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

May 10, 2012

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

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2; BYRON STATION, UNIT NOS. 1 AND 2; CLINTON POWER STATION, UNIT NO. 1; DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3; LASALLE COUNTY STATION, UNITS 1 AND 2; LIMERICK GENERATING STATION, UNITS 1 AND 2; OYSTER CREEK NUCLEAR GENERATING STATION; PEACH BOTTOM ATOMIC POWER STATION, UNITS 2, AND 3; QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2; AND THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - REQUEST TO USE AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE CASE N-789, "ALTERNATIVE REQUIREMENTS FOR PAD REINFORCEMENT OF CLASS 2 AND 3 MODERATE ENERGY CARBON STEEL PIPING FOR RAW WATER SERVICE, SECTION XI, DIVISION 1" (TAC NOS. ME7303, ME7304, ME7305, ME7306, ME7307, ME7308, ME7309, ME7310, ME7311, ME7312, ME7313, ME7314, ME7315, ME7316, ME7317, ME7318, ME7319)

Dear Mr. Pacilio:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated October 7, 2011 (Agencywide Document Access and Management System (ADAMS) Accession No. ML112800669), as supplemented by letters dated November 10, 2011, (ADAMS Accession No. ML113180232) and February 13, 2012, (ADAMS Accession No. ML120440662), Exelon Generation Company, LLC, the licensee, requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) for the repair of Class 2 and 3 Moderate energy carbon steel raw water service system piping at Braidwood Station, Units 1 and 2; Byron Station, Unit Nos. 1 and 2; Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; Lasalle County Station, Units 1 and 2; Limerick Generating Station, Units 1 and 2; Oyster Creek Nuclear Generating Station; Peach Bottom Atomic Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2; and Three Mile Island Nuclear Station, Unit 1.

Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(a)(3)(i), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety. The proposed alternative is based on ASME Code Case N-789, "Alternative Requirements for Pad Reinforcement of Class 2 and 3 Moderate Energy Carbon Steel Piping for Raw Water Service, Section XI, Division 1."

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation (SE), that the proposed alternative fails to meet the regulatory standard of 10 CFR 50.55a(a)(3)(i). However, the staff further concludes that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the ASME, Class 2 and 3, moderate energy carbon steel raw water piping and that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii). Therefore, the NRC staff authorizes the proposed alternative as documented in submittals dated November 10, 2011, and February 13, 2012, for the temporary repair of Class 2 and 3 moderate energy carbon steel raw water service piping at nuclear plants for the 10-year inservice inspection interval as specified in Table 1 of the enclosed SE. The approval of the proposed alternative does not constitute, imply, or infer NRC approval of ASME Code Case N-789. All other ASME Code, Section XI, requirements for which the request was not specified remains applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact Joel S. Wiebe, Senior Project Manager, at (301) 415 6606 or via e-mail at Joel.Wiebe@nrc.gov.

Sincerely,

Jacob I. Zimmerman, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456, STN 50-457,
STN 50-454, STN 50-455, 50-461, 50-237,
50-249, 50-373, 50-374, 50-352, 50-353, 50-219,
50-277, 50-278, 50-254, 50-265, and 50-289

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PROPOSED ALTERNATIVE TO UTILIZE ASME CODE CASE N-789

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNITS 1 AND 2; BYRON STATION, UNIT NOS. 1 AND 2; CLINTON POWER STATION, UNIT 1; DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3; LASALLE COUNTY STATION, UNITS 1 AND 2; LIMERICK GENERATING STATION, UNITS 1 AND 2; OYSTER CREEK NUCLEAR GENERATING STATION; PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3; QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2; AND THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NOS. 50-456, 50-457, 50-454, 50-455, 50-461, 50-237, 50-249, 50-373, 50-374, 50-352, 50-353, 50-219, 50-277, 50-278, 50-254, 50-265, and 50-289

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated October 7, 2011, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112800669), as supplemented by letters dated November 10, 2011, (ADAMS Accession No. ML113180232) and February 13, 2012, (ADAMS Accession No. ML120440662), Exelon Generation Company, LLC, the licensee, requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) for the repair of Class 2 and 3 moderate energy carbon steel raw water service system piping at Exelon fleet of nuclear power plants.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety. The proposed alternative is based on ASME Code Case N-789, "Alternative Requirements for Pad Reinforcement of Class 2 and 3 Moderate Energy Carbon Steel Piping for Raw Water Service, Section XI, Division 1."

Enclosure

2.0 REGULATORY EVALUATION

The licensee requested authorization of an alternative to the requirements Article IWA-4000 of Section XI ASME Code pursuant to 10 CFR 50.55a(a)(3)(i).

Paragraph 10 CFR 50.55a(g)(4) states, in part, that ASME Code Class 1, 2, and 3, components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components."

Paragraph 10 CFR 50.55a(a)(3) states, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if the licensee demonstrates (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above evaluation and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the staff to authorize an alternative proposed by the licensee.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Component Affected

The affected components are all ASME, Class 2 and 3, moderate energy carbon steel raw water piping systems. Raw water is defined as water such as a river, lake, or well, or brackish/salt water used in plant equipment, area coolers, and heat exchangers. In many plants it is referred to as "Service Water." The proposed relief request (RR) applies to Class 2 and 3 moderate energy which is defined as less than or equal to 200 °F (93 degrees C) and less than or equal to 275 psig (1.9 MPa) maximum operating conditions.

3.2 Applicable Code Edition and Addenda

Table 1—Applicable Plants with Associated 10-Year ISI Intervals and ASME Code Editions

PLANT	ISI INTERVAL	ASME CODE EDITION	START	END
Braidwood Station, Units 1 and 2	Third	2001 Edition, through 2003 Addenda	July 29, 2008 October 17, 2008	July 28, 2018 October 16, 2018
Byron Station, Units Nos. 1 and 2	Third	2001 Edition, through 2003 Addenda	January 16, 2006	July 15, 2016
Clinton Power Station, Unit No. 1	Third	2004 Edition	July 1, 2010	June 30, 2020

PLANT	ISI INTERVAL	ASME CODE EDITION	START	END
Dresden Nuclear Power Station, Units 2 and 3	Fourth	1995 Edition, through 1996 Addenda	January 20, 2003	January 19, 2013
Dresden Nuclear Power Station, Units 2 and 3	Fifth	2007 Edition, 2008 Addenda	January 20, 2013	January 19, 2023
LaSalle County Stations, Units 1 and 2	Third	2001 Edition, through 2003 Addenda	October 1, 2007	September 30, 2017
Limerick Generating Station, Units 1 and 2	Third	2001 Edition, through 2003 Addenda	February 1, 2007	January 31, 2017
Oyster Creek Nuclear Generating Station	Fourth	1995 Edition, through 1996 Addenda	October 15, 2002	October 14, 2012
Oyster Creek Nuclear Generating Station	Fifth	2007 Edition, 2008 Addenda	October 15, 2012	October 14, 2022
Peach Bottom Atomic Power Station, Units 2 and 3	Fourth	2001 Edition, through 2003 Addenda	November 5, 2008	November 4, 2018
Quad Cities Nuclear Power Station, Units 1 and 2	Fourth	1995 Edition, through 1996 Addenda	March 10, 2003	April 1, 2013
Quad Cities Nuclear Power Station, Units 1 and 2	Fifth	2007 Edition, 2008 Addenda	April 2, 2013	April 1, 2023
Three Mile Island Nuclear Station, Unit 1	Fourth	2004 Edition	April 20, 2011	April 19, 2022

3.3 Applicable Code Requirement

As shown in Table 1 above, the applicable codes are ASME Code, Section XI, IWA-4400, of the 1995 Edition through 1996 Addenda, 2001 Edition through 2003 Addenda, 2004 Edition, and 2007 Edition though 2008 Addenda. These Code editions and addenda provide requirements for welding, brazing, metal removal, and installation of repair/replacement activities.

3.4 Reason for Relief Request

The licensee proposed an alternative from the requirement for replacement or internal weld repair of wall thinning conditions resulting from degradation in Class 2 and 3 moderate energy carbon steel raw water piping systems in accordance with IWA-4000. The licensee explained that such degradation may be the result of mechanisms such as erosion, corrosion, cavitation, and pitting, but excluded are conditions involving flow-accelerated corrosion (FAC), corrosion-assisted cracking, or any other form of cracking. IWA-4000 requires repair or replacement in accordance with the Owner's Requirements and the original or later Construction Code. Other alternative repair or evaluation methods are not always practicable because of wall thinness and/or moisture issues. The licensee stated that this RR is to permit installation of a technically sound temporary repair to provide adequate time for evaluation, design, material procurement, planning and scheduling of appropriate permanent repair or replacement of the defective piping, considering the impact on system availability, maintenance rule applicability, and availability of replacement materials.

3.5 Proposed Alternative and Basis for Use

The licensee proposes to implement the requirements of ASME Code Case N-789 as a temporary repair of degradation in Class 2 and 3 moderate energy raw water piping systems resulting from mechanisms such as erosion, corrosion, cavitation, or pitting, but excluding conditions involving FAC, corrosion-assisted cracking, or any other form of cracking. The licensee stated that these types of defects are typically identified by small leaks in the piping system or by pre-emptive, non-code required examinations performed to monitor the degradation mechanisms.

The alternative repair technique described in Code Case N-789, which is part of the proposed RR, involves welding a metal reinforcing pad to the exterior of the piping system to reinforce the degraded area and restore pressure integrity. The licensee will use this repair technique when it determines that the temporary repair method is suitable for the particular defect or degradation.

The licensee stated that the Code Case requires that the cause of the degradation be determined, and that the extent and rate of degradation in the piping be evaluated to ensure that there are no other unacceptable locations within the surrounding area that could affect the integrity of the repaired piping. The area of evaluation will be dependent on the degradation mechanism present. The licensee will perform a baseline thickness examination for a completed structural pad, attachment welds, and surrounding area, followed by monthly thickness monitoring for the first three months, with subsequent frequency based on the results of this monitoring, but at a minimum of quarterly. Areas containing pressure pads shall be visually observed at least once per month to monitor for evidence of leakage. If the areas containing pressure pads are not accessible for direct observation, then monitoring will be accomplished by visual assessment of surrounding areas or ground surface areas above pressure pads on buried piping, or monitoring of leakage collection systems, if available.

The licensee stated that the repair will be considered to have a maximum service life at the time until the next refueling outage, when a permanent repair or replacement must be performed. The proposed relief request specifies additional requirements for design of reinforcing pads, installation, examination, pressure testing, and inservice monitoring the same as in Code Case N-789.

The licensee stated that Code Case N-789 was approved by the ASME Board on Nuclear Codes and Standards on June 25, 2011; however, it has not been incorporated into NRC

Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI Division 1," and thus is not available for application at nuclear power plants without specific NRC approval.

3.6 Duration of Proposed Alternative

The licensee proposed to apply Code Case N-789, as needed, for the remainder of each plant's 10-year ISI interval as specified in Table 1 above. Any reinforcing pads installed before the end of the 10-year ISI interval will be removed during the next refueling outage, even if that refueling outage occurs after the end of the 10-year ISI interval.

3.7 Staff Evaluation

Although the NRC has not approved Code Case N-789, 10 CFR 50.55a provides provisions for licensees to submit, for NRC review and approval, alternatives to ASME Code Section XI requirements regarding repair or replacement of degraded pipes. The staff evaluates how the proposed alternative will provide reasonable assurance that the structural integrity of the affected components will be maintained. The staff reviewed the submittals dated November 10, 2011, and February 13, 2012, as they contain the latest revised proposed alternative.

Code Case N-789 as documented in the proposed alternative provides detailed requirements for the initial evaluation, design, installation, examination, pressure testing, and inservice monitoring. The staff finds that the proposed alternative is acceptable except additional clarifications are needed as discussed below.

Applicable Duration

Paragraphs 1(e) and 8(d) of Code Case N-789 state that reinforcing pads, including those installed during a refueling outage, shall not remain in service beyond the end of the next refueling outage. The staff asked the licensee to confirm that both the pressure pad and structural pad are considered as the reinforcing pads and that both pads will not remain in service beyond the end of the next refueling outage. The NRC staff asked the licensee to define the "next refueling outage" if the repair is performed in mid-cycle (e.g., the pad is installed one month before the start of the next refueling outage).

By letter dated February 13, 2012, the licensee confirmed that both pressure pads and structural pads are considered as reinforcing pads. The licensee clarified that neither the pressure pad nor the structural pad may remain in service beyond the end of the next refueling outage after they are installed, unless specific regulatory relief is obtained. If the repair is performed in mid-cycle (e.g., one month before the scheduled refueling outage), the reinforcing pad would be removed no later than the upcoming refueling outage (e.g., in one month) unless specific regulatory relief is obtained. The staff finds that the licensee has clarified the applicable duration of the pressure pad and structural pad.

The NRC staff noted that some piping systems could not be repaired during refueling outages because they are required to be functional during that time. These pipes can be repaired when the unit is operating and the piping systems are on the standby mode. In this case, the service life of the reinforcing pad to the next refueling outage would not be applicable because the installed pad cannot be removed during the next refueling outage. By letter dated February 13, 2012, the licensee responded that in the case cited, the reinforcing pad would need to be removed prior to the conclusion of the next scheduled refueling outage after it was installed. The licensee stated that a similar situation exists with common cooling lines that require a dual-unit outage to remove from service. The licensee stated that in this case, specific regulatory approval would need to be obtained in order to defer removal of a pad beyond the next upcoming refueling outage of either unit. The staff finds that the licensee has clarified the duration for these special case piping and that the installed pad will be removed earlier than the next refueling outage.

The licensee requested NRC approval of the proposed alternative for specific 10-year ISI interval for each unit as shown in the table above. Once the ISI interval reaches the specific end date, the proposed alternative becomes null and void. The staff asked the licensee whether the reinforcing pad will be removed when the end date of an ISI interval falls in mid-cycle in lieu of in a scheduled refueling outage. By letter dated February 13, 2012, the licensee clarified that the end date of the 10-year ISI interval will not necessarily coincide with scheduled refueling outages for each unit. Installation of reinforcing pads in accordance with this RR cannot take place after the end of the 10-year ISI interval for the unit. The licensee further stated that any reinforcing pads installed before the end of the 10-year ISI interval will be removed during the next refueling outage, even if that refueling outage occurs after the end of the 10-year ISI interval as shown in the revised Section 6.0 of the proposed alternative. The reinforcing pad is designed to support a maximum one cycle of operation from one refueling outage to the next refueling outage. The NRC staff finds that absent defect or degradation, the reinforcing pad is acceptable to remain in service beyond the end date of the 10-year ISI interval if that interval end date falls in the mid-cycle and if the pad is removed in the next scheduled refueling outage. Therefore, for this situation, absent defects or degradation, the staff approves the duration of the relief request up to the end of the first refueling outage following the end of the 10-year ISI interval as specified in Table 1 of this safety evaluation.

As shown in Table 1 above, the Dresden Nuclear Plant, Units 2 and 3, Oyster Creek Nuclear Generating Station, and Quad Cities Nuclear Power Station, Units 1 and 2, requested that the proposed RR be applicable to the current (4th) and next (5th) ISI interval because the current ISI interval is about to end for these units. The staff finds that the proposed alternative may be applicable for two consecutive 10-year ISI intervals when the end of the current ISI interval is less than one year away from the date of the NRC approval of the RR. This is because the maximum period of duration for these three plants is about 10.8 years. The staff finds that this additional extension does not significantly affect any technical or regulatory aspect of the RR.

<u>Design</u>

Section 3.1(1) of Code Case N-789 specifies that "...[t]he pressure pads are designed to retain... full structural integrity... assuming a corrosion rate of either two times the actual measurement corrosion rate in that location or four times the estimated maximum corrosion rate for the system..." The staff asked the licensee to clarify (a) how the actual measured corrosion rate will be obtained, (b) whether the maximum (worst) corrosion rate will be used in the pressure pads design, (c) how the corrosion rate is used to design the reinforcing pad, and (d) the acceptance criteria for the full structural integrity of piping.

By letter dated February 13, 2012, the licensee clarified that to measure the corrosion rate at a specific location requires mapping the corroded area (using ultrasonic equipment) a minimum of two times with a distinct time interval between each mapping. Specific time intervals are not defined as they depend on the rate and extent of corrosion; but they would need to be sufficient to measure discernible changes in thickness. The licensee explained that measured point corrosion rates would equal the change in thickness at various points within the mapped area divided by the time interval, revealing a predictable change in configuration of the area over time. The licensee would apply this rate of change in configuration (two-fold for pressure pads) in the design to establish the minimum size of the reinforcing pad.

The licensee further stated that if the actual rate of corrosion is measured at a specific location, then that is what should be applied for predicting further degradation at that location. However, the Code Case imposes a conservative factor of safety of two, requiring that value to be doubled when designing and installing a pressure pad.

According to the licensee, if a repair must be performed without sufficient time to determine the actual rate of corrosion, then a pressure pad design must apply the worst-case corrosion rate observed for the system, plus apply an even more conservative safety factor of four. The staff notes that in this situation the licensee needs to calculate the estimated maximum corrosion rate for the system based on the same degradation mechanism that occurs at the degraded pipe location.

The licensee noted that establishing the corrosion rate defines the changes in configuration of a degraded area over time. Using these defined rates of degradation, one can predict areas of wall thickness that could be less than the minimum thickness required by design by the time of the next refueling outage. The design of reinforcing pads considers such areas the same as if they were holes drilled or cut in the pipe. The reinforcing pad is then designed as a closure for that size hole, using design methodology of the applicable Construction Code (e.g., as a reinforced opening).

The licensee explained that in the context of Code Case N-789, paragraphs 3.1 (a)(1) and (2), "full structural integrity" means the piping maintains full capability to withstand structural (mechanical) loading for which it is designed without need for additional support or reinforcement. Small areas of corrosion can, and do, result in thinning and leakage without impacting the ability of the piping to maintain its structural capabilities. These situations are candidates for pressure pads which provide no added structural support or reinforcement, only pressure retention.

The staff questioned how the actual measured and predicted (estimated or projected) corrosion rates are derived for the design of the pressure pad and structural pad. As the licensee stated above, the actual measured corrosion rate is calculated by dividing the difference in pipe wall thickness measured at two different dates by the time interval. However, Code Case N-789 does not provide specific requirements on how the corrosion rate should be calculated and when the pipe thickness should be measured. The staff notes that the pipe wall thickness of the raw service water piping is not required to be measured frequently; therefore, the exact time of onset of the corrosion occurring cannot be known. A licensee may select a time interval that is fairly long such that the corrosion rate becomes small, or select a location that may not represent the worst degradation. This would affect the accuracy of the actual measured corrosion rate. In addition, there may be uncertainty in the measurement itself.

The licensee's predicted corrosion rate is based on multiplying a factor of 2 on the actual measured corrosion rate of the localized area or a factor 4 on the corrosion rate of the piping system. These factors do not provide staff with reasonable assurance that the predicted corrosion rate used in the pad design is bounding. However, as a compensatory measure, the proposed alternative and the code case do require inservice monitoring to ensure the structural integrity of the repaired pipe. In addition, the proposed repair is limited to a maximum duration of one operating cycle. This relatively short duration of application should limit the degradation. Should the actual corrosion rate exceed the projected corrosion rate during the operating cycle and a leak develop at or around the installed pad, the inservice monitoring as discussed below will be able to detect such leakage and the operator will be able to take corrective actions. Although the staff has concerns regarding the corrosion rate used in the pad design, the staff finds that the inservice monitoring will verify and provide reasonable assurance that the structural integrity and leakage integrity will be maintained during the one-cycle of application.

Sections 3.2 and 6 of Code Case N-789 stipulate the use of the Construction Code or ASME Code, Section III. By letter dated February 13, 2012, the licensee clarified that reconciliation and use of editions and addenda of ASME Section III will be in accordance with ASME Section XI, IWA-4220. Only editions and addenda of ASME Section III that have been accepted by 10 CFR 50.55a may be used. The Code of Record for the specific 10-year ISI interval at each nuclear unit covered under the proposed alternative will be used when applying the various IWA paragraphs unless specific regulatory relief is approved. The staff finds that the proposed alternative follows the appropriate editions and addenda of the ASME Code; therefore, it is acceptable.

Installation

Section 4 of Code Case N-789 discusses requirements for installing the reinforcing pad on water-backed piping. The staff asked the licensee to discuss whether N-789 permits a reinforcing pad be installed on a leaking area of the pipe and how welding will be conducted on a leaking pipe to minimize fabrication defects (e.g., porosity and hydrogen cracking) in the weld. By letter dated February 13, 2012, the licensee responded that Code Case N-789 does permit reinforcing pads to be installed on a leaking area of the pipe; however, the Code Case does not permit welding on wet surfaces. Therefore, a gasket or other sealing material will be applied to prevent moisture from encroaching upon the weld area; refer to paragraph 3.2(1) of Code Case N-789. These pads can be applied to pressurized systems by clamping the pad with gasket against the pipe, and then removing residual moisture by heating prior to welding. The staff finds that the licensee has specific requirements for installing the reinforcing pad on a leaking pipe; therefore, the proposed installation on leaking pipe is acceptable.

Inservice Monitoring

Section 8 of Code Case N-789 stipulates inservice monitoring requirements for the structural pad, but not the pressure pad. In Section 5 of the proposed alternative, the licensee stated that for the pressure pads, inservice monitoring will not be required because the design of pressure pads conservatively assumes two times the actual measured corrosion rate or four times if using an estimated rate. The staff asked the licensee to justify why using either two times the actual measured corrosion rate or four times the estimated maximum corrosion rate for the piping system is adequate to ensure that the pressure pad will not leak or lose structural integrity. The staff noted that even if a conservative corrosion rate is used in the pad design, the licensee needs to justify why the pressure pad does not need inservice monitoring.

By letter dated February 13, 2012, the licensee responded that the ASME Code committee has determined that a factor of safety of two (or four) for pressure pads is very conservative, and ensures excess design margin until the following refueling outage. This is based on several factors, including the fact that pressure pads serve no structural purpose and are only installed for temporary leak prevention. Further, the Code Case is restricted to raw water systems where the primary source of corrosion is microbiological or under-deposit in nature. The licensee stated that these types of corrosion are not expected to accelerate during one refueling cycle by a factor of two for measured rates, or by a factor of four times the worst rate in the system. Also, the consequences of potential leakage at a pressure pad are considered small, since it would most likely begin as pin hole leakage at an attachment weld placed on structurally sound base metal.

The licensee further stated that the degradation beneath a pressure pad and its attachment welds cannot be monitored, although such areas can be monitored in structural pads. This is because the configuration of pressure pad attachment welds is not conducive to ultrasonic examination or thickness measurement of the material beneath the attachment weld. Rates of raw water corrosion do not increase rapidly during a single refueling cycle; and with the conservatism built into the design, and resultant increased size of the pad, there is low likelihood that the corrosion will expand to the structurally-sound attachment weld over that period of time. The licensee explained that this is the reason that the Code Case does not require inspection of pressure pads.

However, based on the staff's concerns, the licensee revised Section 5 of the proposed alternative to require inspection of the pressure pad as follows:

> Areas containing pressure pads shall be visually observed at least once per month to monitor for evidence of leakage. If the areas containing pressure pads are not accessible for direct observation, then monitoring will be accomplished by visual assessment of surrounding areas or ground surface areas above pressure pads on buried piping, or monitoring of leakage collection systems, if available.

As discussed above, the staff believes that aggressive corrosion rates may exist in the raw water system piping and may exceed the predicted corrosion rate with a factor of two or four times the measured corrosion rate. However, the required periodic inservice monitoring of the reinforcing pad will be able to verify the corrosion rate and thus ensure the structural integrity of the repaired pipe. The staff finds acceptable that the licensee will perform inservice monitoring for the pressure pad.

The NRC staff asked the licensee to discuss whether the proposed alternative will be applied to buried piping and how the inspection will be performed on the buried pipes after the reinforcing pad is installed. By letter dated February 13, 2012, the licensee clarified that when used on buried piping, the area of structural pads will need to be accessible for the examinations required by the Code Case, which could necessitate installation of removable barriers at the repair location in lieu of backfilling the pipe at that location. For pressure pads, the monitoring will be based on visual examinations as discussed above. The staff finds that the licensee has adequately addressed the repair and inservice monitoring of the buried pipe.

Compliance with Regulation

As stated above, the licensee submitted the proposed alternative based on 10 CFR 50.55a(a)(3)(i). The licensee stated that the RR is to permit installation of a technically sound temporary repair to provide adequate time for evaluation, design, material procurement, planning and scheduling of appropriate permanent repair or replacement of the defective piping, considering the impact on system availability, maintenance rule applicability, and availability of replacement materials.

To qualify for 10 CFR 50.55a(a)(3)(i), a proposed alternative needs to provide an acceptable level of quality and safety. In most, but not all instances, "acceptable" is interpreted by the staff to mean "equivalent." In the present case, primarily due to the uncertainties associated with predicted corrosion rates, the staff finds that the proposed alternative fails to meet the regulatory standard of 10 CFR 50.55a(a)(3)(i). Alternatively, the staff notes that the licensee could have proposed its alternative under 10 CFR 50.55a(a)(3)(ii). 10 CFR 50.55a(a)(3)(ii) requires that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety

The NRC staff notes that making permanent ASME Code compliant repairs may require piping to be removed from service, which may also require a plant shutdown. The staff considers this to be a hardship. The staff also notes that due primarily to the level of ISIs, making code compliant repairs does not result in an increase in the level of quality and safety commensurate with the hardship. The staff, therefore, finds that the licensees proposal is consistent with the requirements of 10 CFR 50.55a(a)(3)(ii).

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the ASME, Class 2 and 3, moderate energy carbon steel raw water piping and that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii). Therefore, the NRC staff authorizes the proposed alternative as documented in submittals dated November 10, 2011, and February 13, 2012, for the temporary repair of Class 2 and 3 moderate energy carbon steel raw water service piping at nuclear plants for the 10-year ISI interval as specified in Table 1 of this safety evaluation.

The approval of the proposed alternative does not constitute, imply, or infer NRC approval of ASME Code Case N-789. All other ASME Section XI requirements for which relief was not specifically requested and authorized by the NRC staff will remain applicable including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: John Tsao

Date of issuance: May 10, 2012

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the proposed alternative fails to meet the regulatory standard of 10 CFR 50.55a(a)(3)(i). However, the staff further concludes that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the ASME, Class 2 and 3, moderate energy carbon steel raw water piping and that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii). Therefore, the NRC staff authorizes the proposed alternative as documented in submittals dated November 10, 2011, and February 13, 2012, for the temporary repair of Class 2 and 3 moderate energy carbon steel raw water service piping at nuclear plants for the 10-year inservice inspection interval as specified in Table 1 of the enclosed SE. The approval of the proposed alternative does not constitute, imply, or infer NRC approval of ASME Code Case N-789. All other ASME Code, Section XI, requirements for which the request was not specified remains applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact Joel S. Wiebe, Sr. Project Manager, at (301) 415 6606 or via e-mail at Joel.Wiebe@nrc.gov.

Sincerely,

/ ***RA*** /

Jacob I. Zimmerman, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456, STN 50-457, STN 50-454, STN 50-455, 50-461, 50-237, 50-249, 50-373, 50-374, 50-352, 50-353, 50-219, 50-277, 50-278, 50-254, 50-265, and 50-289

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

DISTRIBUTION:
PUBLIC LPL3-2 R/F LPL1-2 R/F
RidsOgcRp Resource
RidsNrrPMBraidwood Resource
RidsNrrPMClinton Resource
RidsNrrPMOysterCreek Resource
RidsNrrLASRohrer Resource
RidsNrrLAABaxter Resource
RidsNrrPMDresden Resource
RidsNrrPMPeachBottom Resource
RidsNrrPMLimerick Resource
RidsNrrPMThreeMileIsland Resource
RidsRgn3MailCenter Resource
RidsNrrDorlDpr Resource
RidsRgn1MailCenter Resource
RidsNrrPMByron Resource
RidsNrrLaSalle Resource
RidsNrrPMQuadCities Resource
RidsAcrsAcnw&mMailCenter Resource
RidsNrrDorlLpl3-2 Resource
RidsNrrDorlLpl1-2 Resource

ADAMS Accession No. ML12121A637 *via email

OFFICE	LPL3-2/PM	LPL3-2/LA	EPNB/BC* (Acting)	LPL3-2/BC	LPL3-2/PM
NAME	JWiebe	SRohrer	DAlley	JZimmerman	JWiebe
DATE	05/ 8 /12	05/ 7 /12	04/17/12	05/ 10 /12	05/ 10 /12

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