UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
Entergy Nuclear Operations, Inc.	)	Docket Nos.
(Indian Point Nuclear Generating	)	50-247-LR
Units 2 and 3)	)	and 50-286-LR

RIVERKEEPER INITIAL STATEMENT OF POSITION REGARDING CONTENTION RK-TC-2 (FLOW ACCELERATED CORROSION)

December 22, 2011

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In accordance with 10 C.F.R. § 2.1207(a)(1), the Atomic Safety and Licensing Board's ("ASLB") July 1, 2010 Scheduling Order,[1] the ASLB's June 7, 2011 Amended Scheduling Order,[2] the ASLB's November 17, 2011 Order,[3] and the ASLB's October 18, 2011 Order clarifying the procedures for evidentiary filings,[4] Riverkeeper, Inc. ("Riverkeeper"), hereby submits this Initial Statement of Position on Riverkeeper Contention TC-2 – Flow Accelerated Corrosion. This statement is supported by the Prefiled Direct Testimony of Dr. Joram Hopenfeld (Riverkeeper Exhibit RIV000003), and exhibits thereto (Riverkeeper Exhibits RIV000004 to RIV000033). Dr. Hopenfeld's testimony and supporting exhibits demonstrate that Entergy does not have an adequate program to manage the aging effects of flow accelerated corrosion ("FAC") during the proposed period of extended operation ("PEO").

BACKGROUND

On or about April 23, 2007, Entergy Nuclear Operations, Inc. ("Entergy") filed a License Renewal Application ("LRA") with the U.S. Nuclear Regulatory Commission ("NRC") seeking 20-year extended operating licenses for Indian Point nuclear generating Units 2 and 3.[5] The LRA purported to include a sufficient and legally acceptable program for managing an aging phenomenon known as flow accelerated corrosion ("FAC") throughout the proposed PEO.

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[1] In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-0247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, Scheduling Order (July 1, 2010), at ¶ K.1.

[2] In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-0247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, Amended Scheduling Order (June 7, 2011), at 3.

[3] In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-0247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, Order (Granting Unopposed Motion by the State of New York and Riverkeeper, Inc. to Amend the Scheduling Order) (November 17, 2011).

[4] In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-0247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, Order (Clarification of Procedures for Evidentiary Filings (October 18, 2011).

[5] Indian Point Energy Center License Renewal Application, *available at*, http://www.nrc.gov/reactors/operating/licensing/renewal/applications/indian-point/ipec_lra_1_2.pdf (hereinafter "Indian Point LRA").

Pursuant to 10 C.F.R. § 2.309 and Federal Register notices published by the NRC,[6] on November 30, 2007 Riverkeeper filed a request for hearing and petition to intervene in the Indian Point license renewal proceedings, proffering, *inter alia*, a contention (Riverkeeper Contention TC-2) challenging the adequacy of Entergy's LRA for failure to demonstrate an effective program for managing FAC at the facility.[7]

In particular, Riverkeeper Contention TC-2 asserts that "Entergy's program for management of FAC is deficient because it has not demonstrated that components in the Indian Point nuclear power plant that are within the scope of the license renewal rule and are vulnerable to FAC will be adequately inspected and maintained during the license renewal term."[8] Riverkeeper Contention TC-2 explains that Entergy's program for managing FAC is inadequate due to reliance on the computer code CHECWORKS without sufficient benchmarking or a track record of performance at Indian Point's power uprate levels, and because Entergy's FAC program does not provide sufficient details to demonstrate that susceptible plant components will be adequately maintained during the PEO.[9]

On July 31, 2008, the ASLB admitted Riverkeeper Contention TC-2 for an adjudicatory hearing, finding that the contention "raises questions regarding the sufficiency of Entergy's AMP [aging management program] to demonstrate that a specific class of components subject to FAC

[6] Entergy Nuclear Operations, Inc., Indian Point Nuclear Generating Unit Nos. 2 and 3, Notice of Acceptance for Docketing of the Application and Notice of Opportunity for Hearing Regarding Renewal of Facility Operating License Nos. DPR-26 and DPR-64 for an Additional 20-Year Period, 72 Fed. Reg. 42,134 (August 1, 2007), as amended by Entergy Nuclear Operations, Inc., Indian Point Nuclear Generating Unit Nos. 2 and 3, Notice of Opportunity for Hearing Regarding Renewal of Facility Operating License Nos. DPR-26 and DPR-64 for an Additional 20-Year Period: Extension of Time for Filing of Requests for Hearing or Petitions for Leave To Intervene in the License Renewal Proceeding, 72 Fed. Reg. 55,834 (Oct. 1, 2007).

[7] Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene in the License Renewal Proceedings for the Indian Point Nuclear Power Plant (November 30, 2007), ADAMS Accession No. ML073410093, at 15-23 (hereinafter "Riverkeeper Petition to Intervene").

[8] Riverkeeper Petition to Intervene at 16.

[9] Riverkeeper Petition to Intervene at 16, 20-23.

will be managed so that their intended functions will be maintained during the period of extended operations."[10] Entergy subsequently filed a motion seeking summary disposition of Riverkeeper Contention TC-2.[11] The ASLB denied this motion "because genuine issues of material fact regarding the adequacy of the Applicant's plan to manage the effects of flow-accelerated corrosion (FAC) during the proposed period of extended operation must be resolved on the merits after an evidentiary hearing."[12]

The applicable law and regulatory requirements, along with the facts, testimony and evidence relating to Riverkeeper Contention TC-2 are described below.

APPLICABLE LEGAL AND REGULATORY REQUIREMENTS

NRC's regulations require nuclear power plant license renewal applicants to have programs for effectively managing the aging of in-scope plant systems, structures, and components. In particular, 10 C.F.R. § 54.21(a)(3) states that for each system, structure and component that is within the scope of NRC license renewal requirements, applicants must "demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB [current licensing basis] for the period of

[10] *See* In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, Memorandum and Order (Ruling on Petitions to Intervene and Requests for Hearing) (July 31, 2008), ADAMS Accession No. ML082130436, at 167-169 (hereinafter "ASLB July 31, 2008 Contention Admissibility Order").

[11] Applicant's Motion for Summary Disposition of Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion) (July 26, 2010), ADAMS Accession No. ML102140430, (hereinafter "Applicant's Motion for Summary Disposition of RK-TC-2").

[12] In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-0247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, Memorandum and Order (Ruling on Entergy's Motion for Summary Disposition of Riverkeeper TC-2 (Flow-Accelerated Corrosion)) (November 4, 2010), at 1.

extended operation." The "ultimate burden of proof on the question of whether the permit or the license should be issued is . . . upon the applicant."[13]

According to applicable guidance contained in NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,* (hereinafter "SRP-LR"), an aging management program ("AMP") sufficient to meet the regulatory standard should include and implement ten separate elements: (1) the scope of the program; (2) actions for prevention and mitigation of aging degradation; (3) parameters for monitoring and inspecting so as to detect the presence and extent of aging effects; (4) detection of aging effects prior to loss of the structure and component intended function; (5) trending activities to provide predictability of the extent of degradation, in order to effect timely corrective and mitigative actions; (6) acceptance criteria against which the need for corrective actions will be evaluated; (7) timely corrective actions when acceptance criteria are not met; (8) confirmation processes to ensure adequate preventative actions, and complete and effective corrective actions; (9) administrative controls to provide for formal review and approval mechanisms; and (10) consideration of plant-specific and industry operating experience.[14]

NRC's NUREG-1801, *Generic Aging Lessons Learned (GALL) Report* (hereinafter "*GALL Report*"), a technical basis document referenced in NUREG-1800, provides license renewal applicants with guidance regarding how an AMP can satisfy the 10 program elements identified in SRP-LR. An AMP that is consistent with the *GALL Report* is acceptable to show

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[13] *Amergen Energy Co.* (Oyster Creek Nuclear Generating Station), CLI-09-7, 69 N.R.C. 235, 269 (2009); *see id.* at 263 (applicant must demonstrate that it satisfies the "reasonable assurance standard" by a preponderance of the evidence).

[14] NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, Rev. 1 (September 2005), at A.1-3 to A.1-7, *available at* http://pbadupws.nrc.gov/docs/ML0521/ML052110007.pdf (hereinafter NUREG-1800, Rev. 1) (Exhibit NYS000195); NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, Rev. 2 (December 2010), at A.1-3 to A.1-7; *available at* http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1800/r2/sr1800r2.pdf (hereinafter NUREG-1800, Rev. 2) (Exhibit NYS000161).

compliance with NRC's regulatory standard in 10 C.F.R. § 54.21(a)(3). However, applicants cannot generically claim consistency with this guidance document, and instead must "provide a reasonably thorough description of its AMP to show conclusively how th[e] program will ensure that the effects of aging will be managed."[15] In contrast, an applicant

> merely stating that its AMP meets NUREG-1801 without any specificity falls short of the required demonstration. . . . [W]hether an applicant is successful depends upon whether it is [sic] has shown that the specific plant details of its AMP have adequately addressed this guidance. But a bald reference to NUREG-1801 fails to show how the recommendations of NUREG-1801 are proposed to be implemented for [the facility] . . . and does not demonstrate that the effects of aging are adequately managed for the plant.[16]

Section XI.M17, "Flow-Accelerated Corrosion" of the *GALL Report*, Revision 1, as well as the more recent *GALL Report*, Revision 2, contain guidance relative to an acceptable FAC AMP.[17] The *GALL Report* indicates that an applicant's FAC program can be based on Electric Power Research Institute ("EPRI") guidelines in the Nuclear Safety Analysis Center (NSAC)-202L-R2 (or R3), and, in summary, "includes performing (a) an analysis to determine critical locations, (b) limited baseline inspections to determine the extent of thinning at these locations, and (c) follow-up inspections to confirm predictions, or repairing or replacing components as necessary."[18]

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[15] *Entergy Nuclear Vermont Yankee* (Vermont Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 870 (Nov. 24, 2008).

[16] *Entergy Nuclear Vermont Yankee*, 68 NRC 763, 871.

[17] *See* NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, Rev. 1 (September 2005), at pp. XI M-61 to XI M-63, *available at* http://pbadupws.nrc.gov/docs/ML0521/ML052110006.pdf (hereinafter "*GALL Report*, Rev 1") (Exhibit NYS000146C); NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, Rev. 2 (December 2010), at pp. XI M17-a to XI M17-4, *available at* http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1801/r2/sr1801r2.pdf (hereinafter "*GALL Report*, Rev. 2") (Exhibit NYS000147D)

[18] *See GALL Report*, Rev. 1 at XI M-61 (Exhibit NYS000146C); *GALL Report*, Rev. 2 at XI M17-1 (Exhibit NYS000147D).

The *GALL Report* indicates that an acceptable FAC program "includes the use of a predictive code, such as CHECWORKS."[19] Such a code "is used to predict component degradation in the systems conducive to FAC, as indicated by plant specific data, including material, hydrodynamic, and operating conditions."[20] Inspections based on the results of such a predictive computer model should provide "reasonable assurance that structural integrity will be maintained between inspections" and "ensure that the extent of wall thinning is adequately determined, that intended function will not be lost, and that corrective actions are adequately identified."[21] In relation to the "acceptance criteria" program element, the *GALL Report* explains that a predictive code such as CHECWORKS is used "to calculate the number of refueling or operating cycles remaining before the component reaches the minimum allowable wall thickness," in order to determine the need for some form of corrective action.[22]

The most recent revision of the *GALL Report*, issued in December 2010, provides elucidation regarding the appropriateness of using a predictive computer code:

> CHECWORKS is acceptable because it provides a bounding analysis for FAC. The *analysis is bounding* because in general the predicted wear rates and component thicknesses are *conservative* when compared to actual field measurements. *It is recognized that CHECWORKS is not always conservative in predicting component thickness; therefore, when measurements show the predictions to be non-conservative, the model must be re-calibrated using the latest field data*.[23]

[19] *See GALL Report*, Rev. 1 at XI M-61 (Exhibit NYS000146C); *GALL Report*, Rev. 2 at XI M17-1 (Exhibit NYS000147D)

[20] *See GALL Report*, Rev. 1 at XI M-61 (Exhibit NYS000146C); *GALL Report*, Rev. 2 at XI M17-1 (Exhibit NYS000147D)

[21] *See GALL Report*, Rev. 1 at XI M-62 (Exhibit NYS000146C); *GALL Report*, Rev. 2 at XI M17-2 (Exhibit NYS000147D)

[22] *See GALL Report*, Rev. 1 at XI M-62 (Exhibit NYS000146C); *GALL Report*, Rev. 2 at XI M17-2 (Exhibit NYS000147D)

[23] *GALL Report*, Rev. 2 at XI M17-1 to XI M17-2 (emphasis added) (Exhibit NYS000147D).

The *GALL Report* otherwise provides further explanation regarding how a license renewal applicant can satisfy the various program elements required for a legally sufficient license renewal AMP.

As reflected in NRC's guidance documents discussed above, the American Society of Mechanical Engineers ("ASME") code requires that licensees maintain minimum design wall thicknesses of nuclear power plant piping during the entire period of plant operations.[24] Additionally, NRC's General Design Criterion 4 requires that plant structures, systems and components be able to "accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents" and "be appropriately protected against dynamic effects . . . that may result from equipment failures and from events and conditions outside the nuclear power unit."[25]

<u>ARGUMENT</u>

I. RIVERKEEPER'S WITNESS

Riverkeeper's witness in support of Riverkeeper Contention TC-2 is Dr. Joram Hopenfeld. Dr. Hopenfeld has submitted pre-filed testimony[26] and an expert report[27] pertaining to Riverkeeper Contention TC-2. Dr. Hopenfeld's professional and educational qualifications are described in his *curriculum vitae*.[28] Dr. Hopenfeld is an expert in the field relating to nuclear

[24] ASME B31.3; ASME Code Section III, Paragraph NB-3200.

[25] 10 C.F.R. Part 50, Appendix A, General Design Criteria for Nuclear Power Plants, *Criterion 4—Environmental and dynamic effects design bases*.

[26] Prefiled Written Testimony of Dr. Joram Hopenfeld Regarding Riverkeeper Contention TC-2 – Flow Accelerated Corrosion, Dec. 21, 2011 (hereinafter "Hopenfeld TC-2 Testimony") (Exhibit RIV000003).

[27] Report of Dr. Joram Hopenfeld in Support of Riverkeeper Contention TC-2 – Flow Accelerated Corrosion (hereinafter "Hopenfeld TC-2 Report") (Exhibit RIV000005).

[28] *Curriculum Vitae* of Dr. Joram Hopenfeld (Exhibit RIV000004).

power plant aging management.[29] Dr. Hopenfeld is a mechanical engineer, holds a doctorate in mechanical engineering, and has 45 years of professional experience in the fields of thermal-hydraulics, material/environment interaction instrumentation, design, project management, and nuclear safety regulation, including 18 years in the employ of the NRC.[30]

Dr. Hopenfeld's extensive professional experience has afforded him knowledge and expertise regarding the material degradation phenomenon known as FAC.[31] He has published numerous peer-reviewed papers in the area of corrosion, and hold patents related to monitoring of wall thinning of piping components.[32] Furthermore, Dr. Hopenfeld has knowledge and expertise regarding the use of the CHECWORKS computer code dating back to 1988, when it was known as CHEC.[33] Most recently, he was a technical consultant and expert witness for the New England Coalition in the Vermont Yankee license renewal proceeding, where he testified at an adjudicatory hearing concerning FAC and CHECWORKS.[34]

Dr. Hopenfeld's testimony and opinions related to Riverkeeper Contention TC-2 are based on his technical expertise in, and experience with, the relevant issues.

II. RIVERKEEPER'S EVIDENCE

The testimony, facts, and evidence provided by Dr. Hopenfeld, as described in detail below, unequivocally demonstrate that Entergy's proposed AMP for FAC fails to comply with 10 C.F.R. § 54.21(a)(3) and other relevant regulations and regulatory guidance, including SRP-

[29] *Curriculum Vitae* of Dr. Joram Hopenfeld (Exhibit RIV000004)

[30] *Curriculum Vitae* of Dr. Joram Hopenfeld (Exhibit RIV000004).

[31] Hopenfeld TC-2 Testimony at 1-2 (Exhibit RIV000003).

[32] *See Curriculum Vitae* of Dr. Joram Hopenfeld (Exhibit RIV000004); Hopenfeld TC-2 Testimony at 2 (Exhibit RIV000003).

[33] Hopenfeld TC-2 Testimony at 2 (Exhibit RIV000003).

[34] *See Curriculum Vitae* of Dr. Joram Hopenfeld (Exhibit RIV000004); Hopenfeld TC-2 Testimony at 2 (Exhibit RIV000003).

LR and the *GALL Report*. The evidence presented shows that Entergy's program does not assure that the aging effects of FAC will be adequately managed, or that Indian Point Units 2 and 3 will operate safely, throughout the proposed 20-year license renewal periods.

A. The Nature and Safety Significant of FAC

As Dr. Hopenfeld explains in detail in his testimony and expert report, FAC is a pipe wall thinning phenomenon in which the thinning rate is accelerated by flow velocity.[35] FAC includes wall thinning by impingement corrosion, electrochemical corrosion, erosion-corrosion, cavitation-erosion, and metal dissolution.[36] The main causes of FAC include turbulence, intensity, steam quality, material compositions, oxygen content, and coolant pH.[37] Wall thinning resulting from FAC is a local phenomenon affected by local geometry, local metal composition, and local turbulences.[38] Once local corrosion has begun, FAC may progress at a non-linear rate.[39] As local turbulence and local flow velocity are not directly measured quantities, it is difficult to identify locations where FAC rates are highest.[40]

FAC poses a significant safety risk at nuclear power plants if left undetected. When FAC reduces wall thickness *below* the minimum design value, the subject component may leak or rupture.[41] A FAC-induced rupture of a high pressure component or pipe may have very serious safety consequences.[42] Dr. Hopenfeld discusses numerous instances of undetected FAC which

[35] Hopenfeld TC-2 Testimony at 3-4 (Exhibit RIV000003); Hopenfeld TC-2 Report at 2-3 (RIV000005).

[36] Hopenfeld TC-2 Testimony at 3-4 (Exhibit RIV000003); Hopenfeld TC-2 Report at 2 (RIV000005).

[37] Hopenfeld TC-2 Testimony at 4 (Exhibit RIV000003); Hopenfeld TC-2 Report at 2 (RIV000005) .

[38] Hopenfeld TC-2 Testimony at 4 (Exhibit RIV000003); Hopenfeld TC-2 Report at 2 (RIV000005).

[39] Hopenfeld TC-2 Testimony at 4 (Exhibit RIV000003); Hopenfeld TC-2 Report at 2 (RIV000005).

[40] Hopenfeld TC-2 Report at 2 (RIV000005).

[41] Hopenfeld TC-2 Testimony at 4 (Exhibit RIV000003); Hopenfeld TC-2 Report at 2-3 (RIV000005).

[42] For this reason, the ASME code, as incorporated into the elements outlined in SRP-LR for an adequate AMP, specifically requires that components and pipes do not operate below design limit wall thicknesses. *See* ASME B31.3; ASME Code Section III, Paragraph NB-3200.

have resulted in catastrophic events, including a pipe rupture at the Surry nuclear power plant in 1986 that resulted in several fatalities and FAC in the secondary loop at the Mihama nuclear power plant that resulted in the deaths of several workers.[43]

B. Entergy's Program for Managing FAC During the PEO

Sections A.2.1.14 and B.1.15 of Entergy's LRA describe Entergy's FAC AMP as "[a]n existing program that applies to safety-related and non-safety related carbon and low alloy steel components in systems containing high-energy fluids carrying two-phase or single-phase high energy fluid ≥ 2% of plant operating time."[44] The LRA explains that Entergy's program is based on EPRI guidelines in Nuclear Safety Analysis Center (NSAC)-202L, (Revision 3), "*Recommendations for an Effective Flow-Accelerated Corrosion Program*,"[45] which outlines how to predict, detect, and monitor FAC in piping and other pressure retaining components.[46] The LRA explains that the FAC program "includes (a) an evaluation to determine critical locations, (b) initial operational inspections to determine the extent of thinning at these locations,

[43] Hopenfeld TC-2 Testimony at 4 (Exhibit RIV000003); Hopenfeld TC-2 Report at 3 (RIV000005); NRC Information Notice No 86-106, "Feed Water Line Break" (December 16, 1986) (Exhibit RIV000006); NRC Bulletin 87-01, "Thinning Pipe Walls in Nuclear Plants" (July 9, 1987) (Exhibit RIV000007); NRC Information Notice 1991-019, "Steam Generator Feed Water Distribution Piping Damage" (March 12, 1991) (Exhibit RIV000008); Monitoring Report 5-93-0042, Steam Generator Feedring Nozzle Through Wall Erosion (June 15, 1993) (Exhibit RIV000009); NRC Information Notice 1997-084, "Rupture of Extraction Steam Piping" (December 11, 1997) (Exhibit RIV000010); NRC Information Notice 2006-008, "Secondary Piping Rupture at Mihama Power Station in Japan (March 16, 2006) (Exhibit RIV000011).

[44] Indian Point LRA § A.2.1.14, p.A-24, *available at*, http://www.nrc.gov/reactors/operating/licensing/renewal/applications/indian-point/ipec-lra-appendix-a.pdf; Indian Point LRA § B.1.15, p.B-54, *available at*, http://www.nrc.gov/reactors/operating/licensing/renewal/applications/indian-point/ipec-lra-appendix-b.pdf.

[45] EPRI, Recommendations for an Effective Flow-Accelerated Corrosion Program, NSAC-202L-R3 (hereinafter "NSAC-202L-R3") (Exhibit RIV000012). While Entergy's LRA states that Entergy's FAC management program is based on NSAC-202L Revision 2, Entergy subsequently amended the program to identify its use of the more recent Revision 3. *See* NL-07-153, Letter From Fred R. Dacimo, Entergy, to NRC Document Control Desk, "Amendment 1 to License Renewal Application (LRA)," Attach. 1 at 46-48 (Dec. 18, 2007), ADAMS Accession No. ML073650195. Entergy explains that, in this respect, Entergy's FAC AMP is not consistent with, and takes "exception" to, the *GALL Report*, Rev. 1 (which was the most recent version of GALL at the time Entergy amended the FAC program).

[46] *See* Indian Point LRA § A.2.1.14, p.A-24; Indian Point LRA § B.1.15, p.B-54.

and (c) follow-up inspections to confirm predictions, or repair or replace components as necessary."[47] Entergy claims that the FAC program at Indian Point includes the ten program elements identified in SRP-LR and the *GALL Report*. This has been memorialized in Entergy's Aging Management Program Evaluation Report, IP-RPT-06-LRD07, Revision 5.[48]

Entergy purports to implement its FAC program, and the guidance and recommendations contained in the *GALL Report* and NSAC-202L-R3 pertaining thereto, via a fleet-wide procedure, EN-DC-315, Revision 3, *Flow Accelerated Corrosion Program* (March 1, 2010).[49] This procedure requires piping and piping component inspections to be conducted, and ultrasonic thickness measurements to be performed to determine pipe wall thickness.[50] Entergy's method of selecting components for wall measurements and determining the time between successive thickness measurements is primarily based on predictions generated from the computer code, CHECWORKS. Entergy asserts that the criteria for selecting components for FAC inspections are consistent with the criteria in NSAC-202L-R3, and that the selection is based on "(1) actual pipe wall thickness measurements from past outages; (2) predictive evaluations performed using the CHECWORKS code; (3) industry experience related to FAC; (4) results from other plant inspection programs; and (5) engineering judgment."[51] As reflected in the guidance contained in NSAC-202L-R3 and in Entergy's implementing procedure, EN-DC-315, Entergy's FAC management program is predominantly based on the use of the computer program

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[47] *See* Indian Point LRA § A.2.1.14, p.A-24; Indian Point LRA § B.1.15, p.B-54

[48] Aging Management Program Evaluation Report, IP-RPT-06-LRD07, Revision 5 (Exhibit RIV000014).

[49] *See* Applicant's Motion for Summary Disposition of RK-TC-2, *supra* Note 11 at 9-10.

[50] EN-DC-315, Rev. 3, *Flow Accelerated Corrosion Program* (March 1, 2010) (Exhibit RIV000015); *see also* Applicant's Motion for Summary Disposition of RK-TC-2, *supra* Note 11 at 10.

[51] *See* Applicant's Motion for Summary Disposition of RK-TC-2, *supra* Note 11 at 10, 12; NSAC-202L-R3 (Exhibit 000012).

CHECWORKS to record plant operating experience and predict timing and locations of wall thinning.

C. The Inadequacies of Entergy's FAC Management Program

Dr. Hopenfeld's testimony and expert report, along with the exhibits in support thereof, demonstrate that Entergy's program for managing FAC at Indian Point demonstrably fails to assure that timely detection of FAC-related wall thinning will occur during the PEO, that Entergy can prevent component wall thickness from being reduced below minimum design values, or that Indian Point will operate safely throughout the proposed license renewal periods. Dr. Hopenfeld testifies that Entergy's AMP for FAC is deficient for the following particular reasons.

i. Entergy's FAC Program Improperly Relies upon CHECWORKS Computer Modeling

Entergy's program for managing FAC at Indian Point during the PEO fails to comply with 10 C.F.R. § 54.21(a)(3), SRP-LR, and the *GALL Report* because it depends upon the CHECWORKS computer code, which is improperly benchmarked and ineffective at predicting FAC-related wall thinning occurrences.

1. The Nature of the CHECWORKS Computer Code

The CHECWORKS computer code is software that "was developed as a predictive tool to assist utilities in planning inspections and evaluating the inspection data to prevent piping failures caused by FAC."[52] As Dr. Hopenfeld explains, because FAC is an unpredictable phenomenon, CHECWORKS is based on statistics, meaning that it is based on a collection of selective data which represents only a fraction of the total flow area.[53] Accordingly,

[52] NSAC-202L-R3 at p.1-1 (RIV000012).

[53] Hopenfeld TC-2 Testimony at 4 (Exhibit RIV000003); Hopenfeld TC-2 Report at 3 (RIV000005).

CHECWORKS will not produce reliable predictive results unless it is adequately calibrated, or benchmarked.[54] The NRC has recognized that benchmarking analytic codes is necessary, stating that "analytical methods and codes are assessed and benchmarked against measurement data. . . . The validation and benchmarking process provides the means to establish the associated biases and uncertainties."[55]

Dr. Hopenfeld discusses that when plant parameters change, re-calibration of the CHECWORKS code to update the model becomes necessary.[56] This is because changes in power output affect various plant parameters, including velocities, temperatures, coolant chemistry, and steam moisture.[57] EPRI has acknowledged that "even small power uprates can have a *significant* affect on FAC rates."[58]

2. <u>The CHECWORKS Computer Code is not Properly Benchmarked and Completely Fails to Provide Reliable Predictive Wall Thinning Results at Indian Point</u>

In light of the fact that CHECWORKS is strictly based on an empirical model and the fact that the operating conditions at Indian Point Units 2 and 3 changed in 2004 and 2005 due to 3.26% and 4.85% power increases, respectively,[59] Riverkeeper Contention TC-2 asserted that Entergy needed to verify that CHECWORKS was properly calibrated and benchmarked, and that

[54] Hopenfeld TC-2 Testimony at 4-5 (Exhibit RIV000003); Hopenfeld TC-2 Report at 3-4 (RIV000005).

[55] Safety Evaluation by the Office of Nuclear Reactor Regulations Related to Amendment No. 229 to Facility Operating License No. DPR-28 Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Vermont Power Station, Docket No. 50-571), at § 2.8.7.1, p. 190, ADAMS Accession No. ML060050028 (Exhibit RIV000013).

[56] Hopenfeld TC-2 Testimony at 4-5 (Exhibit RIV000003); Hopenfeld TC-2 Report at 4 (RIV000005).

[57] Hopenfeld TC-2 Testimony at 4-5 (Exhibit RIV000003); Hopenfeld TC-2 Report at 4 (RIV000005).

[58] NSAC-202L-R3, at p.4-5 (RIV000012).

[59] *See* NRC Staff Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Units 2 and 3, Docket Nos. 50-247, 50-286, NUREG-1930, Volume 2, at pp. 3-25 to 3-29, ADAMS Accession No. ML093170671 (discussing power uprate); *see also* Applicant's Motion for Summary Disposition of RK-TC-2, *supra* Note 11 at 19-21.

approximately 10-15 years would be required to do so.[60] Entergy has consistently maintained that no such calibration is required because the power uprates at Indian Point were small in comparison to, and therefore bounded by, power increases at other plants; since the code was allegedly designed to account for changes in plant parameters; and because data from four to five outages per generating unit would adequately calibrate the CHECWORKS model to account for the power uprate conditions prior to the proposed life extension period.[61] However, the evidence acquired and reviewed since the filing of Riverkeeper's original contention in 2007 clearly refutes Entergy's position, and undeniably demonstrates that the CHECWORKS code is not properly calibrated for effective use at Indian Point during the PEO.

As Dr. Hopenfeld explains, whether or not CHECWORKS is adequately benchmarked can be determined by assessing the degree to which the model can accurately predict wall thinning.[62] The predictive capability of the code can be evaluated by comparing wall thickness predictions generated by CHECWORKS with actual thickness measurements.[63] After performing an extensive analysis of years worth of CHECWORKS comparison data, Dr.

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[60] Riverkeeper Petition to Intervene at 20-21.

[61] *See* Answer of Entergy Nuclear Operations, Inc. Opposing Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene (January 22, 2008), at 60, ADAMS Accession No. ML080300071 ("before Entergy enters the period of extended operation, there will be at least three additional sets of inspection data, based on the current refueling outage, schedule, . . . to calibrate the CHECWORKS models to reflect changes in plant conditions. . . . *each and every* additional set of data serves to improve the accuracy of the IPEC CHECWORKS models" (emphasis in original)); Applicant's Motion for Summary Disposition of RK-TC-2, *supra* Note 11 at 19-21 (CHECWORKS was designed, and has been shown, to handle changes in chemistry, flow rate and or other operating conditions . . . . The 3.26% and 4.85% SPUs at IPEC plainly are bounded by the largest (20%) EPU approved by the NRC to date . . . . Additionally, by the time IP2 and IP3 each enter the PEO, inspection data for at least four to five refueling outages under SPU conditions will be available. Future outage inspection data will be used to calibrate the CHECWORKS predictions to provide a good fit to the post-SPU wear rates at IPEC.").

Notably, Entergy has never supported its position with any data showing that CHECWORKS has been successful in predicting FAC at plants that have had a higher power level change than those at Indian Point. Nor has Entergy ever identified a threshold percent change in plant operating parameters at which there would be no material effect on CHECWORKS results.

[62] Hopenfeld TC-2 Testimony at 5-7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 5 (RIV000005).

[63] Hopenfeld TC-2 Testimony at 5-7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 5 (RIV000005).

Hopenfeld has concluded that the model produces highly unreliable and non-conservative component wear predictions.[64] In Dr. Hopenfeld's expert opinion, this indicates that the CHECWORKS model has never been properly benchmarked, that the model is certainly not currently benchmarked to account for changes in plant operating parameters that occurred at Indian Point Units due to power uprates, and that it will be impossible to properly calibrate the model before Indian Point enters the proposed PEO.[65]

In particular, Dr. Hopenfeld has reviewed more than 6,500 data points in approximately 400 graphs contained in CHECWORKS modeling reports.[66] These data points represent wear predictions of component wall thickness versus actual measurements obtained.[67] Entergy provided this data in relation to FAC inspections performed at Indian Point Unit 2 during refueling outages 14, 16, 17, and 18, and in relation to FAC inspections performed at Indian Point Unit 3 prior to refueling outage 12, as well as during refueling outages 12, 13, 14 and 15.[68] This data generated both before *and* after the changes in plant operating conditions at Indian Point due to power uprates that occurred at Unit 2 in 2004, and at Unit 3 and 2005.[69] However, even though CHECWORKS was introduced in the early 1990s Entergy did not have any CHECWORKS related documentation related to Indian Point Unit 2 generated prior to the year

[64] Hopenfeld TC-2 Testimony at 5-7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 5 (RIV000005).

[65] Hopenfeld TC-2 Testimony at 5-7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 5 (RIV000005).

[66] Hopenfeld TC-2 Testimony at 5-7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 5 (RIV000005). Riverkeeper has excerpted the graphs from 24 CHECWORKS modeling reports provided by Entergy, hereinafter cited as "CHECWORKS Graphs," and compiled them into supporting exhibits RIV00016A and RIV00016B. The covers of each report as well an introductory page that indicates the relevant outage to which the report pertains, is included, and precede the respective data graphs. Additionally, the full titles of each report, including Entergy batestamp designations where provided, are provided in Table 1 of Dr. Hopenfeld's expert report.

[67] Hopenfeld TC-2 Testimony at 5-7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 5 (RIV000005).

[68] *See* CHECWORKS Graphs (Exhibits RIV00016A and RIV00016B).

[69] *See* CHECWORKS Graphs (Exhibits RIV00016A and RIV00016B).

2000.[70] In addition, Entergy did not disclose any CHECWORKS related documentation related to Indian Point Unit 3 generated prior to 2001, because to the extent any such documents existed, locating them "would be extremely burdensome."[71] As a result, the data reviewed by Dr. Hopenfeld represents only a small portion of the total plant data that was allegedly used to benchmark CHECWORKS at Indian Point. According to a discovery ruling by the ASLB, Entergy cannot rely upon any earlier data that was not provided to demonstrate that CHECWORKS is adequately benchmarked or that the program has a track record of performance.[72]

Based on his review and analysis of the CHECWORKS data that was provided, Dr. Hopenfeld has concluded that the computer model as employed at Indian Point is highly inaccurate and produces results that demonstrate a complete lack of correlation between component wear predictions and actual wall thickness measurements.[73] Dr. Hopenfeld explains that if there was a perfect correlation, the data would fall on the 45° line that appears in each graph, but that, instead, the data exhibits a wide scatter.[74] Dr. Hopenfeld further explains that the x-axes of the graphs indicate a prediction of zero wear, and that data points that fall between the 45° line and x-axis, represent non-conservative predictions.[75] Dr. Hopenfeld's review of all of

[70] *See* In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-0247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, Order (Ruling on Riverkeeper's Motion to Compel) (November 4, 2010), at 3.

[71] *See* In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-0247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, Order (Ruling on Riverkeeper's Motion to Compel) (November 4, 2010), at 4.

[72] *See* In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-0247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, Order (Ruling on Riverkeeper's Motion to Compel) (November 4, 2010), at 5.

[73] *See* Hopenfeld TC-2 Testimony at 6 (Exhibit RIV000003); Hopenfeld TC-2 Report at 6 (RIV000005).

[74] *See* Hopenfeld TC-2 Testimony at 6 (Exhibit RIV000003); Hopenfeld TC-2 Report at 6 (RIV000005); *see also* CHECWORKS Graphs (Exhibits RIV00016A and RIV00016B).

[75] *See* Hopenfeld TC-2 Testimony at 6 (Exhibit RIV000003); Hopenfeld TC-2 Report at 6 (RIV000005).

Entergy's plotted data points revealed that CHECWORKS yielded non-conservative predictions about 40-60% of the time.[76] Dr. Hopenfeld's expert report includes Table 1, which details his findings with respect to the degree of non-conservative predictions, in relation to each CHECWORKS report he reviewed.[77] Dr. Hopenfeld has also observed that the CHECWORKS predictions yield widely different measured points, even though with an ideal correlation, each predicted point would have a single measured value.[78]

In addition, Dr. Hopenfeld has explained that the degree of inaccuracy of Entergy's actual wear measurements to CHECWORKS' predictions has been very high. Dr. Hopenfeld observes that there are two lines on every graph designated +50% and -50% that imply that the data within those lines is bounded within 50%.[79] However, Dr. Hopenfeld clarifies that these lines are very misleading, and actually represent a wide margin of error: the +50% line indicates a wear ratio that varies by a factor of .7, while the -50% line indicates a wear ratio that varies by a factor of 2.[80] Dr. Hopenfeld explains how Entergy has not provided any rationale or justification regarding how data within the arbitrary +/-50% lines is acceptable or provides an adequate basis for determining criteria for inspection frequency or component replacement.[81]

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[76] *See* Hopenfeld TC-2 Testimony at 6 (Exhibit RIV000003); Hopenfeld TC-2 Report at 6 (RIV000005).

[77] Hopenfeld TC-2 Report at Table 1, Column (A) (pages 9-12) (RIV000005).

[78] Hopenfeld TC-2 Testimony at 6 (Exhibit RIV000003); Hopenfeld TC-2 Report at 6-7 (RIV000005).

[79] Hopenfeld TC-2 Testimony at 6 (Exhibit RIV000003); Hopenfeld TC-2 Report at 6 (RIV000005); *see also* V.K. Chexal, W.H. Layman, & J.S. Horowitz, Tackling the Single-Phase Erosion Corrosion Issue, To Be Presented at the American Power Conference April 18-20, 1988, Chicago Illinois (EPRI) (stating at the American Power Conference in 1988 that the computer model was "predicting erosion-corrosion rates within a +/-50% band"); *see also* CHECWORKS™ Steam/Feedwater Application, Guidelines for Plant Modeling and Evaluation of Component Inspection Data, 1009599, Final Report, September 2004, IPECPROP00000271, at p.6-7, IPECPROP00000334 (RIV000018).

[80] Hopenfeld TC-2 Testimony at 6 (Exhibit RIV000003); Hopenfeld TC-2 Report at 7 (RIV000005); *see also* Rudolf H. Hausler, Flow Assisted Corrosion (FAC) and Flow Induced Localized Corrosion: Comparison and Discussion, at 1, 8 (RHH Rebuttal, June 2, 2008, Vermont Yankee License Renewal Proceeding, 06-849-03-LR (explaining the misleading nature of the +/-50% lines) (RIV000019).

[81] Hopenfeld TC-2 Testimony at 6 (Exhibit RIV000003); Hopenfeld TC-2 Report at 7 (RIV000005).

Moreover, Dr. Hopenfeld has observed that many data points fall *outside* the +/-50% lines, indicating that CHECWORKS cannot even conservatively bound the data within a factor of 2.[82] Dr. Hopenfeld found many data points for which CHECWORKS over-predicted or under predicted FAC wear by more than a factor of 10. Table 1, column (B) in Dr. Hopenfeld's expert report documents Dr. Hopenfeld's findings with respect to the amount of data that fell outside the already wide uncertainty range represented by the area within +/-50% lines.[83] In Dr. Hopenfeld's opinion, the high degree of over- and under-prediction exhibited by a significant number of components demonstrates that CHECWORKS as used at Indian Point cannot predict FAC with any degree of precision.[84] To the contrary, CHECWORKS can only predict a range of corrosion that is far too wide for practical applications.[85]

Dr. Hopenfeld further explains that each graph includes a "line correction factor" or "LCF."[86] Entergy's documentation that the LCF "indicates the degree to which CHECWORKS over or under-predicts wear."[87] Entergy relies upon LCFs to "compare and adjust CHECWORKS predictions to match inspection data."[88] An LCF of 1 would indicate an exact

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[82] Hopenfeld TC-2 Testimony at 6 (Exhibit RIV000003); Hopenfeld TC-2 Report at 7-8 (RIV000005).

[83] *See* Hopenfeld TC-2 Report at Table 1, Column B (pages 9-12) (RIV000005).

[84] Hopenfeld TC-2 Testimony at 6-7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 7-8 (RIV000005).

[85] Hopenfeld TC-2 Testimony at 7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 8 (RIV000005).

[86] Hopenfeld TC-2 Testimony at 7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 8 (RIV000005).

[87] CSI, Technologies, Inc., Indian Point Unit 3 CHECWORKS SFA Model, Calculation No. 0705.100-01, Revision 2, August 2, 2011, at p.26 (Exhibit RIV000020).

[88] Applicant's Motion for Summary Disposition of RK-TC-2, *supra* Note 11 at Attachment 2, ¶ 48.

agreement between CHECWORKS predictions and actual wall thickness measurements.[89] None of the graphs provided have an LCF of 1, except for those figures with no data in them.[90]

Entergy's documentation states that a "reasonable LCF should be between 0.5 and 2.5."[91] Dr. Hopenfeld explains that Entergy has failed justify the conclusion that this range is acceptable, or how a data plot with an LCF within this range would indicate that CHECWORKS can be used to accurately predict inspection locations.[92] Furthermore, Dr. Hopenfeld has observed that the LCF was reported to be outside this allegedly "acceptable" range *numerous* times.[93] This is a clear demonstration that, even by Entergy's own rubric, CHECWORKS is unreasonably failing to predict wear rates. Table 1, column (C) in Dr. Hopenfeld's expert report memorializes his findings with respect to the number of instances where LCF was outside the range identified by Entergy as acceptable.[94]

In summary, Dr. Hopenfeld's review of all of the available CHECWORKS comparison data demonstrated a complete lack of correlation between predictions and measurements, indicating very poor predictive accuracy of the CHECWORKS model at Indian Point. The CHECWORKS model exhibits highly erratic predictive behavior, making it impossible to determine whether future wall thinning predictions will stay conservative or non-conservative.[95]

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[89] CSI, Technologies, Inc., Indian Point Unit 3 CHECWORKS SFA Model, Calculation No. 0705.100-01, Revision 2, August 2, 2011, at p.26 (Exhibit RIV000020); Hopenfeld TC-2 Testimony at 7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 8 (RIV000005).

[90] *See, e.g.*, CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS SFA Model, CSI Calculation No. 0705.101-01, Revision A, November 17, 2008, Page J-5 of 36) (Exhibit RIV00016A at p. 127).

[91] *See* CSI, Technologies, Inc., Indian Point Unit 3 CHECWORKS SFA Model, Calculation No. 0705.100-01, Revision 2, August 2, 2011, at p.26 (Exhibit RIV000020); Hopenfeld TC-2 Testimony at 7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 8 (RIV000005).

[92] Hopenfeld TC-2 Testimony at 7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 8 (RIV000005).

[93] Hopenfeld TC-2 Testimony at 7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 8 (RIV000005).

[94] Hopenfeld TC-2 Report at Table 1, Column (C) (pages 9-12) (RIV000005).

[95] Hopenfeld TC-2 Testimony at 7 (Exhibit RIV000003); Hopenfeld TC-2 Report at 8 (RIV000005).

Thus, the CHECWORKS code ineffective for objective quantitative assessments. In Dr. Hopenfeld's professional opinion, when a model is not capable of correlating predictions with measurements, whether the predictions are conservative or non-conservative, as a matter of sound engineering and science, such a model cannot be considered a suitable tool for informing predictions related to FAC.[96]

Dr. Hopenfeld observed that CHECWORKS wear predictions over 11 years showed that predictions were not improving at all with time, which demonstrates a lack of benchmarking.[97] Furthermore, Dr. Hopenfeld concludes that the level of correlation produced by the model under post-power uprate conditions is not acceptable, and that there is no way the model will be properly calibrated before Indian Point enters the rapidly approaching proposed PEO.[98] Despite Entergy's claims that the level of power increase at Indian Point is bounded in the computer model by higher power uprates at other plants, CHECWORKS has not been able to account for the changes in plant parameters that have occurred at Indian Point. This completely deflates Entergy's reliance upon the additional outages before the proposed PEO (hardly any of which remain), to adequately calibrate the CHECWORKS model to account for the power uprate conditions.[99]

Based on the foregoing, it is evident that CHECWORKS is not a viable or effective tool for selecting and prioritizing piping and piping component locations at Indian Point for inspections and wall thickness measurements during outages to timely detect and mitigate FAC during the proposed PEO.

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[96] Hopenfeld TC-2 Testimony at 9-10 (Exhibit RIV000003); Hopenfeld TC-2 Report at 13 (RIV000005).

[97] Hopenfeld TC-2 Testimony at 10 (Exhibit RIV000003); Hopenfeld TC-2 Report at 13 (RIV000005).

[98] Hopenfeld TC-2 Testimony at 10 (Exhibit RIV000003); Hopenfeld TC-2 Report at 13 (RIV000005).

[99] Hopenfeld TC-2 Testimony at 10 (Exhibit RIV000003); Hopenfeld TC-2 Report at 13 (RIV000005).

3. The Implications of CHECWORKS' Poor Predictive Accuracy

Dr. Hopenfeld explains that CHECWORKS predictions of wall thinning by FAC at Indian Point plant are too inaccurate to prevent pipe wall thickness from being reduced below minimum design values.[100] Non-conservative predictions affect plant safety because they fail to indicate when a component is reaching a critical wall thickness and thereby result in untimely component inspection and replacement.[101] Dr. Hopenfeld provides examples in his expert report to help explain the safety consequences of relying on CHECWORKS to predict FAC.[102] For example, if a pipe has an initial wall thickness of 0.5 inches, a minimum design thickness of 0.25 inches, and is subject to a wall thinning rate of 4.5 mils (i.e., 0.001 inch) per year, however CHECWORKS predicts a much slower wear rate of 0.4 mils per year, the predicted pipe wall thickness after 60 years would be 0.476 inches.[103] However the actual wall thickness would be 0.23 inches, which dips below the designed minimum and violates the ASME code and NUREG-1801 guidelines.[104] In such a case, an unacceptable amount of wall thinning would go undetected and it is questionable that the component would continue to operate safely.[105] Dr. Hopenfeld explains that even small changes in the corrosion rate can result in unacceptable levels of FAC, and unsafe plant operations.[106] Moreover, the increase in operating life from 40 to 60 years, represents a significant potential for pipe wall thicknesses to fall below designated minimum critical design levels during extended operations, and, in Dr. Hopenfeld's view, it can

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[100] Hopenfeld TC-2 Testimony at 10 (Exhibit RIV000003); Hopenfeld TC-2 Report at 13 (RIV000005); ASME B31.3; ASME Code Section III, Paragraph NB-3200.

[101] *See* Hopenfeld TC-2 Testimony at 10 (Exhibit RIV000003); Hopenfeld TC-2 Report at 13 (RIV000005).

[102] Hopenfeld TC-2 Report at 14 (RIV000005).

[103] Hopenfeld TC-2 Report at 14 (RIV000005).

[104] Hopenfeld TC-2 Report at 14 (RIV000005).

[105] Hopenfeld TC-2 Report at 14 (RIV000005).

[106] Hopenfeld TC-2 Report at 14 (RIV000005).

be expected that an increasing number of components will become prone to failure after 40 years of service.[107] Further, Dr. Hopenfeld's examples were simplified and based on just one set of parameters; Dr. Hopenfeld explains that, in actuality, there is a wide variation in parameters, and that, as a result, the inaccuracy of CHECWORKS is likely to allow many component wall thicknesses to reach critical levels.[108] Therefore, in Dr. Hopenfeld's opinion, the use of CHECWORKS has real safety implications at Indian Point.[109]

Dr. Hopenfeld also explains that even conservative predictions can affect plant safety: Entergy's documentation states that Entergy attributes findings that components have low remaining life and should be replaced) to an "often overpredicted" wear value by CHECWORKS.[110] As Dr. Hopenfeld has explained, the evidence all points to the fact that CHECWORKS predominantly produces non-conservative results. Thus, Entergy's assumption is highly problematic from a safety perspective. As Dr. Hopenfeld explains, if predictions are commonly perceived to be based on conservative estimates, component replacement could be incorrectly postponed, which could result in an excessive, undetected wall thinning.[111]

In conclusion, the use of CHECWORKS at Indian Point has safety implications if the plant continues to operate during the proposed PEO.

[107] Hopenfeld TC-2 Testimony at 10 (Exhibit RIV000003); Hopenfeld TC-2 Report at 14 (RIV000005).

[108] Hopenfeld TC-2 Testimony at 10 (Exhibit RIV000003); Hopenfeld TC-2 Report at 14-15 (RIV000005).

[109] Hopenfeld TC-2 Testimony at 10 (Exhibit RIV000003); Hopenfeld TC-2 Report at 15 (RIV000005).

[110] CSI, Technologies, Inc., Indian Point Unit 3 CHECWORKS SFA Model, Calculation No. 0705.100-01, Revision 2, August 2, 2011, Appendix K – Components with Negative Time to Tcrit, IPEC00238096 (Exhibit RIV000021).

[111] Hopenfeld TC-2 Testimony at 10-11 (Exhibit RIV000003); Hopenfeld TC-2 Report at 15 (RIV000005).

4. CHECWORKS Has No Track Record of Performance in Preventing Unacceptable FAC and Component Failures

The predictive capabilities of the CHECWORKS computer code can also be assessed in terms of the ability of the model to *actually* prevent wall thinning incidents.[112] In light of completely inadequate benchmarking, a demonstrated record of performance is necessary to show that CHECWORKS is able to manage FAC during the PEO, especially in light of the long industry-wide history in which CHECWORKS has been unsuccessful in predicting wall thinning. While Entergy has stated that "CHECWORKS has a demonstrated record of successfully predicting wall thinning at IPEC and other nuclear power plants,"[113] such a position remains unsubstantiated. Dr. Hopenfeld's review of FAC related occurrences at Indian Point indicates that CHECWORKS has no track record of performance at Indian Point, and no track record of performance under post-power uprate conditions.

Generally speaking, CHECWORKS has had questionable effectiveness at nuclear power plants since the program was introduced. For example, in 2005, a member of the Advisory Committee on Reactor Safeguard ("ACRS") Subcommittee on Thermal Hydraulics recognized the poor correlation between CHECWORKS predictions and operating data, stating that "[i]f you look at the data base, you don't really have too much confidence in CHECWORKS."[114] In addition, an assessment in NUREG/CR-6936, PNNL *16186,Probabilities of Failure and Uncertainty Estimate Information for Passive Components* - a *Literature Review* (May 2007),[115] indicates that the rate of failures attributable to FAC actually went up in the period of time after

[112] Hopenfeld TC-2 Testimony at 11 (Exhibit RIV000003); Hopenfeld TC-2 Report at 15 (RIV000005).

[113] *See* Applicant's Motion for Summary Disposition of RK-TC-2, *supra* Note 11 at 22.

[114] Statement by Dr. F. Peter Ford, transcript of January 26, 2005 meeting of the ACRS Subcommittee on Thermal Hydraulics (January 26, 2005), at 198, ADAMS Accession No. ML050400613 (Exhibit RIV000022).

[115] NUREG/CR-6936, PNNL *16186,Probabilities of Failure and Uncertainty Estimate Information for Passive Components* - a *Literature Review* (May 2007) (Exhibit RIV000023).

CHECWORKS was put into use, demonstrating CHECWORKS was not effective in reducing the number pipe failures.[116]

FAC-related failures have persisted in the industry despite the use of CHECWORKS. Dr. Hopenfeld explains that pipe thinning events have occurred in recent years (and since the publication of NUREG/CR-6936) at numerous nuclear power plants across the United States, including Duane Arnold, Hope Creek, Clinton, Braidwood, LaSalle, Peach Bottom, Palo Verde, Palisades, Catawba, Calvert Cliffs, Kewaunee, Browns Ferry, ANO, and Salem.[117] The NRC has acknowledged the seriousness and persistence of FAC throughout the nuclear industry.[118]

Entergy has stated that the use of CHECWORKS has resulted in no fatalities and no "major FAC-caused pipe ruptures in a U.S. nuclear unit for more than 10 years."[119] However, this information does not prove that CHECWORKS had been a success. Just because there have been no FAC-related catastrophes, does not change the fact that FAC documented below minimum acceptable limits has been detected across the industry. As Dr. Hopenfeld testifies, just because a plant does not experience a pipe ruptures is not permission for the plant to operate with pipes of unknown and unacceptable wall thickness, and that the "leak-before-break" concept is not an excuse for operating with excessively worn-out components.[120] Entergy has also pointed to the pipe rupture and resulting fatalities that occurred at Japan's Mihama plant as evidence that CHECWORKS is effective, since that plant did not make use of the computer

[116] NUREG/CR-6936, PNNL *16186,Probabilities of Failure and Uncertainty Estimate Information for Passive Components* - a *Literature Review* (May 2007) (Exhibit RIV000023).

[117] *See* Hopenfeld TC-2 Testimony at 11 (Exhibit RIV000003); Hopenfeld TC-2 Report at 16 (RIV000005); *see also supra* Note 43.

[118] *See supra* Note 43.

[119] *See* Applicant's Motion for Summary Disposition of RK-TC-2, *supra* Note 11 at 23.

[120] Hopenfeld TC-2 Testimony at 11 (Exhibit RIV000003); Hopenfeld TC-2 Report at 16-17 (RIV000005).

model.[121] However, Dr. Hopenfeld clarifies that the occurrence at Mihama is not indicative that CHECWORKS is effective, since it is not clear that CHECWORKS could have modeled the flow of the particular pipe involved.[122] Dr. Hopenfeld further questions how Entergy could draw the conclusion it does, in light of the fact that there is no analysis to show that the use of CHECWORKS would have prevented the accident from happening.[123] All told, there is simply no evidence to suggest that CHECWORKS has been a reliable tool in the industry for predicting and preventing FAC.

In relation to Indian Point, the inability of the CHECWORKS model to produce accurate results, as discussed at length in Dr. Hopenfeld's testimony and expert report,[124] alone undeniably establishes that the model has *no* track record of performance. A history of FAC related incidents confirms that, to date, the code has not been successful in preventing FAC, and further proves that the safety implications of using CHECWORKS at the plant are quite real. In particular, several Entergy documents report numerous leaks and instances of excessive wall thinning in mechanical systems at Indian Point. For example, Entergy's 2008 Operating Experience Review Report, IP-RPT-08-LRD05, Rev. 3, documents various unacceptable wall thinning events which have occurred at the plant.[125] Entergy condition reports document numerous FAC occurrences at Indian Point, including many instances where thinning was reported below the minimum acceptable wall thickness required.[126] These condition reports also

[121] *See* Applicant's Motion for Summary Disposition of RK-TC-2, *supra* Note 11 at 23, fn.133.

[122] Hopenfeld TC-2 Report at 17 (RIV000005).

[123] Hopenfeld TC-2 Report at 17 (RIV000005).

[124] Hopenfeld TC-2 Testimony at 5-10 (Exhibit RIV000003); Hopenfeld TC-2 Report at 5-13 (RIV000005).

[125]Entergy Engineering Report, Operating Experience Review Report, IP-RPT-06-LRD05, Rev. 3 (2008), IPEC00186046 (Exhibit RIV000024).

[126] *See* Daily DER Report, DER-01-01522, April 25, 2001, IPEC00020501 (RIV000025); Entergy Operations, Inc, Condition Report List, IPEC00185743 (RIV000026); Entergy Operations, Inc., Condition Report List,

discuss instances where investigations were undertaken due to component leakage, and subsequent inspections uncovered wall thinning below minimum design limits.[127] Furthermore, in relation to this license renewal proceeding, the Advisory Committee on Reactor Safeguards has also questioned Entergy regarding several incidences in which component wall thinning was found to be below minimum acceptable levels.[128]

Due to the unavailability of any CHECWORKS data predating 2000,[129] it is not possible to assess whether the number of failures has increased since the use of CHECWORKS started at Indian Point. However, what is clear is that there is a demonstrated history of unacceptable FAC-related thinning events that have occurred at Indian Point notwithstanding the use of CHECWORKS at the plant. In Dr. Hopenfeld's opinion, as Indian Point continues to age past 40 years, it is reasonably foreseeable that more and more components will be prone to unacceptable thinning and failure.[130]

Entergy's own documentation irrefutably shows that the CHECWORKS model at Indian Point has not been able to detect levels of FAC before component wall thickness dips below minimum design requirements, in violation of the ASME code, and that there is currently no track record of performance of the code at the plant. Entergy's prospective use of the CHECWORKS code during the proposed PEO, therefore, poses tangible safety related concerns,

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IPEC00092552 (RIV000027); Entergy Condition Report CR-IP2-2001-10525, IPEC00092616 (RIV000028); Entergy Condition Report CR-IP3-2006-02270, IPEC00025699 (RIV000029).

[127] *See* Daily DER Report, DER-01-01522, April 25, 2001, IPEC00020501 (RIV000025); Entergy Operations, Inc, Condition Report List, IPEC00185743 (RIV000026); Entergy Operations, Inc., Condition Report List, IPEC00092552 (RIV000027); Entergy Condition Report CR-IP2-2001-10525, IPEC00092616 (RIV000028); Entergy Condition Report CR-IP3-2006-02270, IPEC00025699 (RIV000029).

[128] Transcript of Meeting of Advisory Committee on Reactor Safeguards (Sept. 10, 2009), ADAMS Accession No. ML092670114, at 90-96 (Exhibit RIV000030).

[129] *See supra* pp.15-16.

[130] Hopenfeld TC-2 Testimony at 10 (Exhibit RIV000003); Hopenfeld TC-2 Report at 14 (RIV000005).

as is apparent from the various leaks that have already occurred at Indian Point due to undetected FAC.

Due to Entergy's failure to demonstrate that CHECWORKS has an appropriate track record of performance at Indian Point, reliance on this compute model cannot be considered an appropriate or useful tool for managing FAC at Indian Point during the PEO.

5. The NRC ASLB Decision in *Vermont Yankee* Concerning the use of CHECWORKS for Managing FAC is Inapposite

In the Vermont Yankee ("VY") license renewal proceeding, an intervenor raised a contention pertaining to FAC, similarly asserting that the licensee (Entergy) improperly relied upon CHECWORKS because the code was not properly benchmarked.[131] The ASLB in that case ultimately determined that a prolonged period of benchmarking of CHECWORKS at VY was not necessary.[132] In the past, Entergy has indicated that the ASLB's findings in the VY case should be dispositive of Riverkeeper's FAC contention in the Indian Point proceeding.[133] Relying upon the findings of the ASLB in the VY proceeding would be inappropriate for numerous reasons.

Generally speaking, the VY license renewal proceeding is a wholly separate and distinct proceeding. As Dr. Hopenfeld explains, safety must be evaluated in each plant separately to account for the differences in flow velocities, temperatures, geometry, material, and coolant chemistry.[134] Notably, there are important differences between the Vermont Yankee and the Indian Point plants. To start, Indian Point is a much larger facility in comparison to Vermont Yankee. Additionally, Indian Point is a different kind of reactor than VY, i.e., a pressurized

[131] *Entergy Nuclear Vermont Yankee*, 68 NRC 763, 854.

[132] *See Entergy Nuclear Vermont Yankee*, 68 NRC 763, 889.

[133] *See* Applicant's Motion for Summary Disposition of RK-TC-2, *supra* Note 11 at 20-21.

[134] Hopenfeld TC-2 Testimony at 9 (Exhibit RIV000003); Hopenfeld TC-2 Report at 19 (RIV000005).

water reactor and not a boiling water reactor, the former of which are known to be significantly more prone to failures from wall thinning due to FAC than the latter.[135] Thus, a site-specific, independent evaluation of Entergy's program for managing FAC, and the use of CHECWORKS, *at Indian Point* is necessary. Indeed, a license renewal applicant must show "that the *specific plant details* of its AMP have adequately addressed [NUREG-1801]"[136] and cannot simply defer to findings related to an allegedly similar program at a different plant.

In any event, the conclusions of the VY ASLB are specific to the continued operation of VY and, therefore, cannot be generically applied in the Indian Point proceeding. No where did the VY ASLB state that their conclusions were universal. That board's decision referenced the role of plant specific inputs and data in the FAC program at VY numerous times, for example, indicating that "[t]o address the adequacy of Entergy's FAC AMP, we reviewed the applicant's description of its Existing FAC Program, explored the details of its inspection plan, evaluated the role of CHECWORKS in its AMP, and investigated the timeliness of Entergy's updates to CHECWORKS with *plant specific* data."[137] This leaves no doubt that the conclusions reached by the VY ASLB are restricted to the VY plant.

Nor could the findings of the VY ASLB be applied to Indian Point, as the circumstances are remarkably different. Dr. Hopenfeld was an expert witness in the VY proceeding and has first-hand knowledge of these important differences, which he testifies about, and discusses in his export report.[138] First, the ASLB in the VY proceeding specifically found that benchmarking was not necessary because Entergy would have three sets of data at the uprated power levels that

[135] *See* NUREG/CR-6936 at p.5.25 (RIV000023).

[136] *See Entergy Nuclear Vermont Yankee*, 68 NRC 763, 871 (emphasis added).

[137] *See Entergy Nuclear Vermont Yankee*, 68 NRC 763, 871-72 (emphasis added).

[138] Hopenfeld TC-2 Testimony at 8-9 (Exhibit RIV000003); Hopenfeld TC-2 Report at 20 (RIV000005).

would "refine the model calibration for the EPU [extended power uprate] prior to the PEO."[139] The VY ASLB did not have the benefit of any data for the VY plant at the uprated power levels because the adjudicatory hearings were held shortly after power uprate occurred. The ASLB could, therefore, not assess the ability of CHECWORKS to detect wall thinning in light of the changed operating conditions. In contrast, at Indian Point, power uprates occurred in 2004 at Unit 2 and 2005 at Unit 3, and so several sets of data at post power uprate conditions for Indian Point Units 2 and 3, have already been collected and are available.[140] Unlike at VY, the success of CHECWORKS in predicting the consequences of changes in plant conditions can be assessed, and, as discussed above, such an assessment of the available data unequivocally demonstrates that the CHECWORKS model remains inaccurate and is not sufficiently benchmarked to account for the new plant conditions.[141] This necessarily renders the conclusions of the VY ASLB regarding the benchmarking of CHECWORKS inapplicable in the instant proceeding.

Second, the ASLB in VY also arrived at the conclusion that prolonged benchmarking of CHECWORKS was not necessary at VY, because the "data collected at VYNPS since 1989" had assisted in calibrating the model.[142] To the contrary, in the Indian Point proceeding, Entergy maintains that data and CHECWORKS modeling at Indian Point prior to the power uprates of 2004 and 2005 are *irrelevant*.[143] Further, Entergy indicated that CHECWORKS documentation related to Indian Point Unit 2 prior to 2000 does not exist, and refused to produce any

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[139] *See Entergy Nuclear Vermont Yankee*, 68 NRC 763, 889.

[140] *See* CHECWORKS Graphs (Exhibits RIV00016A, RIV00016B).

[141] Hopenfeld TC-2 Testimony at 5-10 (Exhibit RIV000003); Hopenfeld TC-2 Report at 5-13 (RIV000005).

[142] *Entergy Nuclear Vermont Yankee* (Vermont Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 894 (Nov. 24, 2008).

[143] *See* Entergy's Answer to Riverkeeper's Motion to Compel Disclosure of Documents (Aug. 13, 2010), at 4-5 (Explaining Entergy's objection "to Riverkeeper's request for additional CHECWORKS documents related to modeling for IP2 prior to outage 2R16 (2004) and for IP3 prior to outage 3R13 (2005) as not relevant to the admitted contention and beyond the scope of this proceeding . . . . FAC reports prepared prior to 1999 are not relevant to the admitted contention.").

CHECWORKS documentation related to Indian Point Unit 3 prior to 2001. Such information would be necessary in order to assess the adequacy of the benchmarking of the CHECWORKS model and/or its predecessor codes since the owners of the plants started using it (ostensibly since the 1980s). Accordingly, Entergy cannot rely upon any earlier data to demonstrate that CHECWORKS is adequately benchmarked,[144] and certainly cannot support an assertion that the CHECWORKS model at Indian Point has been calibrated with decades of data, as the VY ASLB found in the VY license renewal proceeding.

The VY ASLB specifically indicated the need for recalibration of CHECWORKS, stating that "[t]he effectiveness of CHECWORKS improves if the data from these inspections are entered into the model in a timely fashion, and the model re-calibrated for the observed wear rates."[145] In relation to the use of CHECWORKS at VY, the ASLB found that "that data collected at VYNPS since 1989 and the three sets of data for the 4½ years at the uprated power level prior to entering the PEO will be sufficient to assure effective use of the CHECWORKS model in the FAC AMP."[146] In contrast, at Indian Point, there is a smaller universe of historical data, and the post-power uprate data that is available demonstrates that the code has *not* been adequately re-calibrated to account for changed conditions. Ample post-power uprate data demonstrate conclusively that CHECWORKS cannot be used to predict wall thinning *at Indian Point*.

A third reason the VY ASLB's findings cannot be generically applied in the Indian Point proceeding is because of the VY ASLB determination that CHECWORKS had a small

[144] *See* In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-0247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, Order (Ruling on Riverkeeper's Motion to Compel) (November 4, 2010), at 5.

[145] *See Entergy Nuclear Vermont Yankee*, 68 NRC 763, 894.

[146] *See Entergy Nuclear Vermont Yankee*, 68 NRC 763, 894.

contribution to Entergy's overall FAC AMP at VY. In particular a VY operator testified, and the ASLB agreed, that "only one-third of the inspection locations were based on the results from CHECWORKS."[147] On the other hand, Entergy has not specified the exact degree to which the FAC program at Indian Point relies upon CHECWORKS to predict locations for inspections at each outage.[148] Further, as the discussion below shows, it is apparent that the FAC program at Indian Point, in fact, relies primarily upon CHECWORKS.

Fourthly, the ASLB in VY found that the increase in velocities due to the power uprate at VY would have been accounted for in the calibration of CHECWORKS at larger plants.[149] Likewise Entergy has argued that that the power uprates that occurred at Indian Point are bounded by the larger power uprate that occurred at VY somehow rendering the CHECWORKS model automatically benchmarked for Indian Point plant-specific conditions.[150] However, it is improper to assume that the changes in plant operating conditions at Indian Point are accounted for in the calibration of CHECWORKS with data from other plants, including VY. To begin with, in the VY proceeding, Entergy did not demonstrate that the CHECWORKS model had adequately accounted for changed plant conditions from the 20% power uprate; rather the VY ASLB, in part, deferred to future inspection data which it assumed would calibrate the CHECWORKS model sufficiently prior to the period of extended operation.[151] Thus, the magnitude of the power uprate at VY should have no bearing on the instant proceeding whatsoever.

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[147] *See Entergy Nuclear Vermont Yankee*, 68 NRC 763, 881.

[148] Hopenfeld TC-2 Testimony at 8 (Exhibit RIV000003); Hopenfeld TC-2 Report at 20 (RIV000005).

[149] *See Entergy Nuclear Vermont Yankee*, 68 NRC 763, 882-84.

[150] *See* Applicant's Motion for Summary Disposition of RK-TC-2, *supra* Note 11 at 20-21.

[151] *See Entergy Nuclear Vermont Yankee*, 68 NRC 763, 894.

As Dr. Hopenfeld explains, CHECWORKS must be evaluated at each plant separately to account for the unique differences in changed plant conditions, including materials, local flow velocities, temperatures, and water chemistry.[152] Indian Point is much larger than VY and impact of a power uprate on plant conditions is necessarily relative to the size of the particular plant.[153] Indian Point is also a PWR, and more prone to FAC-related failures.[154] Thus, simply because the percent change in power increase at VY was larger than the uprate that occurred at Indian Point does not mean that the impacts on plant conditions would be bounded by what took place at VY or that the VY power uprate would automatically account for all changed conditions at Indian Point.[155] Moreover, Dr. Hopenfeld explains that accessibility for inspections, past history with respect to the number of components and frequency of wall measurements that were used in the calibration of CHECWORKS, the quality of the correlation of predictions with measurements, and the number of component failures from wall thinning, will necessarily vary depending on the facility, further warranting an individual assessment of the use of CHECWORKS at Indian Point.[156] It is illogical to use a generic assessment of CHECWORKS and simply assume that the code is bounding for the power uprate at Indian Point, especially in light of evidence to the contrary. Due regard must be given to how CHECWORKS is implemented at Indian Point, and how it has performed.

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[152] Hopenfeld TC-2 Testimony at 9 (Exhibit RIV000003); Hopenfeld TC-2 Report at 20 (RIV000005).

[153] *See* Hopenfeld TC-2 Testimony at 9 (Exhibit RIV000003); Hopenfeld TC-2 Report at 20 (RIV000005).

[154] *See* NUREG/CR-6936 at p.5.25 (RIV000023).

[155] Hopenfeld TC-2 Testimony at 9 (Exhibit RIV000003); Hopenfeld TC-2 Report at 20 (RIV000005). Thus, the ASLB's questioning of what percent change in plant operating parameters would have a material effect on CHECWORKS results, when it ruled on the admissibility of Riverkeeper Contention TC-2 was completely appropriate. *See* ASLB July 31, 2008 Contention Admissibility Order, *supra* note 10 at 168. Entergy cannot avoid this inquiry by saying the Indian Point power uprate is bounded by the uprate at VY. *See, e.g.*, Entergy Motion for Summary Disposition, *supra* note 11 at 21.

[156] Hopenfeld TC-2 Testimony at 9 (Exhibit RIV000003); Hopenfeld TC-2 Report at 20 (RIV000005).

Based on the foregoing, it would be incorrect for the ASLB in this proceeding to defer in any respect to the findings of a licensing board relating to a plant specific determination at VY.

ii. Entergy FAC Program Largely Relies on the use of CHECWORKS and Lacks Any Other Meaningfully Independent Tools for Addressing FAC

Entergy's has stated that the FAC program at Indian Point will be effective in managing FAC-related aging effects because, even if CHECWORKS is an ineffective tool for predicting FAC, "CHECWORKS is only *one* of several bases used by Entergy to select and schedule in-scope components for inspection."[157] Entergy maintains that inspection scope is also based on (1) actual pipe wall thickness measurements from past outages, (2) industry experience related to FAC, (3) results from other plant inspection programs, and (4) engineering judgment.[158] However, these "additional" tools are not adequate mechanisms to effectively manage FAC throughout the proposed PEO.

As Dr. Hopenfeld testifies, these additional criteria are not independent tools sufficient to establish an accurate FAC inspection scope.[159] In fact, these additional criteria fundamentally depend upon CHECWORKS. For example, Dr. Hopenfeld explains that actual pipe wall thickness measurements from past outages are only useful when used in combination with a predictive tool which would prevent the wall thickness of a given component from being reduced to below the minimum design thickness while in service.[160] Accordingly, this is a required input for the use of CHECWORKS and not a stand-alone "tool" for component selection.[161] Moreover, for components initially selected for inspection by CHECWORKS, any decisions

[157] Applicant's Motion for Summary Disposition of RK-TC-2, *supra* Note 11 at 17.

[158] Applicant's Motion for Summary Disposition of RK-TC-2, *supra* Note 11 at 17, Attach. 2, ¶¶ 39.

[159] Hopenfeld TC-2 Testimony at 13 (Exhibit RIV000003); Hopenfeld TC-2 Report at 21 (RIV000005).

[160] Hopenfeld TC-2 Testimony at 13 (Exhibit RIV000003); Hopenfeld TC-2 Report at 21 (RIV000005).

[161] Hopenfeld TC-2 Testimony at 13 (Exhibit RIV000003); Hopenfeld TC-2 Report at 21 (RIV000005).

regarding future inspection scope based on actual pipe wall thickness measurements and wear rate trending of the actual inspection results, necessarily depends upon use of the CHECWORKS computer model.[162] Industry and plant experience with pipe wall thinning are similarly types of information that feed directly into the CHECWORKS model, and not independent tools for identifying inspection scope during outages.[163] The usefulness of such information for determining future inspections largely rests on how the CHECWORKS model processes the inputs and how such information affects the model over time.[164]

Dr. Hopenfeld testifies that to the extent actual pipe wall thickness, plant and industry experience do not rely upon CHECWORKS in order to meaningfully contribute to inspection scope selection, they can only be properly categorized as inputs which assist in the formulation of an "engineering judgment," and not three independent tools.[165] Rather, of the four additional tools identified by Entergy, only engineering judgment can be considered an "independent" tool for managing FAC.[166] As the EPRI has explained, "engineering judgment cannot substitute for other factors."[167] Dr. Hopenfeld explains how it is commonly recognized in all major industrial plants that engineering judgment alone is not sufficiently reliable to manage FAC.[168] The development of the CHECWORKS computer model itself arose out of the realization by the

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[162] Hopenfeld TC-2 Testimony at 13 (Exhibit RIV000003); Hopenfeld TC-2 Report at 21 (RIV000005).

[163] Hopenfeld TC-2 Testimony at 13 (Exhibit RIV000003); Hopenfeld TC-2 Report at 21 (RIV000005).

[164] Hopenfeld TC-2 Testimony at 13 (Exhibit RIV000003); Hopenfeld TC-2 Report at 21 (RIV000005).

[165] Hopenfeld TC-2 Testimony at 14 (Exhibit RIV000003); Hopenfeld TC-2 Report at 21 (RIV000005).

[166] Hopenfeld TC-2 Testimony at 14 (Exhibit RIV000003); Hopenfeld TC-2 Report at 21-22 (RIV000005).

[167] *See* NSAC-202L-R3 at 2-4 (Exhibit RIV000012) (explaining that "good engineering judgment" requires "that personnel involved in the program be aware of operating experience. . . and receive input from . . . plant operations . . .")).

[168] *See* Hopenfeld TC-2 Testimony at 14 (Exhibit RIV000003); Hopenfeld TC-2 Report at 22 (RIV000005).

nuclear industry that engineering judgment was not enough to be able to detect unacceptable and unsafe component wall thinning.[169]

Dr. Hopenfeld observes that engineering judgment is intrinsically subjective, and that when it is identified as a predictive tool, a very high degree of knowledge is required by those who conduct the assessment and specify the required steps for the prevention of component failures.[170] Even with the same input data, different assumptions could lead to different results because each assessment would depend heavily on the individual skill and experience of the responsible engineer.[171] In order to assess the validity of the use of engineering judgment, it is imperative to fully understand how it is used and all relevant underlying assumptions informing any judgment related determinations.[172]

Dr. Hopenfeld has review the numerous documents provided pertaining to Entergy's FAC program, and opines that Entergy has failed to clearly describe what exactly "engineering judgment" even means in relation to FAC inspections at Indian Point, and what role it actually plays in inspection scope selection.[173] Dr. Hopenfeld has observed that Entergy has not identified any kind of systematic methodology which demonstrates that engineering judgment is a separate predictive tool that would adequately meet applicable regulatory guidelines, including those contained in the *GALL Report*, and which would manage FAC related component degradation throughout the proposed PEO.

Dr. Hopenfeld explains that there are several key elements necessary to form a sound engineering judgment as it relates to FAC at Indian Point, and which Entergy does not appear to

[169] Hopenfeld TC-2 Testimony at 14 (Exhibit RIV000003); Hopenfeld TC-2 Report at 22 (RIV000005).

[170] Hopenfeld TC-2 Testimony at 14 (Exhibit RIV000003); Hopenfeld TC-2 Report at 22 (RIV000005).

[171] Hopenfeld TC-2 Testimony at 14 (Exhibit RIV000003); Hopenfeld TC-2 Report at 22 (RIV000005).

[172] Hopenfeld TC-2 Testimony at 14 (Exhibit RIV000003); Hopenfeld TC-2 Report at 22 (RIV000005).

[173] Hopenfeld TC-2 Testimony at 14 (Exhibit RIV000003); Hopenfeld TC-2 Report at 22 (RIV000005).

espouse: (a) good documentation of historical FAC assessments; (b) good communication between the organization that conducts analytical assessments and plant operators; (c) knowledge of FAC assessment methods; and (d) knowledge or risks and consequences.[174] Regarding the first element, Dr. Hopenfeld explains to ensure well-founded engineering judgment, all aspects of FAC experience (including the accuracy of past predictions, repairs, changes in plant operating conditions like water chemistry, and the like) must be well-documented such that decisions pertaining to FAC management will be based on sound knowledge of plant history.[175] In contrast, Indian Point, it is apparent that more than half of the overall amount of CHECWORKS-related data and documentation has been lost.[176] Dr. Hopenfeld indicates that when such a substantial amount of data and documentation is unavailable, a complete revalidation of the program would be appropriate.[177] Dr. Hopenfeld concludes that the lack institutional history at Indian Point in relation to FAC is a clear hindrance to the ability to form sound engineering judgment.[178]

The second element required to arrive at a sound engineering judgment, good communication, ensures that problems are identified early and appropriate actions are taken to resolve them.[179] It is not apparent to Dr. Hopenfeld that Entergy has the level of communication necessary to make reasoned engineering judgments. This is based on Dr. Hopenfeld's observation that there is an apparent lack of communication between Entergy personnel and the

[174] Hopenfeld TC-2 Testimony at 14 (Exhibit RIV000003); Hopenfeld TC-2 Report at 22-23 (RIV000005).

[175] Hopenfeld TC-2 Testimony at 14-15 (Exhibit RIV000003); Hopenfeld TC-2 Report at 22 (RIV000005).

[176] *See* In the Matter of Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-0247-LR and 50-286-LR, ASLBP No. 07-858-03-LR-BD01, Order (Ruling on Riverkeeper's Motion to Compel) (November 4, 2010), at 3, 4. This may be indicative of a lack of adequate record keeping by the previous owners of Indian Point.

[177] Hopenfeld TC-2 Report at 23 (RIV000005).

[178] Hopenfeld TC-2 Testimony at 15 (Exhibit RIV000003); Hopenfeld TC-2 Report at 23 (RIV000005).

[179] Hopenfeld TC-2 Testimony at 15 (Exhibit RIV000003); Hopenfeld TC-2 Report at 23 (RIV000005).

outside organization that runs the CHECWORKS assessments, CSI Technologies, Inc., concerning the numerous anomalies in the results of CHECWORKS.[180]

In relation to the third element relating to engineering judgment, i.e., knowledge of FAC assessment methods, is critical to understand the engineering model employed, CHECWORKS, since it is the predominant feature of FAC program.[181] One method to understand the validity of the model and gain the knowledge necessary to have a well-founded engineering judgment related to FAC management, is to observe the response of the model to changes in input variables, such as how CHECWORKS responded to the 2004 and 2005 power uprates at Indian Point Units 2 and 3, respectively.[182] However, again, the lack of complete documentation for the majority of FAC related inspections that have occurred at the plant necessarily limits the ability to understand how the model has responded.[183] In addition, Entergy has actually indicated its position that CHECWORKS data that predates the power uprates is *irrelevant*.[184] Dr. Hopenfeld explains, this it is troublesome that past experience is not being used to enhance engineering knowledge on FAC at Indian Point.[185]

In relation to the fourth element required when exercising engineering judgment, that is, knowledge of risks and consequences, it is necessary to understand and take into account the varying safety risks posed by FAC.[186] For example, a rupture of a pipe in the service water system does not pose a risk of a severe reactor accident, while a rupture of a main feedwater or

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[180] Hopenfeld TC-2 Testimony at 15 (Exhibit RIV000003); Hopenfeld TC-2 Report at 23 (RIV000005).

[181] Hopenfeld TC-2 Testimony at 15 (Exhibit RIV000003); Hopenfeld TC-2 Report at 23 (RIV000005).

[182] Hopenfeld TC-2 Testimony at 15 (Exhibit RIV000003); Hopenfeld TC-2 Report at 23 (RIV000005).

[183] Hopenfeld TC-2 Testimony at 15 (Exhibit RIV000003); Hopenfeld TC-2 Report at 23 (RIV000005).

[184] *See* Entergy's Answer to Riverkeeper's Motion to Compel Disclosure of Documents (Aug. 13, 2010), at 4-5

[185] Hopenfeld TC-2 Testimony at 15 (Exhibit RIV000003); Hopenfeld TC-2 Report at 23 (RIV000005).

[186] Hopenfeld TC-2 Testimony at 15 (Exhibit RIV000003); Hopenfeld TC-2 Report at 23 (RIV000005).

steam line pipe may lead to an uncontrolled severe accident.[187] Dr. Hopenfeld observes that he documents and information provided by Entergy reflect no consideration of inspection priorities of FAC-susceptible components relative to the safety risks posed due to a FAC-related failure.[188]

Dr. Hopenfeld has concluded that Entergy has completely failed to demonstrate that engineering judgment alone will safely manage FAC at Indian Point.[189] In turn, it is apparent that Entergy does not employ any meaningful tools that, separate and apart from CHECWORKS, would sufficiently manage the aging effects of FAC at Indian Point. Contrary to Entergy's position that there are other adequate mechanisms employed to manage FAC, in actuality, Entergy's FAC program relies greatly on the undependable CHECWORKS code.

iii. Failure to Address Safety Issues Posed Due to Inadequate Aging Management of FAC During the PEO

Entergy's reliance on an ineffective predictive tool will result in the delay of critical necessary pipe inspections, and corrective actions during the proposed PEO. In Dr. Hopenfeld's opinion, the operation of the plant without an adequate knowledge of the degree to which the strength of various components have been degraded due to FAC-related wear poses significant safety concerns.[190]

Dr. Hopenfeld has expressed concern about FAC-related degradation with respect to sudden transient loads where it may be too late to detect a leak and prevent a component failure.[191] For example, with severely degraded walls the feed water distribution piping ring inside the steam generators, which is subjected to high local velocities and turbulence, may

[187] Hopenfeld TC-2 Testimony at 15 (Exhibit RIV000003); Hopenfeld TC-2 Report at 23 (RIV000005).

[188] Hopenfeld TC-2 Testimony at 15 (Exhibit RIV000003); Hopenfeld TC-2 Report at 23 (RIV000005).

[189] Hopenfeld TC-2 Testimony at 15 (Exhibit RIV000003); Hopenfeld TC-2 Report at 23 (RIV000005).

[190] Hopenfeld TC-2 Testimony at 18-19 (Exhibit RIV000003); Hopenfeld TC-2 Report at 24-25 (RIV000005).

[191] Hopenfeld TC-2 Testimony at 19 (Exhibit RIV000003); Hopenfeld TC-2 Report at 24 (RIV000005).

rupture under transient loads causing damage to other structures within the steam generators.[192] Dr. Hopenfeld observes that Entergy has not provided data on CHECWORKS predictions for components inside the steam generators.

In addition, Dr. Hopenfeld explains that undetected FAC during the proposed PEO also poses a risk of loss of coolant accidents ("LOCA"), which violates NRC's General Design Criterion ("GDC") 4.[193] This criterion requires plant structures, systems and components be able to "accommodate the effects of . . . loss of coolant accidents" and "be appropriately protected against dynamic effects . . . that may result from equipment failures and from events and conditions outside the nuclear power unit."[194] Dr. Hopenfeld explains that when the original Indian Point probabilistic risk assessments ("PRAs") were developed, it was assumed that pipes were in pristine conditions, as the effects of aging were not included.[195] However, when the walls have been degraded, the probability of a pipe failing under a given load will be affected.[196]

Adequate consideration to these safety implications of undetected FAC is especially important at Indian Point because recent risk assessments show that Indian Point is vulnerable to core melts from earthquake loads. In fact, while the area around Indian Point is susceptible an earthquake of up to 7.0 magnitude.[197] An NRC report from August 2010 (in conjunction with

[192] Hopenfeld TC-2 Testimony at 19 (Exhibit RIV000003); Hopenfeld TC-2 Report at 24 (RIV000005).

[193] Hopenfeld TC-2 Testimony at 19 (Exhibit RIV000003); Hopenfeld TC-2 Report at 24 (RIV000005).

[194] 10 C.F.R. Part 50, Appendix A, General Design Criteria for Nuclear Power Plants, *Criterion 4—Environmental and dynamic effects design bases*.

[195] Hopenfeld TC-2 Testimony at 19 (Exhibit RIV000003); Hopenfeld TC-2 Report at 24 (RIV000005).

[196] Hopenfeld TC-2 Testimony at 19 (Exhibit RIV000003); Hopenfeld TC-2 Report at 24 (RIV000005).

[197] Lynn R. Sykes, John G. Armbruster, Won-Young Kim, & Leonardo Seeber, Observations and Tectonic Setting of Historic and Instrumentally Located Earthquakes in the Greater New York City–Philadelphia Area, Bulletin of the Seismological Society of America, Vol. 98, No. 4, pp. 1696–1719, August 2008 (hereinafter "Sykes, *Earthquakes in New York*") (Exhibit RIV000031); *see also* The Earth Institute, Columbia University, "Earthquakes May Endanger New York More than Thought, Says Study: Indian Point Nuclear Power Plant Seen as Particular Risk," Press Release Posted on The Earth Institute website, August 21, 2008, *available at*, http://www.earth.columbia.edu/articles/view/2235 (last visited December 21, 2011) (hereinafter "Columbia Earth Institute Earthquake Study Press Release").

supplemental data regarding power plants not reviewed in the report) indicates that Indian Point Unit 3 has the *highest* risk of seismic related core damage than any other nuclear power plant in the country, and that Unit.[198] Additionally, Dr. Hopenfeld explains that another type of accident for which an understanding of component wall thickness is critical, is station blackouts.[199] In Dr. Hopenfeld's opinion, Entergy should, but has failed to consider how the uncertainty related to pipe wall thickness at Indian Point will affect the integrity of components under transient loads other than plant transients, such as earthquakes and station blackouts.[200] Additionally, Dr. Hopenfeld observes that Entergy has not considered how the operation of Indian Point with such large uncertainties about pipe wall thicknesses will affect the likelihood of components succumbing to the effects of metal fatigue.[201]

Dr. Hopenfeld concludes that, as pipes at Indian Point have already been reduced in strength due to almost 40 years of operation, entering an extended period of operation with no valid tool to predict wall thinning severely limits Entergy's ability to determine the degree of pipe degradation and reduction in strength. There is no evidence to support a conclusion that despite such uncertainty, Indian Point would continue to operate in compliance with GDC 4, and without a severe accident occurring. Entergy has failed to provide any justification that Indian Point can operate safely in spite of the very large uncertainties in CHECWORKS predictions, and the lack of any other meaningful tools to detect FAC during the proposed extended licensing terms.

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[198] *See* Generic Issue 199 (GI-199), Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants Safety/Risk Assessment, August 2010, at Appendix D (Seismic Sore-Damage Frequencies), *available at*, ADAMS Accession Nos. ML100270639, ML100270756 (Exhibit RIV000032); Bill Dedman, *What are the odds? US nuke plants ranked by quake risk*, March 17, 2011, *available at*, http://www.msnbc.msn.com/id/42103936/ns/world_news-asia-pacific/ (last visited Dec.18, 2011) (RIV000033).

[199] Hopenfeld TC-2 Testimony at 19 (Exhibit RIV000003); Hopenfeld TC-2 Report at 24-25 (RIV000005).

[200] Hopenfeld TC-2 Testimony at 19-20 (Exhibit RIV000003); Hopenfeld TC-2 Report at 24-25 (RIV000005).

[201] Hopenfeld TC-2 Testimony at 20 (Exhibit RIV000003); Hopenfeld TC-2 Report at 25 (RIV000005).

D. Entergy's FAC Program Does Not Comply with 10 C.F.R. § 54.21(a)(3), SRP-LR the *GALL Report*, and Other Applicable Regulations and Guidance

The inadequacies identified above with Entergy's FAC program violate applicable NRC regulations and guidance as follows:

i. Entergy's Reliance on CHECWORKS Fails to Comply with Applicable Regulatory Guidance and Standards

It is apparent that Entergy is relying on an outdated version of the *GALL Report*, i.e., Revision 1. There is no doubt that the latest revision, (Revision 2) which clarifies the previous version in relation to the appropriateness of relying on CHECWORKS, should apply to the Indian Point license renewal proceeding. Regarding an appropriate AMP for FAC, the *GALL Report*, Revision 1 states that "CHECWORKS is acceptable because in general it provides a bounding analysis for FAC," and because it was "benchmarked by using data obtained from many plants."[202] the *GALL Report*, Revision 2 actually defines what this means: "The analysis is bounding because in general the predicted wear rates and component thicknesses are conservative when compared to actual field measurements."[203] Revision 2 further indicates that "when measurements show the predictions to be non-conservative, the model must be re-calibrated using the latest field data."[204] As the plain language of this revision makes clear, and as Dr. Hopenfeld testifies, the condition for accepting the use of CHECWORKS to predict FAC is unambiguous: it must provide a conservative results, and if not, be re-calibrated to do so.[205] Notably, the NRC has otherwise recognized that analytic codes need to be benchmarked.[206]

[202] *See GALL Report*, Rev. 1 at XI M-61 to XI M-62 (Exhibit NYS000146C).

[203] *See GALL Report*, Rev. 2 at § XI.M17 (Exhibit NYS000147D).

[204] *See GALL Report*, Rev. 2 at § XI.M17 (Exhibit NYS000147D).

[205] Hopenfeld TC-2 Testimony at 17 (Exhibit RIV000003); Hopenfeld TC-2 Report at 18 (RIV000005).

[206] *See* Safety Evaluation by the Office of Nuclear Reactor Regulations Related to Amendment No. 229 to Facility Operating License No. DPR-28 Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc.

Entergy, on the other hand, bases its use of CHECWORKS on starkly different criteria. As explained above, Entergy believes that CHECWORKS results are valid and appropriate if the results are bounded within arbitrary +/-50% lines, and/or, or if the LCF related to the data is between 0.5 and 2.5.[207] As Dr. Hopenfeld testifies, and as explained above, data within the +/-50% range can actually indicate highly non-conservative under-predictions that vary by a factor of 2.[208] This is not the "bounding analysis" that the *GALL Report* is referring to. To the contrary, the "bounding analysis" discussed in *GALL* is one that provides conservative results.

Under the relevant and applicable rubric of *GALL*, the use of CHECWORKS at Indian Point is not acceptable since it does not produce conservative results, and it cannot be re-calibrated to do so. In particular, as Dr. Hopenfeld testifies, the CHECWORKS model at Indian Point has produced non-conservative results about 50% of the time, and many times data fell outside the broad range that Entergy considered appropriate and "bounding," i.e., the +/-50% lines.[209] This is clear evidence that the model is not properly benchmarked. While *GALL* requires re-calibration, at Indian Point, a prolonged attempt to recalibrate CHECWORKS has not been successful in improving the predictive capability of the code.[210] Dr. Hopenfeld explains that CHECWORKS would have to be recalibrated *continuously* in order to meet the standard in the *GALL Report*, which renders the model useless as a predictive tool.[211] In Dr. Hopenfeld's opinion, because CHECWORKS continues to produce non-conservative results after decades of

(Vermont Power Station, Docket No. 50-571), at § 2.8.7.1, p. 190, ADAMS Accession No. ML060050028 (Exhibit RIV000013).

[207] Hopenfeld TC-2 Testimony at 6, 16-17 (Exhibit RIV000003); Hopenfeld TC-2 Report at 6-8, 18-19 (RIV000005).

[208] Hopenfeld TC-2 Testimony at 6 (Exhibit RIV000003); Hopenfeld TC-2 Report at 6 (Exhibit RIV000005).

[209] *See* Hopenfeld TC-2 Report at 6, 9-12 (Exhibit RIV000005).

[210] Hopenfeld TC-2 Testimony at 17-18 (Exhibit RIV000003); Hopenfeld TC-2 Report at 19 (Exhibit RIV000005).

[211] Hopenfeld TC-2 Testimony at 18 (Exhibit RIV000003); Hopenfeld TC-2 Report at 19 (Exhibit RIV000005).

use, there is no way to ensure appropriate calibration of the model prior to, or even during, the proposed PEO at Indian Point.[212] Therefore, Entergy's reliance upon CHECWORKS, does not demonstrate an AMP for FAC that is consistent and compliant with the *GALL Report*.

Entergy's reliance upon CHECWORKS also runs afoul of the *GALL Report*'s guidance pertaining to acceptance criteria. In particular, with respect to acceptance criteria, the *GALL Report* states that the inspection results are to inputs to a computer code such as CHECWORKS are to calculate the remaining time "*before the component reaches the minimum allowable wall thickness*."[213] As Dr. Hopenfeld has observed, and as Entergy's own documentation plainly demonstrates, CHECWORKS is not capable of accurately calculating the number of operating cycles remaining *before* a component will reach the minimum allowable wall thickness.[214] The inability to ensure the maintenance of minimum design wall thicknesses also violate the ASME code.[215] Entergy's FAC AMP also is inconsistent with the *GALL Report*, because it fails to provide the requisite "reasonable assurance that structural integrity will be maintained between inspections" or "ensure that the extent of wall thinning is adequately determined, that intended function will not be lost, and that corrective actions are adequately identified."[216]

Furthermore, the use of CHECWORKS also fails to meet the guidance of the *GALL Report* because it does not ensure that all forms of FAC will be adequately managed. In particular, while the *GALL Report* does not limit the obligation of licensees to manage wall

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[212] Hopenfeld TC-2 Testimony at 17-18 (Exhibit RIV000003); Hopenfeld TC-2 Report at 19 (Exhibit RIV000005).

[213] *GALL Report*, Rev. 1 at XI M-62 (Exhibit NYS000146C); *GALL Report*, Rev. 2 at XI M17-2 (Exhibit NYS000147D).

[214] *See* Daily DER Report, DER-01-01522, April 25, 2001, IPEC00020501 (RIV000025); Entergy Operations, Inc, Condition Report List, IPEC00185743 (RIV000026); Entergy Operations, Inc., Condition Report List, IPEC00092552 (RIV000027); Entergy Condition Report CR-IP2-2001-10525, IPEC00092616 (RIV000028); Entergy Condition Report CR-IP3-2006-02270, IPEC00025699 (RIV000029).

[215] ASME B31.3; ASME Code Section III, Paragraph NB-3200.

[216] *See GALL Report*, Rev. 1 at XI M-62 (Exhibit NYS000146C); *GALL Report*, Rev. 2 at XI M17-2 (Exhibit NYS000147D).

thinning by FAC, CHECWORKS is limited to predicting FAC caused only by electrochemical reaction.[217] As explained above, there are various other forms of flow-induced corrosion.[218]

In summary, Entergy's reliance upon CHECWORKS fails to demonstrate compliance with the guidance in the *GALL Report*, and that, pursuant to 10 C.F.R. § 54.21(a)(3), that the aging effects of FAC will be adequately managed during the proposed periods of extended operation.

ii. Entergy's FAC Program Lacks Sufficient Detail to Adequately Address all Required Elements Identified in the GALL Report and SRP-LR

Because Entergy's FAC program relies primarily on a method which does not accurately detect FAC, i.e., CHECWORKS, and Entergy has otherwise failed to properly define how it employs other independent tools to sufficiently manage the aging effects of FAC at Indian Point, it is necessary for Entergy to provide the information required by the *GALL Report* and SRP-LR regarding an adequate FAC program. In particular, Entergy must address all the elements identified in the SRP-LR, and the *GALL Report*, including the method for determining component inspections, frequency of such inspections, and attendant criteria for component repair and replacement.[219] In light of Entergy's failure to demonstrate the effective use of CHECWORKS, or other independent tools for managing FAC at Indian Point, Entergy cannot generically claim consistency with NRC's guidance documents, and instead must "provide a reasonably thorough description of its AMP to show conclusively how this program will ensure that the effects of aging will be managed."[220] Indeed, "[f]or an applicant to just illustrate how its

[217] Hopenfeld TC-2 Testimony at 18 (Exhibit RIV000003); Hopenfeld TC-2 Report at 19 (Exhibit RIV000005).

[218] Hopenfeld TC-2 Testimony at 3-4 (Exhibit RIV000003); Hopenfeld TC-2 Report at 2 (Exhibit RIV000005).

[219] SRP-LR at § A.1.2.3 (Exhibit NYS000195, NYS000161); *GALL Report*, Rev. 1 § XI.M17 (Exhibit NYS000146C); *GALL Report*, Rev. 2 § XI.M17 (Exhibit NYS000147D)

[220] *Entergy Nuclear Vermont Yankee*, 68 NRC 763, 870; *see id.* at 871 ("an applicant . . . merely stating that its AMP meets NUREG-1801 without any specificity falls short of the required demonstration [of 10 C.F.R. § 54.21], since

proposed program will, or promises to, follow the same generic program recommendations provided to all plants does not clear the bar required by the regulations."[221]

Likewise, Entergy's fleet-wide procedure, EN-DC-315 and EPRI guidance document NSAC-202L-R3, through which Entergy claims to implement the guidance contained in the *GALL Report*, are focused heavily on the appropriate use of CHECWORKS. Like the *GALL Report*, Revision 2, these documents also imply that CHECWORKS should be properly benchmarked or calibrated. Due to the inadequacy of CHECWORKS as a tool for managing FAC at Indian Point, it is disputable whether Entergy is actually implementing such guidance at the plant. Notably, Entergy cannot rely upon any findings made by the ASLB in the VY license renewal proceeding pertaining to the adequacy of Entergy's fleet-wide implementation document, as such findings are only related to how Entergy's FAC AMP was being implemented at VY.[222] In order to comply with the requirement to adequately manage the aging effects of FAC during the proposed PEO at Indian Point, Entergy cannot simply rely on procedural documents which depend upon the *proper* use of CHECWORKS. Instead, Entergy must provide sufficient details regarding inspection scope, frequency, component replacement and repair criteria, etc., to demonstrate that FAC will be appropriately managed.

CONCLUSION

Entergy has failed to meet its burden to demonstrate that the effects of aging on the intended functions of relevant piping components will be adequately managed during the PEO. In particular, the expert opinion provided by Riverkeeper's witness Dr. Joram Hopenfeld, along

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section XI.M17 of NUREG-1801 consists of less than two pages of narrative evaluating EPRI's guidelines presented in NSAC-202L-R3 with an absence of plant-specific details.").

[221] *See Entergy Nuclear Vermont Yankee*, 68 NRC 763, 870.

[222] *See Entergy Nuclear Vermont Yankee*, 68 NRC 763, 871 (VY ASLB finding after an examination of the FAC program *at VY*, that the relevant guidelines "have been implemented *at VYNPS*.") (emphasis added).

with the documentary supporting evidence, has demonstrated that (1) CHECWORKS is not adequately benchmarked so as to be an effective tool for predicting FAC at Indian Point during an extended period of operation; (2) CHECWORKS has no "track record of performance at Indian Point"; (3) Entergy primarily relies upon the use of CHECWORKS, and has no other tools that are meaningfully independent of CHECWORKS that would sufficiently address FAC at Indian Point, and; (4) given the inadequacy of CHECWORKS, Entergy's FAC program lacks sufficiently detailed information regarding the method and frequency of component inspections and criteria for component repair and replacement, to assure adequate management of FAC during the PEO.

Entergy's program for managing the aging effects of FAC at Indian Point during the period of extended operation are woefully inadequate and fails to comply with 10 C.F.R. §54.21(a)(3), and applicable NRC guidance. Accordingly, Entergy's LRA to renew the operating licenses for Indian Point Units 2 and 3 should be denied.

Respectfully submitted this 22cd day of December 2011.

Signed (electronically) by Deborah Brancato

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