UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 16, 2012

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION – SUBMITTAL OF ANALYTICAL EVALUATION FOR A REACTOR RECIRCULATION LINE WELD (TAC NO. ME5341)

Dear Mr. Pacilio:

By letter dated December 15, 2010, as supplemented on June 27, 2011, and January 25, 2012 (Agencywide Documents Access and Management System Accession Nos. ML103500359, ML111790025 and ML12030A120, respectively), Exelon Nuclear (the licensee) submitted an evaluation of a circumferential indication found in weld NG-E-007 of the reactor recirculation line in scheduled non-destructive examinations conducted during the 2010 refueling outage (1R23) at the Oyster Creek Nuclear Generating Station.

On the basis of the information submitted, the U.S. Nuclear Regulatory Commission (NRC) staff concludes that the licensee has provided reasonable assurance of the structural integrity of the degraded weld, NG-E-007, by demonstrating that the projected size of the circumferential indication in the weld will satisfy the requirements of the American Society of Mechanical Engineers *Boiler and Pressure Vessel* Code, Section XI, IWB-3600, for the remaining years of the renewed operating license at the Oyster Creek Nuclear Generating Station. Enclosed is the NRC staff's safety evaluation. This completes the NRC staff's effort for TAC Number ME5341.

Please contact me at (301) 415-3100 if you have any questions.

Sincerely,

John G. Lamb, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosure: As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING THE FLAW EVALUATION OF WELD NG-E-007

OF REACTOR RECIRCULATION PIPING

EXELON NUCLEAR

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated December 15, 2010, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML103500359), Exelon Nuclear (the licensee) submitted an evaluation of a circumferential indication (flaw) found in weld NG-E-007 of the reactor recirculation line in scheduled non-destructive examinations (NDE) conducted during the 2010 refueling outage (1R23) at the Oyster Creek Nuclear Generating Station. Weld NG-E-007 joins a 90-degree wrought stainless steel elbow to a cast austenitic stainless steel (CASS) valve. By letter dated June 27, 2011, (ADAMS Accession No. ML111790025), the licensee responded to the U.S. Nuclear Regulatory Commission (NRC) staff request for additional information (RAI). By letter dated January 25, 2012 (ADAMS Accession No. ML12030A120), the licensee responded to the NRC staff RAI regarding NDE of the subject weld.

The licensee evaluated the circumferential indication in accordance with the requirements of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel* (B&PV) Code, Section XI, IWB-3600 because the indication exceeded the acceptance standards of IWB-3500.

2.0 REGULATORY EVALUATION

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a, "Codes and standards," Paragraph (g)(4), ASME B&PV Code Class 1, 2, and 3 components (including supports) must meet the requirements set forth in the ASME Code, Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code, incorporated by reference in 10 CFR 50.55a(b), 12 months prior to the start of the 120-month interval, subject to

the conditions listed therein. The 1995 edition and 1996 addenda of the ASME Code, Section XI, is the Code of Record for the current ISI interval.

Paragraph (e) of the ASME Code, Section XI, IWB-3610, "Acceptance Criteria for Ferritic Steel Components 4 in. and Greater in Thickness," states that the evaluation procedures of flaws be subject to approval by the regulatory authority having jurisdiction at the plant site. IWB-3640, "Evaluation Procedures and Acceptance Criteria for Austenitic Piping," states that the flaw evaluation procedures and acceptance criteria are subject to approval of the regulatory authority having jurisdiction at the plant site.

The licensee stated that the design code for the affected weld is ASME Code Section I, 1965 Edition, and American Standards Associations (ASA) B31.1, 1955. The ASME Code, Section XI, 1995 edition through 1996 Addenda is the Code of Record for the repair/replacement activities in the current fourth ISI interval. In the June 27, 2011, letter, the licensee clarified that the fourth 10-year ISI interval is effective from October 15, 2002, through October 14, 2012. The fourth ISI interval was not impacted by the license renewal process and continued to be in effect into the period of extended operation in 2009 as discussed in the "Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station," NUREG-1875, dated April 2007.

The NRC staff notes that although the Oyster Creek operating license has been renewed under 10 CFR 54, pursuant to 10 CFR 54.35, "Requirements during terms of renewed license," the licensee is required to follow the regulations in 10 CFR 50. The NRC staff reviewed the submitted flaw evaluation in accordance with 10 CFR 50.55a.

3.0 NRC STAFF EVALUATION

Flaw Characterization and Examination

During the 2010 refueling outage, the licensee detected a circumferential indication at weld NG-E-007, joining an elbow to a valve in the recirculation piping. The elbow is made of wrought stainless steel SA-403, WP-316NG and the valve is made of CASS, SA-351, CF8M. The weld metal is stainless steel, ER308. The indication is located in the heat affected zone (HAZ) in the CASS valve. The indication length is 1.4 inches and depth is 0.3 inches. The indication is located 0.08 inches from the inside surface of the pipe. Based on the proximity rule of the ASME Code, Section XI, IWA-3310, the licensee considered this as an embedded flaw. The nominal outside diameter (OD) of the weld is 26 inches and the nominal thickness is 1.2 inches. The indication length is about 1.7 percent of the circumference and the depth is about 25 percent of the wall thickness.

In 1986, the licensee applied Induction Heating Stress Improvement (IHSI) to the affected weld joint to mitigate potential intergranular stress-corrosion cracking (IGSCC). In addition, the licensee improved the primary system water chemistry with hydrogen and noble metal chemical addition (NMCA) to minimize IGSCC in the primary coolant system.

By letter dated June 27, 2011, in response to NRC staff's RAI Question Number 1, the licensee stated that weld NG-E-007 was last inspected in 1996 using ultrasonic techniques; however, this indication was not identified at that time. By letter dated January 25, 2012, the licensee

clarified that it performed ultrasonic examinations in 1996 and 2010 using Performance Demonstration Initiative (PDI) qualified techniques developed by the Electric Power Research Institute (EPRI). The examination from 1996 used a General Electric (GE) version of PDI-UT-2, Revision A, and the examination from 2010 used a GE Hitachi version of PDI-UT-2, Revision E. The licensee stated that Revision E contains much more detail than Revision A. The licensee further stated that subtle differences in transducer focusing, scan overlap, equipment and recording criteria are most likely the reason the flaw was either not recorded or not detected in 1996. Based on industry-wide operating experience, the NRC staff agrees that the ultrasonic testing (UT) technology has improved its detection and sizing capability since 1996. Therefore, the licensee was able to detect and size the indication in 2010 as a result of UT technology improvement.

The licensee could only examine the weld from the elbow side, but not from the valve side, because of the valve geometry. The licensee did not note any inspection limitation of the flaw location from the elbow side of the weld. By letter dated January 25, 2012, the licensee explained that the GE Hitachi version of PDI-UT-2, Revision E, is a qualified technique in accordance with the EPRI PDI Program for detection and length sizing of planer inner diameter (ID)-connected flaws. However, the procedure has not been demonstrated (not qualified) to detect or length size embedded flaws such as lack of fusion or slag inclusion, which are considered to be fabrication related.

The licensee explained that the "thru-wall sizing" portion of the UT technique used is also qualified per the EPRI program; however, only for flaws on the same side of the weld as the transducer (i.e., the elbow side of the weld). The licensee noted that the examination procedure used is not qualified for through-wall sizing of planar flaws on the far side of the weld (i.e., the valve side of the weld). The specific flaw in question was located on the far side of the weld (i.e., valve side of the weld). The UT procedure used was not qualified for this specific flaw location; however, it was the best available technology. Embedded flaw sizing guidelines are included in the procedure. The licensee was not able to quantitatively assess the accuracy of the flaw sizing.

For the 2010 examination, the licensee used the shear and refracted longitudinal 45 and 60 degree angle ultrasonic transducers. The licensee performed the UT from the outside of the pipe. The examination could not be performed from the ID surface because the ID of the pipe was inaccessible. The examination volume for flaw detection was, at a minimum, the inner one-third of the pipe wall thickness as required by the ASME Code. The UT was performed to the extent necessary to verify the limit of the flaw's thru-wall dimension. The pipe OD surface was smoothed and flattened in accordance with procedure contouring requirements for better contact with transducers.

The NRC staff asked the licensee to discuss whether a surface examination (dye penetrant or eddy current) was performed to verify that the indication is embedded. If a surface examination was not performed, the licensee was requested to discuss why and how the indication was determined to be embedded. By letter dated January 25, 2012, the licensee explained that a surface examination on the OD surface was not performed and was not required. The licensee could not perform ID surface examination because the ID surface was inaccessible. The GE Hitachi version of PDI-UT-2, Revision E, provides a method to determine flaws as either embedded or ID-connected. The process accurately plots the flaw's location relative to the ID

surface. The licensee explained that by applying the GE Hitachi procedure, this indication clearly did not connect with the ID and, therefore, was characterized as a fabrication or "embedded" reflector.

The NRC staff notes that in accordance with ASME Code, Section XI, IWB-2420, "Successive Inspections," that the licensee is required to perform three successive examinations of the subject indication to confirm the growth (or lack of) of the subject indication. By letter dated June 27, 2011, in response to RAI Question Number 2, the licensee stated that the three successive examinations will begin in the first period of the fifth interval of the ISI program. The subsequent examinations will be conducted in accordance with the ASME Code of Record for the fifth ISI interval, including determining if the component examination schedule may revert to the original schedule of successive inspections. The current fourth ISI interval is scheduled to end on October 14, 2012.

The NRC staff finds that the licensee has inspected, characterized, and dispositioned the flaw in accordance with the requirements of the ASME Code, Section XI, IWB-3000. The NRC staff further finds that the licensee will follow the requirements of IWB-2420(b) to perform successive examinations of the subject indication beginning in the first period of the fifth 10-year ISI interval.

Flaw Evaluation

As stated above, the licensee concluded that the circumferential indication is embedded in weld NG-E-007 and not open to the inside surface of the pipe. The licensee considered fatigue and not IGSCC as a degradation mechanism to calculate the flaw growth.

In Section 4.3 of the flaw evaluation, the licensee stated that the flaw is located in the HAZ of the CASS valve. In RAI Question No. 4, the NRC staff asked the licensee to discuss whether the material properties of CASS were used in the flaw evaluation. In a June 27, 2011, letter, the licensee responded that the material properties of CASS material were used in the flaw evaluation. The licensee noted that thermal aging embrittlement is not addressed in the ASME Code at this time. However, the licensee used the methods which are considered to be bounding data for CASS materials in NUREG/CR-4513, Rev. 1, "Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR [light-water reactor] Systems." The licensee also applied the Z-factor of the submerged arc weld on pipe base material, weld material, and the CASS material. The Z-factor increases the applied load to ensure conservatism in the flaw growth calculation.

The licensee used the combinations of the following loadings: pressure, deadweight, thermal, operating basis earthquake, and safe shutdown earthquake. The fatigue cycle is assumed for the full range of mechanical and thermal load specified. This represents the summation of the deadweight, maximum internal pressure, safe shutdown earthquake, and normal operating thermal expansion loads. The licensee assumed 100 fatigue cycles to occur over the remaining 19 years of plant operation to the end of the license renewal period. This assumes that approximately 11 startup-shutdown cycles occur every 2-year operating cycle and that a seismic event occurs during every startup-shutdown cycle.

The licensee projected that the flaw depth will be 0.324 inches and the length will be 1.424 inches at the end of period of extended operation. The allowable flaw depth and length are 0.9

inches and 8.2 inches, respectively. The margins on allowable flaw depth and length are 2.78 (0.9/0.324) and 5.76 (8.2/1.424), respectively. The flaw is projected to remain embedded at the end of plant life.

The NRC staff asked the licensee whether a measurement uncertainty/error was added to the length and depth of the indication, because the UT used in the 2010 inspection was not qualified to detect embedded flaws. If an uncertainty was not added, the NRC staff asked the licensee to demonstrate that the flaw size used in the flaw evaluation is conservative. By letter dated January 25, 2012, the licensee stated that measurement uncertainty was not added to the indication size; however, conservative assumptions were applied in the fracture mechanics evaluation to accommodate for the fact that the inspection procedure was not qualified, as listed below:

1. The licensee used an edge crack Linear Elastic Fracture Mechanics (LEFM) solution which treats the crack as an infinitely long edge crack yielding a much more conservative result for K, than would be obtained if an internally cracked, finite length, cylinder solution was used or if an embedded crack in a cylinder solution was used. The licensee used the depth for the LEFM calculation as the 2a + S value = 0.3 + 0.08 = 0.38 inches. Thus, the flaw was treated as ID-connected for the stress intensity factor calculation used to calculate FCG (Fatigue Crack Growth).

2. The licensee assumed the fatigue cycle based on the full range of mechanical and thermal load, including the deadweight, maximum internal pressure, Safe Shutdown Earthquake, and normal operating thermal expansion loads. The licensee stated that this assumption is very conservative compared to the anticipated thermal transients for the system such as a sudden start of a cold recirculation loop in that the entire pressure, deadweight, thermal, and seismic loads are being treated as a membrane stress cycle.

3. The licensee assumed 100 fatigue cycles to occur over the remaining 19 years of plant operation. The licensee stated that this assumption is conservative since it assumes approximately 11 startup-shutdown cycles occur every 2 year operating period and that a seismic event occurs during every startup/shutdown cycle.

4. The licensee assumed the weld is applied using a submerged arc weld (SAW) process. This results in application of a bounding Z-factor to account for loss of ductility caused by the SAW process; thus, the fracture mechanics calculation is bounding.

5. The licensee considered a length ratio of 0.1 for the allowable flaw size determination using Tables IWB-3641-1 and IWB-3641-2. The actual length ratio between the flaw circumferential length and the pipe circumference, without applying a length uncertainty, is calculated to be 0.017. The licensee stated that as a result, an effective (allowable) length was conservatively considered in the evaluation, when entering the tables, of 8.16 (i.e., 8.2) inches. This value is 5.8 times larger than the flaw length reported in the NDE report.

The NRC staff does not find the above five assumptions to be particularly conservative beyond those required in accordance with the ASME Code Section XI, IWB-3600. The licensee is required to use the LEFM method to perform the flaw evaluation in accordance with ASME Code, Section XI, IWB 3600. The NRC staff does not consider increasing the measured flaw depth by 0.08 inches to be conservative. Assumption Numbers 2, 3, 4 and 5 are the assumptions that would be considered as general acceptable practice in flaw evaluations and are not overly conservative. However, the NRC staff notes that the licensee's flaw evaluation does comply with IWB-3600.

To be conservative in light of an unqualified UT procedure without an ID surface examination, the licensee should add a measurement uncertainty to the flaw size, assume the flaw is ID-connected, and calculate flaw growth based on the crack growth rate of IGSCC. The NRC staff's concern is that if connected to the ID surface, the flaw will most likely grow to the OD surface in a relatively short period of time by IGSCC and thus may challenge the structural integrity of the weld. However, as a compensatory measure, the ASME Code, Section XI, IWB-2420 requires the licensee to monitor the flaw growth, if any. The successive examinations should monitor and confirm whether the flaw will grow and whether the flaw is ID connected.

The NRC staff finds that the licensee has appropriately used the analysis method in the ASME Code, Section XI, IWB-3600, to demonstrate that the projected final flaw size in the subject weld will be within the allowable size with sufficient margins for the remaining years of the renewed operating license. The NRC staff further finds that the flaw size is not significant to challenge the structural integrity of the piping system. The successive examinations will monitor the flaw growth and confirm the adequacy of and assumptions used in the flaw evaluation.

4.0 CONCLUSION

On the basis of the information submitted, the NRC staff concludes that the licensee has provided reasonable assurance of the structural integrity of the degraded weld, NG-E-007, by demonstrating that the projected size of the circumferential indication in the weld will satisfy the requirements of the ASME Code, Section XI, IWB-3600, for the remaining years of the renewed operating license at the Oyster Creek Nuclear Generating Station. The NRC staff further concludes that the three successive examinations will provide additional assurance that the flaw will not challenge the structural integrity of the subject weld.

Contributor: John Tsao

Date: February 16, 2012

February 16, 2012

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION – SUBMITTAL OF ANALYTICAL EVALUATION FOR A REACTOR RECIRCULATION LINE WELD (TAC NO. ME5341)

Dear Mr. Pacilio:

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Please contact me at (301) 415-3100 if you have any questions.

Sincerely,
/RA/
John G. Lamb, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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