and thermal shock loads.[71] He attributes these alleged deficiencies to industry "confusion" and the NRC's "stove piping" of aging management issues and safety evaluations.[72]

None of these criticisms, however, establishes a genuine material dispute with Amendment 9 to Entergy's LRA. To the contrary, they raise issues beyond the scope of that submission and this proceeding. NYS's and Dr. Lahey's "views of what applicable policies ought to be are not proper for adjudication."[73] If NYS believes that current NRC regulations related to neutron embrittlement, metal fatigue, or leak-before-break are inadequate, then its recourse lies in the rulemaking process—not in this adjudication.[74]

Finally, setting aside its lack of support and materiality, NYS's argument that Entergy must consider the purported synergistic effects of RPV embrittlement and metal fatigue on the RPV and RPV internals is impermissibly late. Section 2.309(f)(2) states that a new or amended contention must be *based on* previously unavailable and materially different information. Entergy's April 2007 LRA identified both embrittlement and metal fatigue as relevant aging mechanisms and described associated aging management activities.[75] NYS submitted separate contentions concerning *embrittlement* (NYS-25) and *metal fatigue* (NYS-26/26A), which the Board admitted.[76] NYS could have raised its concerns about the purported synergistic effects of these two aging mechanisms in its original petition to intervene in this proceeding, which was

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71 *See id.* ¶ 15.

72 *Id.*

73 *Gen. Pub. Utils. Nuclear Corp.* (Three Mile Island, Unit 1) LBP-86-10, 23 NRC 283, 285 (1986).

74 *See Duke Energy Corp.* (Oconee Nuclear Station, Units 1, 2, & 3), CLI-99-11, 49 NRC 328, 345 (1999) ("If petitioners are dissatisfied with our generic approach to the problem, their remedy lies in the rulemaking process, not in this adjudication."); *Conn. Yankee Atomic Power Co.* (Haddam Neck Plant), CLI-03-7, 58 NRC 1, 7 (2003) (stating that "[i]f our safety regulations are in any way inadequate and need revision, the appropriate vehicle to ask the Commission to set a new standard is a petition for rulemaking under 10 C.F.R. § 2.802").

75 *See* LRA at 3.1-3, 3.1-6 to 3.1-7, 4.3-11 to 4.3-12.

76 *See Indian Point*, LBP-018-13, 68 NRC at 129-140.

required to be filed in November 2007. NYS has not shown good cause for its extremely belated identification of this issue, as required by 10 C.F.R. § 2.309(c)(1).[77] Further, they fail to explain how the RVI Program (which does not even address the RPV) could be the source of, or trigger, their newly-proffered claim about synergistic aging effects on the RPV and RPV internals.

B. Entergy Has Provided Specific and Adequate Information Concerning the Timing of Inspections Performed Under the RVI Program

NYS further argues that the RVI Program is inadequate because it does not specify with "meaningful precision" when Entergy will initiate and complete baseline inspections.[78] This argument also fails to support the admission of the Amended Contention because it: (1) lacks an adequate legal or factual foundation, (2) raises issues beyond the scope of this proceeding, and (3) fails to establish a genuine material dispute with Entergy.[79] Indeed, a meaningful analysis of the RVI Program and MRP-227 clearly demonstrates that NYS's claim that the RVI Program is deficient lacks factual support.[80]

MRP-227 states explicitly that it contains guidance for methods, extent, and *frequency* of one-time, periodic, and conditional examinations and other aging management methodologies.[81] For Westinghouse PWRs, that guidance is contained in Table 4-3 (Westinghouse plants Primary components) and Table 4-6 (Westinghouse plants Expansion components).[82] Specifically, the

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77 NYS's belatedness is particularly surprising given the alleged gravity of its claims in the Amended Contention. Dr. Lahey describes the postulated synergistic effects of embrittlement and metal fatigue as a significant safety issue. Lahey Decl. ¶ 15.

78 Amended Contention at 2.

79 *See* 10 C.F.R. § 2.309(f)(1)(iii), (v), (vi).

80 *Millstone*, LBP-04-15, 60 NRC at 95 (denying admission of contention because "the Petitioner's assertion that the applications are deficient is simply based upon a failure to read or perform any meaningful analysis of the applications").

81 MRP-227, App. A at A-2.

82 *See id.* at 4-24 & 4-33. These tables contain columns describing the component, any particular applicability requirement for that component, the degradation effect to be detected, the examination method/frequency, the examination coverage, and any linkage between the Primary and Expansion components. *See id.* at 4-6. *See also, e.g.*, MRP-227 Table 4-3 (Westinghouse plants Primary components). As the RVI Program explains, MRP-227 separates PWR internals components into four groups (Primary, Expansion, Existing Programs, and No Additional

fifth column of Table 4-3, prominently labeled "Examination Method/Frequency," lists the applicable examination frequencies for each of the Westinghouse PWR Primary components.[83] By not addressing this directly-relevant information, NYS fails to meet its obligation under Section 2.309(f)(1)(vi) to explain why the RVI Program is deficient in some material respect.[84]

Dr. Lahey's assertion that baseline inspections should or must occur before the onset of extended operations directly contravenes this Board's admissibility ruling in LBP-08-13. Specifically, in proposed contention NYS-23, NYS argued that the NRC Staff should require Entergy to conduct thorough baseline inspections of IP2 and IP3 *prior to life extension.*[85] In rejecting NYS-23, the Board stated:

> The Board <u>rejects</u> this contention because it is outside the scope of this license renewal proceeding. Part 54 does not require the type of comprehensive baseline inspection desired by NYS, no matter how sensible such a requirement might seem. LRA §§ 2.1 to 2.5 describe the scoping and screening results of the IPAs [Integrated Plant Assessments] required by Section 54.21, and LRA Appendix B provides a discussion of license renewal inspection programs. NYS has not pointed to specific facts to support the conclusion that the IPAs in the LRA—the only plant inspection program required by the regulations—are inadequate. Entergy has done what the regulations require.[86]

Furthermore, while MRP-227 recommends that certain "baseline" inspections be performed before or shortly after entering the period of extended operation, it does not recommend comprehensive, pre-license extension inspections of the type advocated by NYS in NYS-23. Insofar as NYS might argue otherwise, it does so in contravention of current NRC regulations and the Board's prior ruling on this matter.

Measures) depending on (1) their susceptibility to and tolerance of aging effects, and (2) the existence of programs that manage the effects of aging. *See id.* at 3-15 to 3-16.

83 *Id.* at 4-24 to 4-26.

84 *See PPL Susquehanna LLC* (Susquehanna Steam Electric Station, Units 1 & 2), LBP-07-4, 65 NRC 281, 306 (2007).

85 *See* NYS Petition at 219.

86 *Indian Point*, LBP-08-13, 68 NRC at 126.

For the foregoing reasons, the Amended Contention should be dismissed for failing to satisfy 10 C.F.R. § 2.309(f)(1)(iii), (v), and (vi).

C. <u>NYS's Allegations Concerning Reactor Vessel Internals Examination Methods Are Misplaced and Factually Unfounded</u>

NYS alleges that Entergy did not disclose that certain visual examinations (class VT-3 examinations) would be done by remote control, and that other visual examination methods (class VT-1 and class EVT-1) have a greater degree of detection than class VT-3 examinations.[87] In so arguing, NYS points to *purported* discrepancies between the RVI Program and MRP-227. Notably, this argument is not based on Dr. Lahey's declaration or any other expert support.[88] Dr. Lahey's declaration contains no discussion or criticisms of the specific examination methods described in MRP-227 (*see, e.g.*, Section 4.2), MRP-228, or the RVI Program (*see* Program Elements 3 and 4).

In any case, the RVI Program will be implemented using the inspection and evaluation guidelines contained in MRP-227. The Program Description contained in revised LRA Sections A.2.1.41, A.3.1.41, and B.1.42 (Reactor Vessel Internals Program) confirms this fact.[89] It states that MRP-227 and MRP-228 provide the basis of the IPEC RVI Program, and that the program will be implemented in accordance with MRP-227 inspection recommendations and evaluation acceptance criteria.[90] The RVI Program further states that any revisions to MRP-227 and MRP-228, including any changes resulting from the NRC review of the documents (issued as MRP-

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87 *See* Amended Contention at 2-4.

88 *See Crow Butte Res., Inc.* (North Trend Expansion Area), CLI-09-12, slip op. at 35 (June 25, 2009) (stating that "unsubstantiated arguments of counsel . . . do not form the basis for a litigable contention").

89 *See* LRA Amendment 9, at 83 (LRA Sec. A.3.1.41, Reactor Vessel Internals Aging Management Activities), 84-90 (LRA Sec. B.1.42, Reactor Vessel Internals Program).

90 *Id.* at 83. *See also id.* at 90 ("The RVI Program will be effective at managing aging effects since it will incorporate proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls in accordance with MRP-227 and MRP-228 guidelines and current IPEC programs.").

227-A and MRP-228-A), will be incorporated into the RVI Program.[91] Thus, there simply is no basis for NYS's suggestion that the RVI Program is somehow less stringent than the MRP guidelines on which it is expressly based or, as discussed further below, that it does not adequately disclose the bases for Entergy's selection of particular inspection methods.

NYS also suggests that Entergy has not explained why it is using different examination methods for similar components.[92] In particular, NYS alleges that Entergy relies on "less reliable" remote-control VT-3 examinations to examine baffle former assembly plates and edge bolts instead of volumetric ultrasonic testing (UT).[93] This argument (which again is not addressed or supported by NYS's expert) also lacks a factual basis and further reflects NYS's failure to review the relevant documentation, as required by Section 2.309(f)(1)(v) and (vi).

Section 4.2 of MRP-227 squarely addresses this issue. It explains that the different NDE methods described therein (visual examination, surface examinations, volumetric examinations, and physical measurements) are suitable for managing the effects of one or more aging degradation mechanisms for PWR internals, depending upon: (1) tolerance of the component functionality to the progression of particular effects, (2) accessibility of the component by the equipment needed for the examination, and (3) suitability of the equipment for detecting the particular effect.[94] Section 4.2 further states that the selected methods are consistent with those specified in the NRC-approved edition and addenda of ASME Code Section XI.[95]

In other words, extensive industry experience, as explicitly referenced in MRP-227, teaches that different examination methods are appropriate for different components. For

91 *Id.* at 83-84.

92 Amended Contention at 4.

93 *Id.* at 2.

94 *See* MRP-227, at 4-3.

95 *See id.* (*citing* Am. Soc'y of Mech. Eng'rs, *ASME Boiler and Pressure Vessel Code*, Sec. XI (2001 ed.) (Rules for Inservice Inspection of Nuclear Power Plant Components) (including the 2002 and 2003 Addenda)).

example, visual (VT-3) examinations are used for detecting general mechanical and structural degradation of PWR internals subject to ASME Code, Section XI, IWB-2500 B-N-3 requirements (*e.g.*, baffle former assembly plates).[96] In contrast, visual (VT-1) and enhanced visual (EVT-1) examinations are used to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion.[97] Surface examinations, in turn, may be used to confirm or disposition possible indications.[98] Finally, UT is generally used where visual or surface examination is unable to detect the effect of the age-related degradation for some PWR internals due to inaccessibility (*e.g.*, baffle former bolting).[99] Thus, contrary to NYS's assertion, the proposed examination methods described in the IPEC RVI Program are both clearly explained and consistent with MRP-227 guidelines. NYS makes only vague references to these examination methods and does not challenge the suitability of these methods for the specific applications discussed in the RVI Program and MRP-227.

Accordingly, the Amended Contention, as supported by this argument, fails to meet the requirements of 10 C.F.R. § 2.309(f)(1)(v) and (vi). NYS's incomplete review of the RVI Program and MRP-227 does not provide a valid basis for admission of the contention.

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96 *See id.* at 4-4. In particular, MRP-227 states that visual (VT-3) examinations are conducted to determine the general mechanical and structural condition of components by detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion; and by identifying conditions that could affect operational or functional adequacy of components. *Id.* Consistent with this description, the IPEC RVI Program states that periodic visual examinations (VT-3) of the baffle former assembly plates and edge bolts will detect symptoms of distortion due to void swelling or cracking from irradiation-assisted stress corrosion cracking ("SCC"), or IASCC, such as abnormal interactions with fuel assemblies, gaps or displacement along component joints, broken or damaged bolt locking systems, and failed or missing bolts. LRA Amendment 9 at 87 (RVI Program Element 4).

97 MRP-227, at 4-4.

98 *See id.* at 4-5.

99 *See id.* (stating that IASCC in baffle/former bolts may occur under the bolt head (in the shank or threaded region) and will be undetectable by visual or surface examination unless the bolt is removed and subject to examination over its entire length). Again, consistent with MRP-227, the IPEC RVI Program states that volumetric UT examinations will be used to locate potential cracking of baffle former bolting. LRA Amendment 9, at 87 (RVI Program Element 4). Note that MRP-227, on which the RVI Program is based, also states that ASME Code Section XI permits the use of UT as an alternative or supplement to the specified visual examinations for other configurations such as plates and welds. *See* MRP-227, at 4-5.

D. Contrary to NYS's Claim, the RVI Program Does Not Disavow Preventative Actions or Lack Sufficient Details Regarding Future Corrective Actions

NYS states that Entergy "disavows taking any preventative action" in the RVI Program.[100] In reality, Entergy indicates preventative actions are specified for RVI, although not as part of the RVI Program. Specifically, the RVI Program explains that it is a condition monitoring program, which, as defined in the SRP-LR, is intended to inspect for the presence and extent of aging effects.[101] The RVI Program further indicates that the Water Chemistry Control—Primary and Secondary Program provides for preventative action by maintaining primary water chemistry in accordance with EPRI guidelines to minimize the potential for SCC and IASCC.[102] Thus, NYS's allegation is incorrect.

NYS also alleges that Entergy's RVI Program does not provide sufficient objective details concerning implementation of corrective actions, and relies on a "vague commitment to some undefined action in the indefinite future."[103] Again, NYS fails to furnish a factual or legal basis for its contention and account for directly-relevant information in the RVI Program and MRP-227. As such, the Amended Contention does not meet 10 C.F.R. § 2.309(f)(1)(v) and (vi).

MRP-227, Section 6 explains that four principal options are available for the disposition of conditions detected during component examinations that do not satisfy the examination acceptance criteria in Section 5.[104] Those options include, but are not limited to: (1) supplemental examinations, such as a surface examination, to supplement a visual (VT-1) or an enhanced visual (EVT-1) examination; (2) engineering evaluation that demonstrates the acceptability of a detected condition; (3) repair, in order to restore a component with a detected

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100 Amended Contention at 2.

101 *See* LRA Amendment 9, at 86; SRP-LR App. A at A.1-1.

102 *See* LRA Amendment 9, at 86.

103 Amended Contention at 4. *See also* Motion for Leave at 6.

104 *See* MRP-227, at 6-1.

condition to acceptable status; or (4) replacement of a component with an unacceptable detected condition.[105] Section 6 also provides information on methodologies that can be used for the evaluation of detected conditions that exceed the examination acceptance criteria of Section 5.[106]

As MRP-227 further explains, corrective actions following the detection of unacceptable conditions are provided for in each plant's corrective action program ("CAP"), as required by 10 C.F.R. Part 50, Appendix B, with additional guidance contained in ASME Code, Section XI.[107] This is reflected in Program Element 7 (Corrective Action Program) of the RVI Program, which states, in part, that the Entergy (10 C.F.R. Part 50, Appendix B) Quality Assurance Program, including relevant corrective action controls, applies to the RVI Program.[108] It also states that the option of component repair and replacement of PWR internals is subject to the long-standing requirements of ASME Code Section XI, as implemented at IPEC for many years under current plant operating programs.[109] Again, it is unclear what additional details NYS seeks, particularly with respect to now-indeterminable future corrective actions on inspections yet to be performed.

Finally, there is no factual basis for NYS's allegation that there is "growing evidence that Entergy is unable to meet its commitments."[110] The very document that NYS cites in support of this claim—a September 13, 2010 NRC audit report concerning Entergy's Commitment Management System for the Vermont Yankee plant—reaches a different conclusion.[111] It states

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105 *Id.*

106 *See id.* at 6-1 to 6-11.

107 *See id.*, App. A at A-3.

108 LRA Amendment 9, at 88.

109 *See id.*

110 Amended Contention at 2.

111 NYS also references the July 31, 2008 IPEC Independent Safety Report (*available at* http://nyindianpoint.org/images/Full%20Report.pdf) to support the proposition that Entergy will not meet its commitment tracking requirements "due to a backlog of preventative and corrective maintenance work." Amended Contention at 5. NYS, however, does not explain how statements concerning maintenance backlogs in this two-year-old report bear on or impact Entergy's ability to effectively implement the RVI Program during the period of extended operation or to meet applicable IPEC commitments to the NRC.

that Entergy "properly addressed all the regulatory commitments selected for this audit," and that Vermont Yankee has "an effective program to manage changes to regulatory commitment[s]."[112] Thus, NYS presents no factual information to corroborate its suggestion that Entergy is unable or unwilling to meet its regulatory commitments.[113]

Additionally, as a legal matter, docketed licensee commitments provide one acceptable means for meeting certain regulatory obligations.[114] The Commission has long declined to assume that licensees will refuse to meet their obligations,[115] particularly given that licensees remain subject to continuing NRC oversight, inspection, and enforcement authority during the period of extended operation.[116] Moreover, current operational and compliance issues are beyond the scope of a license renewal proceeding, as the Commission recently emphasized in reversing a Board's admission of a "safety culture" contention:

> We stated unambiguously in our License Renewal Rule that "license renewal should not include a new, broad-scoped inquiry into compliance that is separate from and parallel to [our] ongoing compliance oversight activity." We specifically indicated that other broad-based issues akin to safety culture—such as operational history, quality assurance, quality control, management competence, and human factors—were beyond the bounds of a license renewal proceeding. This is because these conceptual issues fall outside the bounds of the passive, safety-related physical systems, structures and components that form the scope of our license renewal review.[117]

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112 Letter from James Kim, NRC Staff, to Site Vice President, Vermont Yankee, "Vermont Yankee Audit of Entergy's Management of Regulatory Commitments (TAC NO. ME4209)," encl. at 3, 4 (Sept. 13, 2010) (Audit Report), *available at* ADAMS Accession No. ML102420206.

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

114 *See, e.g., AmerGen Energy Co., LLC* (License Renewal for Oyster Creek Nuclear Generating Station), LBP-06-7, 63 NRC 188, 207 (2006) (accepting licensee commitment as satisfying regulatory obligation).

115 *Pac. Gas & Elec. Co.* (Diablo Canyon Nuclear Power Plant, Units 1 & 2), CLI-03-2, 57 NRC 19, 29 (2003).

116 *See Hydro Res., Inc.* (P.O. Box 777, Crownpoint, NM 87313), CLI-06-1, 63 NRC 1, 5 (2006) (citation omitted) (assuming licensee noncompliance with commitments "would . . . transmogrify license proceedings into open-ended enforcement actions: that is, licensing boards would be required to keep license proceedings open for the entire life of the license so intervenors would have a continuing, unrestricted opportunity to raise charges of noncompliance.").

117 *Prairie Island*, CLI-10-27, slip op. at 10-11.

VI. CONCLUSION

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

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

Counsel for Entergy certifies that he has made a sincere effort to make himself available to listen and respond to the moving parties, and to resolve the factual and legal issues raised in the motion, and that his efforts to resolve the issues have been unsuccessful.

Respectfully submitted,

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Dated in Washington, D.C.
this 12th day of October 2010

DB1/65758396

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and 50-286-LR
	)	
ENTERGY NUCLEAR OPERATIONS, INC.	)	
	)	
(Indian Point Nuclear Generating Units 2 and 3)	)	
	)	October 12, 2010

CERTIFICATE OF SERVICE

I hereby certify that copies of the "Applicant's Answer to Amended Contention New York State 25 Concerning Aging Management of Embrittlement of Reactor Pressure Vessel Internals" and Supporting Attachments 1 through 4, dated October 12, 2010, were served this 12th day of October, 2010 upon the persons listed below, by first class mail and e-mail as shown below.

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

SUPPORTING ATTACHMENTS TO APPLICANT'S ANSWER TO AMENDED CONTENTION NEW YORK STATE 25 CONCERNING AGING MANAGEMENT OF EMBRITTLEMENT OF REACTOR PRESSURE VESSEL INTERNALS

Filed on October 12, 2010

SUPPORTING ATTACHMENTS
APPLICANT'S ANSWER TO AMENDED NYS-25

Attachment	Description
1	Excerpts from NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, Rev. 1 (Sept. 2005)
2	Excerpts from Vol. 2 of NUREG-1801, *Generic Aging Lessons Learned (GALL) Report* – Tabulation of Results, Rev. 1 (Sept. 2005)
3	Excerpt from NL-10-063, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Amendment 9 to License Renewal Application (LRA) – Reactor Vessel Internals Program" (July 14, 2010)
4	MRP-227, Rev. 0, *Pressurized Water Reactor Internals Inspection and Evaluation Guidelines* (Dec. 2008)

PROPOSED AMENDED CONTENTION NYS-25 (EMBRITTLEMENT OF RPV & RPV INTERNALS)

ATTACHMENT 1

Excerpts from SRP-LR

NUREG-1800, Rev. 1

Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants

Manuscript Completed: September 2005
Date Published: September 2005

Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

3.0 INTRODUCTION TO STAFF REVIEW OF AGING MANAGEMENT

The NRC project manager (PM) responsible for the safety review of the license renewal application (LRA) is responsible for assigning to appropriate NRC Office of Nuclear Reactor Regulation (NRR) divisions the review or audit of aging management reviews (AMRs) or aging management programs (AMPs) identified in the applicant's LRA. The PM should document to which organization each AMR or AMP is assigned. The assigned AMRs and AMPs should be reviewed per the criteria described in Sections 3.1 through 3.6 of this standard review plan (SRP-LR, NUREG-1800) for review of license renewal applications, as directed by the scope of each of these sections.

The NRC divisions that are usually assigned responsibility for the review of AMRs and AMPs are the Division of Engineering (DE), Division of System Safety Analysis (DSSA), and the Division of Regulatory Improvement Program (DRIP) License Renewal and Environmental Impacts Program (RLEP). Typically, the PM will assign DRIP/RLEP to review the AMRs and AMPs that the LRA identifies as being consistent with the GALL Report or NRC-approved precedents. As common exceptions to this assignment, the PM will assign to DE those AMRs and AMPs that address issues identified as emerging technical issues. Usually, AMRs and AMPs that are not in one of the aforementioned categories are assigned to DE.

Review of the AMPs requires assessment of ten program elements as defined in this SRP-LR. The NRC divisions assigned the AMP should review the ten program elements to verify their technical adequacy. For three of the ten program elements (corrective actions, confirmation process, and administrative controls) the NRC division responsible for quality assurance should verify that the applicant has documented a commitment in the FSAR Supplement to expand the scope of its 10 CFR Part 50, Appendix B program to address the associated program elements for each AMP. If the applicant chooses alternate means of addressing these three program elements (e.g., use of a process other than the applicant's 10 CFR Part 50, Appendix B program), the NRC divisions assigned to review the AMP should request that the Division responsible for quality assurance review the applicant's proposal on a case-by-case basis.

3.0.1 Background on the Types of Reviews

10 CFR 54.21(a)(3) requires that the LRA must demonstrate, for systems, structures, and components (SSCs) identified in the scope of license renewal and subject to an AMR pursuant to 10 CRF 54.21(a)(1), that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. This AMR consists of identifying the material, environment, aging effects, and the AMP(s) credited for managing the aging effects.

Sections 3.1 through 3.6 of this SRP-LR describe how the AMRs and AMPs are reviewed. One method that the applicant may use to conduct its AMRs is to satisfy the NUREG-1801 (GALL Report) recommendations. The applicant may choose to use methodology other than that in the GALL Report to demonstrate compliance with 10 CFR 54.21(a)(3).

As stated in the GALL Report:

> The GALL Report is a technical basis document to the SRP-LR, which provides the staff with guidance in reviewing a license renewal application. The GALL Report should be treated in the same manner as an approved topical report that is generically applicable. An applicant may reference the GALL Report in a license renewal application to

demonstrate that the programs at the applicant's facility correspond to those reviewed and approved in the GALL Report and that no further staff review is required, as described in the next paragraph. If the material presented in the GALL Report is applicable to the applicant's facility, the staff should find the applicant's reference to the GALL Report acceptable. In making this determination, the staff should consider whether the applicant has identified specific programs described and evaluated in the GALL Report. The staff, however, should not conduct a re-review of the substance of the matters described in the GALL Report. Rather, the staff should ensure that the applicant verifies that the approvals set forth in the GALL Report for generic programs apply to the applicant's programs. The focus of the staff review should be on augmented programs for license renewal. The staff should also review information that is not addressed in the GALL Report or is otherwise different from that in the GALL Report.

If an applicant takes credit for a program in the GALL Report, it is incumbent on the applicant to ensure that the plant program contains all the elements of the referenced GALL Report program. In addition, the conditions at the plant must be bounded by the conditions for which the GALL Report program was evaluated. The above verifications must be documented on-site in an auditable form. The applicant should include a certification in the license renewal application that the verifications have been completed and are documented on-site in an auditable form.

The GALL Report contains one acceptable way to manage aging effects for license renewal. An applicant may propose alternatives for staff review in its plant-specific license renewal application. Use of the GALL Report is not required, but its use should facilitate both preparation of a license renewal application by an applicant and timely, uniform review by the NRC staff.

In addition, the GALL Report does not address scoping of structures and components for license renewal. Scoping is plant-specific, and the results depend on the plant design and current licensing basis. The inclusion of a certain structure or component in the GALL Report does not mean that this particular structure or component is within the scope of license renewal for all plants. Conversely, the omission of a certain structure or component in the GALL Report does not mean that this particular structure or component is not within the scope of license renewal for any plants.

The GALL Report contains an evaluation of a large number of structures and components that may be in the scope of a typical LRA. The evaluation results documented in the GALL Report indicate that many existing, typical generic aging management programs are adequate to manage aging effects for particular structures or components for license renewal without change. The GALL Report also contains recommendations on specific areas for which generic existing programs should be augmented (require further evaluation) for license renewal and documents the technical basis for each such determination. In addition, the GALL Report identifies certain SSCs that may or may not be subject to particular aging effects, and for which industry groups are developing generic aging management programs or investigating whether aging management is warranted. To the extent the ultimate generic resolution of such an issue will need NRC review and approval for plant-specific implementation, as indicated in a plant-specific FSAR supplement, and reflected in the SER associated with a particular LR application, an amendment pursuant to 10 CFR 50.90 will be necessary.

In this SRP-LR, Subsection 3.X.2 (where X denotes number 1-6.) presents the acceptance criteria describing methods to determine whether the applicant has met the requirements of NRC's regulations in 10 CFR 54.21. Subsection 3.X.3 presents the review procedures to be followed. Some rows (line-items) in the AMR tables (in Chapters II through VIII of the GALL Report, Vol. II) establish the need to perform "further evaluations." The acceptance criteria for satisfying these "further evaluations" are found in Subsections 3.X.2.2. The related review procedures are provided in Subsections 3.X.3.2.

In Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," the NRC has endorsed an acceptable methodology for applicants to structure license renewal applications. Using the guidance described in the aforementioned Regulatory Guide, the applicant documents in the LRA whether its AMR line-item is consistent or not consistent with the GALL Report.

A portion of the AMR includes the assessment of the AMPs in the GALL Report. The applicant may choose to use an AMP that is consistent with the GALL Report AMP, or may choose a plant-specific AMP.

If a GALL Report AMP is selected to manage aging, the applicant may take one or more exceptions to specific GALL Report AMP program elements. However, any deviation or exception to the GALL Report AMP should be described and justified. Exceptions are portions of the GALL Report AMP that the applicant does not intend to implement.

In some cases, an applicant may choose an existing plant program that does not currently meet all the program elements defined in the GALL Report AMP. If this is the situation, the applicant may make a commitment to augment the existing program to satisfy the GALL Report AMP element prior to the period of extended operation. This commitment is an AMP enhancement.

Enhancements are revisions or additions to existing aging management programs that the applicant commits to implement prior to the period of extended operation. Enhancements include, but are not limited to, those activities needed to ensure consistency with the GALL Report recommendations. Enhancements may expand, but not reduce, the scope of an AMP.

An audit and review is conducted at the applicant's facility to evaluate those AMRs or AMPs that the applicant claims to be consistent with the GALL Report. An audit also includes technical assessments of exceptions or enhancements to the GALL Report AMP program elements. Reviews are performed to address those AMRs or AMPs related to emergent issues, stated to be not consistent with the GALL Report, or based on an NRC-approved precedent (e.g., AMRs and AMPs addressed in an NRC SER of a previous LRA). As a result of the criteria established in 10 CFR Part 54, and the guidance provided in SRP-LR, GALL Report, Regulatory Guide 1.188, and the applicant's exceptions and/or enhancements to a GALL Report AMP, the following types of AMRs and AMPs should be audited or reviewed by the NRC staff.

AMRs

- AMR results consistent with the GALL Report
- AMR results for which further evaluation is recommended by the GALL Report
- AMR results not consistent with or not addressed in the GALL Report

AMPs

- Consistent with GALL Report AMPs

- Plant-specific AMPs

FSAR Supplement

- Each LRA AMP will provide an FSAR Supplement which defines changes to the FSAR that will be made as a condition of a renewed license. This FSAR Supplement defines the aging management programs the applicant is crediting to satisfy 10 CFR 54.21(a)(3).
- The FSAR Supplement should also contain a commitment to implement the LRA AMP enhancement prior to the period of extended operation.

3.0.2 Applications with approved Extended Power Uprates

Extended power uprates (EPU) are licensing actions that some licensees have recently requested the NRC staff to approve. This can affect aging management. In a NRC staff letter to the Advisory Committee on Reactor Safeguards, dated October 26, 2004, (ADAMS Accession ML042790085), the NRC Executive Director for Operation states that, "All license renewal applications with an approved EPU will be required to perform an operating experience review and its impact on [aging] management programs for structures, and components before entering the period of extended operation." One way for an applicant with an approved EPU to satisfy this criterion is to document its commitment to perform an operating experience review and its impact on aging management programs for systems, structures, and components (SSCs) before entering the period of extended operation as part of its license renewal application. Such licensee commitments should be documented in the NRC staff's SER written in support of issuing a renewed license. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date. EPU impact on SSCs should be part of the license renewal review. If necessary, the PM will assign a responsible group to address EPU.

PROPOSED AMENDED CONTENTION NYS-25 (EMBRITTLEMENT OF RPV & RPV INTERNALS)

ATTACHMENT 2

Excerpts from GALL Report

NUREG-1801, Vol. 2, Rev. 1

Generic Aging Lessons Learned (GALL) Report

Tabulation of Results

Manuscript Completed: September 2005
Date Published: September 2005

Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

B2. REACTOR VESSEL INTERNALS (PWR) - WESTINGHOUSE

Systems, Structures, and Components

This section addresses the Westinghouse pressurized water reactor (PWR) vessel internals and consists of the upper internals assembly, the rod control cluster assemblies (RCCA) guide tube assemblies, the core barrel, the baffle/former assembly, the lower internal assembly, and the instrumentation support structures. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all structures and components that comprise the reactor vessel are governed by Group A or B Quality Standards.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the extended period of operation are included in IV.E.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A2).

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-1 (R-124)	IV.B2.4-b	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant	Changes in dimensions/ void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-2 (R-123)	IV.B2.4-a	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-3 (R-127)	IV.B2.4-e	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed
IV.B2-4 (R-126)	IV.B2.4-d	Baffle/former assembly Baffle/former bolts	Stainless steel	Reactor coolant	Changes in dimensions/ void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-5 (R-129)	IV.B2.4-h	Baffle/former assembly Baffle/former bolts	Stainless steel	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed
IV.B2-6 (R-128)	IV.B2.4-f	Baffle/former assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-7 (R-121)	IV.B2.3-b	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant	Changes in dimensions/ void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-8 (R-120)	IV.B2.3-a	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-9 (R-122)	IV.B2.3-c	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-10 (R-125)	IV.B2.4-c	Core barrel assembly Baffle/former assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-11 (R-144)	IV.B2.6-b	Instrumentation support structures Flux thimble guide tubes	Stainless steel	Reactor coolant	Changes in dimensions/ void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-12 (R-143)	IV.B2.6-a	Instrumentation support structures Flux thimble guide tubes	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed
IV.B2-13 (R-145)	IV.B2.6-c	Instrumentation support structures Flux thimble tubes	Stainless steel with or without chrome plating	Reactor coolant	Loss of material/ wear	Chapter XI.M37, "Flux Thimble Tube Inspection"	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-14 (R-137)	IV.B2.5-i	Lower internal assembly Clevis insert bolts	Stainless steel; nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed
IV.B2-15 (R-134)	IV.B2.5-f	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel; nickel alloy	Reactor coolant	Changes in dimensions/ void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-16 (R-133)	IV.B2.5-e	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel; nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-17 (R-135)	IV.B2.5-g	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed
IV.B2-18 (R-132)	IV.B2.5-c	Lower internal assembly Lower core plate	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-19 (R-131)	IV.B2.5-b	Lower internal assembly Lower core plate Radial keys and clevis inserts	Stainless steel; nickel alloy	Reactor coolant	Changes in dimensions/ void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-20 (R-130)	IV.B2.5-a	Lower internal assembly Lower core plate Radial keys and clevis inserts	Stainless steel; nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed
IV.B2-21 (R-140)	IV.B2.5-m	Lower internal assembly Lower support casting Lower support plate columns	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/ thermal aging and neutron irradiation embrittlement	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-22 (R-141)	IV.B2.5-n	Lower internal assembly Lower support forging Lower support plate columns	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed
IV.B2-23 (R-139)	IV.B2.5-l	Lower internal assembly Lower support forging or casting Lower support plate columns	Stainless steel	Reactor coolant	Changes in dimensions/ void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-24 (R-138)	IV.B2.5-k	Lower internal assembly Lower support forging or casting Lower support plate columns	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2-25 (R-136)	IV.B2.5-h	Lower internal assembly Lower support plate column bolts	Stainless steel; nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed
IV.B2-26 (R-142)	IV.B2.5-o	Lower internal assembly Radial keys and clevis Inserts	Stainless steel	Reactor coolant	Loss of material/ wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-27 (R-119)	IV.B2.2-e	RCCA guide tube assemblies RCCA guide tube bolts RCCA guide tube support pins	Stainless steel; nickel alloy	Reactor coolant	Changes in dimensions/ void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-28 (R-118)	IV.B2.2-d	RCCA guide tube assemblies RCCA guide tube bolts RCCA guide tube support pins	Stainless steel; nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-29 (R-117)	IV.B2.2-b	RCCA guide tube assemblies RCCA guide tubes	Stainless steel	Reactor coolant	Changes in dimensions/ void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-30 (R-116)	IV.B2.2-a	RCCA guide tube assemblies RCCA guide tubes	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed
IV.B2-31 (R-53)	IV.B2.1-m IV.B2.2-f IV.B2.1-c IV.B2.2-c IV.B2.3-d IV.B2.4-g IV.B2.5-p IV.B2.5-j IV.B2.5-d IV.B2.1-h	Reactor vessel internals components	Stainless steel; nickel alloy	Reactor coolant	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-32 (RP-24)	IV.B2.	Reactor vessel internals components	Stainless steel; nickel alloy	Reactor coolant	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR primary water	No
IV.B2-33 (R-108)	IV.B2.1-d	Upper internals assembly Hold-down spring	Stainless steel	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed
IV.B2-34 (R-115)	IV.B2.1-l	Upper internals assembly Upper core plate alignment pins	Stainless steel; nickel alloy	Reactor coolant	Loss of material/ wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-35 (R-110)	IV.B2.1-f	Upper internals assembly Upper support column	Stainless steel	Reactor coolant	Changes in dimensions/ void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-36 (R-109)	IV.B2.1-e	Upper internals assembly Upper support column	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed
IV.B2-37 (R-111)	IV.B2.1-g	Upper internals assembly Upper support column (only cast austenitic stainless steel portions)	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/ thermal aging and neutron irradiation embrittlement	Chapter XI.M13, “Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)”	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-38 (R-114)	IV.B2.1-k	Upper internals assembly Upper support column bolts	Stainless steel; nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed
IV.B2-39 (R-113)	IV.B2.1-j	Upper internals assembly Upper support column bolts Upper core plate alignment pins Fuel alignment pins	Stainless steel; nickel alloy	Reactor coolant	Changes in dimensions/ void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-40 (R-112)	IV.B2.1-i	Upper internals assembly Upper support column bolts Upper core plate alignment pins Fuel alignment pins	Stainless steel; nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-41 (R-107)	IV.B2.1-b	Upper internals assembly Upper support plate Upper core plate Hold-down spring	Stainless steel	Reactor coolant	Changes in dimensions/ void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
B2	Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
IV.B2-42 (R-106)	IV.B2.1-a	Upper internals assembly Upper support plate Upper core plate Hold-down spring	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" for PWR primary water No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment needs to be confirmed

XI.M13 THERMAL AGING AND NEUTRON IRRADIATION EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS)

Program Description

The reactor vessel internals receive a visual inspection in accordance with the American Society of Mechanical Engineers (ASME) Code Section XI, Subsection IWB, Category B-N-3. This inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) reactor vessel internals. This aging management program (AMP) includes (a) identification of susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (i.e., ferrite and molybdenum contents, casting process, and operating temperature) and/or neutron irradiation embrittlement (neutron fluence), and (b) for each "potentially susceptible" component, aging management is accomplished through either a supplemental examination of the affected component based on the neutron fluence to which the component has been exposed as part of the applicant's 10-year inservice inspection (ISI) program during the license renewal term, or a component-specific evaluation to determine its susceptibility to loss of fracture toughness.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program provides screening criteria to determine the susceptibility of CASS components to thermal aging on the basis of casting method, molybdenum content, and percent ferrite. The screening criteria are applicable to all primary pressure boundary and reactor vessel internal components constructed from SA-351 Grades CF3, CF3A, CF8, CF8A, CF3M, CF3MA, CF8M, with service conditions above 250°C (482°F). The screening criteria for susceptibility to thermal aging embrittlement are not applicable to niobium-containing steels; such steels require evaluation on a case-by-case basis. For "potentially susceptible" components, the program provides for the consideration of the synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement. For each such component, an applicant can implement either (a) a supplemental examination of the affected component as part of a 10-year ISI program during the license renewal term, or (b) a component-specific evaluation to determine the component's susceptibility to loss of fracture toughness.

   Based on the criteria set forth in the May 19, 2000 letter from Christopher Grimes, Nuclear Regulatory Commission (NRC), to Mr. Douglas Walters, Nuclear Energy Institute (NEI), the susceptibility to thermal aging embrittlement of CASS components is determined in terms of casting method, molybdenum content, and ferrite content. For low-molybdenum content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast low-molybdenum steels with ≤20% ferrite and all centrifugal-cast low-molybdenum steels are not susceptible. For high-molybdenum content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast high-molybdenum steels with ≤14% ferrite and centrifugal-cast high-molybdenum steels with ≤20% ferrite are not susceptible. In the susceptibility screening method, ferrite content is calculated by using the Hull's equivalent factors (described in NUREG/CR-4513, Rev. 1) or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values). A fracture toughness value of 255 kJ/m^2 (1,450 in.-lb/in.^{2}) at a crack depth of 2.5 mm (0.1 in.) is used to differentiate between

CASS materials that are nonsusceptible and those that are potentially susceptible to thermal aging embrittlement. Extensive research data indicate that for nonsusceptible CASS materials, the saturated lower-bound fracture toughness is greater than 255 kJ/m^2 (NUREG/CR-4513, Rev. 1).

2. ***Preventive Actions:*** The program consists of evaluation and inspection and provides no guidance on methods to mitigate thermal aging and neutron irradiation embrittlement.

3. ***Parameters Monitored/Inspected:*** The program specifics depend on the neutron fluence and thermal embrittlement susceptibility of the component. The AMP monitors the effects of loss of fracture toughness on the intended function of the component by identifying the CASS materials that either have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) or are determined to be susceptible to thermal aging embrittlement. For such materials, the program consists of either supplemental examination of the affected component based on the neutron fluence to which the component has been exposed, or component-specific evaluation to determine the component's susceptibility to loss of fracture toughness.

4. ***Detection of Aging Effects:*** For reactor vessel internal CASS components that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) or are determined to be susceptible to thermal embrittlement, the 10-year ISI program during the renewal period includes a supplemental inspection covering portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (i.e., ferrite and molybdenum contents, casting process, and operating temperature), neutron fluence, and cracking susceptibility (i.e., applied stress, operating temperature, and environmental conditions). The inspection technique is capable of detecting the critical flaw size with adequate margin. The critical flaw size is determined based on the service loading condition and service-degraded material properties. One example of a supplemental examination is enhancement of the visual VT-1 examination of Section XI IWA-2210. A description of such an enhanced visual VT-1 examination could include the ability to achieve a 0.0005-in. resolution, with the conditions (e.g., lighting and surface cleanliness) of the inservice examination bounded by those used to demonstrate the resolution of the inspection technique. Alternatively, the applicant may perform a component-specific evaluation, including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criterion requires continued use of the supplemental inspection program. For each CASS component that has been subjected to a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV) and is potentially susceptible to thermal aging, the supplement inspection program applies; otherwise, the existing ASME Section XI inspection requirements are adequate if the components are not susceptible to thermal aging embrittlement.

5. ***Monitoring and Trending:*** Inspections scheduled in accordance with IWB-2400 and reliable examination methods provide timely detection of cracks.

6. ***Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3500. Flaw tolerance evaluation for components with ferrite content up to 25% is performed according to the principles associated with IWB-3640 procedures for submerged arc welds (SAW), disregarding the Code restriction of 20% ferrite in IWB-3641(b)(1). Extensive research data indicate that the lower-bound

fracture toughness of thermally aged CASS materials with up to 25% ferrite is similar to that for SAWs with up to 20% ferrite (Lee et al., 1997). Flaw evaluation for CASS components with >25% ferrite is performed on a case-by-case basis by using fracture toughness data provided by the applicant.

7. ***Corrective Actions:*** Repair is performed in conformance with IWA-4000 and IWB-4000, and replacement in accordance with IWA-7000 and IWB-7000. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** The AMP was developed by using research data obtained on both laboratory-aged and service-aged materials. Based on this information, the effects of thermal aging embrittlement on the intended function of CASS components are effectively managed.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2005.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 2001 edition including the 2002 and 2003 Addenda, American Society of Mechanical Engineers, New York, NY.

Lee, S., Kuo, P. T., Wichman, K., and Chopra, O., *Flaw Evaluation of Thermally Aged Cast Stainless Steel in Light-Water Reactor Applications*, Int. J. Pres. Ves. and Piping, pp. 37-44, 1997.

Letter from Christopher I. Grimes, U.S. Nuclear Regulatory Commission, License Renewal and Standardization Branch, to Douglas J. Walters, Nuclear Energy Institute, License Renewal Issue No. 98-0030, *Thermal Aging Embrittlement of Cast Stainless Steel Components*, May 19, 2000, (ADAMS Accession No. ML003717179).

NUREG/CR-4513, Rev. 1, *Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems*, U.S. Nuclear Regulatory Commission, August 1994.

PROPOSED AMENDED CONTENTION NYS-25 (EMBRITTLEMENT OF RPV & RPV INTERNALS)

ATTACHMENT 3

Excerpts from NL-10-063, Letter from Fred Dacimo, Entergy, to NRC, "Amendment 9 to License Renewal Application (LRA) – Reactor Vessel Internals Program" (July 14, 2010)

Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249
Tel (914) 788-2055

Fred Dacimo
Vice President
License Renewal

NL-10-063

July 14, 2010

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Amendment 9 to License Renewal Application (LRA) - Reactor Vessel Internals Program
Indian Point Nuclear Generating Unit Nos. 2 & 3
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

REFERENCES:

1. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application" (NL-07-039)

2. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Boundary Drawings (NL-07-040)

3. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Environmental Report References (NL-07-041)

4. Entergy Letter dated October 11, 2007, F. R, Dacimo to Document Control Desk, "License Renewal Application (LRA)" (NL-07-124)

5. Entergy Letter November 14, 2007, F. R, Dacimo to Document Control Desk, "Supplement to License Renewal Application (LRA) Environmental Report References" (NL-07-133)

Dear Sir or Madam:

In the referenced letters, Entergy Nuclear Operations, Inc. applied for renewal of the Indian Point Energy Center operating license. This letter contains Amendment 9 to the License Renewal Application (LRA) regarding the Reactor Vessel Internals Program.

If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 7-14-2010.

Sincerely,

FRD/dmt

Attachment: 1. Amendment 9 to License Renewal Application – Reactor Vessel Internals Program

cc: Mr. Samuel J. Collins, Regional Administrator, NRC Region I
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
Mr. John Boska, NRR Senior Project Manager
Ms. Kimberly Green, Project Manager
NRC Resident Inspector's Office
Mr. Paul Eddy, New York State Department of Public Service
Mr. Francis J. Murray, President and CEO, NYSERDA

ATTACHMENT 1 TO NL-10-063

Amendment 9 to License Renewal Application -
<u>Reactor Vessel Internals Program</u>

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3
DOCKET NOS. 50-247 AND 50-286
LICENSE NOS. DPR-26 AND DPR-64

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION (LRA)
AMMENDMENT 9

The LRA is revised as described below. (underline – added, strikethrough – deleted)

2.3.1.2 Reactor Vessel Internals

The reactor vessel internals for each unit are described in the reactor coolant system description (Unit 2, Reactor Vessel Internals; Unit 3, Reactor Vessel Internals).

For both units, the lower core support structure, the upper core support structure, and the incore instrumentation support structure are the three major parts of the reactor internals.

Lower Core Support Structure

The major member of the reactor vessel internals is the lower core support structure consisting of the following components included in this evaluation.

core baffle/former assembly: bolts
core baffle/former assembly: plates
core barrel assembly: bolts, screws
core barrel assembly: axial flexure plates (thermal shield flexures), flange, ring, shell, thermal shield, lower core barrel flange weld, upper core barrel flange weld
core barrel assembly: outlet nozzles
lower internals assembly: clevis insert bolt
lower internals assembly: clevis insert
lower internals assembly: intermediate diffuser plate
lower internals assembly: fuel alignment pin
lower internals assembly: lower core plate
lower internals assembly: lower core support plate column sleeves
lower internals assembly: lower core support column bolt
lower internals assembly, lower core support column castings: column cap, lower core support
lower internals assembly: radial key
lower internals assembly: secondary core support (energy absorbing device)
specimen guides (not subject to aging management review)
specimen plugs (installed in IP2 only; not subject to aging management review)

The lower core support structure is supported at its upper flange from a ledge in the reactor vessel. Within the core barrel are a core baffle and a lower core plate, both of which are attached to the core barrel wall. The lower core support structure provides passageways for the coolant flow. The lower core plate at the bottom of the core below the baffle plates provides support and orientation for the fuel assemblies. Fuel alignment pins (two for each assembly) are also inserted into this plate. Columns are placed between the lower core plate and core support casting in order to provide stiffness and to transmit the core load to the core support casting. Adequate coolant distribution is obtained through the use of the lower core plate and a diffuser plate.

Upper Core Support Structure

The "top hat with deep beam features" upper core support structure consists of the following components included in this evaluation.

upper internals assembly, rod control cluster assembly (RCCA) guide tube assembly: bolts
upper internals assembly, RCCA guide tube assembly: guide tube (including lower flange weld), guide plates
upper internals assembly, RCCA guide tube assembly: support pin
upper internals assembly: core plate alignment pin
upper internals assembly: head/vessel alignment pin
upper internals assembly: hold-down spring
upper internals assembly: support column
upper internals assembly, mixing devices: support column orifice base, support column mixer
upper internals assembly: upper core plate, fuel alignment pin
upper internals assembly: support assembly (including ring), upper support plate
upper internals assembly: upper support column bolt

The support columns establish the spacing between the upper support assembly and the upper core plate and are fastened at top and bottom to these plates and beams.

The RCCA guide tube assemblies shield and guide the control rod drive shafts and control rods. They are fastened to the upper support and are guided by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the control rod shroud tube which is attached to the upper support plate and guide tube.

In-Core Instrumentation Support Structure

The in-core instrumentation support structures consist of the following components included in this evaluation.

thermocouple conduit
flux thimble guide tube
bottom mounted instrumentation column

An upper system (thermocouple conduit) is used to convey and support thermocouples penetrating the vessel through the head, and a lower system (flux thimble guide tube) is used to convey and support flux thimbles penetrating the vessel through the bottom.

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to in-line columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system.

Table 2.3.1-2-IP2 and Table 2.3.1-2-IP3 list the mechanical components subject to aging management review and component intended functions for the reactor vessel internals.

Table 3.1.2-2-IP2 and Table 3.1.2-2-IP3 provide the results of the aging management review for the reactor vessel internals.

Table 2.3.1-4-IP2
Reactor Vessel Internals
Components Subject to Aging Management Review

Component Type	Intended Function
Lower Core Support Structure	
Core baffle/former assembly •bolts	Structural support
Core baffle/former assembly •plates	Structural support Flow distribution Shielding
Core barrel assembly •bolts and screws	Structural support
~~Core barrel assembly~~ •~~axial flexure plates~~ •~~flange~~ •~~ring~~ •~~shell~~ •~~thermal shield~~	~~Structural support~~ ~~Flow distribution~~ ~~Shielding~~
Core barrel assembly • axial flexure plates (thermal shield flexures)	Structural support
Core barrel assembly • flange	Structural support
Core barrel assembly • ring • shell • thermal shield	Structural support Flow distribution Shielding
Core barrel assembly • lower core barrel flange weld • upper core barrel flange weld	Structural support

Core barrel assembly •outlet nozzles	Flow distribution
Lower internals assembly •clevis insert bolt •clevis insert •fuel alignment pin •lower core support plate column sleeves •lower core support plate column bolt •radial key	Structural support
Lower internals assembly •intermediate diffuser plate	Flow distribution
Lower internals assembly •lower core plate •lower core support castings •column cap •lower core support •secondary core support	Structural support Flow distribution
Upper Core Support Structure—Upper Internals Assembly	
~~RCCA guide tube assembly~~ ~~•bolt~~ ~~•guide tube~~ ~~•support pin~~	~~Structural support~~
RCCA guide tube assembly •bolt	Structural support
RCCA guide tube assembly •guide tube (including lower flange welds)	Structural support
RCCA guide tube assembly •guide plates	Structural support
RCCA guide tube assembly •support pin	Structural support

Core plate alignment pin	Structural support
Head / vessel alignment pin	Structural support
Hold-down spring	Structural support
Mixing devices •support column orifice base •support column mixer	Structural support Flow distribution
Support column	Structural support
Upper core plate, fuel alignment pin	Structural support Flow distribution
Upper support plate, support assembly (including ring)	Structural support
Upper support column bolt	Structural support
Incore Instrumentation Support Structure	
Bottom mounted instrumentation column	Structural support
Flux thimble guide tube	Structural support
Thermocouple conduit	Structural support

3.1.2.1.2 Reactor Vessel Internals

Materials

Reactor vessel internals components are constructed of the following materials.

- cast austenitic stainless steel
- nickel alloy
- stainless steel

Environment

Reactor vessel internals components are exposed to the following environments.

- neutron fluence
- treated borated water
- treated borated water > 140°F
- treated borated water > 482°F

Aging Effects Requiring Management

The following aging effects associated with the reactor vessel internals require management.

- change in dimensions
- cracking
- cracking – fatigue
- loss of material
- loss of material – wear
- loss of preload
- reduction of fracture toughness

Aging Management Programs

The following aging management programs manage the aging effects for reactor vessel internals components.

- Inservice Inspection
- Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)
- Reactor Vessel Internals
- Water Chemistry Control – Primary and Secondary

3.1.2.2.6 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement and Void Swelling

Loss of fracture toughness due to neutron irradiation embrittlement and change in dimensions (void swelling) ~~could occur~~ in stainless steel and nickel alloy reactor vessel internals components exposed to reactor coolant and neutron flux will be managed by the Reactor Vessel Internals (RVI) Program. The RVI Program will implement the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, MRP-227. The RVI Program will use nondestructive examinations (NDE) and other inspection methods to manage aging effects for reactor vessel internals. ~~To manage loss of fracture toughness in vessel internals components, IPEC will (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval. This commitment is included in the UFSAR Supplement, Appendix A, Sections A.2.1.41 and A.3.1.41.~~

3.1.2.2.9 Loss of Preload due to Stress Relaxation

Loss of preload due to thermal stress relaxation (creep) would only be a concern in very high temperature applications (> 700°F) as stated in the ASME Code, Section II, Part D, Table 4. No IPEC internals components operate at > 700°F. Therefore, loss of preload due to thermal stress relaxation (creep) is not an applicable aging effect for the reactor vessel internals components. However, irradiation-enhanced creep (irradiation creep) or irradiation enhanced stress relaxation (ISR) is an athermal process that depends on the neutron fluence and stress; and, on void swelling if present. ~~Nevertheless~~ Therefore, loss of preload of stainless steel and nickel alloy reactor vessel internals components will be managed by the Reactor Vessel Internals (RVI) Program. The RVI Program will implement the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, MRP-227. The RVI Program will use nondestructive examinations (NDE) and other inspection methods to manage aging effects for reactor vessel internals. ~~to the extent that industry developed reactor vessel internals aging management programs address these aging effects. The IPEC commitment to these RVI programs is included in UFSAR Supplement, Appendix A, Sections A.2.1.41 and A.3.1.41.~~

3.1.2.2.15 Changes in Dimensions due to Void Swelling

Changes in dimensions due to void swelling ~~could occur~~ in stainless steel and nickel alloy reactor internal components exposed to reactor coolant will be managed by the Reactor Vessel Internals (RVI) Program. The RVI Program will implement the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, MRP-227. The RVI Program will use nondestructive examinations (NDE) and other inspection methods to manage aging effects for reactor vessel internals. ~~To manage changes in dimensions of vessel internals components, IPEC will (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as~~

~~applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval. This commitment is included in the UFSAR Supplement, Appendix A, Sections A.2.1.41 and A.3.1.41.~~

3.1.2.2.17 Cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking

Cracking due to stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), and irradiation-assisted stress corrosion cracking (IASCC) ~~could occur~~ in PWR stainless steel and nickel alloy reactor vessel internals components will be managed by the Reactor Vessel Internals (RVI) Program. The RVI Program will implement the EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, MRP-227. The RVI Program will use nondestructive examinations (NDE) and other inspection methods to manage aging effects for reactor vessel internals. ~~To manage cracking in vessel internals components, IPEC maintains the Water Chemistry Control – Primary and Secondary Program and will (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval. The IPEC commitment to these RVI programs is included in UFSAR Supplement, Appendix A, Sections A.2.1.41 and A.3.1.41.~~

A.2.1.41 Reactor Vessel Internals Aging Management Activities

The Reactor Vessel Internals (RVI) Program is a new plant specific program to manage aging effects of reactor vessel internals using the guidance from the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP). The MRP inspection and evaluation (I&E) guidelines for managing the effects of aging on pressurized water reactor vessel internals are presented in MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." The MRP also developed inspection requirements specific to the inspection methods delineated in MRP-227, as well as requirements for qualification of the nondestructive examination (NDE) systems used to perform those inspections. These inspection requirements are presented in MRP-228, "Materials Reliability Program: Inspection Standard for PWR Internals."

MRP-227 and MRP-228 provide the basis of the IPEC Reactor Vessel Internals (RVI) Program. Revisions to MRP-227 and MRP-228, including any changes resulting from the NRC review of the documents (issued as MRP-227-A and MRP-228-A) will be incorporated into the IPEC RVI Program. The RVI Program will monitor the effects of aging degradation mechanisms on the intended function of the internals through periodic and conditional examinations. The RVI Program will detect and evaluate cracking, loss of material, reduction of fracture toughness, loss of preload and dimensional changes of vessel internals components in accordance with MRP-227 inspection requirements and evaluation acceptance criteria.

The IPEC RVI Program will be implemented and maintained in accordance with the guidance in NEI 03-08 [Addenda], Addendum A, "RCS Materials Degradation Management Program Guidelines." Any deviations from mandatory, needed, or good practice implementation requirements established in MRP-227 or MRP-228, will be dispositioned in accordance with the NEI 03-08 implementation protocol. The RVI Program will be implemented prior to the period of extended operation. ~~To manage loss of fracture toughness, cracking, change in dimensions (void swelling), and loss of preload in vessel internals components, the site will (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.~~

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Section B.1.42 of the LRA is completely new.

B.1.42 Reactor Vessel Internals Program

Program Description

The Reactor Vessel Internals Program is a new plant-specific program. Revision 1 of NUREG-1801 includes no aging management program description for PWR reactor vessel internals. NUREG-1801, Section XI.M16, PWR Vessel Internals, instead defers to the guidance provided in Chapter IV line items as appropriate. The Chapter IV line item guidance recommends actions to:

> ". . .(1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval."

The industry programs for investigating and managing aging effects on reactor internals are part of the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP). The MRP developed inspection and evaluation (I&E) guidelines for managing the effects of aging on pressurized water reactor vessel internals. These guidelines are presented in MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." The I&E guidelines include:

- summary descriptions of PWR internals and functions;
- summary of the categorization and aging management strategy development of potentially susceptible locations, based on the safety and economic consequences of aging degradation;
- direction for methods, extent, and frequency of one-time, periodic, and conditional examinations and other aging management methodologies;
- acceptance criteria for the one-time, periodic, and conditional examinations and other aging management methodologies; and
- methods for evaluation of aging effects that exceed the examination acceptance criteria.

The MRP also developed inspection procedure requirements specific to the inspection methods delineated in MRP-227, as well as requirements for qualification of the nondestructive examination (NDE) systems used to perform those inspections. These inspection procedure requirements are presented in MRP-228, "Materials Reliability Program: Inspection Standard for PWR Internals."

MRP-227 and MRP-228 provide the basis of the IPEC Reactor Vessel Internals (RVI) Program. Revisions to MRP-227 and MRP-228, including any changes resulting from the NRC review of the documents (issued as MRP-227-A and MRP-228-A), will be incorporated into the IPEC RVI Program.

The RVI Program will monitor the effects of aging on the intended function of the internals through periodic and conditional examinations. The RVI Program will detect and evaluate cracking, loss of material, reduction of fracture toughness, loss of preload and dimensional changes of vessel internals components in accordance with MRP-227 inspection recommendations and evaluation acceptance criteria.

IPEC will implement and maintain the RVI Program in accordance with the guidance in NEI 03-08 [Addenda], Addendum A, "RCS Materials Degradation Management Program Guidelines." Any deviations from mandatory, needed, or good practice implementation activities established in MRP-227 or MRP-228, will be managed in accordance with the NEI 03-08 implementation protocol.

Evaluation

1. **Scope of Program**

MRP-227 guidelines are applicable to reactor internal structural components. The scope does not include consumable items such as fuel assemblies and reactivity control assemblies which are periodically replaced based on neutron flux exposure. The scope does not include welded attachments to the reactor vessel which are considered part of the vessel, or nuclear instrumentation (flux thimble tubes) which forms part of the reactor coolant pressure boundary. Other programs manage the effects of aging on these components.

MRP-227 separates PWR internals components into four groups depending on (1) their susceptibility to and tolerance of aging effects, and (2) the existence of programs that manage the effects of aging. These groupings include:

- Primary – those internals components that are highly susceptible to the effects of at least one aging mechanism (identified in Table 4-3 of MRP-227);
- Expansion – those internals components that are highly or moderately susceptible to the effects of at least one aging mechanism, but for which functionality assessment has shown a degree of tolerance to those effects (identified in Table 4-6 of MRP-227);
- Existing Programs – those internals components that are susceptible to the effects of at least one aging mechanism and for which generic and plant-specific existing AMP elements are capable of managing those effects (identified in Table 4-9 of MRP-227); and
- No Additional Measures – those internals components for which the effects of aging mechanisms are below the MRP-227 screening criteria (internals components not included in Tables 4-3, 4-6 or 4-9 of MRP-227).

The categorization of internals components for Westinghouse PWRs, as presented in MRP-227, applies to IPEC Unit 2 and Unit 3 vessel internals. The component inspections identified in MRP-227, Tables 4-3 and 4-6 for primary and expansion group components, define the scope of the IPEC RVI Program inspections. Those components subject to aging management by existing programs, as delineated in MRP-227, Table 4-9, are included in the scope of those programs, and are not part of the RVI Program inspections. Components that are not included in Tables 4-3, 4-6 or 4-9 are considered to be within the scope of the program, but require no specific inspections.

2. Preventive Actions

The Reactor Vessel Internals Program is a condition monitoring program that does not include preventive actions. However, primary water chemistry is maintained in accordance with EPRI guidelines by the Water Chemistry Control - Primary and Secondary Program, which minimizes the potential for stress corrosion cracking (SCC) and irradiation assisted stress corrosion cracking (IASCC).

Plant operations also influence aging of the vessel internals. The general assumptions about plant operations used in the development of the MRP-227 guidelines are applicable to the IPEC units. The units are base loaded and implemented low leakage core loading patterns within the first 30 years of operation. IPEC has implemented no design changes to reactor vessel internals beyond those identified in general industry guidance or recommended by Westinghouse.

3. Parameters Monitored or Inspected

The RVI Program will monitor the effects of aging on the intended function of the internals through periodic and conditional examinations and other aging management methods, as required. As described in MRP-227, the program contains elements that will monitor and inspect for the parameters that indicate the progress of each of these effects. The program will use NDE techniques to detect loss of material through wear, identify distortion of components, and locate cracks.

Visual examinations (VT-3) will be used to detect wear. Visual examinations (VT-3) will also detect distortion or cracking through indications such as gaps or displacement along component joints and broken or damaged bolt locking systems. Direct measurements of spring height will be used to detect distortion of the internals hold down spring. Visual examinations (EVT-1) will be used to detect crack-like surface flaws of components and welds. Volumetric (ultrasonic) examinations will be used to locate cracking of bolting. (MRP-227, Tables 4-3 and 4-6)

4. Detection of Aging Effects

The RVI Program will detect cracking, loss of material reduction of fracture toughness, loss of preload and dimensional changes (distortion) of vessel internals components in accordance with MRP-227. The NDE systems (i.e., the combinations of equipment, procedure, and personnel) used to detect these aging effects will be qualified in accordance with MRP-228. The RVI Program will conduct inspections of primary group components as follows (MRP-227, Table 4-3):

- Periodic visual examinations (VT-3) will detect loss of material due to wear from control rod guide tube guide plates and thermal shield flexure plates.
- Periodic visual examinations (VT-3) of the baffle former assembly plates and edge bolts will detect symptoms of distortion due to void swelling or cracking from IASCC. These symptoms include abnormal interactions with fuel assemblies, gaps or displacement along component joints, broken or damaged bolt locking systems, and failed or missing bolts.
- Direct measurements of spring height will detect distortion of the internals hold down spring due to a loss of stiffness. Measurements will be taken periodically, as needed to determine the life of the spring.
- Periodic visual examinations (EVT-1) will detect crack-like surface flaws of the control rod guide tube assembly lower flange welds and the upper core barrel to flange weld.
- Volumetric (UT) examinations will locate cracking of baffle former bolting. Baseline and subsequent measurements will be used to confirm the stability of the bolting pattern.

Indications from EVT-1 or UT inspections may result in additional inspections of expansion group components, as determined by expansion criteria delineated in MRP-227, Table 5-3. The relationships between primary group component inspection findings and additional inspections of expansion group components are as follows.

- Indications from the EVT-1 inspections of the control rod guide tube assembly lower flange welds may result in EVT-1 inspections of the lower support column bodies and VT-3 inspections of bottom mounted instrumentation column bodies to detect cracking.
- Indications from the EVT-1 inspection of the upper core barrel to flange weld may result in EVT-1 inspections of the remaining core barrel welds
- Indications from the UT inspections of baffle former bolting may result in UT inspections of the lower support column bolts and the barrel former bolts for cracking.

5. Monitoring and Trending

The RVI Program uses the inspection guidelines for PWR internals in MRP-227. Inspections in accordance with these guidelines will provide timely detection of aging effects. In addition to the inspections of primary group components, expansion group components have been defined should the scope of examination and re-examination need to be expanded beyond the primary group. Records of inspection results are maintained allowing for comparison with subsequent inspection results.

IPEC will share inspection results with the industry in accordance with the good practice recommendations of MRP-227. The IPEC-specific results will be incorporated into an overall industry report that will track industry progress and will aid in evaluation of potentially significant issues, identification of fleet trends, and determination of any needed revisions to MRP-227 guidelines.

6. Acceptance Criteria

The RVI Program acceptance criteria are from Section 5 of MRP-227. Table 5-3 of MRP-227 provides the acceptance criteria for inspections of the primary and expansion group components. The criteria for expanding the examinations from the primary group components to include the expansion group components are also delineated in MRP-227, Table 5-3. The examination acceptance criteria include: (i) specific, descriptive relevant conditions for the visual (VT-3) examinations; (ii) requirements for recording and dispositioning surface breaking indications that are detected and sized for length by the visual (EVT-1) examinations; and (iii) requirements for system-level assessment of bolted assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits.

7. Corrective Action

Conditions adverse to quality; such as failures, malfunctions, deviations, defective material and equipment, and nonconformances; are promptly identified and corrected. In the case of significant conditions adverse to quality, measures are implemented to ensure that the cause of the nonconformance is determined and that corrective action is taken to preclude recurrence. In addition, the cause of the significant condition adverse to quality and the corrective action implemented is documented and reported to appropriate levels of management. The Entergy (10 CFR Part 50, Appendix B) Quality Assurance Program, including relevant corrective action controls, applies to the RVI Program.

Any detected condition that does not satisfy the examination acceptance criteria must be processed through the corrective action program. Example methods for analytical disposition of unacceptable conditions are discussed or referenced in Section 6 of MRP-227. These methods or other demonstrated and verified alternative methods may be used. The alternative of component repair and replacement of PWR internals is subject to the applicable requirements of the ASME Code Section XI.

8. Confirmation Process

This attribute is discussed in Section B.0.3.

9. Administrative Controls

This attribute is discussed in Section B.0.3.

10. Operating Experience

Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. However, PWR internals aging degradation has been observed in European PWRs, specifically with regard to cracking of baffle-former bolting. For this reason, the U.S. PWR owners and operators created a program to inspect the baffle-former bolting to determine whether similar aging degradation might be expected to occur in U.S. plants. A benefit of this decision was the experience gained with the UT examination techniques used in the inspections.

In addition, the industry began laboratory testing projects to gather the materials data necessary to support future inspections and evaluations. Other confirmed or suspected material degradation concerns that the industry has identified for PWR components are wear in thimble tubes, potential wear in control rod guide tube guide plates, and cracking in some high-strength bolting. The industry has addressed the last concern primarily through replacement of high-strength bolting with bolt material that is less susceptible to cracking and by improved control of pre-load.

The RVI Program established in accordance with the MRP-227 guidelines is a new program. Accordingly, there is no direct programmatic history for IPEC. However, program inspections will use qualified techniques similar to those successfully used at IPEC and throughout the industry for ASME Section XI Code inspections. Internals inspections (VT-3) have been conducted at IPEC in accordance with ASME Section XI Code requirements, with no indications of component degradation. IPEC has appropriately responded to industry operating experience for reactor vessel internals. For example, guide tube support pins (split pins) have been replaced in both units on the basis of industry experience. As with other U.S. commercial PWR plants, cracking of baffle former bolts is recognized as a potential issue for the IPEC units. As a result, IPEC has monitored industry developments and recommendations regarding these components.

Development of the MRP-227 guidelines is based upon industry operating experience, research data, and vendor evaluations. Reactor vessel internals aging degradation incidents in both U.S. and foreign plants were considered in the development of the MRP-227 guidelines. As implemented, this program will account for applicable future operating experience during the period of extended operation.

Conclusion

The RVI Program will be effective at managing aging effects since it will incorporate proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls in accordance with MRP-227 and MRP-228 guidelines and current IPEC programs. The RVI Program will provide reasonable assurance that the effects of aging are managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis through the period of extended operation.