UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

August 12, 2014

EA-14-091

Mr. Scott Batson
Site Vice President
Duke Energy Carolinas, LLC
Oconee Nuclear Station
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION - FINAL SIGNIFICANCE DETERMINATION OF WHITE FINDING, NOTICE OF VIOLATION AND ASSESSMENT FOLLOW-UP LETTER (NRC INSPECTION REPORT NO. 05000269/2014012)

Dear Mr. Batson:

This letter provides you the final significance determination of the preliminary Greater than Green finding discussed in NRC Inspection Report 05000269/2014011. The finding involved your failure to identify and correct a significant condition adverse to quality involving a crack in a weld located in the Unit 1 High Pressure Injection (HPI) system. In 2004, a procedure was developed for augmented in-service inspection program ultrasonic examinations which effectively removed reasonable assurance that HPI nozzle component weld cracking would be identified and corrected. Consequently, an undetected crack developed and propagated, resulting in reactor coolant system pressure boundary leakage which required Unit 1 be shut down until the weld was repaired.

At your request, a Regulatory Conference was held on July 31, 2014, to discuss your views on this finding (ML14211A221). During the meeting your staff described your assessment of the significance of the finding and the corrective actions taken to resolve the issue, including the root cause evaluation. You discussed completed corrective actions, actions to prevent recurrence, and improvements in non-destructive examination program oversight, nuclear oversight, and augmented inspection processes and procedures. You also provided additional information on the risk characterization of the finding. Although the NRC agreed with some of the information you provided, we disagreed with aspects of your data-driven analysis, adjusted conditional rupture probability analysis, and application of common cause failure probability. The details of the NRC's disposition of this information are included as Enclosure 2.

The NRC risk analysts performed a conditional assessment of the risk associated with this finding using: 1) the conditional rupture probability from the probabilistic fracture mechanics analysis ($6x10^{-4}$) after adjusting it to an annualized small break loss of coolant accident (LOCA) frequency, 2) the likelihood for a common cause failure of all injection nozzle welds adjusted to the value referenced in the licensee's report ($7x10^{-4}$), and 3) a split fraction probability of a loss of injection capability on 1 of 4 nozzles from a likelihood of complete failure to a failure probability of 10%, or 0.1. The result of this calculation was a change in core damage frequency (CDF) of $1.33x10^{-6}$ per year. As a confirmation of this conditional assessment, the NRC also performed an event assessment with the leak as the initiating event.

This event assessment used a conditional rupture probability of $2x10^{-3}$ and resulted in a risk of $2.8x10^{-6}$. Both risk assessments resulted in a significance determination above the White threshold.

After considering the information developed during the inspection, the additional information you provided at the regulatory conference, and the results of our risk assessments, the NRC has concluded that the finding is appropriately characterized as White, a finding of low to moderate safety significance.

You have 30 calendar days from the date of this letter to appeal the staff's significance determination for this White finding. Such appeals will be considered to have merit only if they meet the criteria given in the IMC 0609, Attachment 2. An appeal must be sent in writing to the Regional Administrator, Region II, 245 Peachtree Center Avenue NE; Suite 1200; Atlanta, Georgia 30303-1257.

The NRC has also determined that the failure to identify and correct a significant condition adverse to quality is a violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, as cited in the attached Notice of Violation (Notice). The circumstances surrounding the violation were described in detail in the previously referenced inspection report. In accordance with the NRC Enforcement Policy, the Notice is considered an escalated enforcement action because it is associated with a White finding.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC review of your response to the Notice will also determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

Because of this White violation, the NRC determined the performance at Oconee Unit 1 to be in the Regulatory Response Column of the Reactor Oversight Process Action Matrix beginning the second quarter of 2014. Therefore, the NRC plans to conduct a supplemental inspection in accordance with Inspection Procedure 95001, “Supplemental Inspection for One or Two White Inputs in a Strategic Performance Area” when you have notified us of your readiness. The objectives of this supplemental inspection are to 1) provide assurance that the root causes and contributing causes of risk-significant performance issues are understood, 2) provide assurance that the extent of condition and extent of cause of risk significant performance issues are identified and 3) provide assurance that your corrective actions for this risk-significant performance issue are sufficient to address the root and contributing causes, and prevent recurrence.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice and Procedure," a copy of this letter, its enclosures, and your response, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

Sincerely,

/RA/

Victor M. McCree
Regional Administrator

Docket Nos.: 50-269
License Nos.: DPR-38

Enclosures:
1. Notice of Violation
2. Detailed Review of Licensee Risk Information

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ADAMS: X Yes ACCESSION NUMBER:_ ML14224A629______ X SUNSI REVIEW COMPLETE ☐ FORM 665 ATTACHED

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NOTICE OF VIOLATION

Duke Energy Carolinas, LLC.
Oconee Nuclear Station

Docket Nos.: 50-269
License Nos.: DPR-38
EA-14-091

During an NRC inspection completed on June 19, 2014, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. In the case of significant conditions adverse to quality (SCAQ), the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.
>
> Contrary to the above, prior to November 11, 2013, the licensee failed to establish measures to promptly identify and correct a significant condition adverse to quality involving a circumferential crack in weld 1-RC-201-105 located on the Unit 1 High Pressure Injection (HPI) piping-to-cold leg nozzle safe end interface of the 1B2 reactor coolant pump suction pipe, which resulted in non-isolable pressure boundary leakage while the reactor was in Mode 1. Specifically, in 2004, the licensee implemented procedure, NDE-995, "Ultrasonic Examination of Small Diameter Piping Butt Welds and Base Material for Thermal Fatigue Damage," to perform augmented in-service inspection program ultrasonic examinations which did not provide measures to assure that HPI nozzle component cracking would be identified and corrected. Consequently, in 2012, the licensee performed procedure NDE-995 on weld 1-RC-201-105, and did not identify any reportable indications; even though the ≥ 50% through wall circumferential crack was present in the weld. The presence of the crack was confirmed after a re-review of a radiographic film obtained during a non-destructive test of the 1B2 thermal sleeve in 2011. On November 11, 2013, the licensee identified the through-wall circumferential crack in weld 1-RC 201-105 after transitioning Unit 1 to Mode 3 to investigate non-isolable pressure boundary leakage.

This violation is associated with a White significance determination process (SDP) finding.

Pursuant to the provisions of 10 CFR 2.201 Duke Energy Carolinas, LLC is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001 with a copy to the Regional Administrator, Region II, and a copy to the NRC Resident Inspector at the Oconee Nuclear Station, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-14-091" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made publicly available in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be available without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy, or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days of receipt.

Dated this 12th day of August 2014

Detailed Review of Licensee Risk Information

In response to NRC Inspection Report 05000269/2014011, Duke Energy Carolinas, LLC (DEC) attended a Regulatory Conference and provided information to address the preliminary risk characterization of the finding. The NRC agreed with the following points presented by DEC during the Regulatory Conference:

- Dominant Risk Scenario – The dominant risk scenario was a small break loss of coolant accident (SLOCA) with a potential failure of transitioning to high pressure recirculation by the operators. The DEC and NRC cutsets and sequences of concern closely mirrored each other.
- Reactor Coolant Pump (RCP) Seal Modeling – The NRC modified the RCP seal model in SPAR to reflect the current Flowserve (Byron-Jackson) N-9000 package installed at the time of discovery of the Unit 1 weld leak. Revised evaluations were made assuming the newer seal packages.
- Unit 2 and Unit 3 Treatment – Initially the NRC characterized the performance deficiency as having applicability to Unit 2 and 3, but that the risk for these 2 units was one order-of-magnitude less than for Unit 1. DEC disagreed and indicated during the conference that the Unit 1 nozzle weld configuration was an outlier which contributed to the likelihood of the leak. After further analysis, the agency finds DEC's characterization reasonable.
- Medium Loss of Coolant Accident (MLOCA) – The licensee asserted that a MLOCA scenario should not be considered or that, if considered, it would have a negligible contribution. The NRC used existing data from NUREG-1829, "Estimating Loss of Coolant Accident Frequencies" to estimate the likelihood of a guillotine break on the 2.5" line that would result in a MLOCA and considered the probability low enough such that there would be no impact on the final risk determination.
- External Risk – The risk of a seismic event was initially evaluated using the frequency of the Design Basis Earthquake and the assumption that if an earthquake occurred, then a severely flawed pipe would not hold its integrity. The licensee analyzed the 1B2 weld under postulated seismic loads and provided cutsets from their Oconee fire model to address the external risk impact. Based on these results, the NRC concluded that the risk due to external events was negligible.
- Probability of a Reactor Trip – The NRC risk analysis initially estimated the risk of a rapid shutdown or a forced reactor trip caused by a leak as setting TRANS = 1.0, and this was assumed to be an adequate approximation. DEC indicated that this over estimates the risk based on plant procedures and operator performance during simulator scenarios and the NRC found this reasonable.
- Conditional Rupture Probability - The agency reviewed the licensee's probabilistic fracture mechanics calculations to determine the probability of rupture given a leak. This probability was calculated to be $6x10^{-4}$ using a probabilistic fracture mechanics approach. Although the NRC believes this method has some potential for under-estimating the likelihood of rupture, it was assessed by the NRC as having validity because it determined a rupture probability based on a near-through-wall leak, which is what was present prior to the event in November.

The NRC risk analysts performed a conditional assessment of the risk using 1) the conditional rupture probability from the probabilistic fracture mechanics analysis ($6x10^{-4}$) after adjusting it to an annualized small break LOCA frequency, 2) the likelihood for a common cause failure of all injection nozzle welds adjusted to the value referenced in the licensee's report ($7x10^{-4}$), and 3) a split fraction probability of a loss of injection capability on 1 of 4 nozzles from a likelihood of complete failure to a failure probability of 10%, or 0.1. The result of this calculation was a change of core damage frequency (CDF) of $1.33x10^{-6}$ per year. To further confirm the risk analysis, the NRC analysts also performed a sensitivity analysis. An event assessment was performed using the conditional rupture probability from the data driven analysis ($2x10^{-3}$), with a result of $2.8x10^{-6}$.

DEC assessed the risk of this performance deficiency to be in the range of $2x10^{-7}$ to $7x10^{-7}$. This range was the result of using two different approaches to assess the risk, a data-driven analysis and a deterministic fracture mechanics analysis. The NRC disagreed with several aspects of the two approaches.

1. Data-Driven Analysis

   One of the methods DEC used to calculate the change in CDF was done by calculating a change in LOCA frequency and multiplying by a conditional core damage probability (CCDP) for a SLOCA. This calculation did not include the potential for a loss of injection capability due to a broken weld at the 1B2 nozzle.

   This method, which used a change in LOCA frequency based on current and historical data, analyzes the risk of a pre-existing flaw going forward in time and does not capture the total risk to the public during the exposure period in which the performance deficiency existed. The increase in the small-break LOCA event frequency analysis was "forward looking", i.e., it estimated an increase in initiating event frequency using the November 2013 event as a past data point in developing a new frequency distribution. DEC's analysis produced an initiating event frequency distribution which can be applied for updating future baseline risk modeling, but does not provide the necessary information for the Significance Determination Process (SDP) as described in the RASP Manual, Volume 1, Section 2.2, "Definition - Exposure Time." Specifically, the SDP assesses the risk of the finding during the timeframe that it existed (i.e., the exposure time).

   In addition, during the regulatory conference, DEC indicated that there were a total of 26 pipe failures identified in pressurized water reactor (PWR) Class 1 high pressure injection (HPI) piping with 10 coming from Babcock & Wilcox plants and there were 12 leaks and 7 large leaks at PWRs due to vibration fatigue. As a result DEC calculated a very small change in CDF. NRC concluded that the very small sample size was not sufficiently informative for a risk assessment of infrequent events. Consequently the agency considered, but did not significantly credit, the licensee's position that the change in CDF should be much lower than NRC estimates due to the lack of previous loss of coolant accidents which started as leaks.

2. Adjusted Conditional Rupture Probability Analysis

   During the regulatory conference, DEC claimed that implicit within the $6x10^{-4}$ conditional rupture probability were three factors that should make the overall risk result lower:

   1) DEC's analysis provided a range of values for conditional rupture probability (CRP) of which the NRC used the lowest value in the significance determination process ($6x10^{-4}$ probabilistic fracture mechanics analysis of a flawed weld). DEC claimed that the bounding stresses in that analysis (i.e. earthquake motion, HPI full flow injection, etc) are conservative and therefore applied a reduction of $1x10^{-2}$. The NRC agreed that the applied stress values were conservative and that the SDP should use best estimates. However, because DEC did not provide a basis for a reduction in the CRP by two orders of magnitude, it was not adjusted further. In addition, since the NRC used the CRP at the low end of the range of DEC's analyses, any additional reduction would be inappropriate because it would put the CRP near that of an unflawed weld (the unflawed CRP from the DEC supplied PFM analysis is approximately $2x10^{-6}$).

   2) DEC claimed that the $6x10^{-4}$ CRP would represent the rupture probability given that the through-wall crack was present for an operating cycle. This analysis only accounts for the CRP given that the flaw is already through-wall and leaking. In doing so, DEC proposes to credit the increased probability that the leak will be discovered prior to rupturing. However, this analysis is not consistent with the SDP and does not take into account the risk of rupture prior to the flaw progressing to a leak, which would be harder to identify. Therefore, the NRC does not agree that the CRP should be reduced as proposed by DEC.

   3) DEC used a conservative application of ~25% to account for the high degree of uncertainty in the modelled crack geometry in the final results. The NRC risk analysts adjusted for the 25% margin but the result remained greater than the $1x10^{-6}$ threshold.

3. Common Cause Failure Probability

   DEC also asserted that the potential common cause failure (CCF) of other high pressure injection lines was negligible to the overall risk. The agency disagreed with this position in that the licensee's faulty NDE-995 procedure was used on the 1B2 welds and on the other 11 nozzle welds between all 3 Oconee units and, therefore, the potential to misidentify other flaws/cracks existed during prior operating cycles. The Risk Assessment of Operational Events (RASP) manual which serves as guidance to NRC risk analysts states in Section 5.2, "The performance deficiency (PD) that resulted in a failure in the CCCG [common-cause component group] has the potential for CCF of other components in the same CCCG. The performance deficiency identified in an inspection report is assumed to manifest a shared cause of potential failure of other "like" components in a system." As a result, the agency concluded CCF was low but could not be ignored. When adjusted to the maximum value that DEC recommended, CCF had an influential effect on the increase in risk of this event.

Although not included in the analysis, a number of other qualitative factors were considered which would increase the risk significance of this finding, including:

- This performance deficiency existed for a long period of time. As described in the licensee's formal root cause analysis and based on the metallurgical analysis performed after removing the section of piping, the cracked weld at the 1B2 location existed for a number of years and went undetected by the NDE-995 procedure. Normally, per the RASP Manual, Volume 1, "Internal Events," Section 2.7, the maximum exposure time in a condition analysis is limited to one year however, given the lengthy duration, this is a qualitative consideration that would make the risk to the public higher.

- DEC asserted that the crack would have grown in a "stable" manner. However, DEC also indicated that the probabilistic fracture mechanics analysis was highly sensitive to flaw shape and a different shape could yield very different results. Given that failures due to this mechanism can happen in a rapid and unpredictable manner, and the degradation mechanism at work on the 1B2 weld was vibratory fatigue, the agency concluded the rate of crack growth would be difficult to predict under all cases and conditions. As a result, the NRC disagreed with DEC's conclusion that rapid crack propagation could not occur.

- Industry values were used to estimate the probability that a flaw was present, but there have been four leaks at the Oconee site on HPI system welds. This includes a leak on the 2A1 HPI/RCS weld in 1997 due to thermal fatigue and a leak on the 2A HPI line during the HPI full flow test in 2007. The repetitive nature of these weld flaws over an extended period of time would qualitatively add to the risk.