Exelon. Nuclear

Michael P. Gallagher
Vice President
License Renewal

Exelon Nuclear
200 Exelon Way
Kennett Square, PA 19348

Telephone 610.765.5958
Fax 610.765.5658
www.exeloncorp.com
michaelp.gallagher@exeloncorp.com

10 CFR 50
10 CFR 51
10 CFR 54

May 31, 2012

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Limerick Generating Station, Units 1 and 2
Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Subject: Responses to NRC Requests for Additional Information, dated May 18, 2012, related to the Limerick Generating Station License Renewal Application

Reference: 1. Exelon Generation Company, LLC letter from Michael P. Gallagher to NRC Document Control Desk, "Application for Renewed Operating Licenses", dated June 22, 2011
2. Letter from Robert F. Kuntz (NRC) to Michael P. Gallagher (Exelon), "Requests for Additional Information for the review of the Limerick Generating Station, Units 1 and 2, License Renewal Application (TAC Nos. ME6555, ME6556)", dated May 18, 2012

In the Reference 1 letter, Exelon Generation Company, LLC (Exelon) submitted the License Renewal Application (LRA) for the Limerick Generating Station, Units 1 and 2 (LGS). In the Reference 2 letter, the NRC requested additional information to support the staffs' review of the LRA.

Enclosed are the responses to these requests for additional information.

This letter and its enclosure contain no new or revised regulatory commitments.

If you have any questions, please contact Mr. Al Fulvio, Manager, Exelon License Renewal, at 610-765-5936.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 5-31-2012

Respectfully,

Michael P. Gallagher

Michael P. Gallagher
Vice President - License Renewal Projects
Exelon Generation Company, LLC

Enclosure: Responses to Requests for Additional Information

cc: Regional Administrator – NRC Region I
NRC Project Manager (Safety Review), NRR-DLR
NRC Project Manager (Environmental Review), NRR-DLR
NRC Project Manager, NRR- DORL Limerick Generating Station
NRC Senior Resident Inspector, Limerick Generating Station
R. R. Janati, Commonwealth of Pennsylvania

Enclosure

Responses to Requests for Additional Information related to various sections of the LGS License Renewal Application (LRA)

RAI 3.1.1.38-1.1
RAI 3.5.2.11-1.1
RAI 4.2.1-1
RAI B.2.1.28-3

RAI 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.

Exelon Response

1. The “B” and “C” RWCU pumps on LGS Units 1 and 2 are fabricated from CASS material and are classified as nonsafety-related ASME Code Class 3 components. Opportunistic inspection of the pump casings to inspect for cracking, performed only when the pumps are disassembled for maintenance or repair, is consistent with the guidance within the GALL Report and referenced ASME Code Section XI inspection requirements to manage loss of fracture toughness due to thermal aging embrittlement of safety-related ASME Code Class 1 pump casings fabricated from CASS material.

   GALL Report, aging management review line item IV.C1.R-08 is provided to manage loss of fracture toughness due to thermal aging embrittlement of ASME Code Class 1 pump casings and valve bodies fabricated from CASS in a reactor coolant >482 degrees F environment. This line item specifies that the Chapter XI.M1, ASME Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD aging management program be used to manage this aging effect, and it clarifies the requirements as follows; “*for pump casings and valve bodies, screening for susceptibility to thermal aging is not necessary. The ASME Section XI inspection requirements are sufficient for managing the effects of loss of fracture toughness due to thermal aging embrittlement of CASS pump casings and valve bodies.*”

   This is consistent with GALL Report AMP XI.M12, “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)”, which manages loss of fracture toughness for ASME Class 1 components, and includes the following statements:

   - Program Description – “*This inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging embrittlement of cast austenitic stainless steel (CASS) piping components except for pump casings and valve bodies*.”
   - Scope of Program – “*This program manages loss of fracture toughness in potentially susceptible ASME Code Class 1 piping components made from CASS.” “For pump casings and valve bodies, screening for susceptibility to thermal aging is not needed (and thus there are no aging management review line items). For all pump casings and valve bodies, greater than a nominal pipe size (NPS) of 4 inches, the existing ASME Code, Section XI inspection requirements, including the alternate requirements of ASME Code Case N-481 for pump casings, are adequate*.” This position is supported by the letter from Christopher Grimes, Nuclear Regulatory Commission, to Mr. Douglas Walters, Nuclear Energy Institute, “*Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components*,” dated May 19, 2000. Note that Code Case N-481 applied to Table IWB-2500-1, Examination Category B-L-1 inspection of CASS pump casing welds and was annulled in 2004 after Category B-L-1 requirements were deleted from ASME Section XI requirements. The LGS “B” and “C” RWCU pumps do not have casing welds.
   - Parameters Monitored/Inspected – “*The program does not directly monitor for loss of fracture toughness that is induced by thermal aging; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components*.”
   - Detection of Aging Effects – “*For pump casings, valve bodies, and other “not susceptible” CASS piping components, no additional inspection or evaluations are needed to demonstrate that the material has adequate fracture toughness*.”

   ASME Section XI, Table IWB-2500-1, Examination Category B-L-2, Item No. B12.20 applies to ASME Code Class 1 pump casings and specifies that the internal surfaces of pump casings be examined using VT-3 method with reference to Notes 1 and 2. Note 1 states that "*examinations are limited to at least one pump in each group of pumps performing similar functions in the system*." Note 2 states that "*examination is required only when a pump or valve is disassembled for maintenance, repair, or volumetric examination. Examination of the internal pressure boundary shall include the internal pressure retaining surfaces made accessible for examination by disassembly. If a partial examination is performed and a subsequent disassembly of that pump or valve allows a more extensive examination, an examination shall be performed during the subsequent disassembly. A complete examination is required only once per interval*." ASME Section XI, Table IWD-2500-1 applicable to ASME Code Class 3 components, only requires VT-2 system leakage testing of the pressure retaining boundary, and does not require inspection of the internal surfaces of pump casings.

There are no GALL Report aging management review line items provided to manage loss of fracture toughness for ASME Code Class 3 components. Therefore, line item IV.C1.R-08 was selected for the "B" and "C" RWCU pump casings. Since the ASME Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD aging management program does not include internal inspection requirements for ASME Code Class 3 pump casings, a substitute aging management program was selected to implement inspection using the same method at the same frequency specified in GALL Report AMP XI.M12 for ASME Code Class 1 pump casings. The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program, as described in GALL Report AMP XI.M38, was determined to be appropriate to manage loss of fracture toughness of the CASS RWCU pump casings for the following reasons:

- GALL Report AMP XI.M38 is used to manage cracking as specified by GALL Report aging management review line item VII.H2.AP-128. Therefore, it is reasonable to use this program to inspect for cracking of the CASS RWCU pump casings. As discussed above, GALL Report AMP XI.M12 states that loss of fracture toughness is indirectly managed by monitoring for cracking.
- GALL Report AMP XI.M38, Program Description states that inspections can be made "*during the performance of maintenance activities when the surfaces are made accessible for visual inspection*." This is consistent with the inspection frequency for managing loss of fracture toughness in ASME Code Class 1 CASS pump casings as specified by GALL Report AMP XI.M12 and referenced ASME Code Section XI, Table IWB-2500-1, as discussed above, and therefore is adequate for management of Class 3 pump casings.
- GALL Report AMP XI.M38, Detection of Aging Effects, states that "*unless otherwise required (e.g. by the ASME code) all inspections should be carried out using plant-specific procedures by inspectors qualified through plant specific programs. The inspection procedures must be capable of detecting the aging effect(s) under consideration*." Inspection of the CASS RWCU pump internals is not required by ASME Code. The inspections of the CASS RWCU pump casings under the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program will be performed using VT-3 method, consistent with that specified by ASME Section XI, Table IWB-2500-1 for ASME Code Class 1 pump casings. The program bases document for the LGS Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program and the plant-specific implementing procedure will include specifying VT-3 examination method by qualified inspectors.

Opportunistic inspection using VT-3 examination methods is justified to manage nonsafety-related CASS RWCU pump casings for loss of fracture toughness due to thermal aging embrittlement. The inspection method and frequency is consistent with GALL Report guidance and referenced ASME Code Section XI requirements for the management of safety-related ASME Code Class 1 pump casings for loss of fracture toughness due to thermal aging embrittlement, as described above, and therefore is adequate for management of the Class 3 pump casings.

2. Since the LGS Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program is a new program, and enhancements are intended to implement revisions to existing aging management programs to make them consistent with the GALL Report, it is not appropriate to update the UFSAR Supplement to reflect an enhancement to the program to manage loss of fracture toughness of the CASS RWCU pump casings. As discussed in the response to Request 1 above, the proposed aging management of the CASS RWCU pump casings for loss of fracture toughness is consistent with guidance provided in the GALL Report for components within the scope of license renewal having the same material and operating environment. LRA Section A.2.1.26, UFSAR Supplement for the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program, was revised to include aging management for loss of fracture toughness and cracking as part of the Exelon response to RAI B.2.1.26-1 within the letter dated February 15, 2012. Therefore, no further changes to the UFSAR Supplement for the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program are appropriate.

RAI 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.

Request

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.

Exelon Response

1. Potential aging effects due to exposure of the refueling bellows to treated water during refueling outage periods do not warrant aging management for cracking and loss of material, and the basis for not including AMR line items is as follows:

   The refueling bellows are exposed to treated water below 140 degrees F for approximately two weeks each operating cycle (two years) during refueling operations, which is less than two percent of the service time. For 102 out of 104 weeks, the refueling bellows are exposed to their normal air-indoor environment. The refueling bellows are not in low flow or stagnant areas since the area is drained upon completion of refueling, and the air-indoor environment of the Primary Containment is inerted with nitrogen. Heat from the RPV causes rapid evaporation of any moisture remaining on the bellows, resulting in a dry condition. Therefore the conditions described in the cited NUREG and GALL Report item III.A5.T-14 do not apply to the LGS refueling bellows.

Operating experience reviews for LGS have not identified failure of the bellows or leakage from the type of refueling bellows used at LGS. The environmental conditions during normal plant operation - including plant/system operating transients and plant shutdown including refueling operations - were considered in determining aging effects for the stainless steel refueling bellows components requiring aging management. The potential for loss of material due to pitting and crevice corrosion and stress corrosion cracking of the stainless steel as a result of the limited exposure to the low temperature high purity reactor water is not significant and therefore warrants no aging management.

The refueling bellows are constructed from thin stainless steel material such that a postulated crack or other defect that could be detected by visual or surface examination would also likely result in leakage during refueling. LGS is a Mark II concrete containment; therefore any postulated leakage of refueling bellows would not result in a potential for corrosion of inaccessible containment shell areas such as the sand pocket region which is a concern associated with steel (Mark I) containments. Water level in the spent fuel pool and reactor cavity is also monitored during refueling operations. The refueling bellows are provided with leakage detection instrumentation and an alarm in the main control room as described in UFSAR section 9.1.3.5. In the event that refueling bellows leakage occurs, it will be addressed by plant operating procedures and the Corrective Action Program. The refueling bellows design includes a secondary seal designed to limit water loss in the unlikely event of bellows failure.

NUREG/CR-6726 addresses IWE pressure boundary bellows such as piping penetration bellows, vent line bellows, fuel transfer tube bellows, etc. The LGS refueling bellows are not part of the primary containment IWE pressure boundary. The LGS refueling bellows prevent leakage of water from the circular reactor well cavity which is filled with water following removal of the primary containment drywell head and reactor vessel head during refueling operations involving transfer of fuel assemblies.

2. The response to Request 1 addressed consideration of the applicable aging effects that could result from normal plant operation, including plant/system operating transients and plant shutdown for the LGS refueling bellows. There are no applicable aging effects warranting use of the Water Chemistry aging management program.

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.

Exelon Response

There are two neutron fluence calculational methodologies that are used as part of the LGS CLB: 1) the RAMA fluence methodology, used for core shroud evaluations and in the LRA for RPV and internals TLAA evaluations; and 2) the General Electric DORT (discrete ordinates transfer) methodology, which was used in the development of the current 32 EFPY P-T limit curves. Each of these methodologies is described below, along with the bases for their use.

RAMA Fluence Methodology

As prescribed in RG 1.190, the RAMA fluence methodology has been benchmarked against industry standard benchmarks for BWR and PWR designs. In addition, RAMA has been compared with several plant-specific dosimetry measurements and reported fluence from several commercial operating reactors. The results of the benchmarks and comparisons to measurements show that RAMA accurately predicts specimen activities, RPV fluence and vessel component fluence in all light water reactor types.

The RAMA fluence methodology was reviewed in the NRC Safety Evaluation of Proprietary EPRI Reports: "BWRVIP RAMA Fluence Methodology Manual (BWRVIP-114)," "RAMA Fluence Methodology Benchmark Manual (BWRVIP-115)," "RAMA Fluence Methodology – Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5 (BWRVIP-117)," "RAMA Fluence Methodology Procedures Manual (BWRVIP-121)" and "Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1 (TWE-PSE-001-R-001)," dated May 13, 2005 (ML100540367).

Section 4.1 of the SER, *"BWR RPV Neutron Fluence,"* states:

"Based on the staff's review of the BWRVIP-114, -115, -117, and -121 reports, the TWE-PSE-001-R-001 report, and the supporting documentation, the staff concludes that the BWRVIP methodology, as described in these reports, provides an acceptable best-estimate plant-specific prediction of the fast (E ≥ 1.0 MeV) neutron fluence for BWR RPVs."

Section 4.1 of the SER further states:

"With respect to the calculation of BWR RPV neutron fluence, the staff concludes that based on the plant-specific benchmark data presently available, no calculational bias is required for the application of the methodology to plants of similar geometrical design to Susquehanna and Hope Creek, i.e. BWR-IV plants. However, in order to provide continued confidence in the proposed neutron fluence methodology for the BWR RPVs, the acceptance of this methodology is subject to the following conditions for plants which do not have geometries similar to the cited BWR-IVs:

- *To apply the RAMA methodology to plant groups which have geometries that are different than the cited BWR-IVs, at least one plant-specific capsule dosimetry analysis must be provided to quantify the potential presence of a bias and assure that uncertainty is within RG 1.190 limits; and*
- *Justification is necessary for a specific application based on geometrical similarity to an analyzed core, core shroud, and RPV geometry. That is, a licensee who wishes to apply the RAMA methodology for the calculation of RPV neutron fluence must reference, or provide, an analysis of at least one surveillance capsule from a RPV with a similar geometry."*

LGS and Susquehanna RPVs have a similar geometrical design, as described below, so the conditions described above do not apply for use of the RAMA methodology for LGS RPV fluence projections. LGS UFSAR Table 1.3-1, "Comparison of Nuclear Steam Supply System Design Characteristics," shows LGS and Susquehanna reactor vessels each have an inside diameter of 20 ft. 11 inches (251 inches), have 764 fuel bundles, and have 20 jet pumps with two downcomers each. The LGS and Susquehanna vessel shrouds each have an outside diameter of 207 inches. Since LGS is a BWR-IV plant that is similar to the Susquehanna plant in core, shroud, and downcomer-vessel geometry, the NRC conditions for use of RAMA methodology have been met and the RAMA methodology can be applied without a bias for the calculation of RPV neutron fluence, as provided in the SER.

The BWRVIP integrated surveillance program was approved by the NRC in License Amendment No. 167 for Unit 1 and Amendment No. 130 for Unit 2, dated November 4, 2003 (ML032310540) that defines the current licensing basis for computing neutron fluence at LGS. UFSAR Sections 4.1.4.5 and 4.3.2.8 were revised accordingly to state:

"LGS RPV fluence has been evaluated using a method in accordance with the recommendations of RG 1.190. Future evaluations of RPV fluence will be completed using a method in accordance with the recommendations of RG 1.190."

RAMA has since been evaluated for use at LGS in accordance with 10 CFR 50.59, which concluded that the methodology can be used without prior NRC approval since it conforms with the regulatory positions in RG 1.190, consistent with UFSAR Sections 4.1.4.5 and 4.3.2.8. The 57 EFPY neutron fluence values calculated for 60 years of operation using RAMA were used in the following sections of the LRA: Section 4.2.2, "Upper Shelf Energy;" Section 4.2.3, "Adjusted Reference Temperature;" Section 4.2.5, "Axial Weld Inspection;" Section 4.2.6, "Circumferential

Weld Inspection;" and Section 4.2.7, "Reactor Pressure Vessel Reflood Thermal Shock." However, the RAMA fluence values were not used in Section 4.2.4, "Pressure -Temperature Limits," since P-T limit curves have not been revised as part of the LGS LRA.

General Electric DORT Methodology

The other fluence methodology utilized in the CLB at LGS is GE DORT fluence methodology, which was used in developing the 32 EFPY P-T limit curves currently in effect. The current P-T limit curves were initially approved for use by the NRC in Amendment No. 111 for Unit 2 by NRC letter dated 3/23/2001 (ML010540068) and by Amendment No. 155 for Unit 1 by NRC letter dated January 30, 2002 (ML013540132). However, during the staff's reviews, the adequacy of the methodology used to determine the RPV neutron fluence was questioned and, as a result, the P-T limit curves were restricted for use through the end of operating cycle 7 for Unit 2 and operating cycle 10 for Unit 1.

License Amendment Request LG-02-00391, dated June 26, 2002 (ML021910386), and supplement, dated September 12, 2002 (ML022630489), were submitted for removal of the restrictions on the P-T limits, based on recalculation of LGS reactor pressure vessel fluence using the staff-approved GE methodology described in NEDC-32983P-A. The results of this calculation indicated that the existing 32 EFPY P-T limit curves bound the neutron fluence value calculated using the NEDC-32983P-A methodology. NRC letter dated January 2, 2003 (ML030030022) issued Amendment No. 163 for Unit 1 and Amendment No. 125 for Unit 2 that removed the restrictions and extended the validity of the P-T curves to 32 EFPY for each unit. The NRC Safety Evaluation for these license amendments evaluated the results of the updated RPV neutron fluence calculations and determined that the current P-T limit curves have "*a significant margin of conservatism."*

As described in LRA Section 4.2.4, *"Pressure -Temperature Limits,"* Unit 1 is projected to exceed 32 EFPY during operating cycle 19, which begins in the year 2020, and Unit 2 is projected to exceed 32 EFPY during operating cycle 18, which begins in the year 2023. The Reactor Surveillance (B.2.1.21) program is used to ensure that updated P-T limit curves will be submitted to the NRC for approval prior to exceeding 32 EFPY. Until then, the current 32 EFPY P-T limit curves, based on ART values derived using GE fluence methodology, will remain in effect.

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?

Exelon Response

The response to RAI B.2.1.28-2, provided by letter dated April 27, 2012, provided relative cumulative exposure for the coupons compared to the most limiting fuel storage cell utilizing exposure to freshly discharged fuel as an indicator of radiation exposure. This analysis compared the actual location history of the coupons with the maximum possible exposure scenario for the fuel storage cells. Tables 1 and 2 in that letter show that at the end of the proposed five cycles of exposure to being fully surrounded by eight freshly discharged fuel assemblies, the coupons will have been exposed to a greater number of freshly discharged fuel assemblies than the worst case fuel storage cell. This establishes that the coupon exposure will bound that of all fuel storage cells at the end of this five-cycle period, prior to the start of the period of extended operation.

The Safety Evaluation related to Amendment Numbers 82 (Unit 1) and 43 (Unit 2) to the Limerick Facility Operating License defines the current licensing basis for LGS and specifically the strategy for surrounding the coupons with eight freshly discharged fuel assemblies following five consecutive operating cycles. The SER approved by NRC on 11/29/1994 states (in Section 2.4.2): "*This is done to assure that the coupons experience a higher radiation dose than the Boral panels in the storage racks.*" This approach is consistent with GALL program M40, Element 5. In addition, the Safety Evaluation further states that since Boral may generate hydrogen when in contact with water or moisture, a sufficient vent path must be provided to allow potential hydrogen generation to vent from the sheath area. The LGS Boral panels are a vented design (holes at the corners of the sheath areas), thereby providing a vent path for any potential hydrogen generation.

Documented industry research does not differentiate between accelerated and long term exposure effects. In fact, Information Notice 2009-26, *Degradation of Neutron-absorbing Materials in the Spent Fuel Pool*, indicates that *"the degradation mechanisms and deformation rates of the neutron-absorbing materials in the SFP are not well understood. Therefore, for licensees that credit the use of a neutron-absorbing material to maintain subcriticality in their SFP, knowing the condition of the neutron-absorbing material in the SFP and monitoring the SFP for any indications that degradation of the material may be occurring can prevent noncompliance with SFP criticality requirements."* This is the purpose of the B.2.1.28 aging management program, to monitor the condition of the Boral material by performing neutron attenuation testing of the coupons and by evaluating them for physical attributes, dimensional checks, and weight and density characteristics.

EPRI Report 1019110, *Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Transportation and Storage Applications*, 2009 Edition, reviews testing and in-service experience, which indicates that Boral may exhibit corrosion and blister formation. This report suggests that these characteristics may be related to manufacturing practices. For example, formation of weak spots in the Boral passivation oxide film can lead to localized corrosion. Also, blister formation may be the result of water entering the Boral through open porosity at the edges where the material was trimmed to size. Other related EPRI research (EPRI Report 1021051, *Experimental Characterization of Boral Pore Size and Volume Distributions*, dated December, 2010, EPRI Report 1021053, *Reaction-path and Capillary-effects Models of Boral Blistering,* dated December, 2010, and EPRI Report 1016641*, Thermal Analysis of Boral in Storage Racks*), dated November, 2008, also indicate that these characteristics are likely related to the original manufacturing process and the porosity of the material, and not necessarily due to gamma heating. The *Handbook* concludes that that the presence of corrosion has not reduced the neutron loading capability of the material, and that *"blistering of Boral has, to date, proved to be primarily an esthetic effect."*

EPRI Report 1021052, *Overview of BORAL Performance Based Upon Surveillance Coupon Measurement* , dated December, 2010, documents that blistering and corrosion pitting which has been observed in Boral coupons has been analyzed, and has shown no impact on neutron absorption performance. Additionally, the potential issues which can result from blister formation are reduction of free clearances in storage cells, and displacement of water from the flux trap region. The LGS spent fuel storage racks have not exhibited reduction of clearances in storage cells due to blistering (which would prevent fuel bundle placement if present), and do not utilize the flux trap region design.

In summary, the most recent documented analyses of operating experience related to degradation of Boral, indicate that the most likely causes involve manufacturing practices. There are no documented analyses that indicate gamma heating or other effects from radiation exposure are the likely cause of the degradation. The degradation has been characterized as esthetic, possibly leading to minor corrosion and blistering that could lead to reduced clearances between the fuel assembly and the storage cell, with no effect on the intended function of neutron absorption.

GALL AMP Report XI.M40, *Monitoring of Neutron-Absorbing Materials Other Than Boraflex*, states that *"gamma irradiation and/or long-term exposure to the wet pool environment may cause loss of material and changes in dimension that could result in loss of neutron-absorbing capability of the material"*. The AMP indicates that coupon testing is appropriate to detect aging effects, and that periodic verification of boron loss through areal density measurement, and geometric changes in the material (blistering, pitting, and bulging) are acceptable methods to detect aging effects. The approach for relating the measurements to the performance of the spent fuel neutron absorber material should consider "differences in exposure conditions, vented/non-vented test samples, and spent fuel racks, etc". The coupon testing proposed for LGS is consistent with the recommendations within all elements of GALL Report AMP XI.M40, and will effectively monitor the condition of the Boral spent fuel storage cells during the period of extended operation for the following reasons:

- The coupons will be bounding of the most limiting fuel storage cell location relative to radiation exposure as discussed above, and in the response to RAI B.2.1.28-2.

- The coupons are exposed to the same environmental conditions as the Boral panels in the fuel storage cells, relative to being submerged in water within a fuel storage cell. Since the water within the pool is continually circulated, the temperature of the water at the coupons is similar to the temperature of the Boral panels in the fuel storage cells.
- The next coupon test will be performed after the exposure to radiation to the coupons is known to be bounding of the most limiting fuel storage cell, and prior to the start of the period of extended operation.
- Coupon testing will continue to be performed at a frequency not to exceed ten years during the period of extended operation as recommended by GALL AMP XI.M40.
- Coupon testing includes analyses for physical attributes, dimensional checks and neutron attenuation that are designed to identify loss of material, loss of neutron absorption ability, and the types of degradation observed at other plants, as described above.