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

REPORT OF

DR. JORAM HOPENFELD

IN SUPPORT OF

CONTENTION RIVERKEEPER TC-2 – FLOW ACCELERATED CORROSION

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I. Introduction

I have been retained by Riverkeeper, Inc. ("Riverkeeper") as an expert witness in proceedings concerning the application by Entergy Nuclear Operations, Inc. ("Entergy") for the renewal of two separate operating licenses for the nuclear power generating facilities located at Indian Point on the east bank of the Hudson River in the Village of Buchanan, Westchester County, New York, for twenty years beyond their current expiration dates. I was asked to review Entergy's Aging Management Program ("AMP") concerning flow accelerated corrosion ("FAC"), and to assess whether Entergy has demonstrated that this AMP will manage the aging effects of FAC during the proposed period of extended operation ("PEO").

I am an expert in the field relating to nuclear power plant aging management. I am a mechanical engineer and hold a doctorate in mechanical engineering. I have forty-five years of professional experience in the fields of nuclear safety regulation and licensing, design basis and severe accidents, thermal-hydraulics, material/environment interaction, corrosion, fatigue, radioactivity transport, industrial instrumentation, environmental monitoring, pressurized water reactor steam generator transient testing and accident analysis, design, and project management.

My extensive professional experience has afforded me with knowledge and expertise regarding the material degradation phenomenon of FAC. I have published numerous peer-reviewed papers in the area of corrosion, and hold patents related to monitoring of wall thinning of piping components. I have knowledge and expertise regarding the use of the CHECWORKS computer code dating back to 1988, when it was known as CHEC.

I assisted Riverkeeper with the preparation of Contention TC-2, which challenged Entergy's aging management plan for addressing FAC at Indian Point during the proposed extended operating terms. The Atomic Safety and Licensing Board has admitted Riverkeeper Contention TC-2 for an adjudicatory hearing. I have reviewed all of the pleadings related to Riverkeeper's contention, including Entergy's Motion for Summary Disposition of Riverkeeper's Contention TC-2 and supporting attachments thereto.[1] I have also reviewed hundreds of documents identified by Entergy as relevant to Riverkeeper Contention TC-2, including CHECWORKS modeling reports, Entergy FAC program reports, manuals, procedures, audit reports, condition reports, inspection scope reports, operating experience reports, and others. In addition, I have reviewed applicable regulations and NRC guidance documents, technical reports, scientific and scholarly reports and articles, industry guidance documents and reports, and other documents generated by NRC, Entergy, industry groups, and scientific organizations. A list of documents to which I refer to herein is included at the end of this report.

Based on my review of these documents and my forty-five years of professional experience, I have concluded that Entergy has, to date, failed to demonstrate that the aging affects of FAC will be adequately managed at Indian Point during the proposed PEO. The basis for my conclusion is explained forthwith.

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

II. Background

A. The Nature and Safety Significant of FAC

FAC is a pipe wall thinning phenomenon in which the thinning rate is accelerated by flow velocity. Generally, two different mechanisms could lead to such wall thinning: (1) physical removal of metal by mechanical forces (shear or impact), and (2) chemical or electrochemical dissolution of the metal. In many instances both mechanisms occur simultaneously. When the metal is exposed to flowing liquid, flow velocity has a significant effect on metal removal. FAC includes wall thinning by impingement corrosion, electrochemical corrosion, erosion-corrosion, cavitation-erosion, and chemical dissolution. While the mechanism of wall damage is different in each case, all these mechanisms are affected to one degree or another by flow velocities.

The main causes of FAC include turbulence intensity, steam quality, material compositions, oxygen content, and coolant pH, though the behavior of FAC is still not completely understood. Wall thinning resulting from FAC is a local phenomenon in which local geometry, local metal composition, and local turbulences affect corrosion rates. Grooving and the formation of round holes are a manifestation of the interplay between these parameters, and once local corrosion has begun, geometrical changes may further intensify the local turbulence, thereby increasing FAC in a non-linear rate. For example, in a 1986 accident at the Surry nuclear power plant, areas on a failed feed water pipe elbow were almost completely eroded while adjacent areas were much less affected, and J-tubes on the distribution feed ring exhibited a similar phenomenon.[2]

The identification of locations where FAC rates are highest is made difficult by the fact that neither the local turbulence nor local flow velocity are directly measured quantities. As a result, it is necessary for operating plants to use thermal hydraulic computer codes, such as RELAP to calculate average velocities throughout the plant. Because of this indirect method of determining turbulence, considerable data must be collected over a period of time to assure that the locations with the highest propensity for FAC are properly identified.

If insufficiently addressed, FAC poses a significant safety risk at nuclear power plants. While pipes and components susceptible to FAC are designed with a corrosion allowance that is added to the minimum design wall thickness, commonly known as Tcr, in order to ensure that the pipes will operate safely throughout their lifetime, when FAC reduces wall thickness *below* the minimum design value (Tcr), the potential exists for the subject component to rupture. In many such cases, this will be preceded by a leak, which can be detected before the pipe or component ruptures catastrophically. A FAC-induced rupture of a high pressure component or pipe may have very serious safety consequences. For this reason, the American Society of Mechanical Engineers ("ASME") code, specifically requires that components and pipes do not operate below design limit wall thicknesses.[3]

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[2] *See* NRC Information Notice No 86-106, "Feed Water Line Break" (December 16, 1986); NRC Bulletin 87-01, "Thinning Pipe Walls in Nuclear Plants" (July 9, 1987).

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

Numerous instances of undetected FAC have previously resulted in catastrophic events: a feed water pipe elbow rupture at the Surry nuclear power plant in 1986 resulted in several fatalities; in 1993, FAC resulted in failures of feed ring and J-tube components at the San Onofre steam generators; in 1997, extraction steam piping ruptured at the Fort Calhoun Station; and in July of 2004, FAC in the secondary loop at the Mihama nuclear power plant resulted in the deaths of several workers.[4]

B. 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."[5] The developers of CHECWORKS specified that CHECWORKS predictions are limited to wall thinning only by electrochemical reaction. Notably, in many instances, there is no practical way of distinguishing whether wall thinning at a given location has been caused solely by electrochemical reaction or mechanical forces, or by a combination of both.

Because of the inherent unpredictability of FAC, CHECWORKS is based on empiricism, i.e. statistics, rather than on a theoretical model. Rather than being based on a mechanistic model, CHECWORKS is solely based on a collection of selective data which represents only a fraction of the total flow area. As a result, the CHECWORKS computer model is not reliable unless it is adequately benchmarked for each component. NRC has recognized the need to benchmark analytic codes.[6]

CHECWORKS could potentially be used to predict pipe wall thinning *if* all relevant locations are benchmarked for relevant plant parameters, such parameters do not change significantly over time, and benchmark data on relevant plant parameters is collected for a sufficiently long period of time. Data must be based on a sufficiently large number of measured points, be free of errors, and be continuously assessed by impartial experts. For simple geometries and one phase flow in straight pipes where the degree of turbulence is relatively low and stable, it would be reasonable to assume that six years of plant operations would be sufficient to benchmark a code for a given set of parameters. For complex geometries such as elbows and pipe branching areas where

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

[5] EPRI, Recommendations for an Effective Flow-Accelerated Corrosion Program, NSAC-202L-R3, at p.1-1.

[6] *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. (Vermont Power Station, Docket No. 50-571), at § 2.8.7.1, p. 190, ADAMS Accession No. ML060050028 ("Application of NRC-approved Analytical Methods and Codes. . . . In general, the analytical methods and codes are assessed and benchmarked against measurement data, comparisons to actual nuclear plant test data and research reactor measurement data. The validation and benchmarking process provides the means to establish the associated biases and uncertainties. The uncertainties associated with the predicted parameters and the correlations modeling the physical phenomena are accounted for in the analyses.").

turbulence intensity is considerably high, less stable, and less predictable, at least ten to fifteen years of benchmarking the FAC model would be necessary.

Changes in plant parameters necessitate appropriate re-calibration of the CHECWORKS code. Updating the model is especially important in the event of a power increase, since power changes affects various plant parameters, including velocities, temperatures, coolant chemistry, and steam moisture. Indeed, "[i]t is recognized that even small power uprates can have a *significant* affect on FAC rates."[7] In the absence of adequate benchmarking and data, CHECWORKS ceases to be a useful predictive tool, and serves only as a platform for collecting plant data on FAC.

III. Entergy's FAC Aging Management Program

Sections A.2.1.14 and B.1.15 of Entergy's License Renewal Application ("LRA") describe Entergy's FAC AMP as a program based on EPRI guidelines in Nuclear Safety Analysis Center (NSAC)-202L, *Recommendations for an Effective Flow-Accelerated Corrosion Program*,[8] that "includes (a) an evaluation to determine critical locations, (b) initial operational inspections to determine the extent of thinning at these locations, and (c) follow-up inspections to confirm predictions, or repair or replace components as necessary."[9] Entergy claims that the FAC program at Indian Point includes and is consistent with the ten program elements identified in NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, ("SRP-LR") and NUREG-1801, *Generic Aging Lessons Learned (GALL) Report* ("*GALL Report*")[10] as documented in Entergy's Aging Management Program Evaluation Report, IP-RPT-06-LRD07, Revision 5.[11] Entergy implements its FAC program through a fleet-wide procedure, EN-DC-315, Revision 3, *Flow Accelerated Corrosion Program* (March 1, 2010). Entergy's program implementation documents reflect that Entergy's FAC management program is largely based on the use of the computer program CHECWORKS to record plant operating experience and predict timing and locations of wall thinning.

IV. Entergy's Improper Reliance upon CHECWORKS Computer Modeling

Obtaining timely wall thinning measurements is the key activity for ensuring that components and structures subject and susceptible to FAC will continue to operate safely and as originally designed. 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. This will fail to assure that timely detection of FAC-related wall thinning will occur during the proposed PEO, that Entergy can prevent component

[7] EPRI, Recommendations for an Effective Flow-Accelerated Corrosion Program, NSAC-202L-R3, at p.4-5 (emphasis added).

[8] EPRI, Recommendations for an Effective Flow-Accelerated Corrosion Program, NSAC-202L-R3.

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

[10] *See* LRA § B.1.15.

[11] Aging Management Program Evaluation Report – Non-Class 1 Mechanical, IP-RPT-06-LRD07, Revision 5 (March 18, 2009).

wall thickness from being reduced below minimum design values, or that Indian Point will operate safely throughout the proposed license renewal periods.

A. CHECWORKS is Inadequately Benchmarked and Fails to Provide Reliable Predictive Wall Thinning Results at Indian Point

The efficacy of CHECWORKS can be determined by assessing the degree to which the model can accurately predict wall thinning. This can be evaluated by comparing wall thickness predictions generated by CHECWORKS with actual thickness measurements. Entergy has provided and I have reviewed years' worth of CHECWORKS comparison data. My extensive analysis reveals that the CHECWORKS computer code produces highly unreliable and non-conservative component wear predictions. 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 2 and 3 in 2004 and 2005, respectively, due to power uprates of 3.26% and 4.85%, respectively. My analysis of Entergy's data related to FAC further indicates that it will be impossible to properly calibrate the model before Indian Point enters the proposed PEO. The lack of adequate benchmarking renders CHECWORKS an ineffective tool for detecting and managing FAC at Indian Point during the PEO.

I have reviewed more than 6,500 data points contained in CHECWORKS modeling reports generated for Entergy, which were provided to Riverkeeper. In particular, I reviewed CHECWORKS wear predictions of component wall thickness plotted relative to actual measurements subsequently obtained. Entergy provided this data in relation to FAC inspections performed at Indian Point Unit 2 during refueling outages 14, 16, 17, 18, and 19, 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, 15, and 16. These data graphs, along with identifying report covers, have collectively been appended to this report as an exhibit. The data provided by Entergy was generated both before *and* after the changes in plant operating conditions at Indian Point due to power uprates at Unit 2 in 2004, and at Unit 3 and 2005. The data pertains to various systems at the plant, including the condensate, extraction steam, feedwater, moisture separator, moisture preseparator, and reheater drain systems.

Notably, despite the fact that CHECWORKS has been in use since the early 1990s when EPRI sold the program to nuclear power plant operators, Entergy has no CHECWORKS related documentation related to Indian Point Unit 2 generated prior to the year 2000.[12] Further, Entergy did not provide any CHECWORKS related documentation related to Indian Point Unit 3 generated prior to 2001, since "locating such documentation, to the extent it exists, would be extremely burdensome."[13] Thus, the data analyzed represents only a fraction of the total plant data that was allegedly used to benchmark CHECWORKS at Indian Point.

[12] *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.

[13] *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.

In any event, my review and analysis of the available universe of CHECWORKS data that Entergy has provided reveals shocking inaccuracy and a complete lack of correlation between component wear predictions and actual wall thickness measurements.

Entergy's data, presented graphically, depicts data points representing CHECWORKS wear predictions versus actual measured wear values. Entergy provided approximately 400 graphs, representing more than 6,500 CHECWORKS predictions. Each graph also shows three lines: a central 45° line, an arbitrary line designated +50% whereby CHECWORKS over-predicts measured wear by a factor of .7 (i.e., a conservative prediction), and an arbitrary line designated -50% whereby CHECWORKS under-predicts measured wear by a factor of 2 (i.e., a non-conservative prediction). With a perfect correlation, all the data would fall on the 45° line. To the contrary, Entergy's data shows that very few points fall on the 45° line. The wide scatters of data points on the graphs show that the predictions are far from accurate. Indeed, one could draw almost any line through the data on most of these graphs, indicating a complete lack of correlation.

The data points that fall between the 45° line and the abscissa, i.e., the zero wear prediction x-axis, represent non-conservative predictions. A review of all of Entergy's plotted data points reveals that many data points fall below the 45° line, within this non-conservative area. In fact, overall, the data shows that CHECWORKS has yielded non-conservative predictions about 40-60% of the time. For example, in one graph, there are 25 non-conservative points out of a total of 56, about 45%, while in another, there are 23 non-conservative points out of a total of 38 data points, about 61%.[14] Column A of Table 1 below summarizes the number of non-conservative CHECWORKS predictions as a fraction of the total number of measurements, in relation to all of the plots Entergy provided.

Additionally, with an ideal correlation each predicted point would have a single measured value. Again in contrast, Entergy's data shows that a given prediction yields widely different measured points. For example, in one particular graph, for a predicted wear value of 100 mils, there are approximately 17 different actual measurements, varying over a wide range, from 30 mils to over 250 mils.[15] In another graph, for a predicted wear value of 100 mils, about 14 different measurements resulted, again, varying greatly.[16]

The degree to which Entergy's actual wear measurements have deviated from CHECWORKS' predictions has been quite large. Notably, the arbitrary +/-50% lines on each graph imply that when some or most of the data on a given graph falls between these two lines, that

[14] *See* Riverkeeper Exhibit RIV00016A at p.90 (CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS FAC Model, Calculation No. 050714b-01, Revision 1, September 12, 2006, Page I-39 of 396); Riverkeeper Exhibit RIV00016A at p. 55 (CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS SFA Model, CSI Calculation No. 0705.101-01, Revision A, November 17, 2008, Page J-33 of 36).

[15] *See* Riverkeeper Exhibit RIV00016A at p.83 (CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS FAC Model, Calculation No. 050714b-01, Revision 0 (July 5, 2005), Page I-350 of 356).

[16] *See* Riverkeeper Exhibit RIV00016A at p.137 (CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS SFA Model, CSI Calculation No. 0705.101-01, Revision A, November 17, 2008, Page J-15 of 36.

CHECWORKS predictions are +/-50% accurate. The +/-50% lines would appear to give the impression that the code bounds all plant data within +/-50%. EPRI has actually made statements that would also lead one to believe this, stating at the American Power Conference in 1988 that the computer model was "predicting erosion-corrosion rates within a +/-50% band."[17] Likewise, Entergy has indicated its belief that CHECWORKS predictions have a precision of +/-50%, in other words, that CHECWORKS yields predictions with "+/-50% accuracy on predicted wear rate."[18] Thus, it is apparent that to Entergy, data within the +/-50% lines is acceptable.

However, this understanding of the +/-50% lines is incorrect, and such statements are misleading. In fact, this representation has even been characterized as dishonest.[19] When CHECKWORKS is characterized as acceptable because it bounds data within +/-50% it may lead some readers to believe that measurements are within 50% of predictions where in fact it is twice as much. It also leads to the conclusion that there is a large variation in the data. Whether this is tolerable would depend on outliers and risk. In any event, the use of the +/-50% lines as acceptance criteria for CHECWORKS predictions veils the apparent lack of conservatism in the code. In actuality, data points that fall between these two lines represent a range where the measured to predicted wear ratios vary by a factor of .7 to a factor of 2, meaning that even within these two lines, the points represents an over- or under-prediction by as much as .7 and 2, respectively. Entergy has not provided any rationale or justification regarding how prediction accuracy within the +/-50% lines provides an adequate basis for determining criteria for inspection frequency or component replacement.

Moreover, many data points, as much as 50%, fall outside these arbitrary lines, indicating that CHECWORKS cannot even bound the data conservatively within a factor of 2. Overall, the wear data for both Indian Point Units 2 and 3 shows that CHECWORKS can either over-predict or under-predict actual measured FAC by more than a factor of 10. For example, in one graph, a data point furthest to the right indicates a predicted wear of about 10 mils, while the corresponding measured wear was over 100 mils – a value 10 times larger than the predicted value.[20] This same graph shows a data point where CHECWORKS predicted zero wear, while the actual wear was 65 mils, another blatant and gross under-prediction.[21] On the other hand, a different graph shows a data point furthest to the left with a predicted wear of about 215 mils, while the actual measured wear was about 20 mils; the prediction here was 10 times higher than

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[17] 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), at 1.

[18] 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).

[19] 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.

[20] Riverkeeper Exhibit RIV00016A at p.90 (CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS FAC Model, Calculation No. 050714b-01, Revision 1, September 12, 2006, Page I-39 of 396); Riverkeeper Exhibit RIV00016A at p. 55 (CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS SFA Model, CSI Calculation No. 0705.101-01, Revision A, November 17, 2008, Page J-33 of 36).

[21] Riverkeeper Exhibit RIV00016A at p. 55 (CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS SFA Model, CSI Calculation No. 0705.101-01, Revision A, November 17, 2008, Page J-33 of 36).

the measurement.[22] Furthermore, another graph shows a large number of data points where the predicted wear was zero while the measured rate varied between 20 and 180 mils, i.e., no predictions at all. Column B of Table 1 below summarizes the number of CHECWORKS predictions that fell outside the +/-50% lines, meaning instances where the prediction was even worse than the wide uncertainty range represented by the area within the +/-50% lines.

That so many data points fall outside of Entergy's own arbitrary +/-50% lines serves to demonstrate that CHECWORKS cannot even bound the data within a factor of 2. The over-prediction or under-prediction of the data by a factor of 10 exhibited by a significant number of components clearly demonstrates that the CHECWORKS model employed at Indian Point cannot predict FAC to any degree of accuracy or precision. Quite the opposite, CHECWORKS can only predict an overall range of corrosion rate for a given component or a group of components. This range is too wide for practical applications, especially when the consequences of component failure are safety related.

Furthermore, each comparison graph provides, a "line correction factor" or "LCF," which Entergy's CHECWORKS model reports state "indicates the degree to which CHECWORKS over or under-predicts wear."[23] Entergy relies upon LCFs to "compare and adjust CHECWORKS predictions to match inspection data."[24] An LCF of 1 would indicate an exact agreement between CHECWORKS predictions and actual wall thickness measurements. Notably, the *only* graphs showing an LCF of 1, i.e., figures showing perfect agreement between predictions and measurements, are those figures with no data in them.[25]

Entergy's documentation states that a "reasonable LCF should be between 0.5 and 2.5."[26] However, Entergy has failed to provide any justification to support the conclusion that this range is acceptable, or how this range would be an indication that CHECWORKS can be used to accurately predict inspection locations. Nor has Entergy shown how "adjusting" CHECWORKS predictions using a LCF has made, or will make, the model more accurate, since years of modeling reports show consistently inaccurate results, as discussed above.

What's more, the CHECWORKS data reveals *numerous* instances where the LCF was outside the range that Entergy claims is acceptable. It is, therefore, indisputable that CHECWORKS is unreasonably failing to predict wear rates. Column C of Table 1 below summarizes the number of times reported LCFs were outside of Entergy's acceptable range.

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[22] Riverkeeper Exhibit RIV00016A at p.90 (CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS FAC Model, Calculation No. 050714b-01, Revision 1, September 12, 2006, Page I-39 of 396).

[23] *See, e.g.*, CSI, Technologies, Inc., Indian Point Unit 3 CHECWORKS SFA Model, Calculation No. 0705.100-01, Revision 2, August 2, 2011, at p.26.

[24] Applicant's Motion for Summary Disposition of Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion) (July 26, 2010), ADAMS Accession No. ML102140430, at Attachment 2, ¶ 48.

[25] *See, e.g.*, Riverkeeper Exhibit RIV00016A at p. 127 (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). It is difficult to understand Entergy's reason for providing figures containing no data.

[26] *See, e.g.*, CSI, Technologies, Inc., Indian Point Unit 3 CHECWORKS SFA Model, Calculation No. 0705.100-01, Revision 2, August 2, 2011, at p.26.

TABLE 1: Inaccuracy of CHECWORKS Wear Predictions					
Report Title	**Refueling Outage**	**No. of Plots**	**(A) % of Non-Conservative CHECWORKS predictions[27]**	**(B) No. of Data Points Outside +/-50% Lines**	**(C) No. of LCFs outside Entergy's "acceptable" range of 0.5-2.5**
Altran, Flow Accelerated Corrosion Program CHECWORKS Analysis Enhancement, Technical Report No. 00130-TR-001, Revision 0, Prepared for Consolidated Edison of New York, Inc., Indian Point Unit 2, December 2000, IPEC00024129	RO14	44	40-60%	77	13
CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS FAC Model, Calculation No. 050714b-01, Revision 0, July 5, 2005, IPEC00024459	RO16	26	40-60%	86	6
CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS FAC Model, Calculation No. 050714b-01, Revision 1, September 12, 2006, IPEC00168838	RO17	19	40-60%	58	2
CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS SFA Model, CSI Calculation No. 0705.101-01, Revision A, November 17, 2008	RO18	35	33-60%	171	7
CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS SFA Model, CSI Calculation No. 0705.101-01, Revision 1, February 26, 2010, IPEC00216834[28]	RO18	39	40-60%	166	7
CSI Technologies, Inc., Indian Point Unit 2 CHECWORKS SFA Model, CSI Calculation No. 0705.101-01, Revision 2, July 8, 2010, IPEC00236630[29]	RO19	39	40-60%	194	7

[27] There is one single number that represents the ratio of non-conservative points to the total points in each plot. Each percentage was obtained by counting the number of data points in the non-conservative area of a given plot and dividing them by the total number of data points in that plot. The resulting number varies plot to plot, and due to the large number of plots per CHECWORKS modeling report, a single range is used to represent the number of non-conservative predictions during each outage.

[28] While both of these reports indicates that "[c]omponent inspection data through the Refuel Outage 18 was imported to the model," the data graphs appear to indicate that the comparison to wear predictions were for measurements "@CYCLE 19."

[29] While this report indicates that "[c]omponent inspection data through Refueling Outage 19 was imported to the model," the data graphs appear to indicate that the comparison to wear predictions were for measurements "@CYCLE 20."

Report Title	Refueling Outage	No. of Plots	(A) % of Non-Conservative CHECWORKS predictions[30]	(B) No. of Data Points Outside +/-50% Lines	(C) No. of LCFs outside Entergy's "acceptable" range of 0.5-2.5
CSI Technologies, Inc., Indian Point 3 Nuclear Power Plant, Heater Drains CHECWORKS Model, Calculation No. IP3-RPT-HD-00979, Rev. 2, April 18, 2001, IPEC00221266	Pre RO12	4	44%-47%	6	0
CSI Technologies, Inc., Indian Point 3 Nuclear Power Plant, Feedwater CHECWORKS Model, Calculation No. IP3-RPT-FW-00984, Rev. 2, April 18, 2001, IPEC00221107	Pre RO12	2	42%-44%	6	1
CSI Technologies, Inc., Indian Point 3 Nuclear Power Plant, Reheater Drains CHECWORKS Model, Calculation No. IP3-RPT-HD-01144, Rev. 2, April 18, 2001, IPEC002216470	Pre RO12	5	41%-44%	23	5
CSI Technologies, Inc., Indian Point 3 Nuclear Power Plant, Moisture Preseparator Drains CHECWORKS Model, Calculation No. IP3-RPT-HD-00913, Rev. 2, April 18, 2001	Pre RO12	1	25%	1	0
CSI Technologies, Inc., Indian Point 3 Nuclear Power Plant, Moisture Separator Drains CHECWORKS Model, Calculation No. IP3-RPT-MSD-01158, Rev. 2, April 18, 2001	Pre RO12	5	40%-47%	8	5
CSI Technologies, Inc., Indian Point 3 Nuclear Power Plant, Extraction Steam CHECWORKS Model, Calculation No. IP3-RPT-EX-00911, Rev. 2, April 18, 2001, IPEC00220901	Pre RO12	6	38%-57%	8	5

[30] There is one single number that represents the ratio of non-conservative points to the total points in each plot. Each percentage was obtained by counting the number of data points in the non-conservative area of a given plot and dividing them by the total number of data points in that plot. The resulting number varies plot to plot, and due to the large number of plots per CHECWORKS modeling report, a single range is used to represent the number of non-conservative predictions during each outage.

Report Title	Refueling Outage	No. of Plots	(A) % of Non-Conservative CHECWORKS predictions[31]	(B) No. of Data Points Outside +/-50% Lines	(C) No. of LCFs outside Entergy's "acceptable" range of 0.5-2.5
CSI Technologies, Inc., Indian Point 3 Nuclear Power Plant, Extraction Steam CHECWORKS Model, Calculation No. IP3-RPT-EX-00911, Rev. 3, December 2, 2002, IPEC00180979	Pre RO12	6	44%-53%	8	5
Indian Point 3 Nuclear Power Plant, Heater Drains CHECWORKS Model, Report No. IP3-RPT-HD-00979, Rev. 3, March 9, 2003, IPEC00165951	Pre RO12	4	43%-50%	11	1
Indian Point 3 Nuclear Power Plant, Moisture Separator Drains CHECWORKS Model, Report No. IP3-RPT-MSD-01158, Rev. 3, March 9, 2003, IPEC00165622	Pre RO12	5	40%-50%	10	5
CSI Technologies, Inc., Indian Point 3 Nuclear Power Plant, Reheater Drains CHECWORKS Model, Calculation No. IP3-RPT-HD-01144, Rev. 4, May 7, 2004, IPEC00165738	RO12	5	43%-50%	31	3
CSI Technologies, Inc., Indian Point 3 Nuclear Power Plant, Moisture Preseparator Drains CHECWORKS Model, Calculation No. IP3-RPT-HD-00913, Rev. 4, May 7, 2004, IPEC00164944	RO12	1	44%	0	0
CSI Technologies, Inc., Indian Point 3 Nuclear Power Plant, Feedwater CHECWORKS Model, Calculation No. IP3-RPT-FW-00984, Rev. 4, May 7, 2004, IPEC00165451	RO12	3	40%-49%	20	2

[31] There is one single number that represents the ratio of non-conservative points to the total points in each plot. Each percentage was obtained by counting the number of data points in the non-conservative area of a given plot and dividing them by the total number of data points in that plot. The resulting number varies plot to plot, and due to the large number of plots per CHECWORKS modeling report, a single range is used to represent the number of non-conservative predictions during each outage.

Report Title	**Refueling Outage**	**No. of Plots**	**(A) % of Non-Conservative CHECWORKS predictions[32]**	**(B) No. of Data Points Outside +/-50% Lines**	**(C) No. of LCFs outside Entergy's "acceptable" range of 0.5-2.5**
CSI Technologies, Inc., Indian Point 3 Nuclear Power Plant, Condensate CHECWORKS Model, Calculation No. IP3-RPT-COND-00912, Rev. 4, May 7, 2004	RO12	10	40%-50%	15	2
CSI Technologies, Inc., Indian Point 3 Nuclear Power Plant, Extraction Steam CHECWORKS Model, Calculation No. IP3-RPT-EX-00911, Rev. 4, May , 2004, IPEC00181399	RO12	6	38%-53%	16	4
CSI, Technologies, Inc., Indian Point Unit 3 CHECWORKS FAC Model, Calculation No. 050714c-01, Revision 0, October 25, 2005, IPEC00028935	RO13	20	40-60%	54	10
CSI, Technologies, Inc., Indian Point Unit 3 CHECWORKS SFA Model, Calculation No. 0705.100-01, Revision 0, November 14, 2007, IP00055542[33]	RO14	37	40-60%	183	15
CSI, Technologies, Inc., Indian Point Unit 3 CHECWORKS SFA Model, Calculation No. 0705.100-01, Revision 1, February 12, 2010[34]	RO15	37	40-60%	151	13
CSI, Technologies, Inc., Indian Point Unit 3 CHECWORKS SFA Model, Calculation No. 0705.100-01, Revision 2, August 2, 2011[35]	RO16	41	40-60%	150	15

[32] There is one single number that represents the ratio of non-conservative points to the total points in each plot. Each percentage was obtained by counting the number of data points in the non-conservative area of a given plot and dividing them by the total number of data points in that plot. The resulting number varies plot to plot, and due to the large number of plots per CHECWORKS modeling report, a single range is used to represent the number of non-conservative predictions during each outage.

[33] While this report indicates that "[c]omponent inspection data through the Refuel Outage 14 was imported to the model," the data graphs appear to indicate that the comparison to wear predictions were for measurements "@CYCLE 15."

[34] While this report indicates that "[c]omponent inspection data through the Refuel Outage 15 was imported to the model," the data graphs appear to indicate that the comparison to wear predictions were for measurements "@CYCLE 16."

[35] While this report indicates that "[c]omponent inspection data through the Refuel Outage 16 was imported to the model," the data graphs appear to indicate that the comparison to wear predictions were for measurements "@CYCLE 17."

In summary, a review of all of the available CHECWORKS comparison data clearly demonstrates a complete lack of correlation between predictions and measurements, indicating very poor predictive accuracy of the CHECWORKS model at Indian Point. CHECWORKS predictions are at times highly conservative while at other times (40%-60% of the time) they are highly non-conservative. Given the highly erratic predictive behavior of the CHECWORKS model, it is impossible to be sure that future wall thinning predictions will stay conservative or non-conservative, whatever the case may be. Such uncertainty renders the CHECWORKS code useless for objective quantitative assessments. When a model is not capable of correlating predictions with measurements, whether the predictions are predominately conservative or non-conservative, as a matter of sound engineering and science, it simply cannot be considered a suitable tool for informing predictions of where potential pipe failures due to FAC might occur.

CHECWORKS wear predictions covering the period between 2000-2011 revealed no signs that predictions are improving with time. The failure of the CHECWORKS code to produce accurate or useful results at Indian Point *despite decades of use* unmistakably shows that the model continues to lack adequate benchmarking, and will not be properly calibrated before Indian Point enters the rapidly approaching proposed PEO. The level of correlation between CHECWORKS predictions and measured wear following the stretch power uprates at Indian Point is not acceptable, and shows an obvious lack of benchmarking of the code under post power uprate conditions. There is no basis to conclude that the level of power increase at Indian Point is bounded in the computer model by higher power uprates at other plants, since the model is completely failing to produce useful wear predictions at Indian Point. To the contrary, the evidence shows that the code is not capable of accounting for changes in plant parameters at Indian Point.

“Updates” to the CHECWORKS model with have proven unsuccessful: A review of data from numerous post-power uprate outages reveals that the model is not improving in accuracy over time, *at all*, and, in fact, that the model is still unable to provide any kind of useful predictions. There is not enough time to adequately calibrate the CHECWORKS model to account for the power uprate conditions at Indian Point prior to the rapidly-approaching PEO.

Based on the foregoing, 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.

B. The Implications of CHECWORKS' Poor Predictive Accuracy

Entergy's reliance on CHECWORKS to detect FAC at Indian Point during the PEO has important safety implications. Namely, CHECWORKS predictions of wall thinning by FAC at the plant are by far too inaccurate to prevent pipe wall thickness from being reduced below minimum design values. While theoretically conservative predictions would affect plant economics because they would lead to unnecessary or misguided inspections and/or potentially premature component replacement, 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.

The safety consequences of using CHECWORKS can be demonstrated by a few examples. First, assume that a pipe has an initial wall thickness of 0.500" and a critical minimum design thickness (Tcr) of 0.250," and is subject to an actual wall thinning rate of 4.0 mils per year.[36] If CHECWORKS yields a non-conservative, i.e., much slower, wear rate of 0.4 mils per year, the predicted wall thickness of the component after 60 years would be 0.476."[37] The actual wall thickness would be 0.260".[38] In this example, even though CHECWORKS under-predicted the actual thinning, the component and plant will still operate safely because both the predicted and measured values were within the critical design limit of 0.250".

However, now assume that the same pipe is subject to an actual wear rate of 4.5 mils per year, and that, once again, CHECWORKS assumes a much slower wear rate of 0.4 mils per year. In that case, while CHECWORKS would again predict the wall thickness to be 0.476" inches after 60 years, the actual wall thickness would be 0.230",[39] which is below the critical design thickness of 0.250, and out of compliance with the ASME code and NUREG-1801 guidelines. In this example, due to the non-conservative CHECWORKS prediction, an unacceptable level of wall thinning will go undetected. It is questionable that the component and plant could continue to operate safely without intervention, which is rendered "unnecessary" and ostensibly precluded due to use of the CHECWORKS computer model.

These simple examples illustrate that even small changes in the corrosion rate, here 12%, can result in unacceptable levels of FAC, and unsafe plant operations.[40] Notably, in the second example provided above, the critical minimum design wall thickness (Tcr) would not have been reached if the operating time was reduced to a total of 55 years, i.e. an 8% reduction.[41] Thus, the increase in operating life from 40 to 60 years, as contemplated by Entergy's LRA, represents a significant potential for pipe wall thicknesses to fall below designated minimum critical design levels during extended operations, and one would expect that more and more components would become prone to failures after 40 years of service.

Moreover, the above very simplified examples were based on just one set of parameters. In actuality, wear rates, initial pipe wall thickness, and minimum design thickness will vary

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[36] 1 mil = 0.001".

[37] 0.4 mils per year for 60 years is equivalent to a total thinning of 24 mils; 24 mils is equivalent to 0.024 inches; the original thickness of the pipe wall, 0.500", minus a total thinning of 0.024 inches equals a resulting wall thickness of 0.476". *See id.* at x.

[38] The actual wear rate is 4 mils per year; 4 mils per year for 60 years is equivalent to a total thinning of 240 mils; 240 mils is equivalent to 0.24 inches; the original thickness of the pipe wall, 0.500", minus a total thinning of 0.24 inches equals a resulting wall thickness of 0.260".

[39] With an actual wear rate of 4.5 mils per year, 60 years of operation would result in a total thinning of 270 mils; this is equivalent to .27 inches; the original thickness of the pipe wall, 0.500", minus a total thinning of 0.27 inches equals a resulting wall thickness of 0.230". *See id.* at x.

[40] 4 mils per year to 4.5 mils per year is an approximate 12% increase. *See id.* at x.

[41] With an actual wear rate of 4.5 mils per year, 55 years of operation would result in a total thinning of 247.5 mils; this is equivalent to .2475 inches; the original thickness of the pipe wall, 0.500", minus a total thinning of 0.2475 inches equals a resulting wall thickness of 0.2525 inches, which remains above the Tcr of 0.250 inches.

significantly: typically thinning rates will vary between 0.007 mils per year to 50 mils per year; initial component wall thickness can vary between 0.25" to 1.00"; and critical component wall thickness can vary between 0.10" to 0.500". Given such wide variations in parameters, the inaccuracy of CHECWORKS is likely to allow many components to operate outside allowable critical design limits during the PEO. It is, thus, apparent that the non-conservative nature of the CHECWORKS code has very real safety implications at Indian Point.

In addition, conservative predictions may also affect plant safety. Entergy's documentation indicates that Entergy often attributes findings of "negative time to Tcr" (i.e., a finding that a component has low remaining life and should be replaced), to an "often overpredicted," meaning conservative, wear value by CHECWORKS.[42] As a threshold matter, it is far from clear that CHECWORKS generally produces conservative wear values, and in fact, this notion contradicts the findings discussed above that approximately 40%-60% of CHECWORKS wear predictions are *non*-conservative. What's more, this assumption is highly problematic from a safety perspective: if predictions are commonly perceived to be based on conservative estimates, component replacement could be erroneously postponed potentially resulting in an excessive wall thinning.

As the foregoing discussion makes clear, the use of CHECWORKS at Indian Point has safety implications that will likely manifest during the proposed PEO.

C. CHECWORKS Has No Track Record of Performance in Preventing Unacceptable FAC and Component Failures at Indian Point

In addition to evaluating the highly dubious predictive capabilities of the CHECWORKS computer code, the use of the program can also be assessed in terms of the ability of the model to *actually* prevent wall thinning incidents, i.e., an assessment of the number and severity of actual component and pipe failures, including leaks and ruptures, that have occurred at Indian Point over the years since CHECWORKS was introduced. A demonstrated record of performance is necessary to be sure that the model is sufficiently calibrated or benchmarked so as to be an effective predictive tool. This is especially so in light of the long industry-wide history in which CHECWORKS has been unsuccessful in predicting wall thinning.

Entergy has made unsubstantiated claims that "CHECWORKS has a demonstrated record of successfully predicting wall thinning at IPEC and other nuclear power plants."[43] However, a review of FAC related occurrences at Indian Point quickly reveals that CHECWORKS has no track record of performance at Indian Point, and particularly, no track record of performance under the power uprate levels at the plant. In fact, a history of incidents at Indian Point demonstrate that, to date, the code has not been successful in mitigating the effects of FAC, and proves that the safety implications of using CHECWORKS at the plant are far from theoretical, and are, indeed, quite real.

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[42] *See* CSI, Technologies, Inc., Indian Point Unit 3 CHECWORKS SFA Model, Calculation No. 0705.100-01, Revision 2, August 2, 2011, IPEC 00238096, at Appendix K – Components with Negative Time to Tcrit.

[43] *See* Applicant's Motion for Summary Disposition of Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion) (July 26, 2010), ADAMS Accession No. ML102140430, at 22.

Generally, CHECWORKS has a long history of questionably effectiveness at nuclear power plants. In 2005, the Advisory Committee on Reactor Safeguard ("ACRS") Subcommittee on Thermal Hydraulics recognized the poor correlation between CHECWORKS predictions and operating data in relation to the Waterford nuclear power plant, stating that "[i]f you look at the data base, you don't really have too much confidence in CHECWORKS."[44]

The history of CHECWORKS' limited effectiveness to predict wall thinning is further demonstrated in NUREG/CR-6936, PNNL *16186,Probabilities of Failure and Uncertainty Estimate Information for Passive Components* - a *Literature Review* (May 2007) ("NUREG/CR-6936"), which documented the service experience with FAC covering two periods, 1976-1987 and 1988- 2005. Given that CHECWORKS was released to the industry in 1987, and presuming that all plants have been using it, a comparison of the number of pipe failures in the first period with the number of failures in the second period is a measure of the success of CHECWORKS in predicting FAC. The number of through the wall failures in pressurized water reactor ("PWR") plants was 89 and 150 during the 1976-1987 and 1988-2005 periods respectively.[45] This represents an annual failure rate of 8 and 8.8, clearly demonstrating that CHECWORKS was not effective in reducing the number pipe failures.[46]

Additionally, in the years since the nuclear power plant industry began using CHECWORKS, failures due to undetected FAC have persisted. In addition to the specific instances discussed above at San Onofre in 1993, Fort Calhoun in 1997, and Mihama in 2004,[47] 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. The NRC has recognized the seriousness and persistence of FAC throughout the nuclear industry.[48] Given the numerous leaks and pipe ruptures from wall thinning which have occurred at nuclear power plants since its introduction in the late 1980s, the success of CHECWORKS has been questionable. The facts plainly shows that CHECWORKS has a demonstrable history of failing to prevent FAC at nuclear power plants.

Entergy has previously implied that the implementation and 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."[49] However, this information by itself is purely circumstantial, and cannot lead one to conclude that CHECWORKS had been a success. This does not change the reality that FAC that violates applicable standards has been well-documented across the industry, clearly

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[44] 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.

[45] *See* NUREG/CR-6936.

[46] *See* NUREG/CR-6936.

[47] *See supra* p.3.

[48] *See supra* Note 4.

[49] *See* Applicant's Motion for Summary Disposition of Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion) (July 26, 2010), ADAMS Accession No. ML102140430, at 23.

demonstrating that CHECWORKS has *not* been successful. Indeed, the fact that a given plant has not experienced pipe ruptures is not permission for the plant to operate with pipes of unknown and unacceptable wall thickness. The "leak-before-break" concept is not an excuse for operating with excessively worn-out components.

Entergy has also indicated its confidence in the capability of CHECWORKS by emphasizing the fact that "[t]here were five fatalities at Japan's Mihama nuclear plant *not* using CHECWORKS in 2004."[50] However, the pipe rupture at Mihama occurred at low velocity in a straight pipe, downstream of an orifice, and CHECWORKS does not model such flow situations correctly. There is no analysis to show that the use of CHECWORKS at that plant would have prevented the accident from happening. There is simply no evidence to suggest that CHECWORKS has been a reliable tool in the industry for predicting and preventing FAC.

In particular regard to Indian Point, the use of CHECWORKS has, similar to the industry experience with this computer model, proved ineffective. A review of plant-specific data from both before and after the power uprate at the plant, as discussed at length above, clearly demonstrates that CHECWORKS results have been, and continue to be highly unreliable at Indian Point. This alone undeniably establishes a complete lack of an acceptable track record of CHECWORKS results at Indian Point.

Moreover, numerous leaks and reports of excessive wall thinning in mechanical systems at Indian Point indicates that CHECWORKS has not been successful at preventing FAC related occurrences. For example, Entergy's 2008 Operating Experience Review Report, IP-RPT-08-LRD05, Rev. 3, documents numerous unacceptable wall thinning events which have occurred at Indian Point Units 2 and 3 from 2001 to 2006.[51] Various Entergy condition reports memorialize numerous FAC occurrences where inspections uncovered component wall thickness below minimum allowable limits, including leaks from components that resulted from undetected FAC, where subsequent inspections revealed wall thinning below minimum acceptable levels.[52]

Furthermore, the NRC Staff in this license renewal proceeding has also questioned Entergy regarding several incidences of component wall thinning that were below minimum acceptable levels, as memorialized during a meeting of the Advisory Committee on Reactor Safeguards regarding Entergy's LRA.[53]

While it is not possible to assess whether the number of failures has increased since the owners of Indian Point began using CHECWORKS in the 1980s due to the fact that data for years prior

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[50] *See* Applicant's Motion for Summary Disposition of Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion) (July 26, 2010), ADAMS Accession No. ML102140430, at 23, fn.133.

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

[52] *See* Daily DER Report, DER-01-01522, April 25, 2001, IPEC00020501; Entergy Operations, Inc, Condition Report List, IPEC00185743; Entergy Operations, Inc., Condition Report List, IPEC00092552; Entergy Condition Report CR-IP2-2001-10525, IPEC00092616; Entergy Condition Report CR-IP3-2006-02270, IPEC00025699.

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

to approximately 2000 is "unavailable," it is clear that FAC-related failures continue to occur despite the use of CHECWORKS at the plant. As discussed above, as Indian Point continues to age past 40 years, it is reasonably foreseeable that more and more components will be prone to thinning and failure. Furthermore, the number of failures at Indian Point appear to be significantly higher than in the rest of the industry where only 15 leaks were reported in the 18 month period between January 2000 and July 2001.[54]

The evidence irrefutably shows that there is currently no track record of performance of the CHECWORKS code at Indian Point, as the model has not been able to preventatively detect FAC before component wall thickness dips below minimum design requirements. Numerous such instances demonstrate the Entergy's use of CHECWORKS violates the ASME code and NUREG-1801, and poses tangible safety related concerns, as manifested by the various leaks that have occurred at Indian Point due to undetected FAC. Accordingly, CHECWORKS cannot be considered an appropriate or useful tool for managing FAC at Indian Point during the PEO.

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

The newest revision of the *GALL Report* clarifies the previous version in relation to the appropriateness of relying on CHECWORKS. While the *GALL Report*, Revision 1 only 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,"[55] 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."[56] 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."[57] The use of CHECWORKS is only acceptable if it provides conservative results, and if not, that the user re-calibrates it accordingly. NRC has otherwise recognized the need to benchmark analytic codes.[58]

Entergy, on the other hand, bases its use of CHECWORKS on *very different criteria*. In particular, Entergy believes Entergy's results within arbitrary +/-50% lines demonstrate the efficacy of CHECWORKS, or if the LCF is between 0.5 and 2.5. Data within the +/-50% range

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[54] *See* Entergy Engineering Report, Operating Experience Review Report, IP-RPT-06-LRD05, Rev. 3 (2008), IPEC00186046.

[55] *GALL Report*, Rev. 1 at XI M-61 to XI M-62.

[56] *GALL Report*, Rev. 2 at XI M17-1.

[57] *GALL Report*, Rev. 2 at XI M17-2.

[58] *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. (Vermont Power Station, Docket No. 50-571), at § 2.8.7.1, p. 190, ADAMS Accession No. ML060050028 ("Application of NRC-approved Analytical Methods and Codes. . . . In general, the analytical methods and codes are assessed and benchmarked against measurement data, comparisons to actual nuclear plant test data and research reactor measurement data. The validation and benchmarking process provides the means to establish the associated biases and uncertainties. The uncertainties associated with the predicted parameters and the correlations modeling the physical phenomena are accounted for in the analyses.").

can actually indicate highly non-conservative under-predictions by up to a factor of 2. This is not the "bounding analysis" to which the *GALL Report* refers. The "bounding analysis" discussed in *GALL* is one that provides conservative results.

The CHECWORKS model at Indian Point has produced non-conservative results about 50% of the time,[59] and many times data fell outside the broad range that Entergy considered appropriate and "bounding," i.e., the +/-50% lines. The model is, therefore, *not* properly calibrated and, according to *GALL*, Revision 2, it must be re-calibrated. However, years of apparent attempts to recalibrate CHECWORKS at Indian Point have not yielded *any* improvement in the predictive capability of the code, and it continues to yield non-conservative, inaccurate results. CHECWORKS would have to be recalibrated *continuously* in order to meet the standard in the *GALL Report*. When a code has to be calibrated continuously it ceases to be a predictive tool. Calibration of CHECWORKS on a continuous basis would prevent plant operators from being able to determine whether the critical thickness of any given component will be reached before the next cycle, and, in turn, when component repair or replacement is necessary. Because CHECWORKS continues to produce non-conservative results after decades of use, there is no way to ensure appropriate calibration of the model prior to, or even during, the proposed period of extended operations at Indian Point. Thus, according to the *GALL Report*, Revision 2, which only condones the use of CHECWORKS if it produces conservative results or if can be re-calibrated to the extent results are not conservative, the use of CHECWORKS at Indian Point is not acceptable. In other words, Entergy's reliance upon CHECWORKS, does not demonstrate an AMP for FAC that is consistent and compliant with the *GALL Report*.

In addition, CHECWORKS cannot be used to demonstrate that Entergy's FAC program will meet applicable acceptance criteria as discussed in the *GALL Report*, since CHECWORKS is not capable of accurately calculating the number of operating cycles remaining before a component will reach the minimum allowable wall thickness.[60] The inability to ensure the maintenance of minimum design wall thicknesses also violate the ASME code.[61]

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

E. The Use of CHECWORKS at Vermont Yankee is Irrelevant to the Indian Point License Renewal Proceeding

Any information pertaining to the acceptability of the use of CHECWORKS at the Vermont Yankee nuclear power plant is not relevant to the Indian Point license renewal proceeding. This is for several reasons. First, safety must be evaluated in each plant separately to account for the differences in flow velocities, temperatures, geometry, material, and coolant chemistry. Indeed,

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[59] *See supra*, Table 1.

[60] *See* GALL Report, Rev. 1 at XI M-62; GALL Report, Rev. 2 at XI M17-2.

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

there are major differences between the Vermont Yankee and the Indian Point plants. 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 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.[62]

An evaluation of a nuclear power plant operator's use of CHECWORKS necessarily depends upon plant specific data. Such data generated specifically in relation to Indian Point is indicative of how CHECWORKS operates, *at* Indian Point, notwithstanding any findings about the prospective viability CHECWORKS at another plant.

Based on my personal involvement in Vermont Yankee License Renewal Proceeding, it is apparent that there are key differences between the use of CHECWORKS at Entergy's two plants. First, at Vermont Yankee, data from post-power uprates was not yet available when the FAC program was under review. In contrast, at Indian Point ample post-power uprate data shows that CHECWORKS is not adequately benchmarked. Second, at Vermont Yankee, prolonged benchmarking was found not to be necessary because the plant had the benefit of data dating back to 1989. However, at Indian Point, Entergy maintains that historical data is irrelevant, and, in any event, has apparently lost most of the universe of CHECWORKS data. As such, no lengthy record is available to determine the adequacy of the CHECWORKS model at Indian Point, as it was in the case of Vermont Yankee. Further, even if such data was available, given the inaccuracy of the CHECWORKS code at Indian Point, it would be difficult to conclude that such data has ensured that the model has been benchmarked properly. Third, at Vermont Yankee, the operators indicated that only a small fraction of inspection locations were based on the use of CHECWORKS. At Indian Point, as discussed further below, Entergy's FAC program relies integrally upon CHECWORKS for determining inspection locations. Fourth, while the CHECWORKS code may have been somehow "bounding" at Vermont Yankee, the same is not true at Indian Point. In particular, even though the actual percentage of power changes were lower at Indian Point than at Vermont Yankee, Indian Point is a much larger facility, and a different, i.e., more susceptible, kind of reactor. Thus, a plant specific evaluation is necessary. 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. It is simply 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, especially in light of evidence to the contrary. Notably, at Vermont Yankee, it was assumed that the CHECWORKS model would adequately account for the changes in plant operating conditions from the power uprate at that plant, since actual data was not yet available. At Indian Point, it is already apparent that the code is not sufficiently benchmarked to account for the power uprates that occurred at the plant in 2004 and 2005.

[62] *See* NUREG/CR-6936 at p.5.25.

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

Entergy's has indicated that the FAC program at Indian Point will be effective in managing FAC-related aging effects because "CHECWORKS is only *one* of several bases used by Entergy to select and schedule in-scope components for inspection."[63] In particular, Entergy has stated that even if CHECWORKS is an ineffective tool for predicting FAC, the FAC program at Indian Point would still be effective, since 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.[64] However, these "additional" tools for selecting inspection locations do not demonstrate the effectiveness of Entergy's FAC AMP. These additional criteria cannot be viewed as independent tools sufficient to establish an accurate FAC inspection scope. A close examination reveals that these additional criteria largely depend upon, and are an integral part of using, CHECWORKS.

In relation to the first "additional tool," 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. Accordingly, this is a required input for the use of CHECWORKS and not a stand-alone "tool" for component selection. Moreover, for components initially selected for inspection by CHECWORKS, any decisions 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.

In relation to the second and third "additional tools," industry and plant experience (such as information about wall thinning events at Indian Point, changed plant parameters, or severe pipe wall thinning events at other plants, which NRC releases periodically to maintain a knowledge base throughout the industry), are also types of information that feed directly into the CHECWORKS model. These are not necessarily independent tools for identifying and specifying the inspection scope during outages, that is, determining how many and which components should be inspected. Rather, 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.

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.[65] Rather, of the four additional tools identified by Entergy, only

[63] Applicant's Motion for Summary Disposition of Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion) (July 26, 2010), ADAMS Accession No. ML102140430, at 17.

[64] Applicant's Motion for Summary Disposition of Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion) (July 26, 2010), ADAMS Accession No. ML102140430, at 17 and Attach. 2, ¶¶ 39.

[65] EPRI, Recommendations for an Effective Flow-Accelerated Corrosion Program, NSAC-202L-R3, at 2-4 (explaining that "good engineering judgment" requires "that personnel involved in the program be aware of operating experience. . . and receive input from . . . plant operations . . .")).

engineering judgment can be considered an independent tool for managing FAC. However, this is not an *effective* tool by itself. As the EPRI guidance document upon which Entergy's FAC program relies explains, "engineering judgment cannot substitute for other factors."[66] Generally speaking, it is commonly recognized in all major industrial plants that engineering judgment alone is not sufficiently reliable to prevent component failures from wall thinning. The development of the CHECWORKS computer model itself stemmed from the realization by the nuclear industry that engineering judgment alone was no longer enough to be able to detect unacceptable and unsafe wall thinning occurrences.

Engineering judgment is intrinsically subjective. When engineering judgment is identified as an independent 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. However, 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. 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.

To the contrary, Entergy's FAC program fails 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. Entergy has not identified any kind of systematic methodology which demonstrates that engineering judgment is a separate predictive tool that would adequately meet the guidelines contained in the *GALL Report* and manage FAC related component degradation during the period of extended operation.

In particular, there are several elements that are key in forming a sound engineering judgment as it relates to FAC at Indian Point, which Entergy does not appear to 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.

In relation to good documentation, it is reasonable to expect that over forty years of operating, many changes have occurred in the management of FAC at Indian Point as a result of new ownership, and personnel changes, and also due to changed operating conditions. To ensure well-founded engineering judgment, management must guarantee that all aspects of FAC experience (including the accuracy of past predictions, repairs, changes in plant operating conditions like water chemistry, and the like) are well documented so future engineering judgment will be based on sound knowledge of plant history. At Indian Point, it is apparent that more than half of the overall amount of CHECWORKS-related data and documentation has been lost, as Entergy has indicated that such information related to Unit 2 predating the year 2000 is not in Entergy's possession, and that information related to Unit 3 predating 2001 either does not exist or would be too burdensome to generate. When such a substantial amount of data and documentation is unavailable, a complete revalidation of the program would be appropriate. The

[66] EPRI, Recommendations for an Effective Flow-Accelerated Corrosion Program, NSAC-202L-R3, at 2-4.

lack of a complete institutional history concerning the performance of the FAC program at Indian Point is a clear hindrance to the ability to form sound engineering judgment.

In relation to communication, when an outside vendor is hired to run CHECWORKS, generate the analyses, and/or conduct other FAC assessments, it is important that good communication be maintained between the vendor and the plant operator. Good communication ensures that problems are identified early and appropriate actions are taken to resolve them. It is not apparent that adequate communication between Entergy personnel and the outside organization that runs the CHECWORKS assessments, CSI Technologies, Inc. exists at Indian Point. For example, the documents identified by Entergy as relevant to Riverkeeper's Contention TC-2 revealed numerous anomalies in CHECWORKS results; however, the record of documents did not show thorough discussions between CSI, Technologies, Inc. and plant operators regarding the significance of such anomalies. This apparent lack of adequate communication further hinders sound engineering judgment.

In relation to knowledge of FAC assessment methods, it critical to understand the engineering model employed, i.e., CHECWORKS, as it is the predominant feature of FAC program. CHECWORKS is an empirical model, and so only valid within the range of variable for which the model was calibrated. To gain an understanding of the model and the knowledge necessary to have a well-founded engineering judgment related to FAC management, it is important to observe the response of the model to changes in input variables. The changes in velocities and temperatures at Indian Point Units 2 and 3 due to power uprates in 2004 and 2005 provided this opportunity. However, because most of the data and documents prior these power uprates is not available (and/or does not exist), a comparison to past performance, and the model's response to input variables is very limited. Moreover, Entergy has stated that information about the use of CHECWORKS prior to the power uprates is irrelevant for purposes of assessing the validity of CHECWORKS. It, thus, is apparent that past experience with CHECWORKS is not being used to enhance engineering knowledge on FAC at Indian Point.

Lastly, regarding knowledge of risks and consequences, when exercising engineering judgment, it is necessary to understand and take into account the varying safety risks posed by FAC. For example, a pipe rupture of a small pipe in the service water system does not pose a risk that could lead to a severe reactor accident, while a FAC-induced rupture of a main feedwater or steam line pipe may lead to an uncontrolled severe accident. The documents and information provided by Entergy do not reflect adequate consideration of inspection priorities of FAC-susceptible components relative to the safety risks posed due to a FAC-related failure.

Accordingly, Entergy has completely failed to demonstrate that engineering judgment alone will safely manage FAC at Indian Point.

It is, thus, 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. Rather, Entergy's program for managing FAC relies heavily on the unreliable CHECWORKS code.

VI. Entergy's Failure to Address Safety Issues Posed Due to Inadequate Aging Management of FAC During the PEO

The delay of necessary pipe inspections, repairs, replacements, and/or other corrective action due to over-dependence on a demonstrably ineffective predictive tool could result in serious safety issues at Indian Point during the proposed period of extended operation. 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.

This is particularly important when Indian Point is subject to sudden transient loads where it may be too late to detect a leak and prevent a component failure. For example, the feed water distribution piping ring inside the steam generators is subjected to high local velocities, (greater than 20 feet per second), and turbulence. With severely degraded walls, this pipe may rupture under transient loads causing damage to other structures within the steam generators, like tubes. Notably, Entergy has not provided data on CHECWORKS predictions for components inside the steam generators.

In addition, undetected FAC during the extended operating terms at Indian Point also poses a risk of loss of coolant accidents ("LOCA") in violation of NRC's General Design Criterion ("GDC") 4, which 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."[67] Notably, when the original Indian Point probabilistic risk assessments ("PRAs") were developed, the effects of aging were not included, and it was assumed that pipes were in pristine conditions. In actuality, the probability of a pipe failing under a given load will be reduced when the walls have been degraded.

Adequate protection is particularly 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 to earthquakes of up to 7.0 magnitude,[68] an NRC report from August 2010 (in conjunction with supplemental data regarding power plants not reviewed in the report) reveals that Indian Point Unit 3 has the *highest* risk of seismic related core damage than any other nuclear power plant in the country.[69] Another important class of accidents that

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[67] 10 C.F.R. Part 50, Appendix A, General Design Criteria for Nuclear Power Plants, *Criterion 4—Environmental and dynamic effects design bases.*

[68] 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*"); 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 March 24, 2011).

[69] *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; 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).

depends on reliable knowledge of wall thickness of various components are station blackouts, SBOs. The fact that Entergy has not demonstrated that it has any reliable method of predicting component wall thinning casts a doubt about Entergy's risk predictions relating to such accidents.

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. In addition, 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.

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 limits Entergy's ability to determine the degree of pipe degradation and reduction in strength. Entergy has failed to show that despite such uncertainty, Indian Point would continue to operate in compliance with GDC 4, and without a severe accident occurring.

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

Due to reliance on CHECWORKS, Entergy's program for managing FAC during the PEO will not be effective. Entergy's fleet-wide procedure, EN-DC-315[70] and EPRI guidance document NSAC-202L-R3,[71] through which Entergy claims to implement a FAC program that is adequate and which complies with the *GALL Report*, are focused heavily on the appropriate use of CHECWORKS. Like the *GALL Report*, Revision 2, these documents imply that CHECWORKS should be properly benchmarked or calibrated. Since CHECWORKS is woefully inaccurate, and clearly far from properly benchmarked, Entergy's FAC management program is inadequate.

Accordingly, in order to comply with the *GALL Report* and SRP-LR, Entergy must provide sufficient details to address all relevant program elements, including the method for determining component inspections, frequency of such inspections, and attendant criteria for component repair and replacement.[72] 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.

Entergy lacks a sufficiently detailed AMP to demonstrate that the aging effects of FAC will be adequately managed throughout the proposed PEO.

[70] EN-DC-315, Revision 3, *Flow Accelerated Corrosion Program* (March 1, 2010).

[71] EPRI, Recommendations for an Effective Flow-Accelerated Corrosion Program, NSAC-202L-R3.

[72] *See* SRP-LR at § A.1.2.3.

VIII. Conclusion

Entergy's AMP pertaining to FAC during the PEO is inadequate due to the following:

- CHECWORKS is not adequately benchmarked so as to be an effective tool for predicting FAC at Indian Point during an extended period of operation;
- CHECWORKS has no track record of performance at Indian Point;
- 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;
- 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; and
- Entergy fails to consider the very serious safety implications posed by the inadequacies of its FAC program.

Accordingly, Entergy has failed to demonstrate that the aging effects of FAC will be adequately managed for the PEO.

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