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

In the Matter of )  
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)

Entergy Nuclear Operations, Inc. )  
(Indian Point Nuclear Generating )  
Units 2 and 3) )

Docket Nos.  
50-247-LR  
and 50-286-LR

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**RIVERKEEPER OPPOSITION TO ENTERGY'S MOTION  
FOR SUMMARY DISPOSITION OF RIVERKEEPER TECHNICAL  
CONTENTION 2 (FLOW-ACCELERATED CORROSION)**

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Pursuant to 10 C.F.R. § 2.1205(b) and the Atomic Safety and Licensing Board's ("ASLB") Scheduling Order dated July 1, 2010,<sup>1</sup> Riverkeeper, Inc. ("Riverkeeper") hereby submits this answer in opposition to Entergy Nuclear Operations, Inc.'s ("Entergy") Motion for Summary Disposition of Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion). For the reasons set forth below, summary disposition is inappropriate and Entergy's motion must be denied. Riverkeeper's answer in opposition to the instant motion is supported by the attached Counter-Statement of Material Facts (Attachment 1), the Declaration of Dr. Joram Hopfenfeld (Attachment 2), and numerous other supporting attachments (Attachments 3-8).

**BACKGROUND**

The instant proceeding stems from the license renewal application Entergy filed with the Nuclear Regulatory Commission ("NRC") in April 2007 seeking to extend the operating licenses of Indian Point Units 2 & 3 for an additional 20 years. On November 30, 2007, Riverkeeper filed a Request for Hearing and Petition to Intervene in the proceeding, asserting, *inter alia*, a

<sup>1</sup> See 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 11, ADAMS Accession No. ML101820387.

technical safety contention, RK-TC-2, challenging the sufficiency of Entergy's plan to adequately manage an aging phenomenon known as "flow accelerated corrosion" (hereinafter "FAC").<sup>2</sup> The ASLB's ruling on contention admissibility dated July 31, 2008, admitted RK-TC-2 for an adjudicatory hearing.<sup>3</sup>

The ASLB determined that RK-TC-2 raised material "questions regarding the sufficiency of Entergy's AMP to demonstrate that a specific class of components subject to FAC will be managed so that their intended functions will be maintained during the period of extended operations."<sup>4</sup> In particular, as characterized by the ASLB, RK-TC-2,

contends that (1) Entergy's AMP for components affected by FAC is deficient because it does not provide sufficient details (e.g., inspection method and frequency, criteria for component repair or replacement) to demonstrate that the intended functions of the applicable components will be maintained during the extended period of operation; and (2) Entergy's program relies on the results from CHECWORKS without benchmarking or a track record of performance at IPEC's power uprate levels.<sup>5</sup>

Entergy's has now made a motion to summarily dismiss RK-TC-2 claiming that no genuine dispute of material fact exists to litigate. The following amply demonstrates that numerous factual issues remain, warranting complete dismissal of the instant motion.

### STANDARD OF REVIEW

The regulations at 10 C.F.R. § 2.1205 govern summary disposition motions and direct Licensing Boards to "apply the standards for summary disposition set forth in Subpart G."<sup>6</sup>

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<sup>2</sup> See Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene in Indian Point License Renewal Proceeding (November 30, 2007), at 15-23, ADAMS Accession No. ML073410093.

<sup>3</sup> Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3), Docket Nos. 50-247-LR, 50-286-LR, ASLBP No. 07-858-03-LR-BD01, LBP-08-13, Memorandum and Order (Ruling on Petitions to Intervene and Requests for Hearing (July 31, 2008), ADAMS Accession No. ML082130436, at 162-69 (hereinafter "ASLB Contention Admissibility Order").

<sup>4</sup> ASLB Contention Admissibility Order, *supra* note 3, at 167.

<sup>5</sup> *Id.* at 169. The power uprates at occurred at Indian Point Unit 2 and Unit 3 in 2004 and 2005, respectively. See *id.* at 167.

<sup>6</sup> 10 C.F.R. § 2.1205(c).

Under Subpart G, summary disposition is appropriate if the filings in the proceedings, statements of the parties and affidavits, if any, “show that there is no genuine issue as to any material fact and that the moving party is entitled to a decision as a matter of law.”<sup>7</sup> In a motion for summary disposition, the moving party bears the burden to demonstrate the absence of a genuine issue as to any material fact.<sup>8</sup> Any doubt as to the existence of a genuine issue of material fact is resolved against the moving party.<sup>9</sup> “Because the burden is on the moving party, the Board must examine the record in the light most favorable to the non-moving party and give the non-moving party the benefit of all favorable inferences that can be drawn from the evidence.”<sup>10</sup>

A party opposing a motion for summary disposition need not show a likelihood of success on the merits, but rather, only that there is a genuine issue of fact to be evaluated at the evidentiary hearing.<sup>11</sup> Indeed, summary disposition “is not a tool for trying to convince a Licensing Board to decide, on written submissions, genuine issues of material fact that warrant resolution at a hearing.”<sup>12</sup> As the Commission recently elaborated upon, “a licensing board (or presiding officer) should not . . . conduct a ‘trial on affidavits.’ At this stage, ‘the judge’s function is not himself to weigh the evidence and determine the truth of the matter but to

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<sup>7</sup> *Id.* § 2.710(d)(2).

<sup>8</sup> *Id.* § 2.325; *Advanced Med. Sys., Inc.* (One Factory Row, Geneva, Ohio, 44041), CLI-93-22, 38 NRC 98, 102 (1993); *Entergy Nuclear Vermont Yankee LLC* (Vermont Yankee Nuclear Power Station), LBP-06-5, 63 NRC 116, 121 (2006) (quoting *Private Fuel Storage, LLC* (Independent Spent Fuel Storage Installation), LBP-01-39, 54 NRC 497 (2001)).

<sup>9</sup> *Entergy Nuclear Vermont Yankee LLC* (Vermont Yankee Nuclear Power Station), LBP-06-5, 63 NRC 116, 121 (2006) (citing *Advanced Med. Sys., Inc.* (One Factory Row, Geneva, Ohio, 44041), CLI-93-22, 38 NRC 98, 102 (1993)).

<sup>10</sup> *Id.*

<sup>11</sup> *Advanced Med. Sys., Inc.* (One Factory Row, Geneva, Ohio, 44041), CLI-93-22, 38 NRC 98, 102 (1993)

<sup>12</sup> *Entergy Nuclear Vermont Yankee LLC* (Vermont Yankee Nuclear Power Station), LBP-06-5, 63 NRC 116, 121 (2006) (quoting *Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), LBP-01-39, 54 N.R.C. 497, 509 (2001)).

determine whether there is a genuine issue for [hearing].’<sup>13</sup> Accordingly, “[i]f ‘reasonable minds could differ as to the import of the evidence,’ summary disposition is not appropriate.”<sup>14</sup>

As the ASLB has already recognized in this proceeding, when conflicting expert opinions are involved, summary disposition is unsuitable.<sup>15</sup> Indeed, “competing expert opinions present the ‘classic battle of the experts’ and it [is] up to [the finder of fact] to evaluate what weight and credibility each expert opinion deserves.”<sup>16</sup> At the summary disposition stage, “[r]egardless of the level of the dispute . . . it is not proper for a Board” to choose which expert has the better of the argument.<sup>17</sup>

## ARGUMENT

### POINT I: GENUINE MATERIAL FACTS ARE IN DISPUTE CONCERNING ENTERGY’S RELIANCE ON CHECWORKS

#### A. The CHECWORKS Computer Code Lacks Adequate Benchmarking to Assure Reliable Predictive Results Under Post-Stretch Power Uprate (“SPU”) Operating Conditions at Indian Point During the Period of Extended Operation

A genuine dispute exists concerning whether CHECWORKS is adequately benchmarked so as to assure reliable predictive results under post power uprate conditions at Indian Point during the period of extended operation.<sup>18</sup> Notwithstanding Entergy’s various claims that the CHECWORKS model can handle a wide range of operating parameters and that the model has been appropriately “updated” with changed plant parameters as well as actual measured wear

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<sup>13</sup> Entergy Nuclear Generation Co. and Entergy Nuclear Operations, Inc. (Pilgrim Nuclear Power Station), CLI-10-11, 71 NRC \_\_, \_\_ (slip op. at 13) (Mar. 26, 2010).

<sup>14</sup> *Id.*

<sup>15</sup> Licensing Board Memorandum and Order (Ruling on Motions for Summary Disposition) (Nov. 3, 2009), at 1-2, ADAMS Accession No. ML093070521.

<sup>16</sup> See *Entergy Nuclear Vermont Yankee LLC* (Vermont Yankee Nuclear Power Station), LBP-06-5, 63 NRC 116, 121 (2006) (citing *Phillips v. Cohen*, 400 F.3d 388, 399 (6th Cir. 2005)).

<sup>17</sup> *Entergy Nuclear Vermont Yankee LLC* (Vermont Yankee Nuclear Power Station), LBP-06-5, 63 NRC 116, 121 (2006) (citing *Private Fuel Storage, L.L.C. (Independent Spent Fuel Storage Installation)*, LBP-01-39, 54 NRC 497, 510 (2001)).

<sup>18</sup> ASLB Contention Admissibility Order, *supra* note 3, at 167 (ASLB finding that “Riverkeeper has presented sufficient facts and expert opinion to raise a genuine dispute regarding a material issue”).

rates, the actual performance of CHECWORKS demonstrates that it is not a reliable tool for predicting wall thinning at Indian Point.

CHECWORKS modeling reports generated on behalf of Entergy subsequent to the power uprates reveal that CHECWORKS predictions of wall thinning are highly unreliable. Graphs plotting CHECWORKS predictions of wall thickness versus actual measurements for selected plant components, for at least seven different outages at Units 2 and 3 after the power uprates, show an unacceptably large margin of error in CHECWORKS predictions.<sup>19</sup> The wide scatter of data points on such graphs, examples of which are appended hereto as Attachment 7, show that CHECWORKS predictions are far from accurate.<sup>20</sup> Indeed, one could draw almost any line through the data on such graphs, indicating a complete lack of correlation.<sup>21</sup> A straight line parallel to the abscissa would indicate that actual plant observations and computer model predictions are independent of each other.<sup>22</sup> Arbitrary lines are drawn on these graphs to show that some, but not all of the data, can be bound with +/- a factor of two.<sup>23</sup> In fact, a review of such graphs shows that predictions can deviate by as much as factor of +/- 10.<sup>24</sup> Accordingly CHECWORKS can either under-predict or over-predict FAC by 1000%.

It is, thus, apparent that the CHECWORKS model employed at Indian Point cannot predict FAC to any degree of accuracy.<sup>25</sup> A margin of error high as +/- 1000% exhibited by a significant number of components, is not a demonstration of precise and accurate results, as Entergy asserts.<sup>26</sup> On the contrary, CHECWORKS can only predict an overall range of

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<sup>19</sup> See Hopenfeld Declaration (Attach. 2), ¶¶ 12-14; Attach. 7.

<sup>20</sup> See Hopenfeld Declaration (Attach. 2), ¶ 13; Attach. 7.

<sup>21</sup> See Hopenfeld Declaration (Attach. 2), ¶¶ 13, 17; Attach. 7.

<sup>22</sup> See Hopenfeld Declaration (Attach. 2), ¶ 13; Attach. 7.

<sup>23</sup> See Hopenfeld Declaration (Attach. 2), ¶ 13; Attach. 7.

<sup>24</sup> See Hopenfeld Declaration (Attach. 2), ¶ 13; Attach. 7.

<sup>25</sup> See Hopenfeld Declaration (Attach. 2), ¶ 14.

<sup>26</sup> See *id.*



corrosion rates for a given a component or a group of components.<sup>27</sup> This range is too wide for practical applications, especially when the consequences of component failure are safety related.<sup>28</sup> As such, Entergy's apparent position that the level of correlation between the CHECWORKS model predicted wear and the measured wear following implementation of the stretch power uprates at Indian Point is acceptable, is untenable.<sup>29</sup>

Such conclusions are further bolstered by Entergy's arbitrary reliance on a "line correction factor" to "compare and adjust CHECWORKS predictions to match inspection data."<sup>30</sup> As Entergy documentation explains, "[t]he LCF indicates the degree to which CHECWORKS over or under-predicts wear. A reasonable LCF should be between 0.5 and 2.5."<sup>31</sup> Entergy's own documentation reveals numerous instances where the LCF was outside of this range, indicating that CHECWORKS is unreasonably failing to predict wear rates.<sup>32</sup> Moreover, Entergy has provided no justification to support the conclusion that the LCF range of 0.5 to 2.5 is acceptable, or, in particular, how this LCF range would be an indication that CHECWORKS can be used to accurately predict inspection locations.<sup>33</sup> Furthermore, Entergy has failed to show how "adjusting" CHECWORKS predictions using an LCF has made, or will make, the model more accurate, as claimed, since years of modeling reports show consistently inaccurate results, as discussed above.<sup>34</sup>

Based on the foregoing, Riverkeeper disputes Entergy's claim that CHECWORKS is "a viable and effective tool for selecting and prioritizing IPEC piping and piping component

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<sup>27</sup> See *id.* ¶ 12.

<sup>28</sup> See *id.*

<sup>29</sup> See *id.* ¶ 17. Notably, Entergy's experts say that the level of correlation meets their "expectations" without defining what that means. This only serves as further doubt that CHECWORKS results are acceptable, since subjective "expectations" do not necessarily correspond to an acceptable level of performance.

<sup>30</sup> See Entergy Motion for Summary Disposition, Attach. 2 at ¶ 48.

<sup>31</sup> See Attach. 8; see also Entergy Motion for Summary Disposition, Attach. 9 at 4-1.

<sup>32</sup> See, e.g., Attach. 7, Figures 12-15, 17-18.

<sup>33</sup> See Hopenfeld Declaration (Attach. 2), ¶¶ 15-16.

<sup>34</sup> See *id.* ¶¶ 12-14.

locations for inspection to detect and mitigate FAC during the period of extended operation.”<sup>35</sup>

This raises a material and genuine issue of fact regarding whether the CHECWORKS model is adequately calibrated or benchmarked at Indian Point to assure reliable predictions during the period of extended operation.<sup>36</sup>

Entergy attempts to argue that findings of a different Atomic Safety and Licensing Board in the Vermont Yankee license renewal proceeding (“VY ASLB”) should be dispositive in the instant proceeding.<sup>37</sup> Generally speaking, the conclusions of the VY ASLB are specific to the continued operation of VY and, therefore, cannot be generically applied in the instant proceeding. No where did the VY ASLB state that their conclusions were universal. In fact, that board’s decision referenced the role of plant specific inputs and data in the FAC program at VY numerous times, leaving no doubt that the conclusions reached by the VY ASLB are restricted to the VY plant.<sup>38</sup>

Notwithstanding the obvious inappropriateness of relying upon the findings of a licensing board in a wholly separate and distinct proceeding, Entergy points to the VY ASLB’s finding that 10 to 15 years of additional benchmarking of the CHECWORKS model at VY was not necessary because Entergy would have three sets of data at the uprated power levels before that plant entered into its period of extended operation.<sup>39</sup> However, in coming to this conclusion, the VY ASLB did not have the benefit of any data for the VY plant at the uprated power levels. The

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<sup>35</sup> See Entergy Motion for Summary Disposition, Att. 2 at 28-29; Hopenfeld Declaration (Attach. 2), ¶ 8.

<sup>36</sup> The ASLB has already recognized that “neither Entergy nor the NRC Staff [] provided any support for the claim that the inspection data that will be collected during refueling outages prior to the license renewal period will be sufficient to benchmark the model.” ASLB Contention Admissibility Order, *supra* note 3 at 168. Entergy now attempts to claim that no factual dispute exists, but still has no support to demonstrate that CHECWORKS is adequately benchmarked. In light of the substantial evidence presented herein questioning the accuracy of the model, it is patently obvious that a material factual dispute remains.

<sup>37</sup> See Entergy Motion for Summary Disposition at 20-21.

<sup>38</sup> See *Entergy Nuclear Vermont Yankee* (Vermont Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 871-72 (Nov. 24, 2008) (“To address the adequacy of Entergy’s FAC AMP, we [the VY ASLB] reviewed . . . Entergy’s updates to CHECWORKS with plant-specific data”).

<sup>39</sup> See Entergy Motion for Summary Disposition at 20.

circumstances present in the instant proceeding are clearly different: three sets of data at power uprate levels for IP3, and four sets of data at power uprate levels for IP2 have already been collected, and, as discussed above, clearly demonstrate that the CHECWORKS model is not sufficiently benchmarked to account for the new plant conditions.<sup>40</sup> This necessarily renders the conclusions of the VY ASLB regarding the benchmarking of CHECWORKS inapplicable in the instant proceeding.<sup>41</sup>

In coming to the conclusion that 10 to 15 years of benchmarking of CHECWORKS was not necessary at VY, the VY ASLB further reasoned that “data collected at VYNPS since 1989” had assisted in calibrating the model.<sup>42</sup> To the contrary, in the instant proceeding, Entergy maintains that data and CHECWORKS modeling at Indian Point prior to the power uprates of 2004 and 2005 are irrelevant, as evidenced by their position in response to Riverkeeper’s Motion to Compel disclosure of such information.<sup>43</sup> Entergy has refused to provide any CHECWORKS related information dating prior to 2000 for IP2 and 2001 for IP3.<sup>44</sup> Such information would be necessary in order to assess the adequacy of benchmarking/calibration of the CHECWORKS model and/or its predecessor codes since the owners of the plants started using it (ostensibly since the 1980s). In light of Entergy’s unwillingness to admit the relevancy of, or provide such information, Entergy certainly cannot rely upon 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.

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<sup>40</sup> See Hopenfeld Declaration (Attach. 2), ¶ 29.

<sup>41</sup> See *id.*

<sup>42</sup> *Entergy Nuclear Vermont Yankee* (Vermont Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 894 (Nov. 24, 2008).

<sup>43</sup> 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.”).

<sup>44</sup> *Id.*

Entergy further argues 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. Such reasoning is utterly misplaced. 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.<sup>45</sup> Thus, the magnitude of the power uprate at VY should have no bearing on the instant proceeding whatsoever.

In any event, 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.<sup>46</sup> Notably, Indian Point is a much larger facility than VY, and the impact of a power uprate on plant conditions is necessarily relative to the size of the particular plant.<sup>47</sup> Indian Point is also 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.<sup>48</sup> 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

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<sup>45</sup> See *Entergy Nuclear Vermont Yankee* (Vermont Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 894 (Nov. 24, 2008); see also Hopenfeld Declaration (Attach. 2), ¶ 29 (“[T]he VY ASLB did not have the benefit of any data to assess the ability of CHECWORKS to accurately detect wall thinning in light of changed plant operating conditions”).

<sup>46</sup> See Hopenfeld Declaration (Attach. 2), ¶ 28.

<sup>47</sup> See *id.* ¶ 27.

<sup>48</sup> See *id.* (citing *See e.g.*, Entergy Motion for Summary Disposition, Attach. 15 at 5.25).

or that the VY power uprate would automatically account for all changed conditions at Indian Point.<sup>49</sup>

Moreover, 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.<sup>50</sup> Indeed, Entergy produces Indian Point specific CHECWORKS modeling reports, which Entergy repeatedly touts use actual inspection data gathered at the plant and which account for plant specific conditions, such as new conditions due to replaced components. Entergy relies on these plant specific reports for its conclusion that CHECWORKS is an appropriate tool to be used as part of the FAC program at Indian Point. It is, therefore, counterintuitive and downright contradictory to assert that a generic assessment of CHECWORKS, without regard for how it is implemented at a specific plant, is appropriate.

Based on the foregoing, it would be incorrect for the ASLB in this proceeding to simply defer to the findings of a licensing board relating to a plant specific determination at VY, especially in light of the clearly inadequate benchmarking of the CHECWORKS code at Indian Point, as discussed herein.

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<sup>49</sup> 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 Contention Admissibility Order, *supra* note 3 at 168. Entergy cannot simply dismiss this inquiry, saying that it is not necessary to answer this question because the Indian Point power uprate is bounded by the uprate at VY. See Entergy Motion for Summary Disposition at 21.

<sup>50</sup> See Hopenfeld Declaration (Attach. 2), ¶ 28.

## **B. CHECWORKS Patently Lacks a “Track Record of Performance” at the Upgraded Power Levels at Indian Point**

A genuine dispute exists concerning whether Entergy has established that CHECWORKS has a “track record of performance at IPEC’s power upgrade levels,” as characterized by the ASLB.<sup>51</sup> Establishing such a track record is essential since CHECWORKS is entirely based on empirical modeling, meaning that it is solely based on a collection of selective data which represents only a fraction of the total flow area.<sup>52</sup> As such, CHECWORKS requires considerable benchmarking to be used as a reliable predictive tool.<sup>53</sup> Thus, 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.

Entergy’s claim that “CHECWORKS has a demonstrated record of successfully predicting wall thinning at IPEC and other nuclear power plants” is completely unfounded. As the discussion in the foregoing section clearly demonstrates, CHECWORKS results have been highly unreliable at Indian Point since plant conditions changed after the power upgrades.<sup>54</sup> This alone undeniably establishes a dispute of fact regarding the track record of CHECWORKS results at Indian Point.

Additionally, various instances of wall thinning and leaking components at nuclear power plants suggests that, generally speaking, the success of CHECWORKS at detecting FAC related wall thinning has been questionable.<sup>55</sup> At Indian Point in particular, numerous leaks and reports of excessive wall thinning in mechanical systems tend to indicate that CHECWORKS has not been successful at preventing FAC related occurrences. For example, Entergy’s 2007 Operating

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<sup>51</sup> ASLB Contention Admissibility Order, *supra* note 3, at 169.

<sup>52</sup> See Hopenfeld Declaration (Attach. 2), ¶ 9.

<sup>53</sup> See *id.* ¶ 8.

<sup>54</sup> See *id.* ¶¶ 12-18.

<sup>55</sup> See *id.* ¶ 11.

Experience Review Report documents many unacceptable wall thinning events and pipe leaks which occurred between 2001 and 2005.<sup>56</sup> Entergy condition reports document occurrences of leaks from components that resulted from undetected FAC, where subsequent inspections revealed wall thickness measurements that were below acceptable levels.<sup>57</sup> The NRC Staff in this license renewal proceeding has also questioned Entergy regarding incidences of unacceptable wall thinning.<sup>58</sup> Considering that typically, wall thinning rates in pressurized water reactors range from 5 to 50 mills per year, and the wall thickness of the components ranges between 300 to 1000 mills, one would expect that more and more components would become prone to failures after 40 years of service, i.e., during the proposed period of extended operation.<sup>59</sup>

Entergy further implies 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.”<sup>60</sup> However, this information by itself is purely circumstantial, and cannot lead one to conclude that CHECWORKS had been a success. It is, thus, far from clear that CHECWORKS has been successful at predicting FAC at Indian Point. The foregoing undoubtedly demonstrates that a material and genuine issue of a fact regarding whether CHECWORKS has an adequate “track record of performance.”

### **C. Entergy’s FAC Program Relies Largely on the CHECWORKS Computer Code**

A genuine dispute exists concerning Entergy’s assertion 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

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<sup>56</sup> See Hopenfeld Declaration (Attach. 2), ¶ 11; See Attach. 4.

<sup>57</sup> See Hopenfeld Declaration (Attach. 2), ¶ 11; See Attach. 5.

<sup>58</sup> See Hopenfeld Declaration (Attach. 2), ¶ 11; See Attach. 6.

<sup>59</sup> See Hopenfeld Declaration (Attach. 2), ¶ 13.

<sup>60</sup> See Entergy Motion for Summary Disposition at 23.

inspection.”<sup>61</sup> In particular, Entergy maintains that assuming Riverkeeper is correct that 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.<sup>62</sup> Riverkeeper wholly disagrees that Entergy’s identification of these “additional” tools for inspection scope selection demonstrates the effectiveness of Entergy’s FAC aging management program.<sup>63</sup>

Riverkeeper disputes Entergy’s assertion that these additional criteria can be viewed as independent tools sufficient to establish an accurate FAC inspection scope. A close examination reveals that these additional criteria largely depend upon the use of CHECWORKS. For example, 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.<sup>64</sup> Accordingly, this is a required input for the use of CHECWORKS and not a stand-alone “tool” for component selection.<sup>65</sup> 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.<sup>66</sup> Likewise, knowledge of pipe wall thinning events, changed plant parameters, etc., at Indian Point and other plants (i.e., industry and plant experience) are also types of information that feed into the CHECWORKS model.<sup>67</sup>

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<sup>61</sup> *Id.* at 17.

<sup>62</sup> *See id.* at 17; Entergy’s Motion for Summary Disposition, Attach. 2, ¶¶ 39.

<sup>63</sup> *See* Hopenfeld Declaration (Attach. 2), ¶ 19.

<sup>64</sup> *See id.* ¶ 20.

<sup>65</sup> *See id.*

<sup>66</sup> *See id.*

<sup>67</sup> *See id.* ¶ 21.



Thus, the usefulness of such information in determining future inspections rests in part on how the CHECWORKS model processes the inputs.

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.<sup>68</sup> However, Entergy has completely failed to demonstrate that engineering judgment alone will safely manage FAC at Indian Point. 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.<sup>69</sup> 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.<sup>70</sup>

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.<sup>71</sup> Even with the same input data, different assessments could lead to different results because each assessment would depend heavily on the individual skill and judgment of the responsible engineer.<sup>72</sup> Accordingly, 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.<sup>73</sup> To the contrary, Entergy has failed to clearly describe what exactly “engineering judgment” even

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<sup>68</sup> See *id.* ¶¶ 20-21; see Entergy Motion for Summary Disposition, Attach. 9 at 2-4 (EPRI guidance document explaining that engineering judgment requires awareness of operating experience, and input from plant operations, and also that “engineering judgment cannot substitute for other factors”).

<sup>69</sup> See Hopenfeld Declaration (Attach. 2), ¶ 22.

<sup>70</sup> See *id.* ¶¶ 9, 22.

<sup>71</sup> See *id.* ¶ 23.

<sup>72</sup> See *id.*

<sup>73</sup> See *id.*

means in relation to FAC inspections at Indian Point, and what role it actually plays in inspection scope selection.<sup>74</sup> Entergy has not identified any kind of systematic methodology which demonstrates that engineering judgment is a separate predictive tool that would adequately manage FAC related component degradation during the period of extended operation.<sup>75</sup>

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. This clearly disputes Entergy's assertions to the contrary, raising a material and genuine issue of a fact.

Entergy once again improperly relies upon findings of the VY ASLB in the VY license renewal proceeding to bolster its position here. In particular, Entergy points to the VY ASLB's observation that at VY, CHECWORKS played a limited role in the overall FAC program.<sup>76</sup> Entergy attempts to demonstrate that CHECWORKS is employed in the same manner as in VY, and that, likewise, it is only one of many tools used to determine locations for FAC inspections. However, as the above discussion demonstrates, it is disputed whether Entergy has adequately demonstrated any other means by which it *meaningfully* selects inspection points.<sup>77</sup>

Moreover, it would simply be inappropriate to rely upon the conclusions drawn during a completely separate proceeding, and essentially assume that Entergy implements its FAC program at Indian Point in an effective manner, simply because a different licensing board found it did so at a different facility. The implementation of the FAC program at Indian Point necessarily involves site specific considerations, and, as such, the question of the adequacy of the

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<sup>74</sup> See *id.*

<sup>75</sup> See *id.*

<sup>76</sup> Entergy Motion for Summary Disposition at 17-18.

<sup>77</sup> See Hopenfeld Declaration (Attach. 2), ¶ 30.

FAC program is not conducive to a generic determination. Notably, Entergy's attempt to summarily dispose of this issue in the VY proceeding was unsuccessful, and the VY ASLB only reached a determination after a full adjudicatory hearing. In the instant proceeding, at this stage, Riverkeeper need only establish a dispute of fact, which the foregoing, supported by the expert opinion of Dr. Hopenfeld, amply does.

**POINT II: GENUINE MATERIAL FACTS ARE IN DISPUTE CONCERNING THE SUFFICIENCY OF THE FAC PROGRAM AT INDIAN POINT**

**A. Entergy's FAC Program Fails to Adequately Address all Required Elements Identified in the GALL Report and SRP-LR**

A genuine dispute exists concerning whether, in light of the inadequacy of CHECWORKS as a tool for managing FAC at Indian Point during the period of extended operation, Entergy had sufficiently addressed all required elements identified in the SRP-LR. In particular, because Entergy's FAC program relies primarily on a method which does not accurately detect FAC, i.e., CHECWORKS, and Entergy has not otherwise demonstrated that it employs other methods sufficient to manage the aging effects of FAC at Indian Point, it is necessary for Entergy to provide detailed information regarding the method and frequency of component inspections and attendant criteria for component repair and replacement.

In contrast, Entergy merely states that its FAC program is consistent with the SRP-LR and GALL report guidance documents.<sup>78</sup> However, these generic guidance documents focus on the use of a properly calibrated CHECWORKS model. The GALL Report implies that when one uses computer codes to predict wall thinning, the codes must be properly benchmarked at each plant before they can be used as a management tool to control FAC.<sup>79</sup> Because Entergy has

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<sup>78</sup> See Entergy Motion for Summary Disposition at 15.

<sup>79</sup> See Hopenfeld Declaration (Attach. 2), ¶ 10; Entergy's Motion for Summary Disposition, Attach. 7 at XI M-61, XI M-62 ("CHECWORKS is acceptable because it provides a bounding analysis for FAC. CHECWORKS was developed and benchmarked by using data obtained from many plants").

failed to show that CHECWORKS is properly benchmarked to be an effective tool at Indian Point, as discussed above, Entergy has not been successful in implementing a critical aspect of these documents.<sup>80</sup> Moreover, as discussed above, Entergy has failed to properly define how it employs other tools to adequately address FAC in accordance with such guidance.<sup>81</sup>

Accordingly, Entergy cannot generically claim consistency with these 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.”<sup>82</sup>

It is, therefore, clear that Entergy’s FAC program at Indian Point does not adequately addresses the elements outlined in the SRP-LR and GALL Report.

**B. Entergy’s FAC Program Lacks Sufficient Detail to Demonstrate that Relevant Components will be Adequately Inspected and Maintained During the Period of Extended Operation**

Entergy further claims that the FAC program at Indian Point includes sufficient detail “to demonstrate that the intended functions of the applicable components will be maintained during the PEO,” because it implements a fleet-wide procedure, EN-DC-315 and EPRI guidance document (NSAC-202L-R3).<sup>83</sup> Once again, these procedures are focused heavily on the appropriate use of CHECWORKS, and further indicate CHECWORKS should be benchmarked or calibrated.<sup>84</sup> Due to the inadequacy of CHECWORKS as a tool for managing FAC at Indian Point,<sup>85</sup> it is disputable whether Entergy is actually implementing such guidance.<sup>86</sup> Thus, instead

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<sup>80</sup> See Hopenfeld Declaration (Attach. 2), ¶ 25.

<sup>81</sup> See *id.* ¶¶ 19-24.

<sup>82</sup> *Entergy Nuclear Vermont Yankee* (Vermont Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 870 (Nov. 24, 2008); 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 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.”).

<sup>83</sup> Entergy Motion for Summary Disposition at 16.

<sup>84</sup> See Hopenfeld Declaration (Attach. 2), ¶ 25; Entergy Motion for Summary Disposition, Attach. 9, Attach. 11

<sup>85</sup> See Hopenfeld Declaration (Attach. 2), ¶¶ 8-18.

of simply referring to procedural documents which depend upon the *proper* use of CHECWORKS, Entergy must provide sufficient details regarding inspection scope, frequency, etc., such that FAC will be adequately managed during the period of extended operation.

Entergy argues that this is a “settled” issue because the VY ASLB found that EN-DC-315 contained sufficient specificity to show that Entergy had implemented the GALL Report guidelines.<sup>87</sup> However, the VY ASLB only found that the relevant guidelines “have been implemented *at VYNPS*.”<sup>88</sup> This finding does not have general applicability. Indeed, as Entergy even acknowledges, the VY ASLB reached this determination only after thoroughly examining the FAC program *at VY*.<sup>89</sup> In contrast, at this stage of the Indian Point license renewal proceeding, Riverkeeper has highlighted numerous deficiencies with Entergy’s FAC program to question whether a similar conclusion can be drawn here.

Based on the foregoing, there remains a material issue of factual dispute regarding whether Entergy’s program for managing FAC at Indian Point during the period of extended operation contains sufficient specificity to demonstrate that relevant components will be adequately inspected and maintained during the period of extended operation.

### CONCLUSION

The foregoing demonstrates that significant disputes of fact exist regarding the sufficiency of Entergy’s program for managing the aging effects of FAC at Indian Point during the period of extended operation. In particular, Riverkeeper, supported by the expert opinion of

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<sup>86</sup> See *Entergy Nuclear Vermont Yankee* (Vermont Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 870 (Nov. 24, 2008) (“For an applicant to just illustrate how its 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.”).

<sup>87</sup> Entergy Motion for Summary Disposition at 16-17.

<sup>88</sup> See *Entergy Nuclear Vermont Yankee* (Vermont Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 871 (Nov. 24, 2008) (emphasis added).

<sup>89</sup> Entergy Motion for Summary Disposition at 16.

Dr. Joram Hopenfeld, has raised the following issues, which directly controvert Entergy's position that the aging management program to address FAC at Indian Point is adequate:

- Entergy's failure to demonstrate that CHECWORKS is adequately benchmarked so as to be an effective tool for predicting FAC at Indian Point during an extended period of operation;
- Entergy's failure to demonstrate that CHECWORKS has an adequate "track record of performance at Indian Point";
- Entergy's primary reliance upon the use of CHECWORKS, since Entergy has failed to identify any tools that are meaningfully independent of CHECWORKS that would sufficiently address FAC at Indian Point;
- Entergy's failure to demonstrate compliance with applicable regulatory guidance since, given the inadequacy of CHECWORKS, Entergy has failed to provide enough detailed information regarding the method and frequency of component inspections and attendant criteria for component repair and replacement, to assure adequate management of FAC.

In light of numerous material factual disputes, this case boils down to the classic "battle of the experts" for which summary disposition is utterly inappropriate. Accordingly, Entergy's Motion for Summary Disposition must be dismissed in its entirety.

Respectfully submitted,



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Dated: August 16, 2010  
Tarrytown, NY

Riverkeeper Opposition to Entergy's Motion For Summary Disposition of  
Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion)

# Riverkeeper TC-2: Attachment 1

August 16, 2010

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 COUNTER-STATEMENT OF MATERIAL FACTS**

Riverkeeper respectfully submits the following counter-statement of material facts in response to Entergy’s July 26, 2010 Statement of Material Facts. Riverkeeper responds as follows:

GENERAL OBJECTIONS

1. A large portion of what Entergy has submitted as statements of material facts consists of summaries of the contents of documents, statements of law, or legal argument. The referenced documents, law, and arguments, are the best evidence of their content and speak for themselves. Riverkeeper has below disputed only facts; it has largely reserved its counterarguments, including interpretation of documents, for its accompanying memorandum of law.

SPECIFIC RESPONSES AND COUNTERSTATEMENTS<sup>1</sup>

A. Background Concerning FAC, CHECWORKS, and Related Industry Guidance

1. *Flow accelerated corrosion (“FAC”) is a degradation process that attacks carbon steel piping and vessels exposed to moving water or wet steam. This attack occurs under specific water chemistry conditions. If FAC is not detected, then the piping or vessel walls will become progressively thinner until they can no longer withstand internal pressure and other applied loads. Joint Declaration of Jeffrey Horowitz, Ian Mew, and Alan Cox in Support of Entergy’s Motion for Summary Disposition of Riverkeeper Contention TC-2 (Flow-Accelerated Corrosion) ¶ 4 (Attach. 2); EPRI, Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L-R3) at 1-1 (Aug. 2007 (Attach.9)).* **Undisputed that this a general definition of FAC. Disputed to the extent that FAC as used by Entergy in CHECWORKS is limited only to a very specific wall thinning degradation mechanism i.e. due to dissolution of metal in water only. The degradation process also includes wall**

<sup>1</sup> Entergy’s alleged Undisputed Material Facts are reproduced below in Italics, followed by Riverkeeper’s responses in bold.



thinning by electrochemical corrosion, erosion-corrosion and cavitation- erosion. See Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene in the License Renewal Proceeding for the Indian Point Nuclear Power Plant (November 30, 2007), ADAMS Accession No. ML073410093 at 17 (hereinafter "RK Hearing Request"). Moreover, although the main causes of FAC (turbulence intensity, steam quality, material compositions, oxygen content and coolant pH) have been identified, the behavior of FAC is not completely understood. See id.

2. *In December 1986, an elbow in the condensate system at the Surry Unit 2 nuclear plant failed catastrophically, causing steam and hot water to be released into the turbine building. Post accident investigations revealed that FAC was the cause of the degradation to the elbow. At that time, the U.S. nuclear fleet did not have programs in place to deal with single-phase (i.e., water only) piping degradation caused by FAC. Attach. 2, ¶ 5. Undisputed.*

3. *In response to the pipe rupture at Surry in 1986, the Electric Power Research Institute ("EPRI") committed to developing a computer program that would assist utilities in determining the most likely places for FAC damage, and thus key locations to inspect for pipe wall thinning. Attach. 2 ¶ 6; Attach. 9, at 1-1 to 1-2. Undisputed.*

4. *EPRI released the computer program CHEC (Chexal-Horowitz Erosion Corrosion) to U.S. utilities in 1987. In 1989, EPRI replaced CHEC with CHECMATE (Chexal-Horowitz Methodology for Analyzing Two-Phase Environments). In 1993, EPRI replaced CHECMATE with CHECWORKS (Chexal-Horowitz Engineering Corrosion Workstation) in 1993. Each new version of the code built on the previous program and incorporated user feedback, improvements in software technology, and available laboratory and plant data into the algorithms used in the programs. Attach. 2, ¶ 6; Attach. 9, at 1-1 to 1-2. Undisputed.*

5. *In 1993, to help utilities improve and standardize their FAC programs, EPRI's Nuclear Safety Analysis Center ("NSAC") published NSAC-202L, Recommendations for an Effective Flow-Accelerated Corrosion Program. Attach. 2, ¶ 9. Undisputed.*

6. *EPRI issued Revision 3 of NSAC-202L in August 2007. NSAC-202L-R3 describes the elements of an effective FAC program, identifies the need for and suggested scope of program implementation procedures and documentation, recommends specific FAC program tasks, and explains how to develop a long-term strategy for reducing plant FAC susceptibility (e.g., through the use of FAC-resistant materials, improvements in water chemistry, and system design changes). Attach. 2, ¶¶ 31 & 34; Attach. 9. Undisputed that this is an accurate description of the content of NSAC-202L-R3, as characterized by that document.*

7. *Since the release of CHEC and its successor program more than 20 years ago, and the associated development of technology and programmatic guidance on FAC control, there has never been a fatality at any plant using CHEC or its successors. There has not been a major FAC-caused pipe rupture in a nuclear unit in the United States for more than 10 years. At nuclear plants in countries where CHECWORKS is not used, there is approximately one major rupture per year. Attach. 2, ¶ 66. Agree, however dispute implication that FAC has never been an issue at nuclear power plants that have employed CHECWORKS or its*

predecessor programs. For example, in 1997, an extraction steam piping ruptured at the Fort Calhoun Station. See RK Hearing Request at 18; see also Attach. 2, ¶ 11. Moreover, dispute implication that the use of CHECWORKS and its predecessor programs have been effective at adequately managing FAC, or that the fact that no fatalities at plants using these programs has occurred can be directly linked to such use. See id. ¶¶ 8-17. This statement seeks to establish cause and effect by an unsupported correlation.

8. *CHECWORKS is now used in more than 150 nuclear power plant units worldwide, including all U.S. nuclear units, all Canadian nuclear units, and nuclear units in Belgium, the Czech Republic, England, Japan, Korea, Mexico, Romania, Slovenia, Spain, and Taiwan. Attach. 2, ¶¶ 7 & 66. Undisputed.*

9. *Since 2001, the NRC has approved numerous EPU's exceeding 15 percent: Duane Arnold (15.3%), Dresden Unit 2 (17%), Dresden Unit 3 (17%), Quad Cities Unit 1 (17.8%), Quad Cities Unit 2 (17.8%), Clinton (20%), Vermont Yankee (20%), and Ginna (16.8%). There have been no reported failures in any major steam and feedwater system piping components at any of these plants, each of which has continued to use CHECWORKS since implementation of their respective EPU's. Attach. 2, ¶ 67; see also Approved Applications for Power Uprates (Oct. 28, 2009), <http://www.nrc.gov/reactors/operating/licensing/power-uprates/status-power-apps/approved-applications.html> (Attach. 14). **Agree, however, dispute implication that CHECWORKS has fully accounted for changed plant parameters at referenced facilities; such an implication is speculative in light of the information available in this proceeding, and, in any event, not relevant to the instant proceeding which relates specifically to Indian Point. Riverkeeper further disputes the implication that FAC has never been a problem at such facilities; the fact that these facilities have not reported system failures does not preclude the possibility that unacceptable wall thinning may have occurred. Again, the limited information available in this proceeding confines our understanding of whether FAC has occurred at the listed plants since power uprates occurred. Lastly, the level of a power uprate is relative to the size of the particular facility, and the mentioned power uprates do not necessarily have any relevance to power uprates which have occurred at Indian Point. See Attach. 2, ¶¶ 26-27.***

B. Applicable NRC Regulations and Guidance

10. *10 C.F.R. § 54.21(a)(3) requires a license renewal applicant to demonstrate that the effects of aging on structures and components subject to an aging management review ("AMR") will be adequately managed, so that there is "reasonable assurance" that their intended functions will be maintained consistent with the current licensing basis ("CLB") for the period of extended operation ("PEO"). **Agree that 10 C.F.R. § 54.21(a)(3) read in conjunction with 10 C.F.R. § 54.29 requires such a demonstration.***

11. *10 C.F.R. § 54.21(d) requires that the final safety analysis report ("FSAR") supplement for the facility contain a summary description of the programs and activities for managing the effects of aging. **Undisputed.***

12. In reviewing a license renewal application ("LRA"), the NRC Staff uses guidance in NUREG-1800, Rev. 1, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (Sept. 2005) ("NUREG-1800" or "SRP-LR") (Attach. 6), and NUREG-1801, Vol. 2, Rev. 1, Generic Aging Lessons Learned (GALL) Report – Tabulation of Results," (Sep. 2005) ("NUREG-1801" or "GALL Report") (Attach. 7). **Undisputed, and agree that these reports constitute guidance and not binding regulations. See Entergy Nuclear Vermont Yankee (Vermont Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 869 (Nov. 24, 2008).**

13. The GALL Report provides the technical basis for the SRP-LR and identifies generic aging management program ("AMPs") that the Staff has found acceptable based on the experiences and evaluations of existing programs at operating plants during the initial license period. Attach. 6, at 3.0-2 & App. A at A.1-3 to A.1-8. **Undisputed that this is NRC Staff's characterization of the GALL Report as stated in the SRP-LR.**

14. The GALL Report describes each AMP with respect to the ten program elements defined in the SRP-LR: (1) Scope of the Program, (2) Preventative Actions, (3) Parameters Monitored or Inspected, (4) Detection of Aging Effects, (5) Monitoring and Trending, (6) Acceptance Criteria, (7) Corrective Actions, (8) Confirmation Process, (9) Administrative Controls, and (10) Operating Experience. Attach. 2, ¶ 33; Attach. 7, at XI M-61 to XI M-62. **Disputed because the GALL report only generally describes what an AMP should contain. See Entergy Nuclear Vermont Yankee (Vermont Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 869 (Nov. 24, 2008) ("The simple fact is that NUREG-1801 does not contain an AMP, since it merely consists of two pages briefly describing the characteristics of a FAC AMP and specifies ten "evaluation and technical basis" criteria to be used in evaluating a FAC AMP. . . An enumeration of the criteria to be used in evaluating a program, is not itself a program.").**

15. The Commission has stated that a "license renewal applicant's use of an aging management program identified in the GALL Report constitutes reasonable assurance that it will manage the targeted aging effect during license renewal period." AmerGen Energy Co., LLC (Oyster Creek Nuclear Generating Station), CLI-08-23, 68 NRC 641, 468 (2008). **Dispute this characterization of the law. See Entergy Nuclear Vermont Yankee (Vermont Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 871 (Nov. 24, 2008) ("merely stating that its AMP meets NUREG-1801 without any specificity falls short of the required demonstration . . . a bald reference to NUREG-1801 fails to show how the recommendations of NUREG-1801 are proposed to be implemented . . . and does not demonstrate that the effects of aging are adequate managed").**

16. Section XI.M17 of the GALL Report describes the NRC-approved AMP for flow-accelerated corrosion. **Disputed for the reasons stated in ¶ 14. Moreover, the GALL Report is merely a guidance document generated and used by NRC Staff. See ¶ 12. It states that an acceptable FAC program relies on implementation of the EPRI guidelines in NSAC-202L-R2 for an effective FAC program. Attach. 2, ¶ 31; Attach. 7, at XI M-61. Undisputed that this is an accurate description of what the GALL Report states.**

17. *The purpose of a program implemented in accordance with GALL Report Section XI.M17 and EPRI guidelines is to predict, detect, and monitor FAC in plant piping and piping components, such as tees, elbows and reducers. Attach. 2, ¶ 32; Attach. 7, at XI M-61.*  
**Undisputed.**

18. *The program described in GALL Report Section XI.M17 includes performing (1) an analysis to determine critical locations, (2) limited baseline inspections to determine the extent of thinning at these locations, and (3) follow-up inspections to confirm the predictions, or repairing or replacing components as necessary. The program also may include the use of CHECWORKS or similar predictive code that uses the implementation guidance of NSAC-202L to predict component degradation in the systems susceptible to FAC. Attach. 2, ¶ 32; Attach. 7, at XI M-61. Agree that the description of the FAC AMP in the GALL Report is accurately referenced here, but dispute any implication that Entergy's FAC program, including use of CHECWORKS, is implemented in accordance with the referenced guidance or that Entergy's program will adequately manage the aging effects of FAC at Indian Point. See Attach. 2, ¶ 25.*

C. Overview of the Indian Point Energy Center ("IPEC") FAC Program

19. *Chapter 3 of the IPEC LRA summarizes Entergy's detailed assessment, conducted at a structure and component level, to identify those structures and components that require aging management review. Chapter 3 identifies FAC as an applicable aging mechanism for certain plant systems. Attach. 2, ¶ 29; LRA at 3.3-32 & 3.4-3 to 3.4-6, available at ADAMS Accession No. ML071210517. Undisputed.*

20. *The appendices to the LRA contain a description of Entergy's FAC Program. Appendix A presents information required by 10 C.F.R. § 54.21(d) relating to the AMP for FAC that supplements the updated FSAR ("UFSAR") for IPEC. The supplement to the UFSAR, presented in section A.2 of Appendix A, contains a summary description of the program and activities for managing the effects of FAC during the PEO. Appendix A states that this information will be incorporated into the UFSAR following issuance of the renewed operating licenses. Attach. 2, ¶ 29; LRA, App. A at A-1 & A-24, available at ADAMS Accession No. ML071210520. Undisputed that this is an accurate description of the content of the referenced appendix of the LRA.*

21. *Appendix B to the LRA describes those AMPs credited in the integrated plant assessment for managing aging effects. Section B.1.15 describes the IPEC FAC Program and indicates that it is consistent with, and takes no exceptions to, the program described in GALL Section XI.M17. Attach. 2, ¶ 30; LRA, App. B at B-1 & B-54, available at ADAMS Accession No. ML071210523. Undisputed that this is an accurate description of the content of the referenced appendix of the LRA. Dispute that the IPEC FAC program is actually implemented in accordance with NRC and industry guidance. See Attach. 2, ¶ 25.*

22. *LRA Section B.1.15 states that the IPEC FAC Program is based on EPRI guidelines for an effective FAC program contained in NSAC-202L-R2. Attach. 2, ¶ 30; LRA,*

*App. B at B-54. Undisputed that these EPRI guidelines apply. Dispute that the IPEC FAC program is actually implemented in accordance with EPRI guidance. See Attach. 2, ¶ 25.*

23. *Entergy compared the IPEC FAC Program to GALL Report Section XI.M17 with respect to each of the ten program elements. The results of this comparison are documented in the LRA and Entergy's June 2008 AMP Evaluation Report for non-Class 1 mechanical components and show that the IPEC FAC Program elements are consistent with all ten program elements identified in the SRP-LR and the GALL Report. Attach. 2, ¶¶ 33 & 56; LRA, App. B at B-54 to B-55; Entergy Eng'g Report No. IP-RPT-06-LRD07, Rev. 5, Aging Management Program Evaluation Results – Non-Class 1 Mechanical, (Mar. 18, 2009) (Attach. 8) ("AMP Evaluation Report"). **Disputed. Riverkeeper disagrees that the IPEC FAC program elements are consistent with those elements identified in SRP-LR and the GALL Report. See Attach. 2, ¶ 25.***

24. *On December 18, 2007, in response to NRC Audit Item 156, Entergy amended the "scope of program" and "detection of aging effects" program elements to identify its use of Revision 3 of NSAC-202L (NSAC-202L-R3) as an "exception" to GALL Report Section XI.M17, which references the prior Revision 2 of NSAC-202L. Attach. 2, ¶ 34; 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) (Attach. 10). **Undisputed.***

25. *NSAC-202L-R3 incorporates lessons learned and improvements to detection, modeling, and mitigation technologies that arose after the publication of Revision 2. Attach. 2, ¶¶ 34 and 57. It states that the updated recommendations "are intended to refine and enhance those of the earlier versions, without contradiction, so as to ensure the continuity of existing plant FAC programs." Attach. 9, at v. **Undisputed.** Entergy did not take an exception to the GALL Report in its April 2007 LRA because implementing NSAC-202L-R3 does not create program deviations from NSAC-202L-R2. Attach. 2, ¶¶ 34 & 57. **Dispute to the extent this statement implies that that Entergy implements the IPEC FAC Program in accordance with the EPRI guidance. See Attach. 2, ¶ 25.***

26. *The NRC Staff's Safety Evaluation Report concludes that the IPEC FAC program elements, including Entergy's use of NSAC-202L-R3, are acceptable and consistent with all ten program elements in GALL Section XI.M17. Attach. 2, ¶ 35; NUREG-1930, Vol. 2, Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286, Entergy Nuclear Operations, Inc. at 3-22 to 3-30 (Nov. 2009), available at ADAMS Accession No. ML093170671 ("SER"). **Undisputed that this is an accurate description of the content of the referenced NRC Staff SER. Dispute to the extent Entergy is characterizing the substance of NRC Staff's finding as undisputed facts; Riverkeeper disagrees that the IPEC FAC program elements are consistent with those elements identified in SRP-LR and the GALL Report. See Attach. 2, ¶ 25.***

D. IPEC Program for Managing FAC During the Period of Extended Operation

27. *Entergy has maintained a formal FAC inspection program at IPEC based on EPRI and industry guidelines since 1990. The IPEC FAC Program is an existing IPEC program*

that will continue during the PEO. Although the IPEC FAC Program predates EPRI guidelines in NSAC-202L, the program documents have been revised to conform to the recommendations contained in NSAC-202L guidelines. Attach. 2, ¶ 36. **Disputed that Entergy's IPEC FAC Program has been consistent with the referenced EPRI and industry guidelines. See Attach. 2, ¶ 25.**

28. The IPEC FAC Program draws from industry and IPEC operating experience, including NRC information notices, bulletins, and generic letters; inspection data from recent refueling outage inspections and power uprate-related changes in operating parameters; and audits/self-assessments of the IPEC FAC Program. Attach. 2, ¶ 36; LRA, App. B at B-54 to B-55; SER Vol. 2, at 3-29 to 3-30. **Dispute that Entergy's IPEC FAC Program "draws from" such mechanisms in a manner which effectively addresses FAC at Indian Point. See Attach. 2, ¶¶ 19-24.**

29. Entergy has implemented the IPEC FAC Program in accordance with its fleet-wide procedure EN-DC-315, "Flow Accelerated Corrosion Program, Rev. 3 (Mar. 1, 2010) (Attach. 11), which governs the FAC programs at all of Entergy's nuclear power plants. EN-DC-315 implements the recommendations of the GALL Report and the more detailed EPRI NSAC-202L-R3 guidelines. In developing EN-DC-315, Entergy reviewed best practices for the FAC Program at all Entergy sites and included guidance from the EPRI CHECWORKS Users Group ("CHUG"). Attach. 2, ¶ 37. **Disputed. Whether Entergy has implemented the IPEC FAC Program in accordance with its fleet-wide procedure is a subjective assessment and statement of opinion at best, not a fact. Further disagree that the IPEC FAC program is actually implemented in accordance with the GALL Report and EPRI NSAC-202L-R3 guidelines. See Attach. 2, ¶ 25.**

30. The IPEC FAC Program applies to carbon and low-alloy steel piping systems and includes feedwater heater and moisture separator re-heater ("MSR") shells susceptible to FAC. It includes inspections of single-phase and two-phase piping components for both safety-related and nonsafety related systems. Attach. 2, ¶ 38; LRA, App. B at B-54. **Undisputed that this accurately reflects the language describing the IPEC FAC Program in the LRA.**

31. Ultrasonic testing ("UT") thickness measurements performed in accordance with approved procedures are the primary method used to determine pipe wall thickness. Attach. 2, ¶ 38; Attach. 11, at 19-23. **Undisputed.**

32. EN-DC-315 states that FAC inspections are to be conducted during scheduled refueling and maintenance outages. Attach. 2, ¶¶ 38 & 59; Attach. 11 at 3 & 10. **Undisputed that this accurately reflects the language in Entergy's procedural document, EN-DC-315.**

33. The IPEC FAC Program includes specific criteria or guidance for selecting components for inspections, performing the inspections, evaluating inspection data, dispositioning component inspection results, conducting re-inspections, addressing components that fail to meet initial screening criteria, expanding the sample to other components similar to those failing to meet acceptance criteria, repairing or replacing degraded components. Attach. 2, ¶ 59; Attach. 9, at 4-1 to 4-28; Attach. 11, at 15-26 & 34. **Disputed to the extent this**

statement implies that the IPEC FAC Program “criteria” and “guidance” for the above activities is adequate to effectively manage FAC at Indian Point. See Attach. 2, ¶¶ 19-24.

34. *The IPEC criteria for component selection for FAC inspection during outages are consistent with those cited in NSAC-202L-R3, with the selection being based principally on: (1) 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. The planning process for future inspections at IPEC also considers the consequences of failure of a particular component with respect to personnel safety and plant availability, and the margin of nominal wall thickness versus code minimum wall thickness. EN-DC-315 provides additional guidance on component selection. Attach. 2, ¶¶ 39-40; Attach. 11, at 16-17. Disputed. Generally, this alleged undisputed fact merely states a subjective judgment regarding the consistency of Entergy’s FAC program with EPRI guidance. Disagree that Entergy’s IPEC FAC Program is consistent with EPRI guidance. See Attach. 2, ¶ 19-25. Criteria for component selection during outages and the scope of inspection in the IPEC FAC program are inadequate because they are based on questionable CHECWORKS predictions and rely on “engineering judgment” that Entergy has failed to describe with sufficient specificity. Id. ¶¶ 8-24.*

35. *The IPEC FAC Program also includes specific criteria for the disposition of inspection results, including the criteria for component repair and replacement. Using the inspection results, the wear rate and predicted thickness at a future inspection date (usually the next refueling outage) is calculated and compared to the component nominal thickness ( $t_{nom}$ ) (i.e., wall thickness equal to the ANSI standard thickness). Specific actions are taken based on the results of this comparison. The component may be found acceptable for continued service, subjected to a structural evaluation in accordance with pipe code stress requirements, or immediately repaired and replaced (in accordance with Section 5.13 of EN-DC-315). Attach. 2, ¶ 41; Attach. 9, at 4-17 to 4-27; Attach. 11, at 23-26 & 35. Disputed. “Specific” is ambiguous and subject to varying interpretation by different experts. No conclusion can be reasonably drawn or legitimately inferred from this statement; dispute to the extent this statement implies that the IPEC FAC Program criteria cited is adequate to effectively manage FAC at Indian Point. See Attach. 2, ¶¶ 18-24.*

36. *If a component is found that has a current or projected wall thickness less than the minimum acceptable wall thickness, then Entergy will perform additional inspections of identical or similar piping components in a parallel or alternate train, as necessary, to bound the extent of thinning. Section 5.12 of EN-DC-315 describes the sample expansion protocol. Attach. 2, ¶ 42; Attach. 11, at 25-26. Undisputed that this accurately reflects the language in Entergy’s procedural document, EN-DC-315. But disputed to the extent this statement implies that the IPEC FAC Program section referenced is adequate to effectively manage FAC at Indian Point. See Attach. 2, ¶¶ 18-24.*

37. *Entergy has replaced certain IPEC piping components susceptible to FAC previously with FAC-resistant materials (e.g., stainless steel, chromium-molybdenum steel). Sufficient concentrations of certain alloying elements, particularly chromium, make steels immune to FAC. Undisputed, however, Entergy has not specified the extent to which FAC-*

resistant materials have replaced FAC-susceptible materials. Therefore, no conclusion can be reasonably drawn or legitimately inferred from this statement.

38. *Entergy also maintains water chemistry to inhibit corrosion of FAC-susceptible piping and piping components. In accordance with the Secondary Water Chemistry Program, IPEC utilizes an all volatile treatment ("AVT") that includes the addition of monoethanolamine ("ETA") and hydrazine to the condensate to control pH control [sic] and oxygen levels. Under the Secondary Water Chemistry Program, corrosion products of iron and copper typically are reduced to less than 1 part per billion ("ppb") and 0.01 ppb, respectively. These concentrations are below the industry recommended limits specified in EPRI's PWR secondary water chemistry guidelines for feedwater iron (5 ppb) and feedwater copper (1 ppb) during full power operation. Attach. 2, ¶ 44. Undisputed, however, Entergy has not provided analyses of the performance of the FAC inhibiting water chemistry program. Therefore, no conclusion can be reasonably drawn or legitimately inferred from this statement.*

E. Use and Updating of CHECWORKS Models at IPEC

39. *The decision to repair or replace piping or components at IPEC is based on actual inspections of plant piping and piping components for wall thinning. Attach. 2, ¶ 49; Attach. 11, at 21-26; SER Vol. 2, at 3-27 to 3-29. Undisputed, however object to the extent this statement attempts to minimize the role CHECWORKS plays in this process, since actual inspection point locations are chosen in the first instance in large part because of CHECWORKS. See Attach. 2, ¶ 20.*

40. *CHECWORKS is a multi-purpose computer program designed to assist FAC engineers in identifying potential locations of FAC vulnerability. It is designed for use by plant engineers as a tool for identifying piping locations susceptible to FAC, predicting FAC wear rates, planning inspections, evaluating inspection data, and managing inspection data. Attach. 2, ¶ 45; Attach. 9, at 1-1. Undisputed to the extent this is a general description of CHECWORKS and its intended use, but disputed that CHECWORKS is an effective tool for identifying appropriate inspection locations. See Attach. 2, ¶¶ 8-18.*

41. *At IPEC CHECWORKS is used in conjunction with trend data from actual inspections, relevant information from other plant programs, industry or plant operating experience, and engineering judgment. Attach. 2, ¶ 49; Attach. 11, at 16-17; SER Vol. 2, at 3-29. Disputed to the extent this statement implies that such measures, as implemented by Entergy, sufficiently manage the effects of FAC at Indian Point. See Attach. 2, ¶¶ 19-24.*

42. *The CHECWORKS user constructs a mathematical model of the FAC-susceptible piping systems, similar in concept to a piping stress model or flow model. The input to the CHECWORKS modeling program includes plant operating parameters such as flow rates, pipe material, operating temperatures and piping configuration, as well as measured wall thicknesses from FAC Program components. Based on this input, CHECWORKS predicts the rate of wall thinning and remaining service life on a component-by-component basis. Attach. 2, ¶ 46. Undisputed to the extent this is a general description of CHECWORKS and its intended*



use, but disputed that CHECWORKS is an effective tool for predicting “the rate of wall thinning and remaining service life” of plant components. See Attach. 2, ¶¶ 8-18.

43. CHECWORKS uses two types of evaluations in determining the susceptible locations for FAC and predicting wear rates. The first evaluation, called a “PASS-1 Analysis,” is performed to report predicted wear rates based on plant operating characteristics that do not incorporate actual pipe thicknesses from plant inspections. This evaluation is normally used to generate a list of components for inspections when plant data are not available. Attach. 2, ¶ 47; Attach. 9, at 4-1 to 4-2; Attach. 11, at 8 & 11. **Undisputed to the extent this is a general description of how CHECWORKS is run and its intended use, but disputed that CHECWORKS is an effective tool for “determining the susceptible locations for FAC and predicting wear rates.”** See Attach. 2, ¶¶ 8-18.

44. The second evaluation, called a “PASS-2 Analysis,” incorporates measurements from actual inspections of plant piping and components. The model then compares the results to the initial predicted values and adjusts the FAC calculations to account for actual wall thickness through the use of a “line correction factor” (“LCF”). If the model-predicted wear rate is less than the actual wear rate, then the predicted wear rates are increased (multiplied by the LCF) to match the inspection data. Attach. 2, ¶ 48; Attach. 9, at 4-1 to 4-2; Attach. 11 at 8 & 11; Attach. 12, at 15. **Undisputed to the extent this is a general description of how CHECWORKS is used, but disputed that CHECWORKS is an effective predictive tool, or that Entergy’s use of LCF’s render CHECWORKS predictions accurate.** See Attach. 2, ¶¶ 8-18.

45. The piping system locations at IPEC with areas of high flow velocity and high turbulence are expected to be most susceptible to FAC. These locations have been confirmed through two decades of inspections performed under the FAC program. Attach. 2, ¶ 50; SER Vol. 2, at 3-26 to 3-27; see also NL-08-004, Letter from Fred R. Dacimo, Entergy, to NRC Document Control Desk, “Reply to Request for Additional Information Regarding License Renewal Application (Steam Generator Tube Integrity and Chemistry),” Attach. 1, at 3 (Jan. 4, 2008) (Attach. 13). **Disputed. Whether the IPEC FAC Program has been appropriately implemented so as to accurately identify locations most susceptible to FAC is subject to varying interpretation by different experts.**

46. The CHECWORKS model is updated after every outage with the latest chemistry, operating, and inspection data. Through this process, changes due to replacement or repair of piping and piping components, adjustments in water chemistry, and post-power uprate operations are incorporated into the IPEC CHECWORKS models. Attach. 2 ¶ 51; Attach. 11, at 15-16; Attach. 12, at 15-17; SER Vol. 2, at 3-27 to 3-28. **Disputed that the CHECWORKS model has been sufficiently calibrated or benchmarked to account for the changed operating parameters at Indian Point following the power uprates.** See Attach. 2, ¶¶ 8-18.

47. The NRC approved stretch power uprates (“SPUs”) of 3.26% and 4.85% for IP2 and IP3 in October 2004 and March 2005, respectively. Attach. 2, ¶ 52; Attach. 14, at \*3. **Undisputed.**

48. *Entergy updated the IPEC CHECWORKS models to account for changes to plant operating parameters resulting from the SPUs. Specifically, before the SPUs were performed, Entergy entered the new operating parameters (e.g., flow rates, temperatures, pressures, and steam quality) into the IP2 and IP3 CHECWORKS databases and ran the CHECWORKS models to calculate new wear rates. These evaluations were complete in March 2005. The results of the updated CHECWORKS results were used in the inspection planning for the subsequent outages. Attach. 2, ¶¶ 52 & 62; Attach. 12, at 15; SER Vol. 2, at 3-26. Disputed that the CHECWORKS model has been sufficiently calibrated or benchmarked to account for the changed operating parameters at Indian Point following the power uprates. See Attach. 2, ¶¶ 8-18.*

49. *Consistent with EN-DC-315, Rev. 3 and NSAC-202L-R3, Entergy uses UT inspection results obtained during plant outages to assess the accuracy of the CHECWORKS wear predictions and to perform re-baselined CHECWORKS analyses. Attach. 2, ¶ 53; Attach. 11, at 15-16 & 27; SER Vol. 2, at 3-28 to 3-29. Disputed that the IPEC FAC program is "consistent with" with referenced guidance. See Attach. 2, ¶ 25. Further disputed that Entergy's use of inspection results adequately calibrates the CHECWORKS model to produce accurate predictions. See id. ¶¶ 8-18.*

50. *Under the current IPEC outage schedule, Entergy expects that at least four IP2 refueling outages and five IP3 refueling outages will have occurred between implementation of the SPU and expiration of the respective plant operating licenses. Attach. 2, ¶¶ 53 & 63. Undisputed, however, dispute implication that collection of data during these outages will adequately benchmark/calibrate CHECWORKS for effective use under the power uprate conditions at Indian Point during the period of extended operation. See Attach. 2, ¶¶ 8-18.*

51. *Entergy has updated the IP2 CHECWORKS model to incorporate inspection data from the 2R16 (2005), 2R17 (2006), 2R18 (2008) and 2R19 (2010) outages. Attach. 2, ¶ 54. Dispute that such updates have adequately benchmarked/calibrated CHECWORKS for effective use under the power uprate conditions at Indian Point during the period of extended operation. See Attach. 2, ¶¶ 8-18.*

52. *Entergy has updated the IP3 CHECWORKS model to incorporate inspection data from the 3R13 (2005), 3R14 (2007), 3R15 (2009) outages. Attach. 2, ¶ 54. Dispute that such updates have adequately benchmarked/calibrated CHECWORKS for effective use under the power uprate conditions at Indian Point during the period of extended operation. See Attach. 2, ¶¶ 8-18.*

53. *Comparison of measured wear and CHECWORKS model-predicted wear indicates a level of correlation following SPU implementation that is consistent with industry and plant expectations relative to the performance of CHECWORKS. Attach. 2, ¶¶ 55, 63 & 76. Disputed. Comparison of measured wear and CHECWORKS predicted wear indicates a level of correlation that is unacceptable, and which demonstrates CHECWORKS is not sufficiently calibrated/benchmarked to post power uprate conditions at Indian Point. See Attach. 2, ¶ 13.*

54. *The NRC Staff's SER concludes that the IPEC FAC Program is adequate to manage FAC during the PEO because: (1) the CHECWORKS code is considered to be a self-benchmarking code that is capable of modeling, predicting, and tracking the results of the ultrasonic inspections that are performed in accordance with the applicant's FAC Program; (2) the self-benchmarking feature of CHECWORKS makes prolonged benchmarking of CHECWORKS unnecessary; (3) the applicant uses the actual UT inspection results to confirm the predictive modeling of the CHECWORKS analyses and to perform re-baselined CHECWORKS analyses; (4) the applicant does not use the CHECWORKS computer code as the sole basis for establishing which steel piping, piping components, or piping elements at IP2 and IP3 will be inspected; and (5) the program includes acceptable program elements for managing flow-accelerated corrosion that are consistent with the program element criteria in GALL AMP XI.M17 or with the acceptable alternative to use EPRI Report NSAC-202L-R3 as the implementation guideline for this program. SER Vol. 2, at 3-29. Undisputed that this is an accurate description of the content of the referenced NRC Staff SER. Disputed to the extent Entergy is characterizing the substance of NRC Staff's finding as undisputed facts. Riverkeeper does not agree with the NRC's Staff's findings or conclusions in the SER in regards to Entergy's program for managing FAC at Indian Point, consistent with the specific disputes already identified herein.*

55. *The NRC Staff's SER also concludes that Entergy's LRA, including the FAC Program, satisfies the applicable requirements of 10 C.F.R. Part 54, including those contained in 10 C.F.R. § 54.21(a)(3), 10 C.F.R. § 54.21(d). SER Vol. 2, at 3-31. Undisputed that this is an accurate description of the content of the referenced NRC Staff SER. Disputed to the extent Entergy is characterizing the substance of NRC Staff's finding as undisputed facts. Riverkeeper does not agree with the NRC's Staff's findings or conclusions in the SER that Entergy's LRA satisfies the requirements of Part 54 in regard to Entergy's program for managing FAC at Indian Point, as discussed in Riverkeeper's accompanying Memorandum of Law.*

Respectfully submitted,



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Dated: August 16, 2010  
Tarrytown, NY

Riverkeeper Opposition to Entergy's Motion For Summary Disposition of  
Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion)

## Riverkeeper TC-2: Attachment 2

August 16, 2010

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
	)	

**DECLARATION OF DR. JORAM HOPENFELD**

Joram Hopenfeld, hereby declares under penalty of perjury that the following is true and correct:

1. I have been retained by Riverkeeper, Inc. as an expert witness in proceedings concerning the application by Entergy Nuclear Operations, Inc. ("Entergy") for a renewal of the 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.

2. I submit this declaration in opposition to Entergy's July 26, 2010 Motion for Summary Disposition that seeks the dismissal of Riverkeeper Technical Contention 2 concerning Flow-Accelerated Corrosion (hereinafter "Entergy's Motion for Summary Disposition").

3. My professional and educational qualifications are described in the *curriculum vitae* appended as Attachment 3. Briefly summarized, 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 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 U.S. Nuclear Regulatory Commission.

4. My extensive professional experience has afforded me with knowledge and expertise regarding the material degradation phenomenon known as “flow-accelerated corrosion” (hereinafter referred to as “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. Most recently, I was a technical consultant and expert witness for the New England Coalition in the Vermont Yankee license renewal proceeding, where I testified at an adjudicatory hearing concerning FAC and CHECWORKS.

5. I reviewed the April 30, 2007 License Renewal Application submitted by Entergy to renew the operating licenses for Indian Point Units 2 and 3, and assisted Riverkeeper with the preparation of Contention TC-2, which articulates Entergy’s failure to provide for adequate aging management of FAC.

6. I have reviewed the pertinent sections of the NRC Staff’s August 12, 2009 Safety Evaluation Report, numerous documents provided by Entergy pursuant to mandatory disclosure obligations of 10 C.F.R. § 2.336, and Entergy’s Motion for Summary Disposition together with its attendant declarations and attachments. After a review of these documents, for the reasons explained more fully below, it remains my professional opinion that Entergy’s proposed aging management program for FAC fails to provide reasonable assurance that Indian Point Units 2 and 3 will operate safely through their proposed license renewal periods.

7. A discussion of various assertions in Entergy's filing, sufficient to establish that Entergy's arguments are by no means dispositive and that technically credible and substantial disputes of fact remain, follows below:

### **Entergy's Misplaced Reliance on CHECWORKS**

8. I disagree with Entergy's assertion that CHECWORKS is "a viable and effective tool for selecting and prioritizing IPEC piping and piping component locations for inspection to detect and mitigate FAC during the period of extended operation." See Entergy Motion for Summary Disposition, Attach. 2 at 28-29. In particular, I continue to maintain that CHECWORKS is not a mechanistic model, and therefore it requires considerable benchmarking to be used as a reliable predictive tool.

9. Following the 1987 catastrophic pipe rupture accident at the Surry nuclear power plant, the nuclear industry funded the development of a computer program, today known as CHECWORKS, to predict wall thinning rates of critical reactor components that are exposed to high velocity single phase water. Wall thinning by wet steam, cavitation, or by abrasion are not included in the model. CHECWORKS is entirely based on empirical modeling, meaning that it is solely based on a collection of selective data which represents only a fraction of the total flow area. Accordingly, CHECWORKS must be calibrated or benchmarked separately at each individual power plant and recalibrated when plant conditions change.

10. NRC's guidance report, NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, further implies that when one uses computer codes to predict wall thinning, the codes must be properly benchmarked at each plant before they can be used as a management tool to control FAC. See Entergy Motion for Summary Disposition, Attach. 7 (GALL Report at XI.M17).

11. It is difficult to quantify the overall success of CHECWORKS since no formal comparison of data from nuclear power plants that use CHECWORKS and those that do not, is available. Generally speaking, 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 at best. *See generally* Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene in the License Renewal Proceeding for the Indian Point Nuclear Power Plant (November 30, 2007), ADAMS Accession No. ML073410093 at 21-23. At Indian Point in particular, numerous leaks and reports of excessive wall thinning in mechanical systems tend to indicate that CHECWORKS has not been successful at preventing FAC related occurrences. For example, Entergy's 2007 Operating Experience Review Report (relevant excerpts appended as Attachment 4) documents various unacceptable wall thinning events which occurred between 2001 and 2005. *See* Attach. 4. Also by way of example, Entergy condition reports appended as Attachment 5, document occurrences of leaks from components that resulted from undetected FAC, where subsequent inspections revealed wall thinning was below minimum acceptable levels. *See* Attach. 5. The NRC Staff in this license renewal proceeding has also questioned Entergy regarding 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, the relevant excerpt of which is appended as Attachment 6. *See* Attach. 6.

12. My review of numerous reports generated on behalf of Entergy in relation to CHECWORKS modeling at Indian Point has revealed that CHECWORKS predictions of wall thinning are highly unreliable. CHECWORKS can only predict the overall range of corrosion



rates for a given a component or a group of components. This range is too wide for practical applications, especially when the consequences of component failure are safety related.

13. For the purposes of demonstrating the highly unreliable predicative capability of CHECWORKS, I have collected graphs excerpted from seven individual CHECWORKS modeling reports, which plot CHECWORKS predictions of wall thickness versus actual measurements for selected plant components. Riverkeeper has labeled these graphs Figures 1 through 18 and, they are appended together as Attachment 7. These graphs were generated based on CHECWORKS/FAC data from Indian Point Unit 2 refueling outages 16 (2005), 17 (2006), 18 (2008), and 19 (2010), and from Indian Point Unit 3 refueling outages 13 (2005), 14 (2007), and 15 (2009), as indicated by cover sheets in Attachment 7. All of these outages occurred after the operating conditions at the Units 2 and 3 changed due to power uprates in 2004 and 2005, respectively.

If CHECWORKS predictions were completely accurate, all data points in Figures 1 through 18 would fall on a 45 degree line, i.e., the center line in the graphs. To the contrary, the wide scatters of the plotted points on the example graphs demonstrate that this is not the case. *See Att. 7.* Indeed, one can draw almost any line through the data in these graphs, indicating a complete lack of correlation. *See id.* A straight line parallel to the abscissa would indicate that actual plant observations and computer model predictions are independent of each other. *See id.* The two lines drawn on these graphs above and below the center line were drawn completely arbitrarily to show that most, but not all of the data, can be bound with +/- a factor of two from the straight 45 degree line. *See id.*

Many of these graphs show that points outside these two lines can deviate by as much as a factor of +/- 10. *See Attach. 7.* For example, the data point furthest to the left in Figure 4

represents an actual measured wall thickness of a given component of about 20 mills (abscissa) while the corresponding CHECWORKS prediction was over 200 mills. On the other hand, Figure 7 shows that the predicted wall thickness of the data point furthest to the right in the figure was about 10 mills while the corresponding measured value was over 100 mills. Thus, CHECWORKS can either under-predict or over-predict FAC by a factor of 10 or 1000%. It is, thus, apparent, and my expert opinion, that CHECWORKS cannot predict FAC to any degree of accuracy. Considering that typically, wall thinning rates in pressurized water reactors range from 5 to 50 mills per year, and the wall thickness of the components ranges between 300 to 1000 mills, one would expect that more and more components would become prone to failures after 40 years of service. In my professional opinion, the margin of error in CHECWORKS predictions is too large and, therefore, CHECWORKS is not a reliable tool for identifying locations for inspections.

14. In discussing the alleged success of CHECWORKS in predicting wall thinning, Entergy maintains that “while CHECKWORKS sometime underestimates wear rates, it also yields precise and accurate results.” *See Answer of Entergy Nuclear Operations, Inc. Opposing Riverkeeper Inc.’s Request for Hearing and Petition to Intervene (Jan. 22, 2008), at 54, ADAMS Accession No. ML080300149.* I did not find any data that would support this conclusion. In my opinion, a margin of error high as +/- 1000% exhibited by a significant number of components is not a demonstration of precise and accurate results.

15. Entergy uses a “line correction factor” (“LCF”) to compare and adjust CHECWORKS predictions to match inspection data. *See Entergy Motion for Summary Disposition, Attach. 2 at ¶ 48.* According to Entergy documentation related to CHECWORKS, the relevant excerpt of which is appended as Attachment 8, “[t]he LCF indicates the degree to

which CHECWORKS over or under-predicts wear. A reasonable LCF should be between 0.5 and 2.5.” *See* Attach. 8. My review of Entergy’s reports related to CHECWORKS for the above referenced outages has revealed numerous instances where the LCF was outside of this range. *See, e.g.*, Attach. 7, Figures 12-15, 17-18. Thus, Entergy’s own data indicates that CHECWORKS is unreasonably failing to predict wear rates.

16. Moreover, I am unaware of a justification to support the conclusion that the LCF range of 0.5 to 2.5 is acceptable. I attended a tutorial on CHECWORKS where Entergy witness Dr. Jeffrey Horowitz explained that the LCF is obtained by comparing the predicted amount of wall thinning with the measured results for each component and then using proprietary statistical methods to determine the LCF, which is applied to all components in a given pipe line. Based on this explanation and my review of relevant documents provided by Entergy, it is evident that Entergy has failed to demonstrate how the stated LCF range would be an indication that CHECWORKS can be used to predict inspection locations. In my opinion, one acceptance criteria for the LCF must be the consequences of component failure.

17. The foregoing directly controverts Entergy’s apparent position that the level of correlation between the CHECWORKS model predicted wear and the measured wear following implementation of the stretch power uprates at Indian Point is acceptable. *See* Entergy Motion for Summary Disposition, Att. 2 at ¶¶ 55, 63, 75. The foregoing further contradicts Entergy’s conclusion that CHECWORKS is adequately benchmarked under post power uprate operating conditions, and that it “is a suitable tool for informing predictions of where potential pipe failures due to FAC might occur.” *See id.* at ¶¶ 74, 75.

18. Entergy claims that the CHECWORKS model is updated after every plant refueling outage, and takes into account inspection data, as well as changed parameters, thus allowing for

more accurate predictions over time. *See* Entergy Motion for Summary Disposition, Attach. 2 at ¶¶ 48, 51-54, 61-65, 75. My review of Indian Point related CHECWORKS modeling reports encompassing the previous five years of refueling outages at Units 2 and 3, as discussed above, demonstrates the such claims are highly disputable.

#### **Entergy's Other Inspection Point Selection Criteria**

19. Entergy's Motion for Summary Disposition vaguely identifies other "tools" or criteria Entergy allegedly relies upon in addition to CHECWORKS for selecting components for FAC inspections. Specifically, Entergy states that, in addition to the use of CHECWORKS, component selection is "based principally 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. *See* Entergy's Motion for Summary Disposition at Attach. 2 ¶ 39. I disagree with Entergy's characterization of these "additional" criteria as independent tools that demonstrate the effectiveness of Entergy's FAC aging management program irrespective of the use of CHECWORKS. *See* Entergy's Motion for Summary Disposition at 17-18.

20. 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 or for the formulation of an engineering judgment, and not a stand alone "tool" for component selection. Moreover, obviously, 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.

21. Industry and plant experience also cannot be properly categorized as independent “tools” for component selection. Rather, knowledge of pipe wall thinning events at Indian Point and other plants are simply types of information that feed into the CHECWORKS model, and/or contribute to ones ability to formulate a judgment regarding proper inspection scope. In other words, they are merely inputs into engineering judgment and/or CHECWORKS.

22. To the extent engineering judgment can be considered an independent tool for selecting components for FAC inspections, Entergy has failed to demonstrate that this alone will safely manage FAC at Indian Point. It is commonly recognized in all major industrial plants, (power, chemical, oil) that engineering judgment alone is not sufficiently reliable to prevent component failures from wall thinning. For this reason, many plants supplement that judgment with either computer modeling and/or direct or indirect continuous on line wall thinning measurements. 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.

23. 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. Engineering judgment is intrinsically subjective. Even with the same input data, different assessments could lead to different results because each assessment would depend heavily on the individual skill and judgment of the responsible engineer. Accordingly, 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 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. Entergy has not identified any kind of systematic methodology which demonstrates that engineering judgment is a separate predictive tool that adequately manages FAC related component degradation.

24. Based on the foregoing, and in direct contradiction to Entergy’s assertions otherwise, 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. Rather, Entergy’s program for managing FAC relies heavily on the unreliable CHECWORKS code.

#### **Entergy’s Improper Reliance on Guidance Documents**

25. Because Entergy has failed to show that CHECWORKS will be an effective tool for adequately managing the effects of FAC at Indian Point during the period of extended operation, or that it has other methods sufficient to manage such aging effects, it is necessary for Entergy to provide detailed information regarding the method and frequency of component inspections and attendant criteria for component repair and replacement. In contrast, Entergy merely states that its FAC program relies on and is consistent with various guidance documents. *See* Entergy Motion for Summary Disposition, Attach. 2 ¶¶ 56-59. However, these generic guidance documents and fleet wide procedure focus heavily on the use of a properly calibrated CHECWORKS model. *See, e.g. id.* at Attach. 9. Because Entergy has failed to show that CHECWORKS is properly benchmarked so as to be an effective tool at Indian Point, as discussed above, Entergy has not been successful in implementing the essential elements of the

referenced documents. Moreover, as discussed above, Entergy has failed to properly define how it employs “engineering judgment” or other “tools” to adequately address FAC in accordance with such guidance. Accordingly, mere reference to guidance and fleet wide procedure is not a demonstration that Entergy had developed an effective program for safely managing FAC at Indian Point.

### **Entergy’s Improper Comparison to the Vermont Yankee License Renewal Proceeding**

26. Entergy’s Motion for Summary Disposition references various findings of a different Atomic Safety and Licensing Board (“ASLB”) in a different license renewal proceeding concerning the Vermont Yankee nuclear power plant (“VY”), in order to demonstrate compliance with applicable regulatory requirements. For example, Entergy relies upon the VY ASLB’s findings relating to the need to benchmark CHECWOKS for use at VY, and the overall sufficiency of Entergy’s “other tools” for managing FAC at VY. I have reviewed the decision made by the Atomic Safety and Licensing Board in the Vermont Yankee license renewal proceeding (“VY ASLB”). Based on this review, as well as my participation in the VY license renewal proceeding as an expert witness, I disagree that any of the specific findings in the VY proceeding can be generically applied to the instant license renewal proceeding involving Indian Point. The VY ASLB’s conclusions are restricted to the VY plant.

27. Generally speaking, major differences between VY and the Indian Point plants underscore the need to perform an independent assessment of the efficacy of Entergy’s program for managing FAC at Indian Point. This includes the very small size of the VY plant (with a gross thermal output of 1912 MWth) in comparison to the large Indian Point plants (with gross thermal output on the order of 3200 MWth). Moreover, VY is a boiling water reactor, while the Indian Point plants are pressurized water reactors. The latter are known to be significantly more

prone to failures from wall thinning due to FAC than the former. *See e.g.*, Entergy Motion for Summary Disposition, Attach. 15 at 5.25.

28. The VY ASLB's findings regarding the adequacy of the benchmarking of the CHECWORKS computer code at VY cannot be applied here. The use of CHECWORKS must be evaluated at each plant separately to account for the unique differences in materials, local flow velocities, temperatures, water chemistry, accessibility for inspection, 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 at the specific plant.

29. Moreover, in the VY license renewal proceeding, hearings were held shortly after the VY plant changed its operating power. Therefore, the VY ASLB did not have the benefit of any data to assess the ability of CHECWORKS to accurately detect wall thinning in light of changed plant operating conditions (including changed velocities, temperatures, and coolant chemistry). Rather, the VY ASLB relied upon the fact that data would be collected prior to the VY license extension period which it assumed would adequately calibrate the CHECWORKS model. In contrast, data from seven post power uprate outages at Indian Point Units 2 and 3 is already available to assess the ability of CHECWORKS to account for changed plant conditions. As discussed above, this data shows almost a complete failure of CHECWORKS to predict wall thinning in light of new plant operating parameters. This necessarily renders the conclusions of the VY ASLB regarding the benchmarking of CHECWORKS inapplicable in the instant proceeding.

30. The VY ASLB's findings in relation to the minimal role CHECWORKS plays in Entergy's FAC program at VY, and the attendant sufficiency of the detail of Entergy's FAC



program at VY are likewise inapplicable in the instant license renewal proceeding, since, as the discussion above demonstrates, Entergy relies primarily on CHECWORKS in order to address FAC at Indian Point.

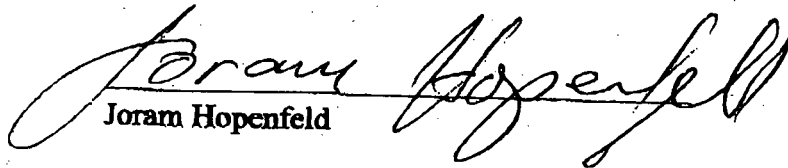
### **Conclusion**

31. The foregoing demonstrates that significant disputes of fact exist regarding the sufficiency of Entergy's program for managing the aging effects of FAC at Indian Point during the period of extended operation. In particular, my testimony herein supports the following findings and conclusions:

- Entergy has failed to demonstrate that CHECWORKS is adequately benchmarked so as to be an effective tool for predicting FAC at Indian Point during an extended period of operation;
- Entergy's program for managing FAC is largely reliant upon the use of CHECWORKS, since Entergy has failed to identify any tools that are meaningfully independent of CHECWORKS that would sufficiently address FAC at Indian Point;
- In the absence of CHECWORKS, Entergy has failed to provide detailed information regarding the method and frequency of component inspections and attendant criteria for component repair and replacement, sufficient to assure adequate, safe management of FAC;
- The findings of the VY ASLB are not applicable to the instant proceeding.

In accordance with 28 U.S.C. §1746, I declare under penalty of perjury that the foregoing is true and correct.

Executed on Aug. 13, 2010.

  
Joram Hopenfeld

## References

Entergy Condition Reports CR-IP2-2001-10525, CR-IP3-2006-02270

Entergy Fleet Procedure, EN-DC-315, Rev. 3, *Flow Accelerated Corrosion Program* (March 1, 2010)

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

Entergy Operating Experience Review Report, Engineering Report No. IP-RPT-06-LRD05, Rev. 1 (June 2007)

Indian Point Unit 2 CHECWORKS FAC Model, Calculation No. 050714b-01, Rev. 0 (July 5, 2005)

Indian Point Unit 2 CHECWORKS FAC Model, Calculation No. 050714b-01, Rev. 1 (Sept. 12, 2006)

Indian Point Unit 2 CHECWORKS SFA Model, Calculation No. 0705.101-01, Rev. A (Nov. 17, 2008)

Indian Point Unit 2 CHECWORKS SFA Model, Calculation No. 0705.101-01, Rev. 1 (Feb. 26, 2010)

Indian Point Unit 3 CHECWORKS FAC Model, Calculation No. 050714c-01, Rev. 0 (Oct. 25, 2005)

Indian Point Unit 3 CHECWORKS SFA Model, Calculation No. 0705.100-01, Rev. 0 (Nov. 14, 2007)

Indian Point Unit 3 CHECWORKS SFA Model, Calculation No. 0705.100-01, Rev. 1 (Feb. 12, 2010)

NL-08-004, Letter from Fred Dacimo, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding License Renewal Application (Jan. 4 2008) (ADAMS Accession No. ML080160123)

Nuclear Safety Analysis Center (NSAC)-202L-R2, Rev. 2, *Recommendations for an Effective Flow Accelerated Corrosion Program*

Nuclear Safety Analysis Center (NSAC)-202L-R3, Rev. 3, *Recommendations for an Effective Flow Accelerated Corrosion Program*

NUREG-1801, *Generic Aging Lessons Learned (GALL) Report* (Sept. 2005), Section XI.M17

NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants* (Sept. 2005).

Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286, Entergy Nuclear Operations, Inc. (Nov. 2009), NUREG-1930 (ADAMS Accession No. ML093170671).

Transcript of Meeting of Advisory Committee on Reactor Safeguards (Sept. 10, 2009)

Riverkeeper Opposition to Entergy's Motion For Summary Disposition of  
Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion)

# Riverkeeper TC-2: Attachment 3

## **Curriculum Vitae for Dr. Joram (Joe) Hopfenfeld**

1724 Yale Pl., Rockville, MD 20850

Tel: 301 340 1625

### **A. Professional Expertise:**

- a. **Nuclear Safety and Licensing** (design basis/severe accidents)
- b. **Thermal/Hydraulics** ( Transient Boiling, Jet Mixing, Reentry Heat transfer, molten metal/coolant interactions, pool fires, computer code developments)
- c. **Materials/Environment Interaction** (corrosion, erosion, stress corrosion, fatigue, cavitation, fouling)
- d. **Radioactivity Transport (10 CFR Part 100 )**
- e. **Industrial Instrumentation and Environmental Monitoring.**

### **B. Current Position - CEO, Noverflo, Inc**

### **C. Education - Engineering- University of California at Los Angeles: BS 1960, MS 1962, Ph.D 1967.**

### **D. Summary of Work Experience**

#### **1. Nuclear Plant Related Experience**

I have 45 years of experience in industry and government primarily in the areas of thermal hydraulics, materials, corrosion, radioactivity transport, instrumentation, PWR steam generator transient testing and accident analysis. I have managed major international programs on steam generator performance during steam generator tube ruptures, steam line and feed line breaks. Following a decade of studies and several Advisory Committee on Reactor Safety hearings, the Nuclear Regulatory Commission, ("NRC") adopted my position regarding the safety consequences of operating with degraded steam generator

tubes. In 2001 the NRC initiated a major program on the effects of steam generator tube degradation on plant safety ( see NRC website). I have consulted to law firms and citizen groups regarding Steam Generators, Thermal Hydraulics, Corrosion , and Material Fatigue in connection with license renewals and a power upgrades.

## **2. Non Nuclear Related Experience**

I am the owner and the CEO of a small Maryland company, Noverflo, Noverflo is developing advanced fiber optic sensors for the oil & gas and the environmental monitoring industries. In 2004 Noverflo has completed a three year program which was sponsored by the U.S. Department of Energy. The program produced a new system for automatic tank gauging, which will be presented at the 2006 National Petrochemicals and Refiners Association Maintenance Conference.

In 1994-1996 Noverflo has developed and commercialized a shutoff valve for fuel tanks to comply with new EPA regulations.

## **E. Brief Employment History**

### **A. Recent Consulting**

#### **1. Winston & Strawn , 1400 L St. Washington D.C**

**2001**

Provided assistance in connection with the February 2000 steam generator event at Indian Point.

#### **2. C-10 Research and Education Foundation, Inc. 44Merrimac St. Newburyport, MA**

**2002-2003**

Provided assistance in the preparation of a 2.206 petition to the NRC and other matters in connection with steam generator problems at the Seabrook Station

#### **3. California Earth Corps (Sabrina D. Venskus, Attorney at Law, Santa Monica, CA)**

**2005**

Provided testimony to the Public Utility Commission of the State of California on behalf of California Earth Corps in connection with the San Onofre steam generator replacement project.

**4. New England Coalition (Raymond Shadis, Edgecomb, Maine 04556)**

**2005-2006**

Technical consultant and expert witness in connection with Vermont Yankee power uprate and life extension hearings before the Atomic Safety and Licensing Board. Prepare contentions and testify before the Board.

**B. Industry and Government Employment**

1962- 1971 –Corrosion testing of materials for the design and operation of liquid metal cooled nuclear reactors. Modeling Transient Boiling in water and sodium. Modeling Sodium Fires. Modeling destruction of SNAP fuel rods on reentry into the earth atmosphere. Atomic International, Canoga Park, Calif.

1971- 1973- Participated in the resolution of design issues as related to material behavior in the Breeder reactor environment. Atomic Energy Commission

1973 – 1978 Project Manager for the safety evaluation and testing of steam generators for liquid metal reactors. Managed the development of thermal –hydraulic computer codes such as COBRA. ERDA/Department of Energy. Responsible for testing material compatibility and cavitation damage in sodium. Development of acoustic leak detection systems for sodium/water reactions.

1978 – 1982 Project Manager for the development of materials and instrumentation for high temperature steam generators for fossil plants. Responsible for the resolution of issues relating to corrosion/erosion and NO<sub>x</sub> /SO<sub>x</sub> emissions, Department of Energy.

1982 – 2001 Program manager for the resolution of various, thermal hydraulics, material corrosion and safety issues primarily in relation to PWR steam generators. Nuclear Regulatory Commission.

**Publications**

In addition to numerous reports, I have published 15 papers in peer-reviewed technical journals in the areas of thermal-hydraulics, corrosion/ erosion, steam generator dose releases during accidents, steam explosions, sensors and ECM machining.

#### **Peer Reviewed**

- 1 "New Fiber Optic Based Technology for Automatic Tank Gauging", *Sensors*, December 2006
2. "Distributed Fiber Optic Sensors for Leak Detection In Landfills", *Proceeding of SPIE Vol 3541 (1998)*
3. "Continuous Automatic Detection of Pipe Wall Thinning", *ASME Proceedings of the 9th, International Conference on Offshore Mechanics and Arctic Engineering*, Feb. 1990
- 4 "Iodine Speciation and Partitioning in PWR Steam Generators", *Nuclear Technology*, March 1990
5. Comments on "Assessment of Steam Explosion Induced Containment Failures" Letter to the Editor, *Nuclear Science and Engineering*, Vol. 103, Sept. 1989
6. "Experience and Modeling of Radioactivity Transport Following Steam Generator Tube Rupture", *Nuclear Safety*, 26,286, 1985
7. "Simplified Correlations for the Predictions of Nox Emissions from Power Plants". *AIAA Journal of Energy*, Nov.-Dec., 1979
8. "Grain Boundary Grooving of Type 304 Stainless Steel in Armco Iron Due to Liquid Sodium Corrosion", *Corrosion*, 27, No.11, 428, 1971
9. "Corrosion of Type 316 Stainless Steel with Surface Heat Flux in 1200 Flowing Sodium", *Nuclear Engineering and Design*, 12; 167-169, 1970
10. "Prediction of the One Dimensional Cutting Gap in Electrochemical Machining", *ASME Transaction, J. of Engineering for Industry*, p100 (1969)
11. "Electrochemical Machining- Prediction and Correlation of Process Variables", *ASME Transactions, J. of Engineering for Industry*, 88:455-461, (1966)
12. "Laminar Two-Phase Boundary Layers in Subcooled Liquids", *J. of Applied Mathematics and Physics (ZAMP)*, 15, 388-399 (1964)



13. "Onset of Stable Film Boiling and the Foam Limit", International j. of Heat Transfer and Mass Transfer, 6; 987-989 (1963) ) (co-author)
14. "Operating Conditions of Bubble Chamber Liquids", The Review of Scientific Instruments, 34, 308-309. (1963); co-author
15. "Similar Solutions of the Turbulent Free Convection Boundary Layer for an Electrically Conducting Fluid in the Presence of a Magnetic Field," AIAA J. 1:718-719 (1965)

**Not Peer Reviewed (Recent Publications Only )**

1. **New Fiber Optic Based Technology for Automatic Tank Gauging ( ATG), NPRA – 2006 Reliability and Maintenance Conference, May 23-26, San Antonio, TX**
2. **Automatic Tank Gauging: A New Level of Accuracy; A New Device Promises Greater Accuracy for Custody Transfer by Combining Fiber- Optic Sensing with a Pressure. Sensors Magazine, 12/01/06**
3. **PlasticOptical Fibers Sensors for Industrial Process Controls and Environmental Monitoring, POF World West 2007, June 25-27. 2007**

**List of Patents**

1. Automatic Shut-Off Valve for Liquid Storage Tanks, 5,522,415
2. Method and Apparatus for Detecting the Presence of Fluids, 5,200,615
3. Sensors For Detecting Leaks, 5,187,366
4. Method for Monitoring Thinning of Walls and Piping Components 4,922,74
5. Method for Monitoring Thinning of Pipe Walls, 4,779,453
6. Looped Fiber Optic Sensor for the Detection of Substances (5,828,798)
7. Coated Fiber Optic Sensor for The Detection of Substances (5,982,959)
8. Method and Apparatus for Analyzing Information of Sensors Provided Over Multiple Waveguides (6,870,607)

### **Honors**

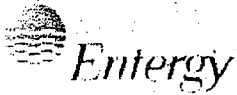
1. **Engineer of Distinction – Published by Engineers Joint Council**
2. **American men and Women in Science**
3. **The Blackwall Award for Machine Tools**
4. **Member Sigma-Xi**

### **Professional Activities**

1. **Reviewed papers for the ASME Journal and the Journal of Sensors and Actuators**
2. **Taught a class on Diesel Engines at Montgomery College, Rockville, MD.**
3. **Served as a member of a Railroad Committee that development a standard for locomotive Fueling**
4. **Funded and sponsored research and development work at the Engineering Department of the University of Virginia. The research produced a novel method of measuring pipe wall thinning from erosion/corrosion**

Riverkeeper Opposition to Entergy's Motion For Summary Disposition of  
Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion)

# Riverkeeper TC-2: Attachment 4



**ENERGY NUCLEAR**  
Engineering Report Cover Sheet

**Engineering Report Title:**

Operating Experience Review Report

**Engineering Report Type:**

New  Revision  Cancelled  Superseded

**Applicable Site(s)**

IP1  IP2  IP3  JAF  PNPS  VY  WPO   
ANO1  ANO2  ECH  GGNS  RBS  WF3

DRN No.  N/A:

**Report Origin:**  Entergy  Vendor

Vendor Document No. \_\_\_\_\_

**Quality-Related:**  Yes  No

Prepared by: Andrew C. Taylor *Andrew C. Taylor*  
Responsible Engineer (Print Name Sign)

Date: 6/7/2007

Design Verified/ N/A  
Design Verifier (if required) (Print Name Sign)

Date: \_\_\_\_\_

Reviewed by: Jack Orlicek *Jack Orlicek*  
Reviewer (Print Name Sign)

Date: 6/7/2007

Reviewed by\*: \_\_\_\_\_  
ANI (if required) (Print Name Sign)

Date: \_\_\_\_\_

Approved by: Michael Stroud *Michael Stroud*  
Supervisor (Print Name Sign)

Date: 6-14-2007

\*\* For ASME Section XI Code Program plans per ENN-DC-120, if required.

**Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems**

Item	Issue	Evaluation
CR-IP2-2001-01924	200101924 – UHT-10-248 (auxiliary steam trap) is a bucket trap that has a plug on the top with an allen wrench center that has a leak. The leak appears to be a through wall leak in the middle of the plug.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water or steam.
CR-IP2-2001-01994	200101994 – Discovered excessive steam leaks on Dock Steam line, 5' section of Utility Tunnel. The area of one leak was a one foot section corroded almost completely through.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water or steam.
CR-IP2-2001-02051	200102051 – Location: 15' south side of loading well by janitor supply cage.  Elbow in aux steam line leaking.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water or steam.
CR-IP2-2001-02140	200102140 – Through wall piping leak on Main Steam line from 22A Moisture Separator Reheater Vent Chamber to 26A Feedwater Heater approximately 1 foot from tie to MS line from 21A MSR Vent Chamber (leak located close to 26A FWH, near valve MS-645).	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water or steam.
CR-IP2-2001-02187	200102187 – The 2" City Water supply piping from the 12" City Water Header, in the Unit 1 Water Factory, to the retired Resin Storage Tank is CORRODED & HAS 2 CLAMPS AND BLACK ELECTRICAL TAPE HOLDING IT TOGETHER.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water.
CR-IP2-2001-02451	200102451 – Through wall leak between valves 387 and 310 inside valve gallery PAB. Noted fresh boron buildup on top of piping in between 310 and 387. Found no other source of leakage above the piping that could have dripped on the pipe.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in treated water.
CR-IP2-2001-02482	200102482 – The 1/2 pipe downstream of CT-843 connected to 3EX-10-1, route stop 23b feed water heater drain. The pipe is welded to 3EX-10-1 and is leaking condensate around the weld.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water or steam.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP2-2001-03046	200103046 – 21 Emergency Diesel Generator Fuel Oil Storage tank fill valve cover iron frame is no longer attached to the concrete pad, repair as required. The iron frame is bent and rusted and should be replaced with new material.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in indoor air.
CR-IP2-2001-03608	200103608 – While performing daily rounds found on 21 House service boiler, soot blower drain line upstream of valve AS-1408 dripping. Could not pinpoint exact location of leak due to insulation/lagging on the pipe.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water and steam.
CR-IP2-2001-03681	200103681 – Steam leak on the crossunder inlet pipe to 21A Moisture Separator Reheater has evidence of steam condensate dripping on the floor coming off the lagging. The leak is located at 53' turbine hall building, north of 21A MSR.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water and steam.
CR-IP2-2001-03734	200103734 – While walking down upcoming jobs I noticed a steam leak on a one inch line leaking from the one inch union located between valve tag number, (HD&V 5EX-512 HDT LC-5004s root stop) and (HDTLC LC-5004s)	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water and steam.
CR-IP2-2001-03887	200103887 – There is a leak of several drops per second that appears to be coming from the pipe cap on the downstream side of valve 5EX-35-15. The valve is located on the underside of Moisture Separator Drain Tank 21A's level controller (LC-1105S).	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water and steam.
CR-IP2-2001-04083	200104083 – Air leak from AOM-9. Found EDG building louvers open. Air leak may be enough to keep these louvers open. No tag found on air motor.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in treated air.
CR-IP2-2001-04232	200104232 – 24A FWHTR outlet temperature indicator thermowell has small leak. Repair/replace thermowell as required.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel in treated water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP2-2001-04776	200104776 – While performing PI-M2 VC inspection found boron encrustation on the packing gland needs to be cleaned for valve 955D.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel in treated water.
CR-IP2-2001-04959	200104959 – The Auxiliary Condensate return header just upstream of valve UW-88 near the Toolroom on 15 foot elevation has a through the wall leak at the twelve o'clock position. Please repair.	Loss of material and cracking are aging effects identified in the mechanical tools for carbon steel or stainless steel in treated water.
CR-IP2-2001-05054	200105054 – On 5/18/01, while attempting to perform a soot blow of 21 HSB, it was noted that there was a through wall leak in the piping upstream of AS-1408 condensate drain for the front soot lance on 21 HSB.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water.
CR-IP2-2001-05904	200105904 – Seal Oil Vacuum Pump Oil sampled on the Seal Oil Vacuum Pump Gear Box indicating large amount of built up oxidized sludge. Oxidized sludge promotes corrosion and deterioration of the oil. The gearbox oil is changed every three months.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in lubricating oil.
CR-IP2-2001-05968	200105968 – While doing a survey in the utility tunnel I noticed that the discharge pipe to the river is scaled with rust in one section. This is the pipe that contaminated the tunnel in the past.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in treated water.
CR-IP2-2001-06470	200106470 – LCV-1127D has a through wall leak. This is a large Heater Drain Tank dump to 23 condenser. This valve is being isolated. Initiate work order to repair.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water.
CR-IP2-2001-06476	200106476 – Hydrogen cooler #22 south section inlet relief valve SWT-62 located on the east side of 36' near the Service Water manual throttle valves has a service water through wall leak at elbow weld.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in raw water.

**Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems**

Item	Issue	Evaluation
CR-IP2-2001-06740	200106740 – 24 Service Water Pump Vacuum Breaker SWN-9-3 Failed its PMT due to leak at threaded fitting at top of valve body.  Reference CR# 200100463.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in raw water.
CR-IP2-2001-07107	200107107 – Following performing chlorination on Circulating Water Pump bays 21,22, & 23, it was discovered that PCV-7979 has evidence of leakage (residue on side of valve, residue stalagmite forming on bottom of valve, and residue on pump casing).	Loss of material is an aging effect identified in the mechanical tools for plastic in raw water.
CR-IP2-2001-07232	200107232 – LCV-1127C Heater Drain Tank Large Dump Valve to 22 Condenser has a through wall leak on the east side of the valve body. This is the second of the three large dump valves to have a leak. (LCV-1127D is already isolated CR#01-06470)	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water.
CR-IP2-2001-07951	200107951 – LCV-1127D Heater Drain Tank Large Dump to 23 Condenser, has through wall leak on piping located directly below the valve.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water.
CR-IP2-2001-08270	200108270 – GT-3, # 4 basket fuel oil supply line coupling closest to the nozzle is leaking at 1 drip/minute during Gas Turbine operation, investigate and repair.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in fuel oil.
CR-IP2-2001-09058	200109058 - While performing an Environmental/Safety tour of the Utility Tunnel, it was discovered that the 20" Fuel Oil Fill Line is being severely eroded by in leakage of "sweet" water.	Loss of material is an aging effect identified in the mechanical tools for carbon steel with outside air on external surfaces, which is presumed to include moisture.
CR-IP2-2001-09241	200109241 – During the Annual Walkdown on the Gas Turbines it was found on the GT3 Blackstart diesel that its exhaust stack has a minor exhaust leak located at the joint between the expansion joint and the lower end of the muffler	Loss of material is an aging effect identified in the mechanical tools for carbon steel in exhaust gas, indoor air or outdoor air.



**Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems**

Item	Issue	Evaluation
CR-IP2- 2001-09482	200109482 - #12 ignition oil pump is leaking from the bottom of the pump casing.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in fuel oil.
CR-IP2- 2001-09593	200109593 – DPI-5000S, 21 Service Water Strainer Differential Pressure low side impulse line has a minor leak at the threaded connection on the housing, this is causing corrosion of surrounding components, repair same.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.
CR-IP2- 2001-09653	200109653 – There is a thru wall pipe leak downstream of AS-1076. Pipe is severely corroded and needs to be changed from the steam trap and to include the check valve and AS-1076.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2- 2001-09659	200109659 – RW-132 heat exchanger 12 tube side drain and RW-126 heat exchanger 11 tube side drain have both broken off the heat exchangers. The threaded nipples corrode and rot away.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.
CR-IP2- 2001-09743	200109743 – Leaks exist at three locations on 22 House Service Boiler (22 HSB) at the interface of the mud drum and Firebox outer casing. Each leak is approximately two drops per second.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2- 2001-09797	200109797 – Hot water return inlet stop UH-344 to heat exchanger HE-1 in MOB HVAC room is severely corroded and has a packing leak.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water.
CR-IP2- 2001-09821	200109821 – The 10" piping just below LCV-1127C has a thru wall steam leak. This leak is in addition to the thru wall body leak on the valve.  NOTE: This condition is the same as the steam leak on LCV-1127D which is to be worked under 01-22963.	Loss of material is an aging effect identified in the mechanical tools for carbon-steel in treated water.
CR-IP2- 2001-10925	200110925 – Request a more immediate response to repair of Dock Steam. The condensate return line associated with this is corroded and leaks badly. This is rendering Dock Steam out of service.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel or carbon steel in steam and treated water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP2-2001-11779	200111779 – In an effort to reduce the effect of Flow Accelerated Corrosion in the secondary piping, the Wet Steam Piping Replacement Project was created as part of the IP2 Flow Accelerated Corrosion (FAC) Program.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2001-11861	200111861 – The inside fish spray header on #22 TSC has a thru wall leak at the elbow just down stream of the stop valve WW-103.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in raw water.
CR-IP2-2002-00013	200200013 – The 2 inch Trough drain waste water line threaded fitting on 23 Charging pump is leaking at a rate of 1 drip/ 10 seconds at the 90 degree elbow before the vertical section of piping causing a housekeeping issue in the cell.	Loss of material is an aging effect identified in the mechanical tools for stainless steel or carbon steel in treated water.
CR-IP2-2002-00818	200200818 – While performing Corrective Maintenance (NP-98-05617 Replace Piping upstream of FP-2) there was excessive corrosion observed on the inside of the fire piping. The wall thickness of the piping is heavily degraded.	Loss of material is an aging effect identified in the mechanical tools for stainless steel or carbon steel in treated water.
CR-IP2-2002-01055	200201055 – The welded elbow downstream of AS-1075 has a significant steam leak as well as a union downstream of this elbow. There exists a deficiency tag on or around AS-1075 however I was unable to read it due to steam impinging on the tag.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel or carbon steel in treated water or steam.
CR-IP2-2002-01199	200201199 – Leak at end bell on CCC side of Heat Exchanger. Leak appeared during isolation of heat exchanger to change out zinc plugs.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water.
CR-IP2-2002-01457	200201457 – Drain plug for BFD-4-2 (26C Feedwater Heater outlet stop) has a small drain plug leak.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel or carbon steel in treated water or steam.

**Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems**

Item	Issue	Evaluation
CR-IP2-2002-01628	200201628 – This condition was found during SAO-141 Walkdown. Water is leaking from a threaded cap connection downstream of valve 5EX-23 onto floor. Valve 5EX-23 is a Heater Drain & Vent dump line header drain stop valve located on the 5' elevation of the turbine building.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel or carbon steel in treated water or steam.
CR-IP2-2002-01820	200201820 – MSR-22A Vent Chamber line drain stop leaks- by. It is located on the 15' el. by moisture pre-separator tank.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel or carbon steel in treated water or steam.
CR-IP2-2002-02368	200202368 – The steam supplied wall heater on the South wall of the Ignition Oil Tank Room has an elbow leak at the outlet of the heater. Condensate was leaking onto the floor and under the tanks.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel or carbon steel in treated water or steam.
CR-IP2-2002-02864	200202864 – During field inspections noted 21 Main Boiler Feed Pump suction piping had water dripping out of the insulation about three feet above the pump casing. Possible through the wall leak on the suction piping of 21 MBFP.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water.
CR-IP2-2002-04664	200204664 – Insulation needs to be removed from in between 5EX-4 and 5EX-3 in order to identify possible thru wall leak on extraction steam line.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel or carbon steel in treated water or steam.
CR-IP2-2002-04951	200204951 – Small steam leak located at 23B MSR inlet inspection/access port at north end of MSR 53' Turbine Hall. Leak is located under lagging. Maximo work order # 02-02851.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel or carbon steel in treated water or steam.
CR-IP2-2002-05463	200205463 – SWN 77-6 has a thru wall leak at the vent valve.	Loss of material is an aging effect identified in the mechanical tools for stainless steel or carbon steel in raw water.
CR-IP2-2002-05472	200205472 – There is a thru wall leak on the piping approx. 3 feet to the west FW-226.	Loss of material is an aging effect identified in the mechanical tools for stainless steel or carbon steel in raw water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP2-2002-05524	200205524 – A through wall leak has developed on 12 house tank fill pump suction piping immediately downstream of suction valve FP-68.12 house tank fill pump was already out of service via tagout 2000N-14395, which isolated the pump discharge.	Loss of material is an aging effect identified in the mechanical tools for stainless steel or carbon steel in treated water.
CR-IP2-2002-06004	200206004 – The piping between WW-178 and PI-6987 has a 0.5 gpm leak when 28 traveling screen is washing. The insulation needs to be removed to determine the location of the leak. Most likely the elbow weld is leaking.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in raw water.
CR-IP2-2002-06358	200206358 – Noted during field inspections small leak (2-3 drops/minute) on the suction piping going to #23 Charging pump. Thru wall leak is down stream of valve 284 (suction stop) on weld just upstream of C-7 drain valve.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in treated water.
CR-IP2-2002-07210	200207210 – During PM of R-46 four of the casing studs and nuts were found unacceptable due to corrosion. Couplings for the upper and lower manifolds are unacceptable due to damaged threads.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in condensation and indoor air.
CR-IP2-2002-07731	Found 22 HZFP (22 Hydrazine Feed Pump) leaking at pump casing onto floor. About 500 cc of dilute Hydrazine from Chemical addition tank has leaked onto floor. Pump was taken out of service and valves isolated, chemical spill tape was placed around tank.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in steam and treated water.
CR-IP2-2002-08136	FAC (Flow Accelerated Corrosion) component FAC-1B-VCD17 has wall thickness readings below allowable limits per FAC procedure SE-SQ-12.318 (T <sub>min</sub> = 0.135", T <sub>meas</sub> = 0.110"). Component is a 3" 90-degree elbow directly downstream of valve HCV-5068B on the MSR.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP2-2002-08370	FAC (Flow Accelerated Corrosion) component FAC-2B-VCD39 has wall thickness readings below allowable limits per FAC procedure SE-SQ-12.318 ( $T_{min} = 0.135"$ , $T_{meas} = 0.134$ ). Component is a 3" elbow directly downstream of valve MS-618.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2002-08676	PI-M9 aboveground petroleum tanks inspection failed due to oil leaks. Main boiler feed pump wrt IP2-02-02798, Main turbine oil conditioner wrt IP2-02-02713, IP2-02-02714, IP2-02-02794, IP2-02-00349, AND CR200203880, 200201990, 200202090, AND 006360.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in fuel oil.
CR-IP2-2002-08823	Piping downstream of steam trap AST-20 is leaking. Leak seems to be coming from coupling downstream of steam trap. Steam trap is located in front of 15' elev Tool Room.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water or steam.
CR-IP2-2002-08858	22 Containment Spray Pump continues to show evidence of large amounts of copper in the oil from the pump reservoir. Inspection was performed in July with no visible anomalies identified. Copper is still being generated from an unknown source.	Loss of material is an aging effect identified in the mechanical tools for copper in lube oil.
CR-IP2-2002-09024	While examining the zinc anodes on #21 EDG JWC & LOC it was observed that the recently replaced expansion joint SWN-66 has a thru wall leak of less than one drop per min.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in raw water.
CR-IP2-2002-09073	Found 24 SWP Zum strainer blowdown piping flange bolts badly rusted during performance of PI-3Y13. This is the 3"-150 psig pressure boundary flange downstream of the strainer and upstream of SWN-594. All the other strainers have stainless steel bolting.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP2-2002-09074	During performance of PI-3Y13 for 24 SWP epoxy delamination was found downstream of SWN -3-3 along the entire spool piece from flange to flange. The air pocket created by delamination is causing the discharge piping to corrode at an accelerated rate.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in indoor air.
CR-IP2-2002-09076	Corrosion and evidence of thru wall leakage was observed on EDG 21 SW stainless steel expansion sleeve SWN-66-3. This examination was performed as part of the Extent of Condition response for CR 20002-09024 to inspect the lower SW expansion sleeves.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in raw water.
CR-IP2-2002-09115	Calculated wear rate (see EVAL# 15P-MST-24(b)) indicated the predicted thickness (Tp) for component MST-24 will reach the component's minimum required thickness (Tmin) within the next operating cycle. It is recommended that the component be replaced.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2002-09781	Valve 897B has at least one stud that has some amount of degradation found during the Section XI bolted Connection Inspection Program. The back side of the valve is inaccessible and therefore has not been inspected.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in treated water.
CR-IP2-2002-09869	16" Flange face that mates to FCV-1112 has an area of degradation approximately 1/8" depth, 4-1/2" long around inner circumference at 8 o'clock position, and 1" wide. This is on the west side flange of the valve.	Loss of material is an aging effect identified in the mechanical tools for carbon steel, stainless steel, and nickel alloy in raw water.
CR-IP2-2002-09949	18" lower flange that mates to SWN-39 has an area of concrete lining that is missing on the interior. The missing area of concrete is about a 3 inch width right near the flange face and extends about half way around the circumference of the pipe.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP2-2002-10043	Visual inspection performed of piping at location of SWN-2 which was removed for a valve PM. This inspection revealed a cement lining defect on the upstream flange at 7 o'clock position.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water.
CR-IP2-2002-10920	There is a crack in the air line leading to MPS-758 (MPS Tank B Non Return Inlet check valve). The crack is in the first elbow upstream of the valve. MPS-758 is located in the 32' elev mezzanine of the Turbine Bldg on the north side under the HP Turbine.	Cracking is an aging effect identified in the mechanical tools for stainless steel in treated air.
CR-IP2-2002-10965	Today while conducting a test on the fire supply to the Service Center the pre test flush discharged significant amounts of rust and other debris. Such debris could potentially clog fire nozzles and damage equipment.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water.
CR-IP2-2002-11024	#2 Fuel Oil Header in the utility tunnel is degraded. The lagging and flashing were removed from the carbon steel fuel oil header in the utility tunnel.	Loss of material is an aging effect identified in the mechanical tools for carbon steel with outside air on external surfaces, which is presumed to include moisture.
CR-IP2-2002-11154	The CPD sample cooler for the sodium and hydrazine analyzer has a shell leak. This cooler has a service water cooling supply.	Loss of material is an aging effect identified in the mechanical tools for stainless steel or copper alloys in treated water.
CR-IP2-2002-11159	The Boric Acid Building Make-up Air Unit Heater Coil is leaking. UH-684 is closed and water is still coming out of the base of the fan and forming a puddle on the floor. The water then seeps through the floor and puddles on the lower elevation.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel or copper alloys in treated water or steam.
CR-IP2-2002-11169	Extraction Steam line to EST-18 down stream of 3EX-37-4 is leaking from under the insulation and has increased from the original report (see WRT-IP2-02-01676, 10/4/02). There is steam and water coming from the insulation at one end and water dripping from the other.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water or steam.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP2-2002-11194	At the union to 26B Feedwater Heater low level column there is a pinhole leak in the weld.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water or steam.
CR-IP2-2002-11229	Found pin hole leak thru a weld on valve 5EX-48-1, HDT Dump To Cond. 22 Drain Stop, on 5ft. of the Turbine Hall. This condition could have possible dissolved oxygen level increase.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water or steam.
CR-IP2-2002-11266	Drain valve 5EX-48-1, the drain on LCV-1127C (HDT large condenser dump to 22 condenser) was reported in CR 200211229 to have a pin hole leak in the weld.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water or steam.
CR-IP2-2002-11594	There is approximately a 1 drop per second leak upstream of SWT-47-10 (22 Hydrogen Cooler North Section Vent Stop). The leak is dripping from the first elbow out of the 22AHC hydrogen cooler.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in raw water.
CR-IP2-2003-00003	LW-828 Sphere Foundation pump discharge drain valve upstream piping is corroded. Failure of this pipe will cause CSB 14' to be flooded. LW-828 is where the back-up Air Driven pump discharge is connected.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.
CR-IP2-2003-00088	Attempted to flush 21 Condenser Vacuum Pump moisture separator tank. First flush brought down the sodium to 9 PPB from 40 PPB.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel in steam and water.
CR-IP2-2003-00341	Strainer downstream of AS-1261, atomizing steam to 22 HSB, failed due to a through wall crack from the bottom threads to middle of body. This failure caused the room to fill with steam while I & C personnel were in room troubleshooting 21 HSB.	Cracking is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2003-00587	During the restoration of the unit 1 dock steam header the following leaks were identified in the utility tunnel AS-27 had a leak on a welded union. The dock steam aux condensate header in the utility tunnel had a pinhole leak.	Loss of material and cracking are aging effects identified in the mechanical tools for carbon steel or stainless steel in steam and treated water.



Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP2-2003-00941	24 inch service water lines 405 and 409 have corrosion buildup near the tops of the pipes. The ceiling of the steam generator blowdown tank room shows no evidence of leakage directly above the pipes.	Loss of material is an aging effect identified in the mechanical tools for carbon steel with condensation on external surfaces.
CR-IP2-2003-02016	A through-wall leak was found on the stator winding cooling water system. The leak is on line 2"YRCF at 45 degree elbow just North of the staircase near the weir. Presently about one drop per second is leaking and appears to be increasing.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in treated water.
CR-IP2-2003-02020	Lube oil valve, LO-1, outlet flange leaks. Documented under PI-M9 (DEC tank inspection).	Loss of material is an aging effect identified in the mechanical tools for carbon steel in lube oil.
CR-IP2-2003-02310	21 House Service Boiler steam drum leaking slightly. The water is evaporating before it reaches the floor.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2003-02794	Steam Generator Blowdown Tank outlet pipe down stream of SWN-53 has a small through wall leak at the weld where the pipe is connected to the Service Water Outlet. Leak is a couple of drops per minute.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2003-02798	Noted through wall leak at elbow downstream of MS-102-63 (MST-45 inlet stop). Insulation has been removed previously due to water accumulation in area of MS-102-63.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2003-02870	Request for carbon steel bolt inspection. During the performance of a safety injection pump surveillance, we observed that the carbon steel flange bolts for FE-950 (Recirculation to refueling water storage tank) are rusted.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in outdoor air.
CR-IP2-2003-03384	During tours noted a thru wall steam leak just up stream of MS-20B on 22 Main Steam lead 36' elevation north end of Turbine Hall in overhead. (PT-1134-3 root stop)	Loss of material and cracking are aging effects identified in the mechanical tools for carbon steel in steam and treated water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP2-2003-03849	An epoxy coating defect was found on 21EDLC when performing the 6 month clean/inspect PM per work order IP2-02-42966. The defect is approximately a 1 inch long by 1/16 inch wide chip in the epoxy at the under side of the channel end divider plate.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water.
CR-IP2-2003-04031	There is a minor steam leak just upstream of MSR-29 (21A MSR LP Inlet Press Root Stop) at a flanged connection. The leak is evident from the top and bottom of the flange (east side) through the insulation.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2003-04633	WO IP202461 and CR 200304031 written on 6/22/03 describes the LP steam inlet (hi press turbine exhaust) to 21A MSR flange leak. The leak appears to have worsened.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2003-05306	A rusty nipple was found to be leaking on the bottom of the 10-inch Service Water Return line for the station EDG's just upstream of the 1176 valves. It is believed that the nipple used to belong to SWN-76-1.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water.
CR-IP2-2003-06224	During the tagout of 11 fresh water cooling heat exchanger to replace drain valve RW-127 under work order IP2-02-04630, it was discovered that 11 fresh water heat exchanger has tube leakage.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel or copper in raw water or treated water.
CR-IP2-2003-06567	During the performance of changing out the zincs and endbell gasket on 23SIP lube oil cooler under work order 03-19020, a small crack was found in the lower zinc hole. The crack was thru wall and down the length of the threads.	Cracking is an aging effect identified in the mechanical tools for carbon steel or stainless steel in treated water.
CR-IP2-2004-00213	Ultrasonic thickness reading taken on line 405, outlet piping from #21 CCW HT ETX was found to be below 87.5% (.328") of the nominal wall. Reading as low as .250" were observed on the first elbow downstream from SWN-35.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water or treated water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP2-2004-01401	Replacement piping and Victualic coupling on city water line, 43' Unit 1 Utility Tunnel is extremely corroded. This is due to the same ground water action that necessitated the original piping replacement.	Loss of material is an aging effect identified in the mechanical tools for carbon steel with outside air on external surfaces, which is presumed to include moisture.
CR-IP2-2004-01738	Section of pipe upstream of valve SWN-62-4 is coated with rust. Location is the bottom of the pipe between the 1 <sup>st</sup> & 2 <sup>nd</sup> elbow downstream of the service water header.	Loss of material is an aging effect identified in the mechanical tools for carbon steel with condensation on external surfaces.
CR-IP2-2004-02281	There is a pin hole leak on 21 SJE-C first stage ejector for 21 SJAE located about 6 inches below the elbow going to SJAE condenser. The leak is a steady stream and has created a 2 scfm air leak.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2004-02954	During Flow Accelerated Corrosion (FAC) examination (WO IP2-03-26606) FAC point 214-25P, wall thinning was noted.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2004-04010	Air In-Leakage is elevated at Unit 2. Latest air in-leakage results from 8/29/04 indicate a total of 9.85 scfm. (21 Condenser air in-leakage - 7.5 scfm, 22 Condenser air in-leakage - .65 scfm and 23 Condenser air in-leakage - 1.7 scfm)	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2004-04011	While flushing 22 BATP prior to hanging PTO 2-CVCS-22BATP REBUILD/VARIOUS WORK REV 0-0, water was noted coming out of the insulation under valve 355B.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in treated water.
CR-IP2-2004-04446	During the Service Water Radiography of Line 410, Weld F-1574, one area of degradation in the form of erosion was identified. The area of erosion measured approx. 1" wide by 2" long.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water.
CR-IP2-2004-04556	Service water is leaking from the nipple downstream of valve SWT 823 which is on the outlet side of HPPFW sample cooler at the SWAP.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP2-2004-04565	Main steam thru wall leak downstream of valve MS-667-X1 Inlet Isolation Valve on FT-5058 on 23A MSR. This is located on the south end of 23A MSR on the Main Steam Inlet piping.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2004-04691	While performing 2Y inspection of the CCR HVAC UNIT21 MTR under WO IP2-02-64800 it was discovered that the flare nut on the TXV equalizing line has a crack its entire length and thru wall.	From the CR description, it does not appear that either of the known cracking mechanisms for bolting (stress corrosion cracking and fatigue) was present. Since no other examples of cracking of non-Class 1 bolting materials was found, this isolated case is judged to reflect a manufacturing defect in this flare nut.
CR-IP2-2004-05358	During UT inspection of component MS-1B26 (90 ELBOW) in the main steam line wall thinning was detected below the administrative screening criteria of 70% of nominal wall thickness. The nominal thickness and the screening criteria of the component is 0.432.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2004-05794	CR written to document results of evaluation performed on piping between valves 6EX-3 and 6EX-4, Extraction steam non-return check valves for 26FWH. When the valves were opened up for inspection, a rust bloom was found in bottom of pipe between the two valves.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2004-06150	During performance of PWT# IP2-04-15373, Inservice inspection for leakage of various service water system piping, valves, and components, identified the following; On 22 EDG Service Water Supply from the 1-2-3 header, located on a horizontal piping run upstream of SWN-62-4, there is a considerable build up of rust on the underside of the pipe.	Loss of material is an aging effect identified in the mechanical tools for carbon steel with condensation on external surfaces.
CR-IP2-2004-06162	Chemical trends indicate an active corrosion mechanism in the Unit 2 CCW. Copper and iron concentrations have increased significantly.	Loss of material is an aging effect identified in the mechanical tools for carbon steel and copper in treated water.

**Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems**

Item	Issue	Evaluation
CR-IP2-2004-06238	This is to record and track the as-found condition of the main boiler feed water pump lube oil coolers 22FPLOC and 21FPLOC. There was some corrosion damage on the channel heads of both of them.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water, lube oil and indoor air.
CR-IP2-2004-06741	During the Extent Of Condition inspection of the Service water line welds to the EDGs today, it was determined that the 6" line from the 1-2-3 header to 21 EDG has a weld below minimum thickness that will need to be removed and replaced.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water.
CR-IP2-2004-06830	Flange below LC-5206-2S is leaking.	Loss of material and cracking are aging effects identified in the mechanical tools for carbon or stainless steel in treated water.
CR-IP2-2004-06776	The preliminary UT reports for the 9 welds upstream of valve SWN-62-6 on the 1-2-3 header of the 23 EDG has indications below the minimum wall calculated.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water.
CR-IP2-2004-06796	Unit Heater 246 has a steam leak in the coil area.	Loss of material and cracking are aging effects identified in the mechanical tools for copper or stainless steel in steam and treated water.
CR-IP2-2004-06847	During a routine PM of Vacuum Breaker SWN-9-3 (IP2-02-32450) the nuts and studs attaching the Stainless Steel elbow to the 24 Service Water discharge header was found to be severely corroded. No leakage is present.	Loss of material on bolting is an aging effect identified in the mechanical tools for carbon steel or stainless steel bolting with condensation on external surfaces.
CR-IP2-2005-00162	Through Wall leak on weld for CD-98-1, 21 MBFP suction line vent valve. Leak can be isolated by removing 21 MBFP from service and applying PTO.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP2-2005-00294	Flange upstream of MS-1064 has a steam leak.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.

**Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems**

Item	Issue	Evaluation
CR-IP3-2001-01045	During an erosion/corrosion examination wall thinning was noted on piping downstream of HD-LCV-7003. This inspection was required as the valve is leaking and was noted by performance test personnel via PFM-59.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2001-01096	During an erosion/corrosion examination (WR 00-04379-07), wall thinning was noted on MSR Vent Chamber Drain piping downstream of MSR 32B, located 2'6" south and 11'6" west of F/20, approx. el. 45'.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2001-01285	During an erosion/corrosion examination, (WR 00-04379-09, 01-PT-08), wall thinning was noted on piping downstream of valve MS-HCV-146-2. The inspection was required as a pinhole leak was discovered at a weld in upstream piping, and similar valves are cur	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in steam and treated water.
CR-IP3-2001-01322	During an erosion/corrosion examination, (WR 00-04523-02, 01-PT-24) wall thinning was noted on piping downstream of Main Steam Trap MST-80 (Main Steam Balancing Line).	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2001-01514	During R09, R10, and pre-R11 NRC GL 89-13 NDE inspections of insulated carbon steel Service Water piping in the VC, a condition has routinely been found of heavy metal exfoliation on the exterior of the 10" FCU supply and return lines.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in indoor air.
CR-IP3-2001-01593	During the Generic Letter 89-13 inspections, location EOC-26, on the 24" line 408 in the room with the rock area, was found to have wall thinning. Minimum code thickness was .151", while a 1" length was found to be 0.132".	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.
CR-IP3-2001-01749	During inspections of the fan cooler units, pin hole leakage or evidence of pin hole leakage was found on six of the ten valves that serve as isolation valves to the fan cooler unit motor coolers. The valves in question are: SWN-520; SWN-521; SWN-523.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP3-2001-01887	A forced plant outage was necessary in Jan. 1997 due to feedwater heater tube leaks in the #31 FWH's. Most of the tube damage found was in the form of OD thinning at the bottom of the inlet passes.	Loss of material is an aging effect identified in the mechanical tools for copper alloy or carbon steel in steam and treated water.
CR-IP3-2001-01921	No piping replacement is required.  During an erosion/corrosion examination, (WR 00-04521-01, 01-PT-5), wall thinning was noted on three (3) piping segments downstream of MS-PCV-1152 and MS-196.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2001-01985	No piping replacement required.  During an erosion/corrosion examination, (WR 00-05234-09, RHD-02.6B-01E), wall thinning was noted on an 8" elbow downstream of RHD-LCV-1105B.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2001-02124	At the 5/15 day to night SW (Service Water) turnover, dayshift reported that during the extent of condition flange face inspection the engineer noticed a missing piece of concrete liner on the pipe near the flange.	Loss of material is an aging effect identified in the mechanical tools for nickel alloy or stainless steel in raw water.
CR-IP3-2001-02319	During the Service Water ISLT the following items were noted:  PID 01065 – SWT-238 Blowdown Hx 4 relief valve inlet piping leak.  PID 01067 – SWT-80 31 Exciter Air Cooler Inlet Isol. pipe leak.  PID 01068 – 31 MBFP Oil Cooler head has small leak.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.
CR-IP3-2001-02320	A pin-hole type leak was discovered just downstream of the downstream flange-to-pipe weld at valve SWT-24. Leak rate is approx. 2-3 drops/sec. Leak is in a non-safety-related, non-ISI section of the service water system.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.

**Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems**

Item	Issue	Evaluation
CR-IP3-2001-02324	PID 03521 stated:  Boron buildup / leak on Swagelok between SP-AOV- 956c and SP-AOV-956d, where IVSWS ties on PZR liquid space sample line.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in treated water.
CR-IP3-2001-02419	Main Generator H2 leakage is above the action limit of 500 SCFD. Based on the last 3 days the trend is up, with leakage increasing from 600 CFD to 900 CFD.	Loss of material is an aging effect identified in the mechanical tools for carbon steel, stainless steel or copper alloys in raw water.
CR-IP3-2001-02489	Plant personnel discovered that an elbow on a 2" drain line from the 1A (northeast) moisture preseparator to the heater drain tank is leaking steam. Elbow is located about halfway between the line isolation valve MS-125-3 and the check valve MS-126-3.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2001-02534	While measuring air in leakage on 32 condenser, it was observed that on 32 Air Ejector Loop Seal Check Valve CV-49 had a thru wall leak. Leakage thru the valve was approx. one (1) drop every 4 seconds.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2001-02567	1/2" to 1" thick buildup of corrosion products was found on the inside of valve bodies removed from MW-337 and MW-338 under corrective maintenance WRs 00-02670-00 and 00-02672-00.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water.
CR-IP3-2001-02620	While performing RE-CCI-030 "Electrical Generator Hydrogen Survey" the chemistry technician found a significant leak around the bottom of 32 hydrogen cooler.	Loss of material is an aging effect identified in the mechanical tools for carbon steel, stainless steel and copper alloy in raw water.
CR-IP3-2001-02710	During a routine Shift Manager tour 5HD-2-5 was found leaking. The valve, "33A Moisture Separator Drain Tank to HDT" check valve has a through wall leak on the side of the valve.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.



**Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems**

Item	Issue	Evaluation
CR-IP3-2001-02751	During rounds, NPO discovered a small pinhole, through-wall leak in the service water pipe header to the CCW heat exchangers. This hole appears to be at the toe of the weld on the cross-tie tee connection (between valves SWN-31 and SWN-33-2).	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.
CR-IP3-2001-02817	During routine rounds the conventional NPO identified a small leak on the MST-64 strainer.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2001-03181	While replacing 31 Potable Water Booster pump which had a through wall leak on the casing, the mechanics bumped into the adjacent 32 Potable Water Booster pump. The discharge line on the pump completely sheared off probably due to corrosion.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless in steam and treated water.
CR-IP3-2001-03440	During removal of a PTO, an NPO discovered small service water leaks on 32A and 32B condenser heads at the piping welds. The leaks were approximately 2 to 6 drops/minute.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water.
CR-IP3-2001-04148	Pinhole service water leak discovered on outlet of piping from 33 FCU.  The leak is approx. 1 drop per minute between the containment wall and the Containment isolation valve.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.
CR-IP3-2001-04449	Eddy current inspections were performed on the tube side (Service Water side) of the #31 & #32 CCW heat exchangers as part of scheduled PMs under WRs 99-04460-01 & 99-04461-01. In both heat exchangers, ID corrosion pitting was found resulting in the plugging of 2 tubes in #31 HTX and 5 tubes in # 32 HTX.	Loss of material is an aging effect identified in the mechanical tools for copper alloy in raw water.
CR-IP3-2002-00068	During a walkdown, the pipe sleeve for Weld Channel Zone 4A was found corroding in the drain trench. The sleeve for this flow-thru test station protects the embedded weld channel and piping as shown on dwg. 9321-F-70333, Detail No. 3.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.

**Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems**

Item	Issue	Evaluation
CR-IP3-2002-02254	The City Water Line that leads to the EDG Expansion Tanks is corroded on the outside. The corroded section is just as it leaves the wall in the valve pit in the EDG Valve Room. The corrosion has been caused by occasional discharges from a vacuum breaker.	Loss of material is an aging effect identified in the mechanical tools for carbon steel with outside air on external surfaces, which is presumed to include moisture.
CR-IP3-2002-02751	One through wall pin hole leak and one through wall 2" circumferential crack on the body weld were found on the CT-LCV-1158-2. This Valve is CAT I and Seismic Class I.	Loss of material and cracking are aging effects identified in the mechanical tools for carbon steel in treated water.
CR-IP3-2002-02793	The following conditions were found during replacement of 36 CWP Motor cooling coil under WO# I3-020087100:  *Numerous corrosion-induced pinhole leaks were noted during as-found testing of the installed stainless steel cooling coil.	Loss of material is an aging effect identified in the mechanical tools for stainless steel in raw water.
CR-IP3-2002-02886	Steam trap EST-4 downstream piping "T" has pinhole steam leak.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2002-03132	Service Water leakage discovered at threads where SWT-63-1, -3, -4, -5, and -6 (Carbon Steel) threads onto the Bus Duct Cooling piping (Stainless Steel).	Loss of material is an aging effect identified in the mechanical tools for carbon steel and stainless steel in raw water.
CR-IP3-2002-03263	A steam leak was identified on the 16" drain line from the moisture pre-separators to the heater drain tank.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2002-03622	Found pinhole leak on weld upstream of SWN-34-1 (inlet to 31 CCW heat exchanger).	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water.
CR-IP3-2002-03811	A leak was identified on the 31 sparging pump at the seal water connection tap.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP3-2002-05086	Engineering area for improvement identified by WANO team: Long term degradation of the service water system and of the Circulating Water Pump LCI drives has challenged operators. This has been caused, in part, by the lack of comprehensive and aggressive	Loss of material and cracking are aging effects identified in the mechanical tools for carbon steel and stainless steel in raw water.
CR-IP3-2003-00409	34 MSIV Flange is leaking water/steam from its west side and leaking steam (2 feet steam plume) from its east side. The top 2 most west bolts are also leaking water/steam slightly.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2003-00423	During FIN investigation of WRT IP3-03-01697 City Water line 001-JND-6" was found to have a slight leak. After further investigation it was determined that an approx. 30' section of this pipe should be replaced.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in treated water.
CR-IP3-2003-00508	The 31 PAB Heating Coil has developed a leak and was removed from service. The leak was a 3 foot steam plume and filled the area with approximately 3 inches of standing water.	Loss of material and cracking are aging effects identified in the mechanical tools for stainless steel or copper alloys in steam and treated water.
CR-IP3-2003-00556	Brass plug between SWN-123 and PCV-1271-2 galvanically corroded to SS bushing. Problem discovered during performance of 3PT-R185B. Plug needs to be removed and replaced with like sized plug of SS material.	Loss of material is an aging effect identified in the mechanical tools for stainless steel or copper alloys in raw water.
CR-IP3-2003-01071	During a FAC examination (WO I3-010447602, 03-PT-03), wall thinning was noted on an elbow downstream of VCD-PCV-7009 (32A MSR); specifically the elbow downstream of the Westinghouse control section at the entrance to the 31 condenser.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam.
CR-IP3-2003-01176	Continuous Chlorination tank appears to be degrading. Pieces of fiberglass coating found floating in tank. Within the last week the continuous chlorination system became plugged with material.	Loss of material is an aging effect identified in the mechanical tools for carbon steel, stainless steel or plastic in treated water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP3-2003-01186	During inspection of the Sodium Hypochlorite (NaOCl) tank, it was noted that bits of what appeared to be pieces of fiberglass resin was found floating in the tank. It appears that the tank may be degrading and is likely related to the recent clogging.	Loss of material is an aging effect identified in the mechanical tools for carbon steel, stainless steel or plastic in treated water.
CR-IP3-2003-01327	During a FAC examination (WO IP3-02-23675, 03-PT-25), wall thinning was noted on piping downstream of valve 5HD-LCV-1107 (31B MS Drain Tank drain) at the Drains Collecting Tank.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2003-01346	During inspection of 31 EDG east and west air start systems i.e. air start motor, carbon steel pipe, pressure regulators and strainers, rust particles were found in the east strainer cap and a small amount of water was found in the west air regulator.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in treated air.
CR-IP3-2003-01362	Nondestructive examination of Service Water erosion corrosion location IS-19 (line # 1086) identified wall thickness readings below the acceptance criteria (0.135") of work order IP3-02-21094. Two localized areas of degradation below the criteria were identified.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in raw water.
orCR-IP3-2003-01366	The sodium hypochlorite tank was inspected and a three and a half ft crack was found on the interior wall. In addition the fiberglass is delaminating in the area of the crack. A four inch portion of the crack appears to be almost through wall.	Loss of material is an aging effect identified in the mechanical tools for carbon steel, stainless steel or plastic in treated water.
CR-IP3-2003-01927	During a FAC examination (WO IP3-02-24847, EX-02.9-02P), wall thinning was noted on an elbow on the line from the Moisture Separator 1B to the extraction steam header.	Loss of material is an aging effect identified in the mechanical tools for carbon steel steam and treated water.
CR-IP3-2003-02161	Visual inspection of the #35 & #36 main condenser inlet waterbox tubesheets revealed that several of the existing tube plugs were either missing the brass expanding screws or the screws showed signs of corrosion degradation.	Loss of material is an aging effect identified in the mechanical tools for copper alloys in raw water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP3-2003-02298	It appears that there is corrosion on feedwater line #7 at the whip restraint just inside the containment wall, at elevation 57'-6" near column line 10.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in indoor air.
CR-IP3-2003-02319	During a FAC examination, (WO IP3-03-10074, EX-02.2-02T) wall thinning was found on a 10" X 18" tee in the line from the 2A preseparator to the extraction steam header.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2003-03812	While adjusting Batching Tank Aux Steam flow via the PCV Bypass a large spout of water gushed through the wall of the outlet line for Batch Tank Aux. Steam Relief valve. I immediately shut the valve and inspected the area of the leak.	Loss of material and cracking are aging effects identified in the mechanical tools for carbon steel or stainless steel in steam and treated water.
CR-IP3-2003-04048	During heat trace trouble shooting on line DF-1055 (fuel oil supply line to 31 EDG day tank) a pinhole leak was found in the pipe wall approximately 10 inches from check valve DF-15-1.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in fuel oil.
CR-IP3-2003-04266	When removing the PTO for 31 EDG oil lines the cap for the drain valve DF-10-1 could not be put back. The nipple on DF-10-1 is corroded and has to be cut out and a new one welded in.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in fuel oil.
CR-IP3-2003-04873	During the internal tank inspections of #31 Fire Water Storage Tank (FWST) performed on 8/26 and 8/27/03, several areas of localized coating failure and iron nodules with underlying pitting were identified on the tank floor. Many nodules were removed.	Loss of material and cracking are aging effects identified in the mechanical tools for carbon steel in treated water.
CR-IP3-2003-05443	On rounds the nuclear NPO found a through wall hole in the aux steam to boric acid batch tank relief line. The hole is about 1.5 inches long and .5 inch wide and is located directly behind PI-1370.	Loss of material and cracking are aging effects identified in the mechanical tools for carbon steel or stainless steel in steam and treated water.

Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems

Item	Issue	Evaluation
CR-IP3-2003-05491	During the performance of the 5 year inspection (WO IP3-03-14198) of the 32 FWST (FP-T-2), the tank interior was found to exhibit general coatings deterioration and localized failures.	Loss of material and cracking are aging effects identified in the mechanical tools for carbon steel in treated water.
CR-IP3-2004-00179	UH-T-599-8, Condensate Return from Unit Htr HSB-UH-9 Aux. Steam Trap, has a through-wall steam leak with a 7" plume.	Loss of material and cracking are aging effects identified in the mechanical tools for carbon steel or stainless steel in treated water.
CR-IP3-2004-01448	It was reported following inspection that leak rate on SI-733B, 31 Residual Heat Removal Heat Exchanger Discharge Line Relief Valve was 8 ml/min. This leakage is due to a cracked bellows is water from RWST and not RCS. Operability Evaluation 04-09 germane.	Cracking is an aging effect identified in the mechanical tools for stainless steel in treated water.
CR-IP3-2004-01579	During the replacement of 5EX-SOV-1252C found removed valve had excessive erosion and steam cuts to body and seat areas.	Loss of material and cracking are aging effects identified in the mechanical tools for carbon steel or stainless steel in steam and treated water.
CR-IP3-2004-02902	Ecologem watch noticed a crack on the 6" PVC inlet flange to the carbon bed.	Loss of material is an aging effect identified in the mechanical tools for plastic in treated water.
CR-IP3-2004-03378	MW-473, City Water to north loading well hose connection isolation, has a body leak of about one drop per minute and has significant rust and corrosion at the sight of the leakage.	Loss of material is an aging effect identified in the mechanical tools for carbon steel or stainless steel in treated water.
CR-IP3-2004-03540	During the 33 CCW pump PTO removal the pump casing was noted to be leaking.	Loss of material is an aging effect identified in the mechanical tools for gray cast iron in treated water.
CR-IP3-2005-00163	There is an Aux. Steam leak downstream of UH-516 within the confines of Air Handling Unit RS-AH-1 at the far side the heating coils near the floor. The leak is causing multi-level flooding on the 73', 55', and 41' RAMS bidg.	Loss of material and cracking are aging effects identified in the mechanical tools for carbon steel, stainless steel or copper alloys in steam and treated water.

**Table 3.1.1 Operating Experience Applicable to Non-Class 1 Mechanical Systems**

Item	Issue	Evaluation
CR-IP3-2005-00235	During the installation of temp indication for the power uprate, found the threads on MS-287, Moisture Preseparator 1B Test Connection severely corroded with the last two threads on the pipe connection completely gone.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2005-00613	During 3R13P FAC UT inspection of the 10" X 14" expander downstream of valve 5HD-LCV-1127B, (Heater Drain Tank Bypass to Condenser 33 LCV), wall thinning was detected below the administrative screening criteria of 70% of the nominal wall thickness (0.175").	Loss of material is an aging effect identified in the mechanical tools for carbon steel in steam and treated water.
CR-IP3-2005-01101	During inspection of 31 main turbine lube oil cooler. The outlet pipe flange face has 4 areas of crevice corrosion.	Loss of material is an aging effect identified in the mechanical tools for carbon steel, stainless steel or copper alloys in lube oil, raw water or treated water.
CR-IP3-2005-01366	The tube bundle in the Main Boiler Feed Pump Lube Oil Cooler #32 is severely degraded due to corrosion pitting per eddy current inspection (see iTi Report No. PR No. 32-134, dated 3-21-05). The vendor recommended tube bundle replacement.	Loss of material is an aging effect identified in the mechanical tools for carbon steel, stainless steel or copper alloys in lube oil, raw water or treated water.
CR-IP3-2005-03088	During inspection of the fuel oil supply pipe to 32 EDG it was discovered that line 1053 had wall thickness loss in multiple areas of up to 0.056" due to corrosion.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in fuel oil.
CR-IP3-2005-05466	Inspection of the 31 FCU HX waterbox shows that the previously identified deterioration of the cover plates by crevice corrosion has progressed to the point that repairs are necessary to seal the waterboxes.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water or treated water.
CR-IP3-2005-05558	Inspection of the 32 FCU HX waterbox shows widespread pitting corrosion of cover plates to the point that repairs are necessary to seal the waterboxes.	Loss of material is an aging effect identified in the mechanical tools for carbon steel in raw water or treated water.
CR-IP3-2005-05832	Steam leak from flange on SW side of Magnatrol, leak is audible, and visually verified, hi level alarm is not in.	Loss of material and cracking are aging effects identified in the mechanical tools for carbon steel and stainless steel in steam or treated water.

Riverkeeper Opposition to Entergy's Motion For Summary Disposition of  
Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion)

# Riverkeeper TC-2: Attachment 5



**Originator:** MALONE, HAZEL

**Originator Phone:** 0

**Originator Site Group:** IP2 ENG P&C-Code Programs Staff

**Operability Required:** Y

**Supervisor Name:** SCHWARTZ, GEOFFREY

**Reportability Required:** N

**Discovered Date:** 10/31/2001 00:00

**Initiated Date:** 10/31/2001 00:00

**Condition Description:**

CR Date: 10/31/2001 12:33

CR Entered Date: 10/31/2001 14:21

UT inspections were performed on sections of Crossunder piping as the result of a pinhole leak found on the MSR21A inlet piping during the cycle (see CR# 200103681). Areas on the expansion joints and piping upstream of MSR21A show measured thickness below or close to allowable minimum wall (0.247") based on UT results taken during the mid-cycle outage. The cause of these thinned areas is believed to be Flow Accelerated Corrosion (FAC). It is recommended that these areas be repaired at this time. Design Engineering has been notified and temporary repair of thinned areas are being performed. See drawings attached for location of thinned area and the measured thickness readings of these areas.

**Immediate Action Description:**

Evaluated thinned areas and worked with Design Engineering to develop temporary repair for degraded areas.

**Suggested Action Description:**

**EQUIPMENT:**

<u>Tag Name</u>	<u>Tag Suffix Name</u>	<u>Component Code</u>	<u>Process System Code</u>
21AMSR			MS

**REFERENCE ITEMS:**

<u>Type Code</u>	<u>Item Desc</u>
CR	200103681
CR	200103681
CR	200110521
CR	200110526
DETECTION	SI
LOCATION	Turbine
WON	01-23886
WON	01-23886 Y
WON	01-24370
WON	01-24370 Y

**TRENDING (For Reference Purposes Only):**

<u>Trend Type</u>	<u>Trend Code</u>
PR	PR-CORROSION & EROSION (ALL WALL-THINNING PROCESSE)
OR	OR-DESIGN ENGINEERING
EQ	EQ-AH
KA	KA-IN

CA Number: 1

	Site	Group	Name
Assigned By:	IP2	CA&A Staff	E-CAPTAIN, CRS
Assigned To:	IP2	ENG P&C-Code Programs Mgmt	Azevedo,Nelson F
Subassigned To :	IP2	ENG P&C-Code Programs Staff	MALONE, HAZEL

Originated By: E-CAPTAIN, CRS 11/1/2001 00:00:00  
Performed By: Azevedo,Nelson F 11/29/2001 00:00:00  
Subperformed By: MALONE, HAZEL 11/26/2001 00:00:00  
Approved By:  
Closed By: E-CAPTAIN, CRS 11/29/2001 00:00:00

Current Due Date: 12/01/2001 Initial Due Date: 12/01/2001

CA Type: DISP - CORR ACTION

Plant Constraint: NONE

**CA Description:**

Please evaluate to determine apparent cause and recommend corrective actions. (cbh)

**CA REFERENCE ITEMS:**

<u>Type Code</u>	<u>Description</u>
CRS ID	254719

Response:

**Subresponse :**

11/09/2001 Assigned to: MALONE, HAZEL Status: Closed + Approved

Action Requested: Hazel, please evaluate this SL2 and provide corrective actions as required. Nelson  
Assignee Response: See SL Report

**Reviewer Comment:**

Although this CR was classified as an SL2, the wall thinning detected during the mid-cycle inspection was a result of Flow Accelerated Corrosion (FAC) which is a well understood degradation mechanism and it is well modeled in industry and in IP2 FAC analysis. No causal factors or any other evaluation was performed on this CR because it was considered provide no additional value to this already, well understood phenomena. Based on this, it is requested that this CR be downgraded to an SL3 and closed as such.

Nelson Azevedo

11/29/01

**Significance Level 3 Report**

UT inspections were performed on the 26.5" ID vertical riser section of crossunder piping leading to MSR21A as the result of a pinhole leak which occurred in this section during the cycle (See CR# 200103681). Results of the UT inspections performed found an additional thinned area on the same expansion joint containing the pinhole leak and a thinned area on a pup piece adjacent to that expansion joint (See attachment for locations of thinned areas). The thinned areas found measured below or close to allowable minimum wall thickness (See wear rate/structural evaluation for each component inspected for details).

Cause of wear is believed to be the result of Flow Accelerated Corrosion (FAC). FAC is the process whereby the protective oxide layer on carbon steel piping is dissolved by flowing water or wet steam which results in the wearing away of the underlying metal. Main Steam exhausted from the High Pressure Turbine enters the Crossunder piping as a wet steam, that can contain as much as 20% moisture. This wet steam mixture combined with the high fluid velocity and high temperature of this piping system can result in extremely high wear rates and establishes Crossunder piping as one of the most susceptible systems to FAC. Though the IP2 FAC program history shows that all crossunder piping was completely weld overlaid with Stainless Steel to prevent wear, certain areas near expansion joints were not weld overlaid due to restrictions on welding near the expansion joints links (dogbones). Certain areas near these expansion joint dogbones may have been consequently left as Carbon Steel due to these restrictions and are still vulnerable to the effects of FAC. The leak that occurred during the cycle 15, as well as the thinned areas found during the mid-cycle outage were all located in the vicinity of the expansion joints. Thinned areas located during the mid cycle inspection will be temporarily repaired externally per WO# 01-23886 until they can be visually inspected internally during the 2002 refueling outage and permanently repaired. Internal inspections are recommended to determine the location of the expansion dogbones and also to locate areas not weld overlaid that may need repair. Internal inspections of crossunder piping will be added to the scope of the 2002 refueling outage as part of the FAC outage inspection scope (see FAC Master Inspection List for details) and may include inspection of parallel trains or similar expansion joint areas.

Permanent repair of thinned areas will be performed under work order 01-24370. Methods of permanent repair as well as expansion joint replacement options should be researched and planned for accordingly prior to the outage to ensure proper scheduling.

-----  
Thinned areas located during the mid cycle inspection will be temporarily repaired externally per WO# 01-23886 until they can be visually inspected internally during the 2002 refueling outage and permanently repaired. Internal inspections are recommended to determine the location of the --- see attachment for rest ---

**Closure Comments:**

reject per CAG quality review

**Attachments:**

Subresp Description

11/09/2001 Assigned to: MA

CA Number: 2

	Site	Group	Name
Assigned By:	IP2	CA&A Staff	E-CAPTAIN, CRS
Assigned To:	IP2	ENG P&C-Code Programs Mgmt	Azevedo,Nelson F

Subassigned To :

Originated By: E-CAPTAIN, CRS 12/4/2001 00:00:00

Performed By: Azevedo,Nelson F 12/4/2001 00:00:00

Subperformed By:

Approved By:

Closed By: Azevedo,Nelson F 12/4/2001 00:00:00

Current Due Date: 12/06/2001

Initial Due Date: 12/06/2001

CA Type: DISP - CORR ACTION

Plant Constraint: NONE

**CA Description:**

reject per CAG quality review

Downgraded from SL2 to SL3 per Joe Barlok. Due date changed to 12/6/01...12/4/01 MK

**CA REFERENCE ITEMS:**

Type Code

CRS ID

Description

260983

**Response:**

## Significance Level 3 Report

UT inspections were performed on the 26.5" ID vertical riser section of crossunder piping leading to MSR21A as the result of a pinhole leak which occurred in this section during the cycle (See CR# 200103681). Results of the UT inspections performed found an additional thinned area on the same expansion joint containing the pinhole leak and a thinned area on a pup piece adjacent to that expansion joint (See attachment for locations of thinned areas). The thinned areas found measured below or close to allowable minimum wall thickness (See wear rate/structural evaluation for each component inspected for details).

Cause of wear is believed to be the result of Flow Accelerated Corrosion (FAC). FAC is the process whereby the protective oxide layer on carbon steel piping is dissolved by flowing water or wet steam which results in the wearing away of the underlying metal. Main Steam exhausted from the High Pressure Turbine enters the Crossunder piping as a wet steam, that can contain as much as 20% moisture. This wet steam mixture combined with the high fluid velocity and high temperature of this piping system can result in extremely high wear rates and establishes Crossunder piping as one of the most susceptible systems to FAC. Though the IP2 FAC program history shows that all crossunder piping was completely weld overlaid with Stainless Steel to prevent wear, certain areas near expansion joints were not weld overlaid due to restrictions on welding near the expansion joints links (dogbones). Certain areas near these expansion joint dogbones may have been consequently left as Carbon Steel due to these restrictions and are still vulnerable to the effects of FAC. The leak that occurred during the cycle 15, as well as the thinned areas found during the mid-cycle outage were all located in the vicinity of the expansion joints. Thinned areas located during the mid cycle inspection will be temporarily repaired externally per WO# 01-23886 until they can be visually inspected internally during the 2002 refueling outage and permanently repaired. Internal inspections are recommended to determine the location of the expansion dogbones and also to locate areas not weld overlaid that may need repair. Internal inspections of crossunder piping will be added to the scope of the 2002 refueling outage as part of the FAC outage inspection scope (see FAC Master Inspection List for details) and may include inspection of parallel trains or similar expansion joint areas.

Permanent repair of thinned areas will be performed under work order 01-24370. Methods of permanent repair as well as expansion joint replacement options should be researched and planned for accordingly prior to the outage to ensure proper scheduling.

-----

Thinned areas located during the mid cycle inspection will be temporarily repaired externally per WO# 01-23886 until they can be visually inspected internally during the 2002 refueling outage and permanently repaired. Internal inspections are recommended to determine the location of the expansion dogbones and also to locate areas not weld overlaid that may need repair. Internal inspections of crossunder piping will be added to the scope of the 2002 refueling outage as part of the FAC outage inspection scope (see FAC Master Inspection List for details) and may include inspection of parallel trains or similar expansion joint areas.

Permanent repair of thinned areas will be performed under work order 01-24370. Methods of permanent repair as well as expansion joint replacement options should be researched and planned for accordingly prior to the outage to ensure proper scheduling.

**Subresponse :****Closure Comments:**

N/A

CA Number: 3

	Site	Group	Name
Assigned By:	IP2	CA&A Staff	E-CAPTAIN, CRS
Assigned To:	IP2	ENG SYS-Balance of Plant Staff	Ray,Bryan J

Subassigned To :

Originated By: E-CAPTAIN, CRS 11/1/2001 00:00:00

Performed By: Ray,Bryan J 11/3/2001 00:00:00

Subperformed By:

Approved By:

Closed By: Ray,Bryan J 11/3/2001 00:00:00

Current Due Date: 11/08/2001

Initial Due Date: 11/08/2001

CA Type: CRS - FYI

Plant Constraint: NONE

CA Description:

For your information on equipment with your system.

CA REFERENCE ITEMS:

Type Code	Description
CRS ID	254720

Response:

Subresponse :

Closure Comments:

N/A

CA Number: 4

	Site	Group	Name
Assigned By:	IP2	CA&A Staff	E-CAPTAIN, CRS
Assigned To:	IP2	ENG P&C-Code Programs Mgmt	Azevedo,Nelson F

Subassigned To :

**Originated By:** E-CAPTAIN, CRS 12/4/2001 00:00:00  
**Performed By:** Azevedo,Nelson F 2/1/2002 00:00:00  
**Subperformed By:**  
**Approved By:**  
**Closed By:** Azevedo,Nelson F 2/1/2002 00:00:00

Current Due Date: 02/17/2002

Initial Due Date: 02/17/2002

CA Type: CR CLOSURE REVIEW

Plant Constraint: NONE

CA Description:

Follow up on corrective action assignments

CA REFERENCE ITEMS:

Type Code	Description
CRS ID	261193

Response:

Since the corresponding ICAs have been adequately implemented, this CR is ready for closure.

Subresponse :

Closure Comments:

N/A

CA Number: 5

	Site	Group	Name
Assigned By:	IP2	ENG P&C-Code Programs Mgmt	Azevedo,Nelson F
Assigned To:	IP2	ENG P&C-Code Programs Staff	MALONE, HAZEL

## Subassigned To :

Originated By: Azevedo,Nelson F 12/4/2001 00:00:00

Performed By: MALONE, HAZEL 1/18/2002 00:00:00

## Subperformed By:

## Approved By:

Closed By: Azevedo,Nelson F 1/18/2002 00:00:00

Current Due Date: 01/18/2002

Initial Due Date: 01/18/2002

CA Type: PERFORM CA

Plant Constraint: NONE

## CA Description:

Add internal inspection of MSR vertical risers to FAC Master Inspection List (MIL) for 2002 refueling outage.

## CA REFERENCE ITEMS:

Type Code	Description
CRS CLASS	1
CRS ID	261191

## Response:

Internal inspection of 21A&amp;B, 22A&amp;B and 23A&amp;B MSR vertical risers will be added to FAC Master Inspection List (MIL) for 2002 refueling outage. Final MIL is due for release 1/31/2002

## Subresponse :

## Closure Comments:

Please reflect the fact that the inspection locations have been added to the inspection list even though the final list will not be issued until 1/31/02.



CA Number: 6

	Site	Group	Name
Assigned By:	IP2	ENG P&C-Code Programs Mgmt	Azevedo,Nelson F
Assigned To:	IP2	ENG P&C-Code Programs Staff	MALONE, HAZEL

Subassigned To :

Originated By: Azevedo,Nelson F 12/4/2001 00:00:00

Performed By: MALONE, HAZEL 1/18/2002 00:00:00

Subperformed By:

Approved By:

Closed By: Azevedo,Nelson F 1/18/2002 00:00:00

Current Due Date: 01/18/2002

Initial Due Date: 01/18/2002

CA Type: PERFORM CA

Plant Constraint: NONE

CA Description:

Research methods of permanent repairs for previously temporary repaired areas of crossunder piping. Also, research the cost and resource information for the replacement of crossunder expansion joints to determine if this option is practicable.

CA REFERENCE ITEMS:

Type Code	Description
CRS CLASS	I
CRS ID	261192

**Response:**

There have been many different methods used to permanently repair these areas of crossunder piping. The following are descriptions of these methods:

**1. Welding under expansion dogbones**

One option of repair that has been performed at plants such as Surry is to remove the expansion dogbones, perform Stainless steel cladding to expose piping underneath and replace dogbones. Advantages to this repair method are that the entire area would be protected (including area under dogbones) and thinned areas of piping can be brought back up to nominal thickness by the overlay. Disadvantages to this repair is that it is not a recommended repair method of Westinghouse because it may jeopardize the flexibility of the expansion joint and that extra time will be needed to perform the engineering to analyze this concern.

**2. Stainless Steel Tubing Repair**

This repair option was used at St. Lucie Unit 1 & 2 to repair eroded area under dogbones. A stainless steel piece of tubing was placed next to the dogbones and welded in place. The tubing was then deformed to ensure a secure fit up to the dogbone (see attachment for details). Advantages to this repair are that there is no welding to the dogbone and no arc strikes on the dogbone as recommended by Westinghouse. Disadvantage to this repair is that if there is not a consistent bond between the tubing and the dogbone, steam can still get into that area and erode the piping underneath.

**3. Stainless Steel Covering over dogbones**

During the last CHECWORKS Users Group (CHUG) meeting (January 14&15, 2002), it was mentioned the Point Beach has welded stainless steel covers over the dogbones in their crossunder piping. At this time, Point Beach FAC engineer has provided no additional information.

**4. Welding on Outside of crossunder piping**

A 1995 letter to Westinghouse on the subject of repairing eroded areas of crossunder piping from the outside yield the following response:

These eroded areas beneath the expansion joint link (dogbones) can be repaired from the outside of the pipe. The eroded area beneath the link should be ground out and weld repaired using 309 stainless as per PS 600374 for the first few passes to provide an erosion resistant inner surface. The remaining cavity should be built-up with carbon steel weld material as per PS 600945 Part 1-1-1-B. Post weld heat treatment should not be performed for any of the above welding processes.

The expansion joint link should not be subjected to any arc strikes during the welding process. It is recommended that a copper backing plate be placed between the link and the area of repair to protect the link during welding.

A carbon steel backing plate can be welded over the repaired area on the outside diameter of the pipe, if desired. This welding should be performed to PS 600945 1-1-1-B. No PWHT is to be performed.

Advantages of this repair are that it is a recommended repair from the expansion joint vendor (Westinghouse) and that it would restore original pipe thickness. Would need to obtain welding procedure from Westinghouse if similar procedure is not available.

Replacement of Expansion Joint sections (21A & 21B Vertical Risers) was also investigated for this RES. At the time of this response, Westinghouse (Siemens) had not replied with an estimate for replacing both 21A and 21B vertical risers. The following information was provided from proposals of IP3 1997 Crossunder replacement of their 31A and 31B vertical risers and IP2 1995 Crossunder replacement of 32" expansion joints under the HP turbine.

**IP2 32" Expansion Joints under HP Turbine (1995)**

- o Material cost - \$24,160 (8, 32" Plates of 1-1/4 Chrome)
- o Heat Treatment of 32" Expansion Joints - \$17,086.76 (15 days to perform)

**IP3 31A & 31B Vertical Riser Replacement (1997)**

- o Lump sum quote for design/engineering of all MSR vertical risers and --- see attachment for rest ---

**Subresponse :****Closure Comments:**

N/A

**Attachments:**

Resp Description

There have been many different

CA Number: 7

	Site	Group	Name
Assigned By:	IP2	ENG P&C-Code Programs Mgmt	Azevedo,Nelson F
Assigned To:	IP2	ENG P&C-Code Programs Staff	MALONE, HAZEL

Subassigned To :

Originated By: Azevedo,Nelson F 1/18/2002 00:00:00  
 Performed By: MALONE, HAZEL 1/18/2002 00:00:00

Subperformed By:

Approved By:

Closed By: Azevedo,Nelson F 1/18/2002 00:00:00

Current Due Date: 01/18/2002 Initial Due Date: 01/18/2002

CA Type: PERFORM CA

Plant Constraint: NONE

**CA Description:**

Please reflect the fact that the inspection locations have been added to the inspection list even though the final list will not be issued until 1/31/02.

**CA REFERENCE ITEMS:**

Type Code	Description
CRS CLASS	1
CRS ID	270378

**Response:**

Internal inspections of 21A&B, 22A&B and 23A&B MSR vertical risers have been added to FAC Master Inspection List (MIL) for the 2002 refueling outage. The final MIL is due for release 1/31/2002

Subresponse :

**Closure Comments:**

N/A

**Originator:** Lizzo, Nicholas

**Originator Phone:** 8277

**Originator Group:** Operations Watch Mgmt

**Operability Required:** N

**Supervisor Name:** Cramer, Thomas A

**Reportability Required:** N

**Discovered Date:** 07/23/2006 04:49

**Initiated Date:** 07/23/2006 04:58

**Condition Description:**

The line downstream of the 1104 valves ("A" reheaters) to 36 FWH shell side is leaking approximately 1/2 gpm. WRT IP3-06-18192. Rapid Response activated.

**Immediate Action Description:**

SM notified - rapid response activated

**Suggested Action Description:**

Repair.

**EQUIPMENT:**

<u>Tag Name</u>	<u>Tag Suffix Name</u>	<u>Component Code</u>	<u>Process</u>	<u>System Code</u>
1104				HD

**REFERENCE ITEMS:**

<u>Type Code</u>	<u>Description</u>
#LEVEL OF DEFENSE	Observation
KEYWORDS	leak
TEAM 3C	tt
WON	IP3-06-18192

**TRENDING (For Reference Purposes Only):**

<u>Trend Type</u>	<u>Trend Code</u>
REPORT WEIGHT	4
EM	MAMM
KEYWORDS	KW-LEAKS-WATER
	EF11

CA Number: 1

**Group****Name**

Assigned By: CRG/CARB/OSRC

Harrison,Christine B

Assigned To: P&amp;C Eng Codes Mgmt

Azevedo,Nelson F

Subassigned To: P&amp;C Eng Codes Staff

Hartjen,Harry G

Originated By: Harrison,Christine B

7/25/2006 11:27:08

Performed By: Azevedo,Nelson F

8/15/2006 14:22:53

Subperformed By: Hartjen,Harry G

8/10/2006 09:27:02

Approved By:

Closed By: Azevedo,Nelson F

8/15/2006 14:22:53

Current Due Date: 08/16/2006

Initial Due Date: 08/16/2006

CA Type: DISP - CA

Plant Constraint: #NONE

**CA Description:**

Please review and assign further corrective actions as required.

**Response:**

UT thickness measurements were taken around the leaking area on this pipe (UT Report 06UT171). The thinned area was found to be localized on one side of the pipe which is downstream of a tee. A pipe clamp (Team Inc.) was installed over the leak and thinned area under work order IP3-06-18192. Leak repair was successful and the leak has stopped. CA 00002 has been issued to review the FAC program and determine if additional inspections are warranted as a result of this leak.

**Subresponse :**

Issued CA#2

**Closure Comments:**

CA Number: 2

**Group**

**Name**

Assigned By: P&C Eng Codes Staff

Hartjen, Harry G

Assigned To: P&C Eng Codes Staff

Hartjen, Harry G

**Subassigned To :**

Originated By: Hartjen, Harry G

8/10/2006 09:24:00

Performed By: Hartjen, Harry G

10/30/2006 11:28:34

**Subperformed By:**

Approved By:

Closed By: Hartjen, Harry G

10/30/2006 11:28:34

**Current Due Date:** 10/31/2006

**Initial Due Date:** 10/31/2006

**CA Type:** ACTION

**Plant Constraint:** #NONE

**CA Description:**

Review both Unit 2 and Unit 3 FAC Programs and determine if similar locations to this leak have been inspected for wall thinning. Also determine if and when additional inspections are required to determine if wall thinning is occurring at these similar locations. If additional inspections are required, incorporate inspections into work week schedule, or FAC Program outage scopes.

**Response:**

A review of both Unit 2 and Unit 3 FAC programs was performed to determine if similar locations to the current Reheater Drain line leak have been inspected for wall thinning and determine if and when additional inspections are required to determine if wall thinning is occurring at these similar locations.

Review of the Unit 2 FAC inspection history found that all similar locations have been recently inspected or replaced. No additional inspections are recommended for Unit 2 at this time.

Review of the Unit 3 FAC inspection history found that some similar locations do not have recent inspections and should be inspected. A total of 9 additional inspections were identified.

Details of this review, along with the recommended additional inspections are attached.

No on-line exams are recommended due to the high temperature (385 degrees F) of these components, and the congestion of piping in these areas. Therefore the 9 additional inspections are to be performed during the 3R14 refuel outage.

CA 3 is issued to generate work orders and 3R14 scope add forms for these inspections.

**Subresponse :**

**Closure Comments:**

**Attachments:**

Resp Description

Review of FAC Inspections due to leak in U3 RHD

Riverkeeper Opposition to Entergy's Motion For Summary Disposition of  
Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion)

# Riverkeeper TC-2: Attachment 6



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UNITED STATES OF AMERICA  
NUCLEAR REGULATORY COMMISSION

+ + + + +

565TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

THURSDAY,

SEPTEMBER 10, 2009

+ + + + +

ROCKVILLE, MARYLAND

The Advisory Committee met at the Nuclear  
Regulatory Commission, One White Flint North,  
Commissioner's Conference Room, 11555 Rockville Pike,  
at 8:30 a.m., Dr. Mario V. Bonaca, Chairman,  
presiding.

COMMITTEE MEMBERS:

- MARIO V. BONACA, Chairman
- SAID ABDEL-KHALIK, Vice Chairman
- GEORGE E. APOSTOLAKIS, Member
- J. SAM ARMIJO, Member-at-Large
- SANJOY BANERJEE, Member
- CHARLES H. BROWN, Member
- MICHAEL L. CORRADINI, Member
- OTTO L. MAYNARD, Member

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COMMITTEE MEMBERS (Continued):

DANA A. POWERS, Member

HAROLD B. RAY, Member

MICHAEL T. RYAN, Member

WILLIAM J. SHACK, Member

JOHN D. SIEBER, Member

JOHN W. STETKAR, Member

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Adjourn

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1 Thank you.

2 MS. GREEN: I would like to move on to the  
3 flow-accelerated corrosion program and the operating  
4 experience.

5 During the ACRS Subcommittee meeting in  
6 March, an ACRS member questioned why the inspection  
7 frequency did not change for instances where the  
8 minimum measured wall thickness was near or below  
9 minimum acceptable wall thickness. At that time, the  
10 staff did not answer the ACRS member's question. So I  
11 would like to try to address that now.

12 During the audit, the staff questioned the  
13 applicant about the incidences of wall thinning that  
14 were reported in the license renewal application.  
15 Specifically, there was an IP3 vent chamber drain  
16 piping, IP3 high-pressure turbine drain piping. There  
17 is a 2-inch diameter line and a three-quarter-inch  
18 diameter line, and the IP2 steam trap piping. These  
19 were, I think, the four cases that the ACRS member was  
20 referring to in the staff's audit report.

21 In response to the audit question, as well  
22 as a few others that were related to the flow-  
23 accelerated corrosion program, the applicant stated  
24 that the piping and affected components were included  
25 in the flow-accelerated corrosion program prior to the

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1 inspections. As the wall thinning of these components  
2 was discovered, the applicant replaced the components  
3 with like-for-like materials or FAC-resistant  
4 materials.

5 The applicant also stated that, if a  
6 component is discovered that has a current or  
7 projected wall thickness less than the minimum  
8 acceptable wall thickness, then additional inspections  
9 of identical or similar piping components in a  
10 parallel or alternate train is performed to bound the  
11 extent of thinning. When the inspections of  
12 components detects significant wall thinning, then the  
13 sample size for that line is increased.

14 One of the examples I would like to talk  
15 about to explain this is the IP3 vent chamber  
16 drainpipe thinning. During the refueling outage '13,  
17 Entergy did an inspection of an elbow immediately  
18 downstream of the moisture separator reheater and  
19 found wall thinning less than the minimum acceptable  
20 wall thinness, requiring replacement of the elbow.

21 Based on the results of that inspection,  
22 the applicant performed a sample expansion to  
23 determine the extent of condition for this pipe  
24 thinning. The expansion included corresponding  
25 components on the other moisture separator reheaters

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1 with a configuration similar to that of the elbow  
2 displaying the thinning.

3 Energy then performed four additional  
4 inspections. These inspections also found wall  
5 thinning less than the minimum acceptable thickness  
6 requiring replacement of the components.

7 The sample expansion was continued until  
8 no additional components were detected with  
9 significant wear. Energy performed four additional  
10 inspections downstream of the worn elbows. The  
11 results of this expansion did not find significant  
12 wear, and the sample expansion was then terminated by  
13 Energy. The applicant updated and adjusted the  
14 Checkworks model to incorporate the inspection data.

15 MEMBER BROWN: Before you go on, I guess I  
16 asked that question. So I will ask it again.

17 I'm trying to draw a conclusion from your  
18 answer that, No. 1, they replaced them with more  
19 erosion-resistant or flow-accelerated corrosion-  
20 resistant materials when they did the replacements.  
21 Is that correct?

22 MS. GREEN: For that particular line, they  
23 were planning to replace with Chrome-Moly, but for  
24 other lines --

25 MEMBER BROWN: That doesn't mean anything;

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1 I'm not a metallurgist. Is it better or worse?

2 MS. GREEN: It's better.

3 MEMBER BROWN: Okay. Thank you.

4 MS. GREEN: Sorry.

5 That is more FAC-resistant. For other  
6 lines, they did a replacement of like-for-like  
7 material.

8 MEMBER BROWN: Okay. The second question  
9 was they had found the wall thicknesses considerably  
10 less. There were a number of other locations also  
11 that had less than the minimum acceptable wall  
12 thickness.

13 So the second part of the question about,  
14 if they just did it like-for-like, what do you do to  
15 your inspection process to make sure you don't  
16 encounter a circumstance that you now find you've got  
17 less than minimum wall thickness again, which means  
18 increased frequency? That part I didn't understand  
19 the answer. Or was there an answer?

20 MS. GREEN: I am not a flow-accelerated  
21 corrosion program expert. So I would have to ask Matt  
22 Yoder from the staff to address your question.

23 MEMBER MAYNARD: I believe we have  
24 somebody coming to answer that.

25 We need a portable microphone, I believe.

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1 MR. YODER: Okay, Matt Yoder, NRR staff.

2 So, when these instances were found, the  
3 data is then fed back into your Checkworks model. So  
4 that, for future planning of inspections and UT, your  
5 model is going to predict a greater wear rate at those  
6 locations, and it should then be scheduled for more  
7 frequent UT inspection.

8 MEMBER BROWN: Okay. So there was an  
9 explanation of the Checkworks thing in, I think, the  
10 applicant's answer back, which I read, not being a  
11 Checkworks expert.

12 So the point being that the information of  
13 the increased wear rate is then fed back into this  
14 model, so that it gets into a periodic inspection that  
15 is more frequent than before? It is not like you go  
16 change a chart somewhere, but you do it based on the  
17 predictions of the model?

18 MR. YODER: That is correct. The model is  
19 continuously updated with actual field data.

20 MEMBER BROWN: Okay. All right, thank  
21 you.

22 MEMBER SHACK: How long has the Checkworks  
23 program been in place at Indian Point?

24 MR. YODER: I will have to defer to  
25 Entergy.

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WASHINGTON, D.C. 20005-3701



1 MR. AZEVEDO: My name is Nelson Azevedo.  
2 I'm the Supervisor of Programs at Indian Point.

3 We first started using the Checkworks  
4 models when it was first issued by EPRI, which I  
5 believe was the early nineties. I don't know the  
6 exact date.

7 MEMBER SHACK: It hasn't reached steady-  
8 state yet?

9 MEMBER MAYNARD: Okay, let's go.

10 MS. GREEN: Okay. I would just like to  
11 cover briefly the staff's evaluation of the  
12 applicant's flow-accelerated corrosion program.

13 In the license renewal application, the  
14 applicant stated that its flow-accelerated corrosion  
15 program is consistent with the GALL AMP XI.M17 with  
16 one exception, that exception being the use of EPRI  
17 NSAC-202L, Revision 3, in lieu of Revision 2, which is  
18 recommended in the GALL report. The staff reviewed  
19 the exception and found that the use of Revision 3 is  
20 acceptable.

21 Based on the staff's audit and review, it  
22 determined that all other program elements are  
23 consistent with the GALL report AMP.

24 The applicant's program includes updated  
25 inputs for the power operating parameter changes with

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1 steam flow rates and temperatures and such. It also  
2 identified piping systems and components that are  
3 currently the most susceptible to the loss of material  
4 due to FAC.

5 Corrective actions that are in place  
6 include re-evaluation, repair, or replacement. Based  
7 on the review of the applicant's program, the staff  
8 concluded that it is adequate to manage the effects of  
9 aging, and therefore, acceptable.

10 During the March ACRS Subcommittee, ACRS  
11 Member Brown asked the staff to explain the various  
12 criteria for Charpy upper-shelf energy. At the time,  
13 the staff did not provide a full explanation, and  
14 therefore, Chairman Maynard asked us to provide an  
15 explanation of the criteria, which I will attempt to  
16 do now.

17 10 CFR 50, Appendix G, requires that  
18 reactor vessels must maintain Charpy upper-shelf  
19 energy values of no less than 50-foot pounds, unless  
20 it can be demonstrated that lower values of upper-  
21 shelf energy will provide margins of safety against  
22 fracture equivalent to those required by Appendix G of  
23 Section 11 of the ASME Code.

24 Appendix K of the ASME Code, Section 11,  
25 and ASME Code Case N-512 provide criteria for

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Riverkeeper Opposition to Entergy's Motion For Summary Disposition of  
Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion)

# Riverkeeper TC-2: Attachment 7

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Indian Point Unit 3 CHECWORKS FAC Model, Calculation No. 050714c-01, Rev. 0 (Oct. 25, 2005)	IP3 RO13 (2005)	Figure 10 Figure 11 Figure 12
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**Indian Point Unit 2  
CHECWORKS FAC Model**

**Calculation No. 050714b-01  
Revision 0  
Issued For-Use**

**July 5, 2005**

*prepared for:*

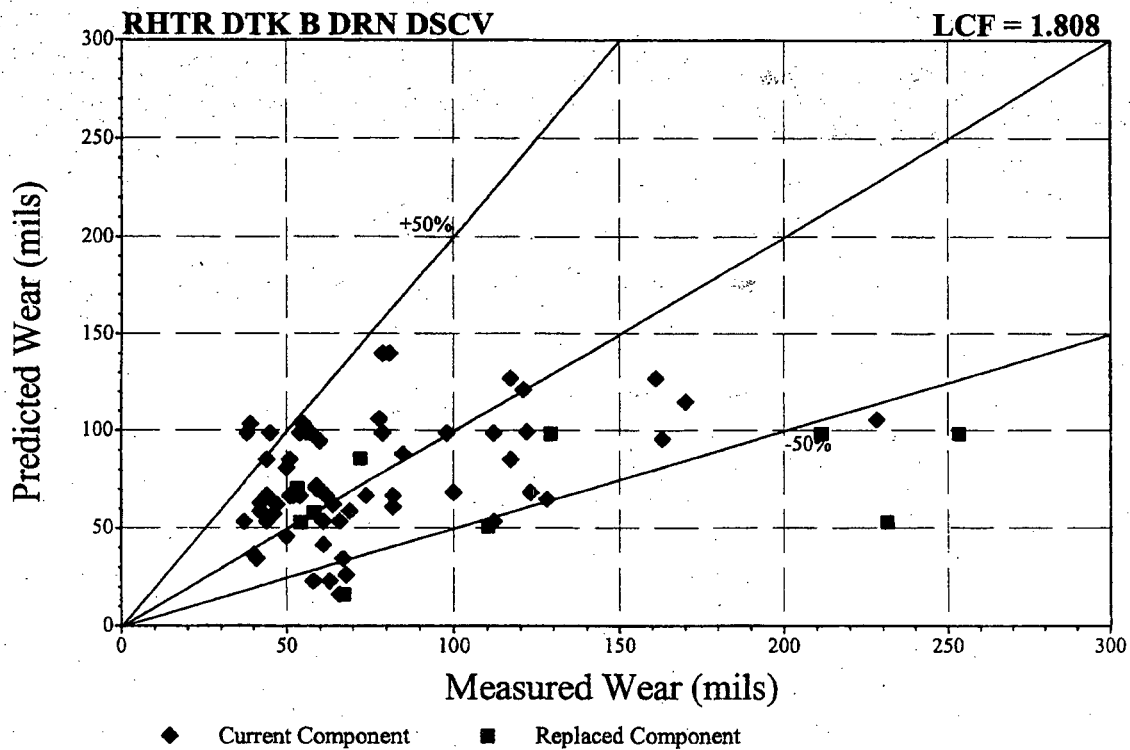
**Entergy Nuclear Northeast  
295 Broadway Suite 3  
PO Box 308  
Buchanan, NY 10511-0308**

*prepared by:*

**CSI TECHNOLOGIES, INC.  
1051 E. Main St., Suite 215  
East Dundee, IL 60118**

FIGURE 3

### Comparison of Wear Predictions



**Indian Point Unit 2  
CHECWORKS FAC Model**

**Calculation No. 050714b-01  
Revision 1  
Issued For-Use**

**September 12, 2006**

*prepared for:*

**Entergy Nuclear Northeast  
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PO Box 308  
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*prepared by:*

**CSI TECHNOLOGIES, INC.  
1051 E. Main St., Suite 215  
East Dundee, IL 60118**

FIGURE 4

### Comparison of Wear Predictions

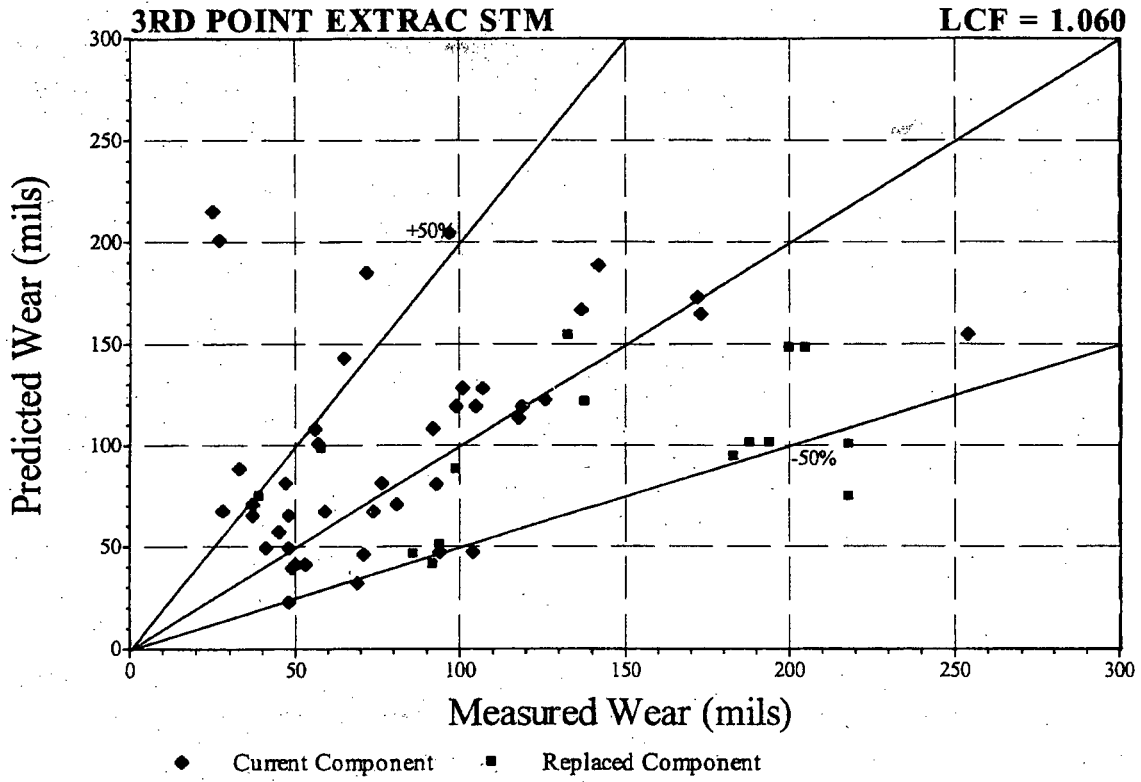
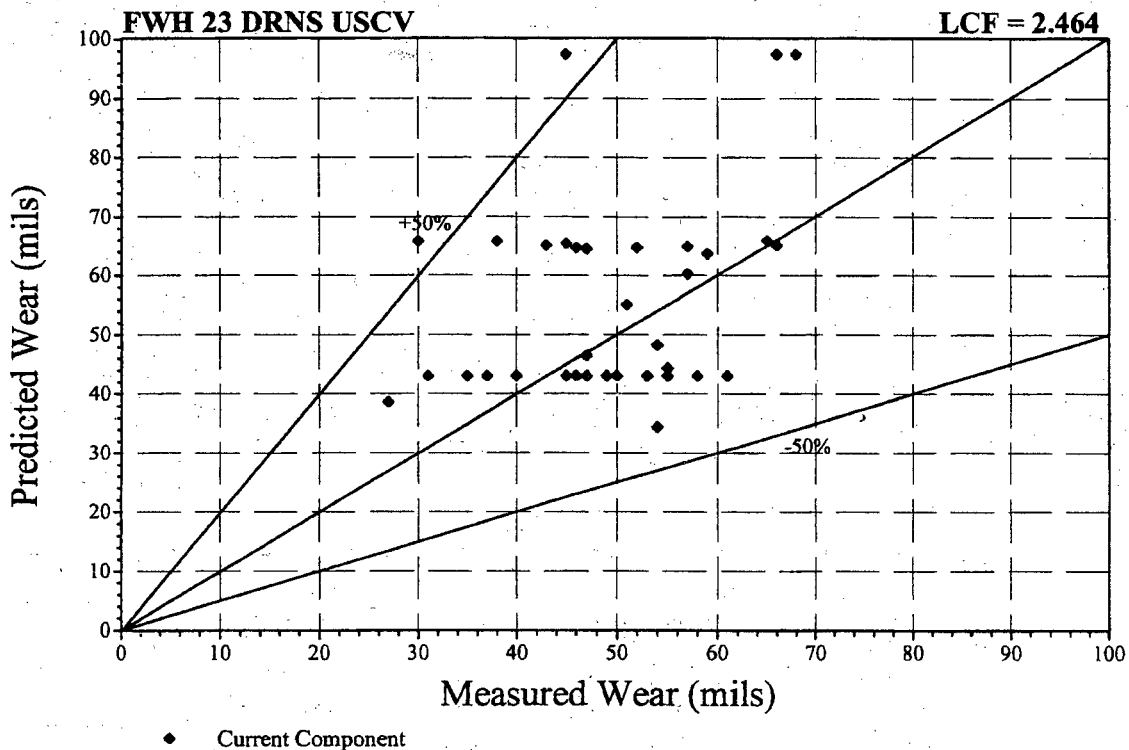




FIGURE 5

### Comparison of Wear Predictions



**Indian Point Unit 2  
CHECWORKS SFA Model**

**CSI Calculation No. 0705.101-01  
Revision A  
Issued For-Use**

**November 17, 2008**

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One Douglas Avenue, Suite 300  
Elgin, IL 60120**

Plot J.27: RHTR DRN TK 21A USCV

Comparison of Wear Predictions - RHTR DRN TK 21A USCV @CYCLE 19

LCF = 1.03313

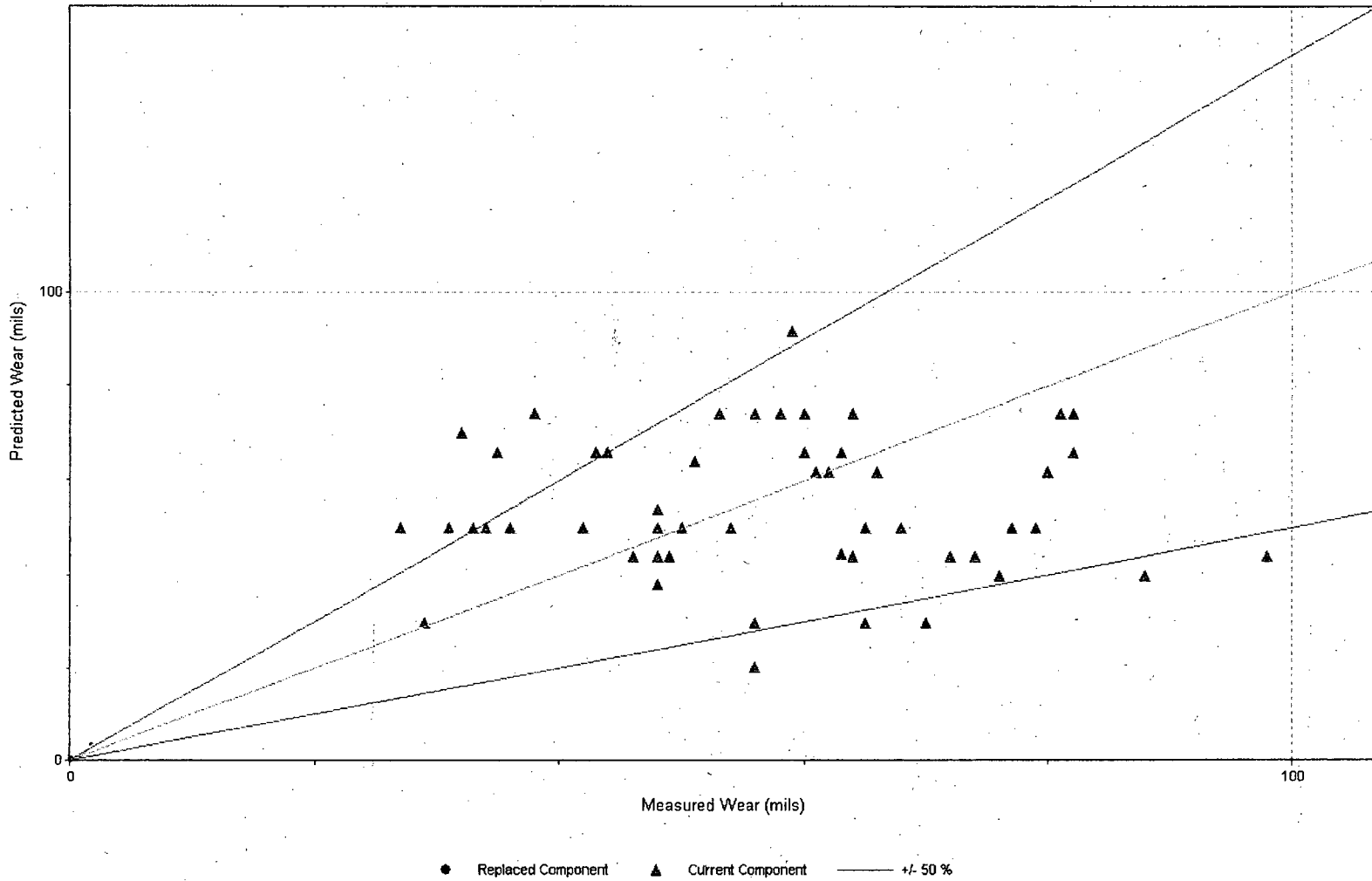


FIGURE 6

Plot J.32: RHTR DRN TK 23B USCV

Comparison of Wear Predictions - RHTR DRN TK 23B USCV @CYCLE 19

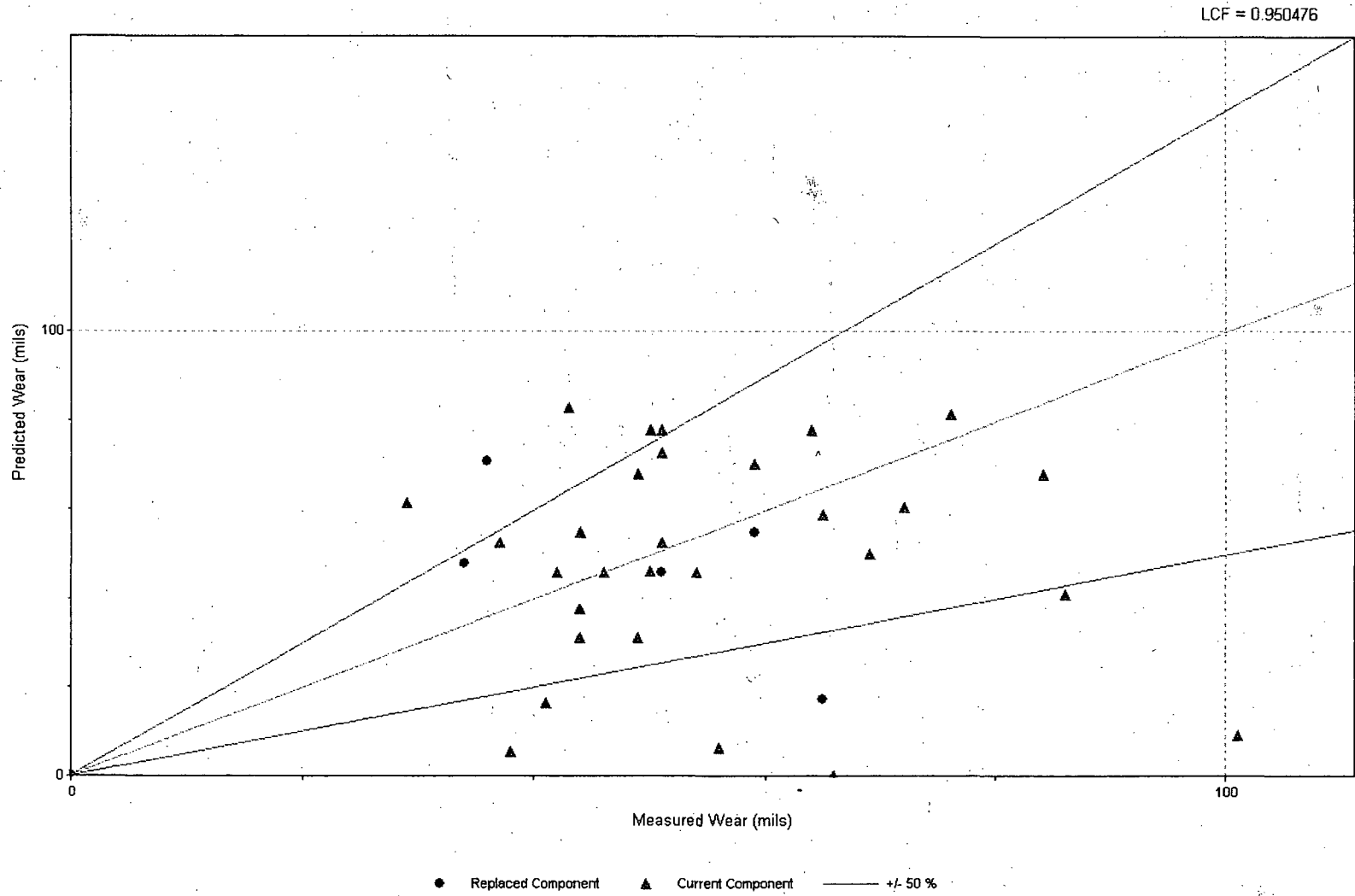


FIGURE 7

**Indian Point Unit 2  
CHECWORKS SFA Model**

**CSI Calculation No. 0705.101-01  
Revision 1  
Issued For-Use**

**February 26, 2010**

*prepared for:*

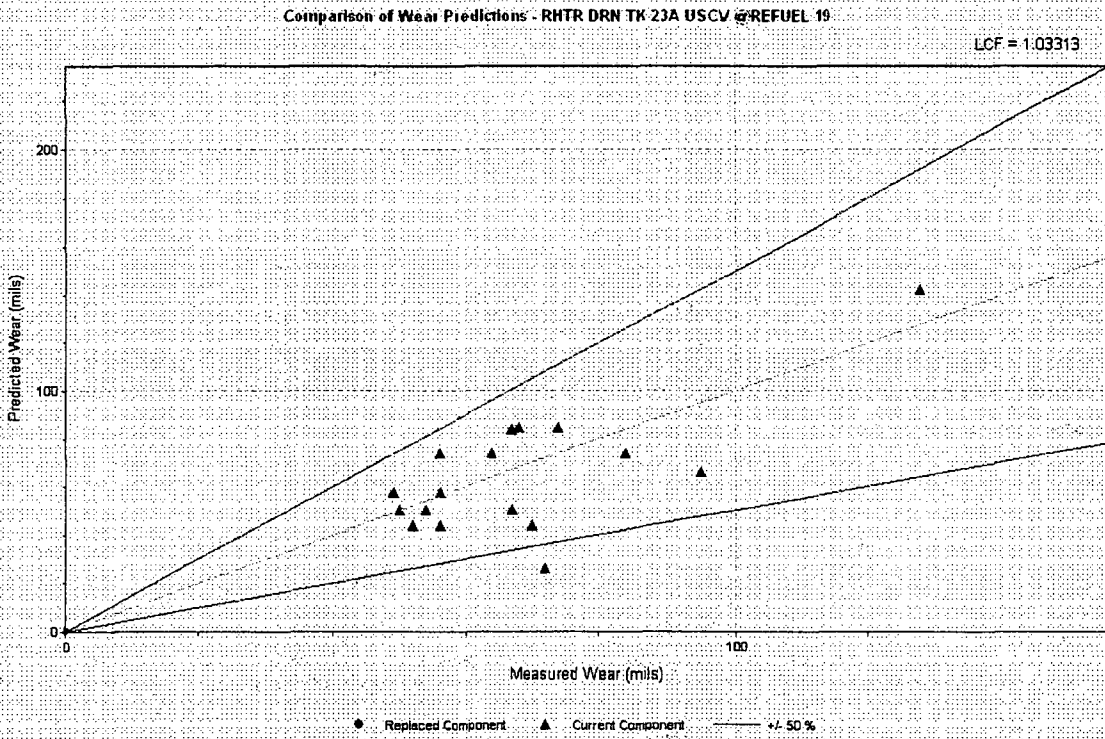
**Indian Point Unit 2  
295 Broadway Suite 3  
PO Box 308  
Buchanan, NY 10511-0308**

*prepared by:*

**CSI TECHNOLOGIES, INC.  
One Douglas Avenue, Suite 300  
Elgin, IL 60120**

# FIGURE 8

### Plot J.35: RHTR DRN TK 23A USCV



### Plot J.36: RHTR DRN TK 23B USCV

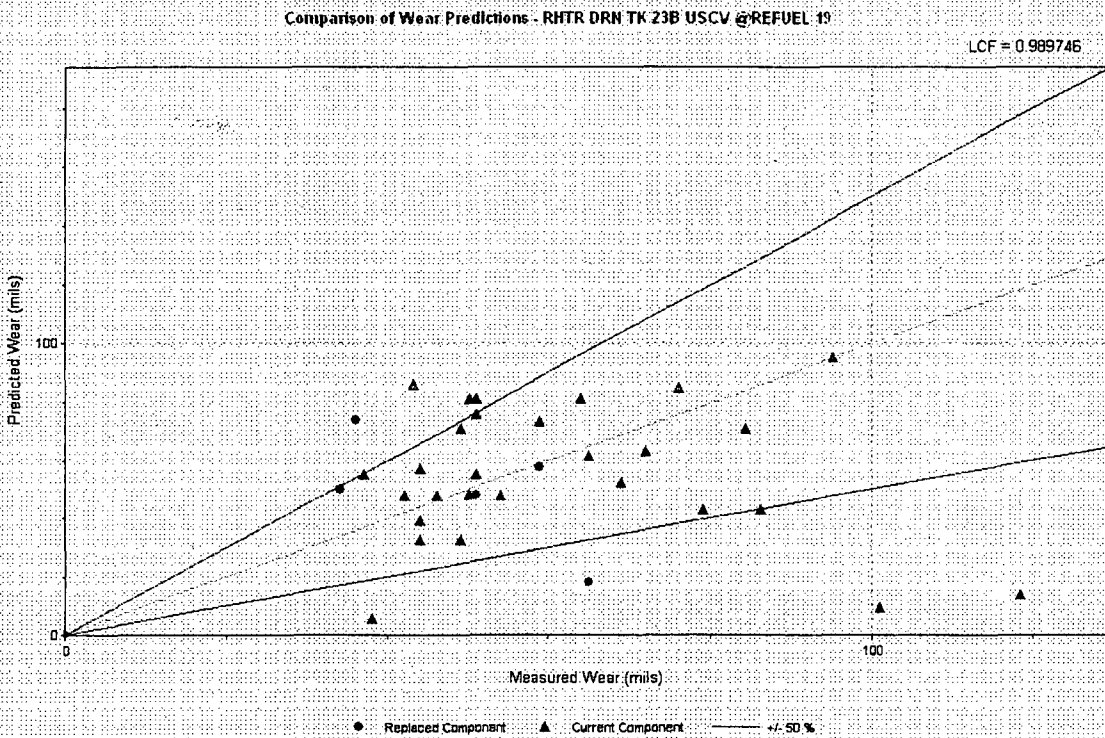
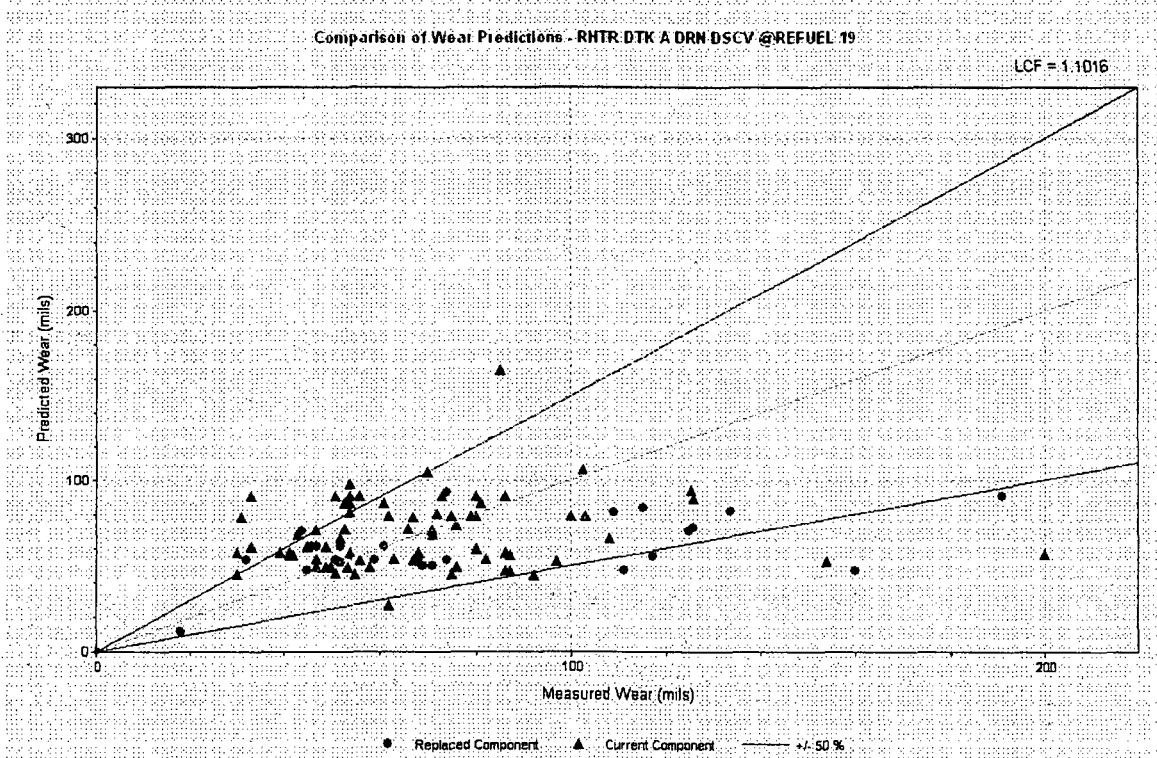
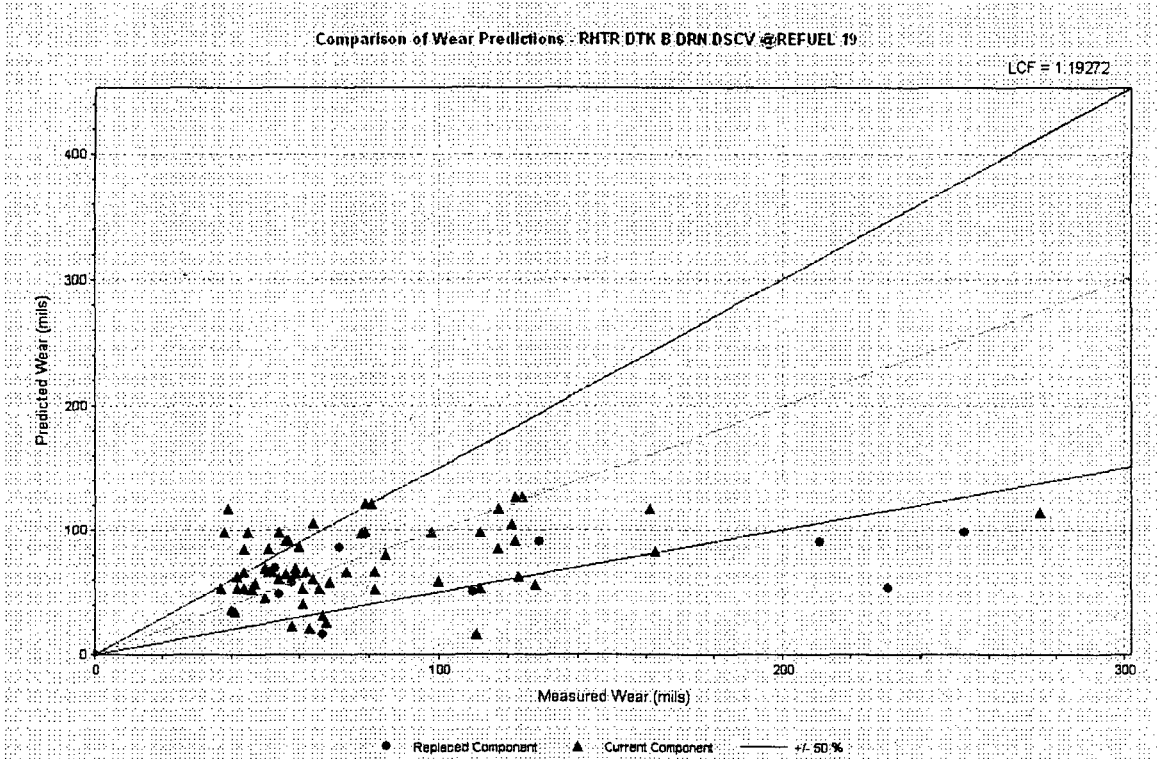


FIGURE 9

Plot J.37: RHTR DTK A DRN DSCV



Plot J.38: RHTR DTK B DRN DSCV



**Indian Point Unit 3  
CHECWORKS FAC Model**

**Calculation No. 050714c-01  
Revision 0  
Issued For-Use**

**October 25, 2005**

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*prepared by:*

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East Dundee, IL 60118**



FIGURE 10

### Comparison of Wear Predictions

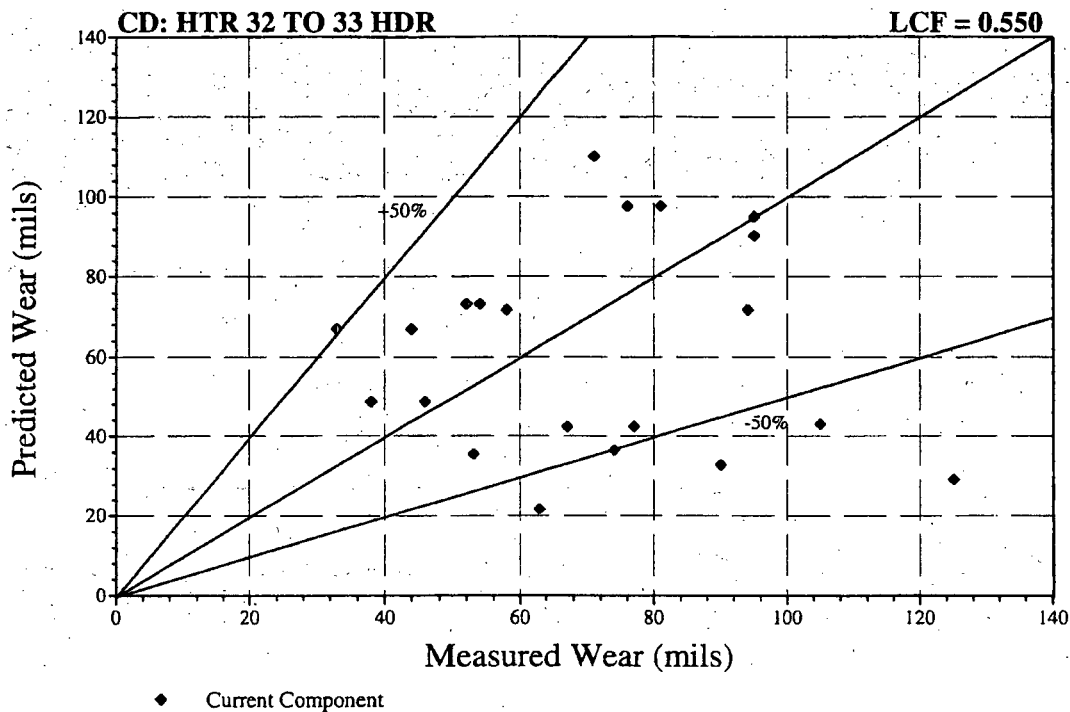


FIGURE 11

### Comparison of Wear Predictions

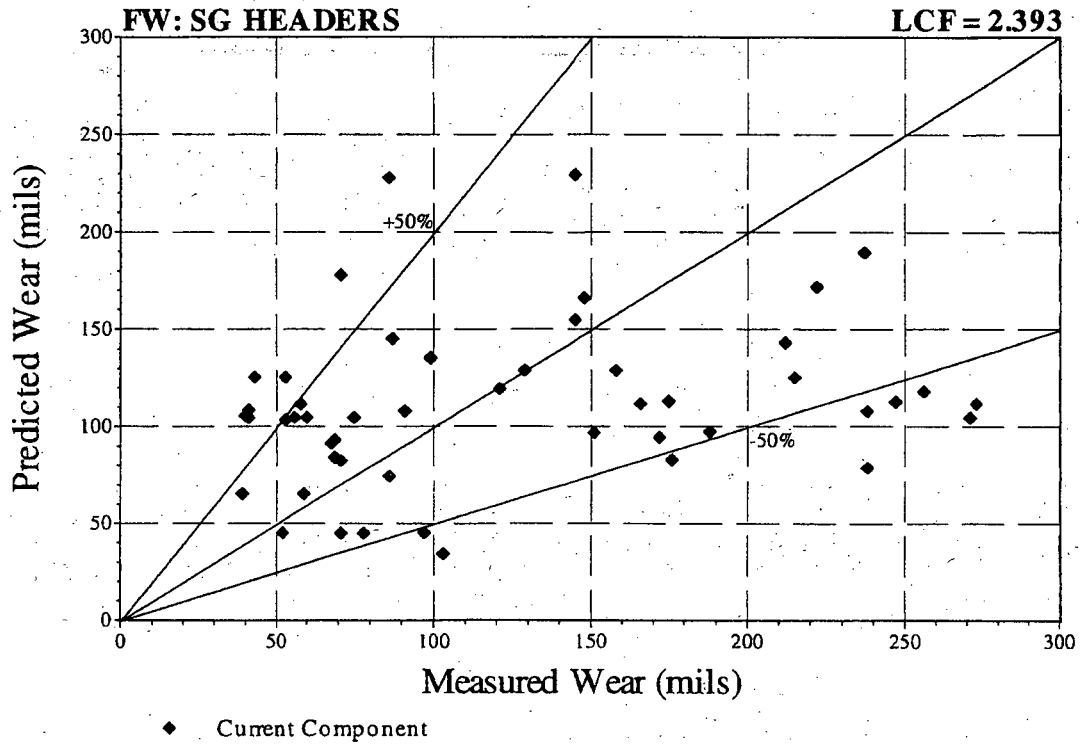
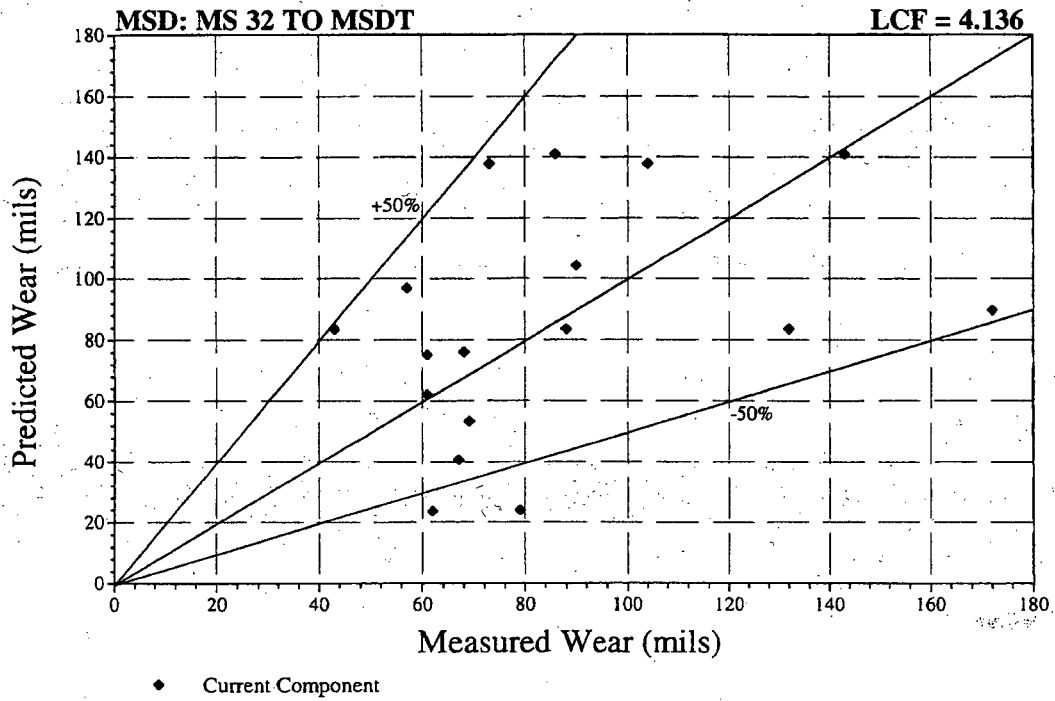


FIGURE 12

### Comparison of Wear Predictions



**Indian Point Unit 3  
CHECWORKS SFA Model**

**Calculation No. 0705.100-01  
Revision 0  
Issued For-Use**

**November 14, 2007**

*Prepared for:*

**Entergy Nuclear Northeast  
295 Broadway Suite 3  
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Buchanan, NY 10511-0308**

*Prepared by:*

**CSI TECHNOLOGIES, INC.  
1051 E. Main St., Suite 215  
East Dundee, IL 60118**

Comparison of Wear Predictions - FW: SG HEADERS @Cycle 15

LCF = 3.42342

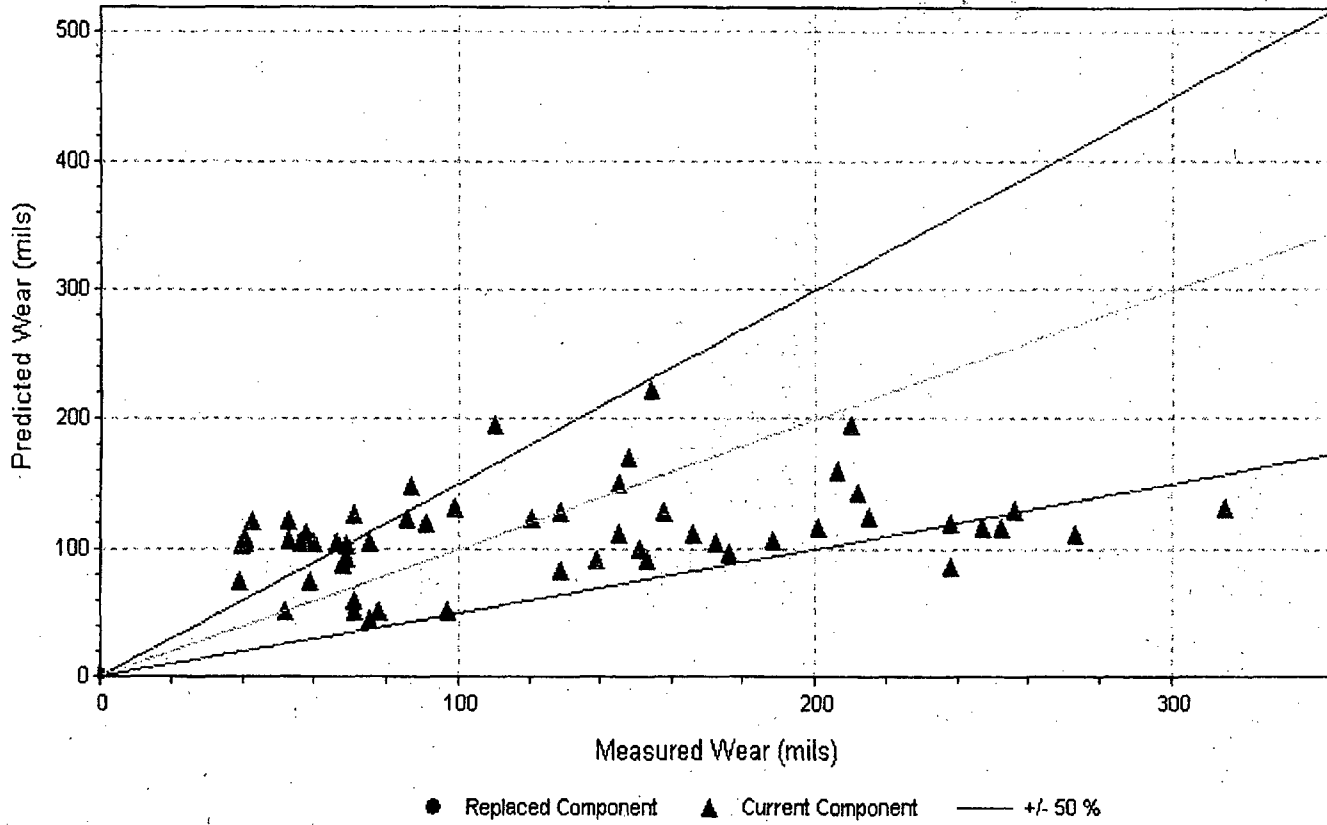


FIGURE 13

Comparison of Wear Predictions - MSD: MS 32 TO MSDT @Cycle 15

LCF = 13.9913

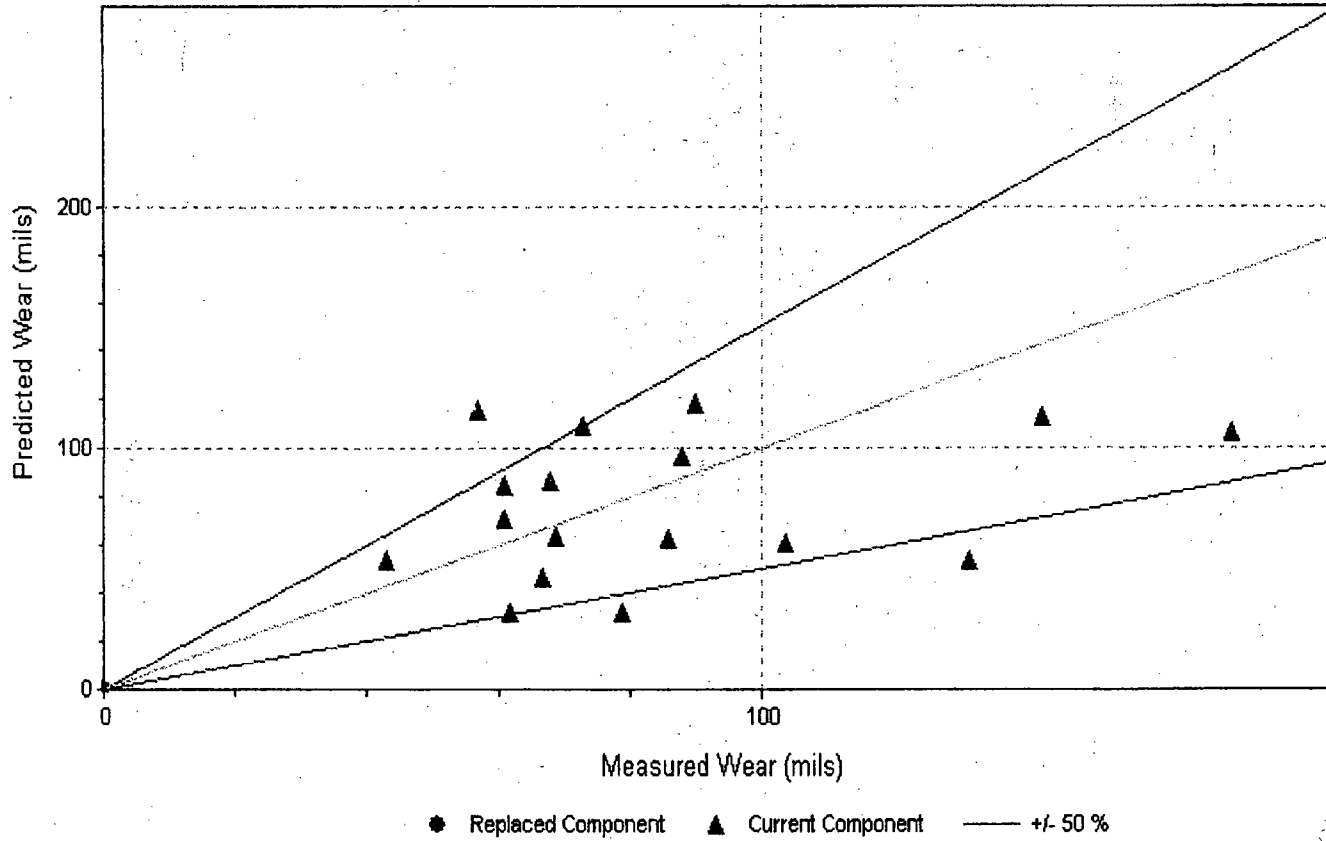


FIGURE 14

Comparison of Wear Predictions - MSD: MSDT 33 TO HDT @ Cycle 15

LCF = 3.76998

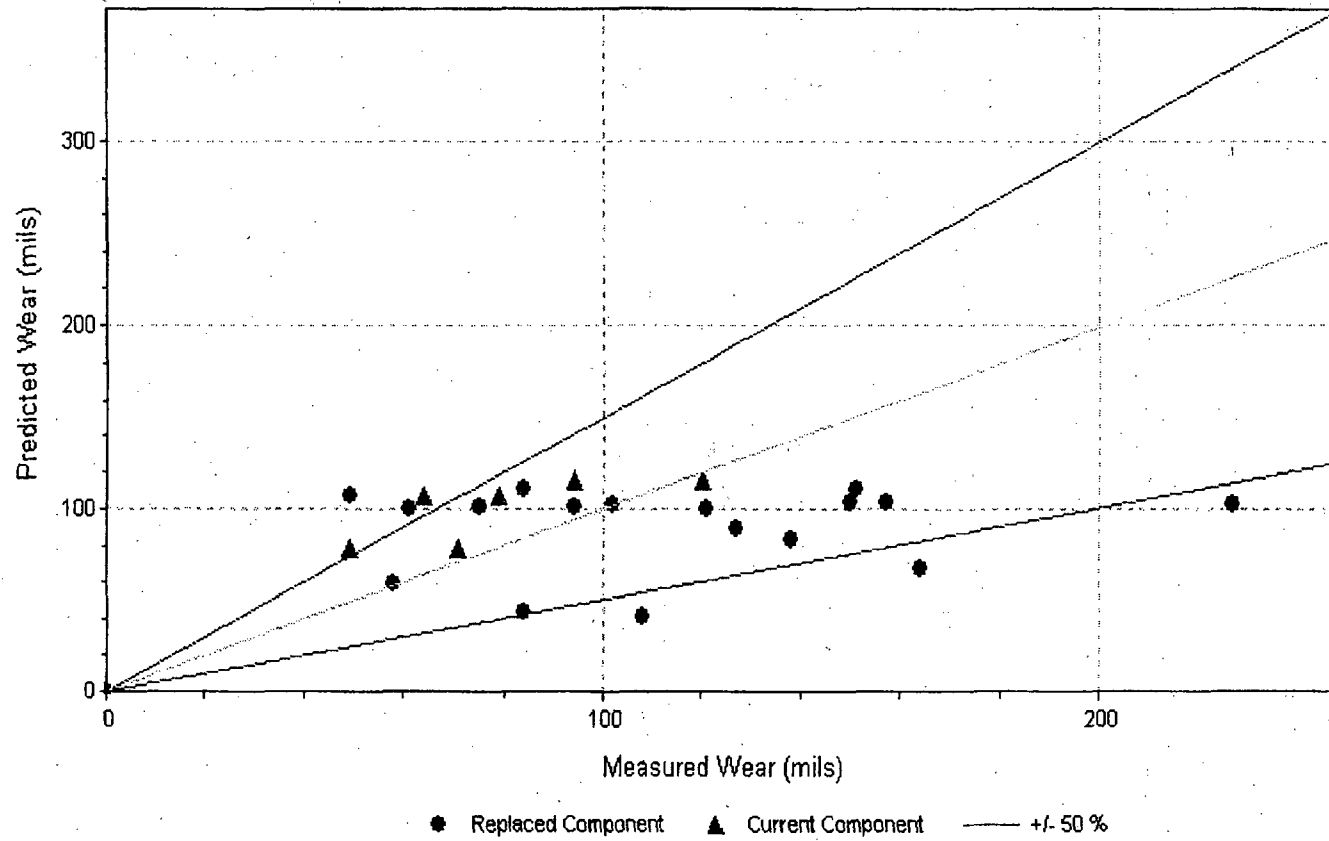


FIGURE 15

**Indian Point Unit 3  
CHECWORKS SFA Model**

**Calculation No. 0705.100-01  
Revision 1  
Issued For-Use**

**February 12, 2010**

*prepared for:*

**Entergy Nuclear Northeast  
Indian Point Unit 3  
295 Broadway Suite 3  
P.O Box 308  
Buchanan, NY 10511-0308**

*prepared by:*

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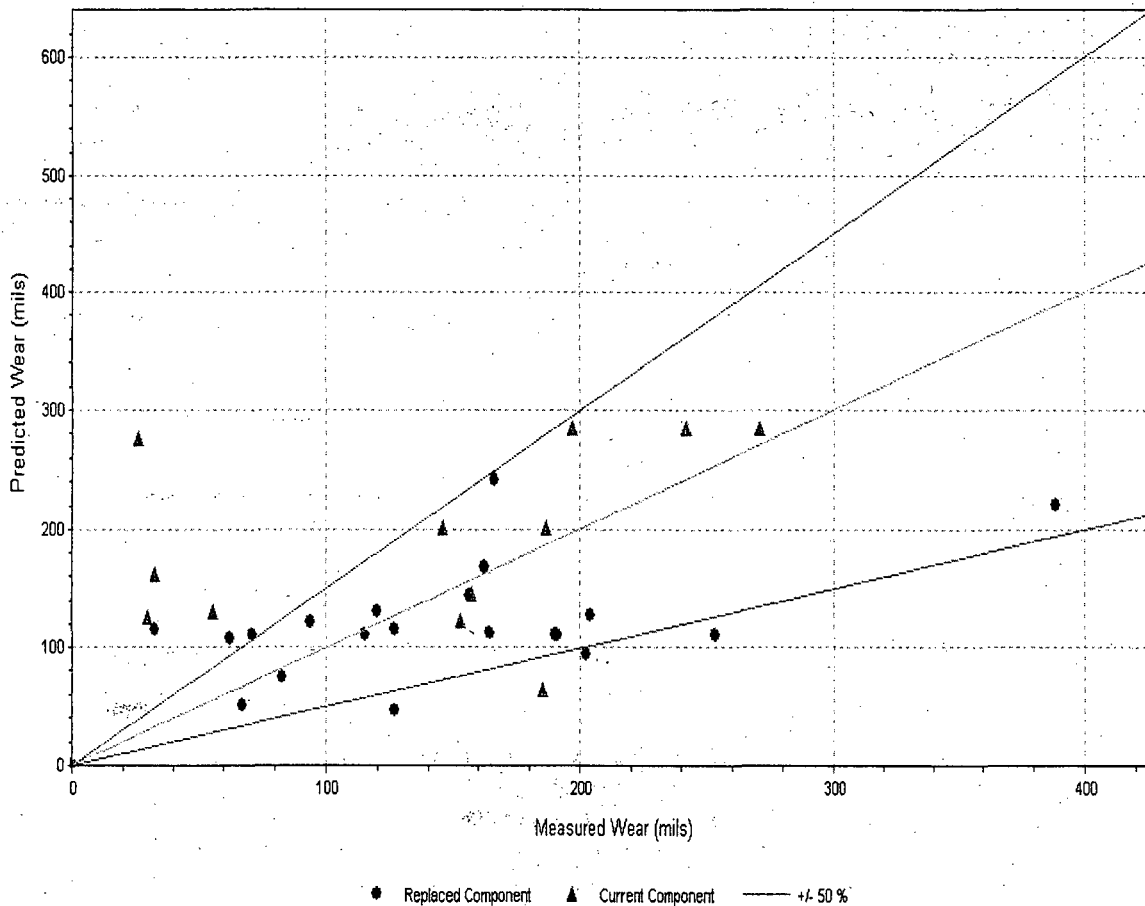


FIGURE 16

Plot J.19: ES: PRESEP TO 35 HDR

Comparison of Wear Predictions - ES: PRESEP TO 35 HDR @Cycle 16

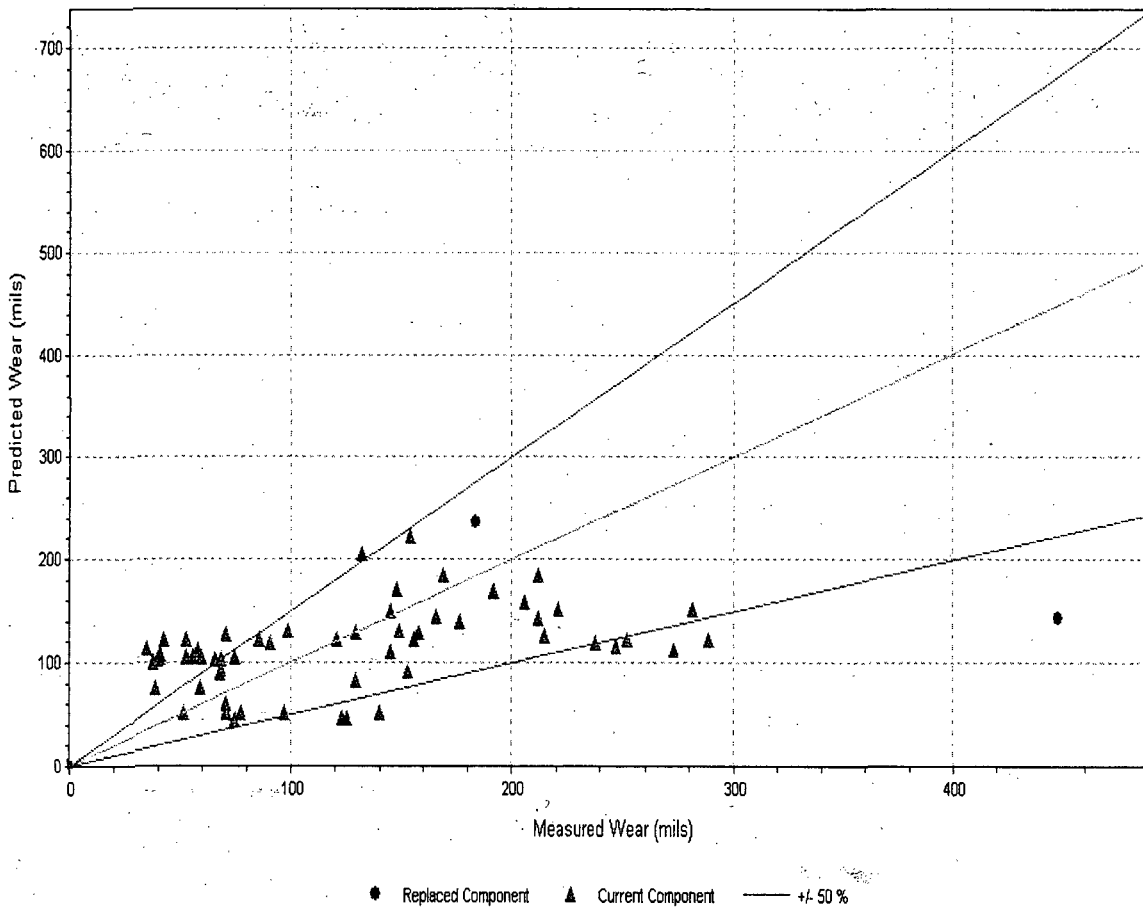
LCF = 2.06052



Plot J.23: FW: SG HEADERS

Comparison of Wear Predictions - FW: SG HEADERS @Cycle 16

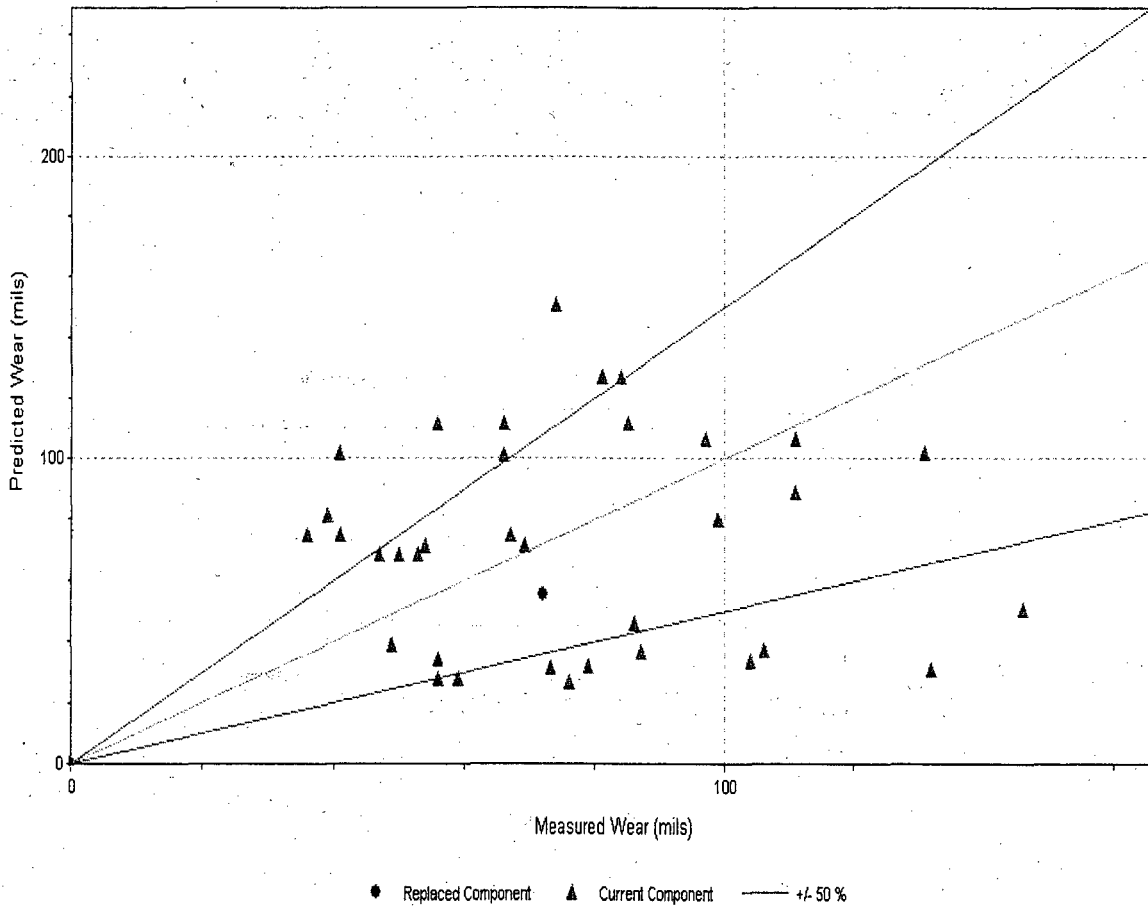
LCF = 3.42342



Plot J.41: RHD: RH 32B TO HDR

Comparison of Wear Predictions - RHD: RH 32B TO HDR @Cycle 16

LCF = 2.93243



Riverkeeper Opposition to Entergy's Motion For Summary Disposition of  
Riverkeeper Technical Contention 2 (Flow-Accelerated Corrosion)

## Riverkeeper TC-2: Attachment 8

**Indian Point Unit 2  
CHECWORKS FAC Model**

**Calculation No. 050714b-01  
Revision 1  
Issued For-Use**

**September 12, 2006**

*prepared for:*

**Entergy Nuclear Northeast  
295 Broadway Suite 3  
PO Box 308  
Buchanan, NY 10511-0308**

*prepared by:*

**CSI TECHNOLOGIES, INC.  
1051 E. Main St., Suite 215  
East Dundee, IL 60118**

CHECWORKS FAC allows a number of options to determine the value of the minimum measured thickness ( $T_{meas}$ ) of an inspected component. "Min. Meas Thickness from Region of Max. Wear" (GW) uses the smallest thickness value from the region that has the highest wear. This option is selected by default if the wear calculation uses the band, blanket, or area methods. The second option used, "Minimum Measured Thickness" (MT), uses the smallest thickness value from any region. MT was chosen for subcomponents that had counterbore, for baseline inspections, when wear was calculated using the point-to-point method, and when the MT value was over 0.040" less than the GW value.

Since the MT method uses the minimum reading from the entire UT inspection grid and the GW method uses the minimum reading from the region where wear is maximum, the  $T_{meas}$  value calculated by MT will be less than or equal to the value calculated by GW in all cases. Thus MT is the more conservative method. However, conservatism is not always the best option in the CHECWORKS model. Because the CHECWORKS model contains many components, using an overly conservative method to calculate the remaining life of one component may cause that component to be selected for inspection at the expense of another. Therefore, the method used was to model components as realistically as possible. See Section 4.1.1 for further discussion on conservatism in the CHECWORKS model.

For inspected components, the  $T_{meas}$  value listed in the "Wear Rate Analysis: Wear Predictions Report" in the Pass 2 Analysis, Appendix I, may not match the measured minimum thickness from the UT readings. In all cases, the  $T_{meas}$  values should not conflict by more than 0.040". Note that the "Wear Rate Analysis: Wear Predictions Report" in Appendix I lists the  $T_{meas}$  method, MT or GW, that was used.

#### **5.4.6. Pass 2 Wear Rate Analyses (WRA) and Line Correction Factor (LCF)**

Pass 2 Wear Rate Analysis was performed on the Wear Rate Analysis Runs as defined with one change: the Analysis Option, "Do Not Use Measured Wear" was deselected. As in Pass 1 WRA, Pass 2 WRA will generate for each component a predicted wear rate, and a predicted remaining service life. During Pass 2 WRA, CHECWORKS also generates a Line Correction Factor (LCF) for each WRA Run in the following way. For each inspected component in the run where the option "Do Not Use for LCF" is not chosen, CHECWORKS generates a ratio of the calculated wear to the predicted wear. The LCF for a run is defined as the median value of these ratios. CHECWORKS multiplies the Pass 1 wear predictions by the LCF to generate the Pass 2 wear predictions.

The LCF indicates the degree to which CHECWORKS over or under-predicts wear. A reasonable LCF should be between 0.5 and 2.5 [7.8]. An LCF outside this range may be the result of inaccuracies in the model (e.g., incomplete chemistry history) or non-representative inspection data.

August 16, 2010

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

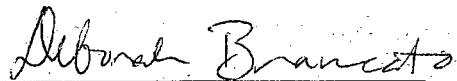
In the Matter of )	
)	
Entergy Nuclear Operations, Inc. )	Docket Nos.
(Indian Point Nuclear Generating )	50-247-LR
Units 2 and 3) )	and 50-286-LR
)	

**CERTIFICATE OF SERVICE**

I certify that on August 16, 2010, copies of the foregoing "Riverkeeper Opposition to Entergy's Motion for Summary Disposition of Riverkeeper Technical Contention 2 (Flow Accelerated Corrosion)" along with eight (8) accompanying attachments, were served on the following by first-class mail and e-mail:

Lawrence G. McDade, Chair Atomic Safety and Licensing Board Panel Atomic Safety and Licensing Board U.S. Nuclear Regulatory Commission Washington, D.C. 20555 E-mail: <a href="mailto:Lawrence.McDade@nrc.gov">Lawrence.McDade@nrc.gov</a>	Judge Kaye D. Lathrop 190 Cedar Lane East Ridgeway, CO 81432 E-mail: <a href="mailto:Kaye.Lathrop@nrc.gov">Kaye.Lathrop@nrc.gov</a>
Richard E. Wardwell Atomic Safety and Licensing Board U.S. Nuclear Regulatory Commission Washington, D.C. 20555 E-mail: <a href="mailto:Richard.Wardwell@nrc.gov">Richard.Wardwell@nrc.gov</a>	Michael J. Delaney, V.P. – Energy New York City Econ. Development Corp. 110 William Street New York, NY 10038 E-mail: <a href="mailto:mdelaney@nycedc.com">mdelaney@nycedc.com</a>
John J. Sipos, Esq. Assistant Attorney General Office of the New York Attorney General for the State of New York The Capitol Albany, NY 12224 E-mail: <a href="mailto:John.Sipos@oag.state.ny.us">John.Sipos@oag.state.ny.us</a>	Kathryn M. Sutton, Esq. Paul M. Bessette, Esq. Jonathan M. Rund, Esq. Morgan, Lewis & Bockius, LLP 1111 Pennsylvania Ave. N.W. Washington, D.C. 20004 E-mail: <a href="mailto:pbessette@morganlewis.com">pbessette@morganlewis.com</a> <a href="mailto:ksutton@morganlewis.com">ksutton@morganlewis.com</a> <a href="mailto:jrund@morganlewis.com">jrund@morganlewis.com</a>

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<p>Sherwin E. Turk  Beth N. Mizuno  Brian G. Harris  David E. Roth  Andrea Z. Jones  Office of General Counsel  Mail Stop: 0-15D21  U.S. Nuclear Regulatory Commission  Washington, D.C. 20555-0001  E-mail: <a href="mailto:Sherwin.Turk@nrc.gov">Sherwin.Turk@nrc.gov</a>;  <a href="mailto:Beth.Mizuno@nrc.gov">Beth.Mizuno@nrc.gov</a>; <a href="mailto:brian.harris@nrc.gov">brian.harris@nrc.gov</a>;  <a href="mailto:David.Roth@nrc.gov">David.Roth@nrc.gov</a>; <a href="mailto:andrea.jones@nrc.gov">andrea.jones@nrc.gov</a>;</p>	<p>Sean Murray, Mayor  Village of Buchanan  Municipal Building  236 Tate Avenue  Buchanan, NY 10511-1298  E-mail: <a href="mailto:vob@bestweb.net">vob@bestweb.net</a>,  <a href="mailto:SMurray@villageofbuchanan.com">SMurray@villageofbuchanan.com</a>,  <a href="mailto:Administrator@villageofbuchanan.com">Administrator@villageofbuchanan.com</a></p>



Deborah Brancato

August 16, 2010