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

June 25, 2012

LICENSEE: Exelon Generation Company, LLC

FACILITY: Limerick Generating Station

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE CALL HELD ON MAY 16, 2012, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND EXELON GENERATION COMPANY, LLC, CONCERNING REQUESTS FOR ADDITIONAL INFORMATION PERTAINING TO THE LIMERICK GENERATING STATION, LICENSE RENEWAL APPLICATION (TAC. NOS. ME6555 AND ME6556)

The U.S. Nuclear Regulatory Commission (NRC or the staff) and representatives of Exelon Generation Company, LLC held a telephone conference call on May 16, 2012, to discuss and clarify the staff's requests for additional information (RAIs) concerning the Limerick Generating Station license renewal application. The telephone conference call was useful in clarifying the intent of the staff's RAIs.

Enclosure 1 provides a listing of the participants and Enclosure 2 contains a listing of the RAIs discussed with the applicant, including a brief description on the status of the items.

The applicant had an opportunity to comment on this summary.

Robert F. Kuntz, Senior Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353

Enclosures:
1. List of Participants
2. List of Requests for Additional Information

cc w/encls: Listserv

TELEPHONE CONFERENCE CALL
LIMERICK GENERATING STATION
LICENSE RENEWAL APPLICATION

LIST OF PARTICIPANTS
May 16, 2012

PARTICIPANTS	AFFILIATIONS
Robert Kuntz	Nuclear Regulatory Commission (NRC)
Seung Min	NRC
John Wise	NRC
Ata Istar	NRC
James Medoff	NRC
On Yee	NRC
Matthew Yoder	NRC
Aloysius Obodoako	NRC
John Hufnagel	Exelon Generation Company, LLC (Exelon)
Gene Kelly	Exelon
Al Fulvio	Exelon
Jim Jordan	Exelon
Mary Kowalski	Exelon
Mike Guthrie	Exelon
Dave Clohecy	Exelon
Mark Miller	Exelon
Ron Hess	Exelon

DRAI 3.1.1.38-1.1

Background

The response to RAI 3.1.1-38 provided by letter dated February 16, 2012, addressed how the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program will manage loss of fracture toughness due to thermal aging embrittlement of the cast austenitic stainless steel (CASS) pump casings described in license renewal application (LRA) item 3.1.1-38 and Table 3.3.2-21. The response indicated that the program manages the aging effect by implementing opportunistic visual inspections for evidence of cracking in the CASS pump casings of the reactor water cleanup (RWCU) system.

Issue

For metallic piping components or elements whose internal surfaces are not lined or coated with polymeric or ceramic materials, the "scope of program" element of GALL Report AMP XI.M38 states that the program is used only for the detection of mechanisms that can lead to loss of material in the components. GALL Report AMP XI.M38 does not define any criteria on how visual inspections can be used to manage drops in the fracture toughness property of CASS piping components or elements, as performed only on a qualitative periodic surveillance or maintenance schedule basis.

The basis for using the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program to manage loss of fracture toughness of the CASS RWCU pump casings has not been identified as an exception to the "scope of program" element. The basis also does not establish exactly which type of visual inspections and inspection frequency will be performed to detect crack indications as an indirect measure for determining whether loss of fracture toughness is occurring in the RWCU pump casings. In addition, the program does not address how the visual inspection method and frequency will be capable of detecting and resolving flaw sizes that are less than the limiting lower bound critical flaw size for the RWCU pump casings, as assessed for limiting thermal aging embrittlement conditions. Thus, the staff needs additional information for concluding that the program (LRA Section B.2.1.26) will be capable of managing thermal aging embrittlement of the CASS RWCU pump casings.

Request

1. Justify why opportunistic inspections and inspection methods are sufficient to manage loss of fracture toughness of the pump casings through timely detection of a flaw before it grows to the size that can lead to rapid unstable crack propagation due to thermal aging embrittlement.

   As part of the response, clarify which type of visual inspection method (e.g., EVT-1, VT-1 or VT-3) will be used to detect flaws in the components. In addition, justify why the performance of these visual inspections on an opportunistic basis is considered to be capable of detecting and resolving a flaw prior to unstable crack propagation in the components (e.g., the basis for concluding that the visual inspection method and frequency will be capable of detecting and resolving a flaw smaller than the critical crack size of the component under reduced fracture toughness conditions, as induced by thermal aging embrittlement).

2. Provide an update to the updated final safety analysis report (UFSAR) supplement to reflect an enhancement to the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program to manage loss of fracture toughness of pump casings.

Discussion: The applicant indicated that the request is clear. This DRAI will be sent as a formal RAI.

DRAI 3.5.2.11-1.1

Background

The refueling bellows assembly accommodates the movements of the reactor vessel caused by operating temperature variations and seismic activities as well as prevents leakage from the reactor well during refueling operations. The NRC issued NUREG/CR-6726 "Aging Management and Performance of Stainless Steel Bellows in Nuclear Power Plants" in May 2001, summarizing information on how to evaluate bellows for age-related degradations including aging mechanism results in loss of bellows functionality during the current operations or for the period of extended operations (PEO).

The LRA states that the refueling bellows assembly is evaluated within the license renewal Primary Containment Structure. The LRA Table 3.5.2-11 "Primary Containment" identified the stainless steel portion of the refueling bellows assembly as subject to loss of material in the treated water environment, and referenced line item III.A5.T-14 from the GALL Report (NUREG-1801), which states that loss of material and cracking of spent fuel pool liner components should be managed with the Water Chemistry program and monitoring of the spent fuel pool water level and leakage from the leak chase channels.

The response to RAI 3.5.2.11-1, provided by letter dated March 13, 2012, revised LRA Table 3.5.4-11 and deleted the aging management review (AMR) items associated with the treated water environment, because the refueling bellows assembly is normally exposed to air and is exposed to treated water only during refueling outages.

The SRP-LR, Section A.1 "Aging Management Review," Subsection A.1.2.1 "Applicable Aging Effects," item 7 states that the applicable aging effects to be considered for license renewal include those that could result from normal plant operation, including plant/system operating transients and plant shutdown.

The program description of SRP-LR Section XI.M2 "Water Chemistry" states that the water chemistry program may not be effective in low flow or stagnant flow areas. Therefore, a verification of the effectiveness of the chemistry control program should be conducted of selected components at susceptible locations in the system to ensure that significant degradation is not occurring and the component's intended function is maintained during the PEO.

Issue

LRA Table 3.5.2-11 was revised in response to RAI 3.5.2.11-1 by deleting the line items associated with stainless steel refueling bellows assemblies components that are exposed to treated water during refueling outages; however, the SRP-LR states that the management of aging should consider the environment during plant shutdown. Considering the guidance in GALL Report item III.A5.T-14, the staff believes that the stainless steel refueling bellows components exposed to treated water should be managed for both cracking due to stress corrosion cracking and loss of material due to pitting and crevice corrosion.

Requests

1. Provide a summary description of the actions that will be taken to manage the aging effects on refueling bellows in a treated water environment. If no actions are planned, provide the basis for not including AMR items to address management of cracking due to stress corrosion cracking and loss of material due to pitting and crevice corrosion for the stainless steel refueling bellows components exposed to treated water, as described in GALL Report item III.A5.T-14 (which was referenced in the original LRA Table 3.5.2-11 for the stainless steel bellows).

2. If the program selected to manage the effects of aging is water chemistry control, as indicated in the original application, provide a summary description of the actions that will be taken to verify the effectiveness of the water chemistry control program to ensure that significant degradation does not occur. If no verification actions are planned, provide the basis for not considering verification of the susceptible locations of stainless steel portion of the refueling bellows assembly in a low flow or stagnant water.

Discussion: The applicant indicated that the request is clear. This DRAI will be sent as a formal RAI.

RAI 4.2.1-1

Background

LRA Section 4.2.1 provides the basis for calculating the 57 effective full power years (EFPY) neutron fluence values that are time-dependent inputs to the neutron irradiation embrittlement time-limited aging analysis (TLAA) for the Limerick Generating Station (LGS) reactor pressure vessel (RPV) beltline shell, nozzle and weld components. Fifty-seven EFPY is the expected value associated with LGS, Units 1 and 2 power operations through the proposed extended period of operation (60 years). The neutron fluence values for 60 years of operation calculated in the fluence TLAAs are referenced and used in the following sections of the LRA: Section 4.2.2, "Upper Shelf Energy"; Section 4.2.3, "Adjusted Reference Temperature"; Section 4.2.4, "Pressure – Temperature Limits"; Section 4.2.5, "Axial Weld Inspection"; Section 4.2.6, "Circumferential Weld Inspection," and Section 4.2.7, "Reactor Pressure Vessel Reflood Thermal [Analysis]."

LRA Section 4.2.1 indicates that RAMA Code methodology was used to derive the 57 EFPY neutron fluence values for high energy neutrons with kinetic energies greater than 1.0 MeV (E > 1.0 MeV) and that the RAMA Code methodology conforms to the NRC's recommended regulatory position in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [March 2001], for applying neutron fluence methodologies to these type of TLAA calculations.

Issue

LRA Section 4.2.1 references use of the RAMA Code to calculate TLAAs such as the P-T Limits. However, the applicant's CLB (Technical Specification 3.4.6.1) references a different methodology to calculate its P-T Limits—General Electric (GE) Company Report No. NEDC-32983P-A. The LRA does not provide an explanation of this discrepancy.

Request

Discuss and provide the basis for referencing the use of RAMA Code methodology to calculate neutron fluence for TLAAs in the LRA, when such methodology has not been previously identified (e.g., in accordance with 10 CFR 50.90 or 50.59) as part of the LGS, Units 1 and 2 CLB.

As part of the response, also clearly identify, by document reference number, title, and date, all neutron fluence methodologies that are being adopted in the LGS CLB to conform with the regulatory position in RG 1.190 and clarify whether the neutron fluence methodologies adopted in the CLB have been endorsed for use by the NRC. As part of the response, clarify how any and all relevant limitations and conditions that have been placed on implementation of the fluence calculation methodologies adopted in the LGS CLB have been addressed.

Discussion: The applicant indicated that the request is clear. This DRAI will be sent as a formal RAI.

RAI 4.6.6-1.1

Background

The response to RAI 4.6.6-1, provided by letter dated January 24, 2012, stated that the TLAA for the jet pump restrainer bracket pad repair clamps had been re-evaluated to show that the fluence value used to determine the loss of preload in the 40-year design analysis will not be exceeded during the period of extended operation. The response stated that the fluence value of 1.0 E+19 n/cm^2 was used in the design analysis to determine loss of preload, and this value is five percent higher than the fluence value of 9.5 E+18 n/cm^2 calculated for a 40-year service life. The response also changed the disposition of the TLAA to 10 CFR 54.21(c)(1)(i), that the analysis remains valid for the period of extended operation.

Issue

In accordance with 10 CFR 54.21(c)(1)(i), an applicant must demonstrate that the analysis remains valid for the period of extended operation. The response to RAI 4.6.6-1 stated that the design of the jet pump restrainer bracket pad repair clamps was based on the 9.5 E+18 n/cm^2 fluence value calculated for a 40-year service life; however, the response did not provide further demonstration to show that the fluence projected through the period of extended operation will not exceed this value.

Request

Provide an explanation to demonstrate that the design fluence value of 9.5 E+18 n/cm^2 will not be exceeded during the period of extended operation.

Discussion: The applicant clarified that the values provided in its original response to 4.6.6-1 are from its design analysis for the jet pump restrainer bracket pad repair clamps. The applicant provided an explanation of how these fluence values were calculated in the design analysis. This DRAI will not be sent as a formal RAI.

RAI B.2.1.28-3

Background

The response to RAI B.2.1.28-2, provided by letter dated April 27, 2012, stated that plant documentation on fuel pool inventory was reviewed, and it was determined that the actual number of cycles that the coupons were completely surrounded by freshly discharged fuel for LGS, Unit 2 is five (first five cycles following rack installation), and for LGS, Unit 1 is two (first two cycles following rack installation). The coupons in each spent fuel pool (SFP) were subsequently relocated to a representative location. The response then stated that surrounding the test coupons by eight freshly discharged fuel bundles for five future cycles (ending 2024 and 2023 for LGS, Units 1 and 2, respectively) will ensure that the test coupons will be leading indicators for other individual fuel storage cells.

The response stated that an analysis was performed on the spent fuel pool inventory relative to the test coupons to predict when the exposure of the coupons to freshly discharged fuel would be equal to the exposure of the limiting storage cells to freshly discharged fuel. It was concluded that the coupons in the SFP will be exposed to the same number of freshly discharged fuel assemblies as the theoretical worst case cell in 2020 for LGS, Unit 1 and 2021 for LGS, Unit 2.

Issue

Although the response provided a path forward for coupon exposure such that the coupons would be the leading indicator for other individual fuel storage cells for LGS, Units 1 and 2, it did not provide the relative cumulative exposure of the coupons compared to the most limiting storage cell.

Request

Discuss the relative cumulative exposure for the coupons compared to the most limiting storage cell at the end of the proposed five cycles of exposure to freshly discharged fuel. Also, discuss the impact of an accelerated exposure to freshly discharged fuel versus a long term exposure to representative conditions. Is the degradation mechanism understood well enough to say that cumulative exposure is the primary driver?

Discussion: The applicant indicated that the request is clear. This DRAI will be sent as a formal RAI.

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NAME	YEdmonds	RKuntz	DMorey	RKuntz
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