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10 CFR 50
10 CFR 51
10 CFR 54

January 24, 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: Response to NRC Request for Additional Information, dated December 15, 2011, 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 License Renewal Application (TAC Nos. ME6555, ME6556)", dated December 15, 2011.

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. 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 enclosures contain no 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 1-24-2012

Respectfully,

Michael P. Gallagher

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

Enclosures: A: Responses to Request for Additional Information
B: Updates to affected Limerick LRA sections

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 A

Responses to Request for Additional Information related to various sections of the Limerick Generating Station, Units 1 and 2, License Renewal Application (LRA)

RAI 4.2.6-1
RAI 4.6.5-1
RAI 4.6.6-1
RAI 4.6.6-2
RAI 4.6.6-3
RAI 4.6.9-1

RAI 4.2.6-1

Background

License renewal application (LRA) Table 4.2.6-1 (page 4-50) summarizes the pertinent parameters for Limerick Generating Station (LGS), Units 1 and 2 to estimate the probability of failure for the circumferential welds.

Issue

The values for fluence at 0T and shift in Reference Temperature ΔRT_{NDT} (°F) (without margin) do not agree with the corresponding values found in Tables 4.2.1-1 and 4.2.1-3 for fluence and Tables 4.2.3-1 and 4.2.3-3 for the temperature shift.

Request

Justify the fluence values in LRA Tables 4.2.3-1 and 4.2.3-3 in light of the apparent discrepancy discussed above.

Exelon Response

In LRA Table 4.2.6-1, the 0T fluence value for Unit 1 should have been 8.35 E+17 n/cm^2 rather than the 8.35 E+19 n/cm^2 value originally shown. The 0T fluence value for Unit 2 should have been 8.02 E+17 n/cm^2 rather than the 8.02 E+19 n/cm^2 value originally shown. The corrected values match the corresponding values in LRA Tables 4.2.1-1 and 4.2.1-3. Enclosure B provides a revised LRA Table 4.2.6-1 to make these corrections.

The values for Shift in Reference Temperature ΔRT_{NDT} (°F) (without margin) in LRA Table 4.2.6-1 are based upon 0T fluence values, consistent with the NRC Staff Circumferential Weld Assessment, also shown in LRA Table 4.2.6-1. These values are comparable to the ΔRT_{NDT} (°F) values shown in LRA Tables 4.2.3-1 and 4.2.3-3 that represent the change due to fluence without a margin term. The ΔRT_{NDT} (°F) values shown in LRA Tables 4.2.3-1 and 4.2.3-3 were computed based upon the 1/4 T fluence, consistent with the requirements of Regulatory Guide 1.99, Revision 2, so the values are lower than those shown in Table 4.2.6-1 due to the reduced fluence at 1/4T.

RAI 4.6.5-1

Background

LRA Section 4.6.5 describes the time-limiting aging analysis (TLAA) for the jet pump auxiliary spring wedge assemblies. The LRA states that the original design analysis included an evaluation for relaxation of bolt preload due to integrated neutron fluence over 40 years. The applicant determined that the analysis remains valid for the period of extended operation based on the use of Radiation Analysis Modeling Application (RAMA) fluence projections for the jet pump riser brace weld, RS-9 location, which are bounding for all locations on the jet pump, including where the auxiliary spring wedge assembly is installed. The LRA states that the RS-9 weld attaches the riser brace to the riser pipe, located at approximately the 304 inch elevation, while the auxiliary spring wedge assembly is located at approximately the 230 inch elevation, where the fluence values are lower.

Issue

The applicant did not provide sufficient information to describe how it determined the existing analysis is valid for the period of extended operation. Specifically, the staff's review identified the following issues.

(a) The RS-9 weld is located on the riser brace, which, according to LRA Table 2.3.1-3, is part of the jet pump assembly of the reactor vessel internals. The NRC staff approved use of RAMA for boiling water reactor pressure vessel fluence projection applications, but use of RAMA for reactor internals applications is subject to NRC staff review on a case-by-case basis. Therefore, it is not clear why use of RAMA is appropriate for fluence projections on the RS-9 weld, because it is part of the reactor vessel internals, not the reactor pressure vessel.

(b) It's not clear why the fluence values at the location of the auxiliary spring wedge assembly are lower than the fluence values at the location of the RS-9 weld.

Request

1. Justify the use of RAMA for obtaining fluence projections at the location of the RS-9 weld.

2. Justify why the fluence values at the RS-9 weld location are bounding for all locations on the jet pump, including the location lower on the jet pump where the auxiliary spring wedge assembly is installed.

Exelon Response

1. 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)," and "RAMA Fluence Methodology Procedures Manual (BWRVIP-121)" and "Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1 (TWE-PSE-001-R-001)" (TAC No. MB9765), dated May 13, 2005 evaluated the use of RAMA fluence methodology for BWR reactor pressure vessel (RPV) and internal components of BWR plants.

   Section 4.2 states that "...the submittal does not include any benchmarking for reactor internals' neutron fluence calculations. Therefore, the staff will review qualification of RAMA for reactor internals applications on a case-by-case basis, based on consideration of C/M values and the associated accuracy requirements." It further states that "Licensees who wish to use the RAMA methodology for the calculation of neutron fluence at reactor internals locations must reference, or provide, an analysis which adequately benchmarks the use of the RAMA methodology for uncertainty and calculational bias based on the consideration of: (1) the location at which the neutron fluence is being calculated, (2) the geometry of the reactor, and (3) the accuracy required for the evaluation. In addition, if a licensee qualifies RAMA for calculating, for example, helium generation at one location (e.g. the core shroud), this qualifies RAMA for the same reactor and purpose at other reactor internals locations (e.g. at the location of the jet pumps)."

The BWRVIP-145 report, "BWR Vessel and Internals Project, Evaluation of Susquehanna Unit 2 Top Guide and Core Shroud Material Samples Using RAMA Fluence Methodology (ML 100260948)," was submitted to the USNRC for the purpose of supporting generic regulatory improvements in order to enable the evaluation of reactor internal component degradation by providing a methodology to determine fast neutron fluence values for BWR internal components. The BWRVIP-145 report was based upon using RAMA fluence methodology to calculate reactor internals fluence values. The methodology was applied to core shroud and top guide samples removed from Susquehanna Unit 2 after 11 cycles of irradiation. The report compares the actual dosimetry results from the Susquehanna samples with the corresponding RAMA fluence values.

The staff reviewed the BWRVIP-145 report for its suitability in applying the RAMA fluence methodology to calculation of fast neutron fluence values for BWR reactor vessel internals, specifically for the core shroud and top guide. The staff concluded for the core shroud comparison that "the calculated values are in excellent agreement with the measured values, thus, the RAMA fluence methodology in this particular case performs very well." The staff concluded that "the calculated values for the top guide dosimetry are in reasonable agreement with the measured values and that the results are acceptable for determining fast neutron fluence values in the core shroud and top guide." The staff further concluded that "although the benchmarking guidance in RG1.190 was not achieved, there is reasonable agreement between the calculated and measured dosimetry values of the limited data provided." Therefore, the staff found that "for applications related to IASCC, crack propagation rates, and weldability determinations, the existing data provides adequate justification for applying the RAMA methodology to determine the fast neutron fluence values in the core shroud and top guide."

Therefore, the BWRVIP-145 report is referenced for satisfying the requirements specified in the SER for BWRVIP-114, -115, -117, and -121 with respect to using the RAMA fluence methodology to predict fast neutron fluence for core shroud, top guide, and jet pump locations in determining the extent of loss of preload as a function of neutron fluence. The Limerick Unit 1 and Unit 2 reactors are of the same BWR/4 design as Susquehanna Unit 2, with 251-inch inside diameter reactors, 764 fuel bundles, and core shrouds with a 207-inch outside diameter. Therefore, since the RAMA methodology may be used at Susquehanna for core shroud and top guide applications, it may also be used at Limerick for these applications. In addition, since this report qualifies RAMA for calculating fast neutron fluence on the core shroud, it is also qualified for calculating fast neutron fluence for the jet pumps, as provided in the SER for BWRVIP-114, -115, -117, and -121.

2. The jet pump welds are located at five different elevations in the reactor vessel, as shown in the following table, listed from highest to lowest elevation. 57 EFPY RAMA fluence values are provided for each jet pump weld. The elevations for the Top of Active Fuel, Bottom of Active Fuel, and Core Midplane are also provided for reference. Since the jet pump weld fluence values do not include the 230-inch elevation where the auxiliary spring wedge assemblies are located, fluence values from the outer surface of the core shroud V17 vertical weld are provided to demonstrate the fluence profile in the lower portion of the core.

Unit 1 Component Description	Elevation (inches)	Unit 1 57 EFPY Jet Pump Weld Fluence (n/cm^2)	Unit 1 57 EFPY Core Shroud Vertical Weld V17 Fluence (n/cm^2)
Top of Active Fuel	366		
Jet Pump Weld RS-3	321	1.80 E+20	
Jet Pump Weld RS-9	304	1.82 E+20	
Jet Pump Weld RS-8	300	1.79 E+20	
Top of Core Shroud Vertical Weld V17	299		3.47 E+20
Core Mid-Plane	291		3.32 E+20
Auxiliary Spring Wedges	230		1.45 E+20
Bottom of Active Fuel	216		4.75 E+19
Jet Pump RS-2 Weld	191	1.07 E+17	1.81 E+17
Jet Pump RS-1 Weld	181	1.96 E+15	

Unit 2 Component Description	Elevation (inches)	Unit 2 57 EFPY Jet Pump Weld Fluence (n/cm^2)	Unit 2 57 EFPY Core Shroud Vertical Weld V17 Fluence (n/cm^2)
Top of Active Fuel	366		
Jet Pump Weld RS-3	321	1.76 E+20	
Jet Pump Weld RS-9	304	1.80 E+20	
Jet Pump Weld RS-8	300	1.76 E+20	
Top of Core Shroud Vertical Weld V17	299		3.09 E+20
Core Mid-Plane	291		2.95 E+20
Auxiliary Spring Wedges	230		1.27 E+20
Bottom of Active Fuel	216		4.20 E+19
Jet Pump RS-2 Weld	191	1.04 E+17	1.62 E+17
Jet Pump RS-1 Weld	181	1.87 E+15	

The jet pump weld fluence profile shows that fluence decreases as elevation decreases below the 304-inch elevation where the RS-9 weld is located. Fluence values for the core shroud vertical weld V17, located in the lower half of the core shroud, show a similar trend. The data demonstrates that fluence for the RS-9 weld, located at the 304-inch elevation of the jet pumps, bounds the fluence at the 230-inch elevation of the jet pumps where the auxiliary spring wedges are located.

The 60-year RAMA fluence values shown above are for 57 EFPY. However, the Unit 1 auxiliary spring wedge assemblies were installed in March 2004 and the Unit 2 assemblies were installed in March 2005. As a result, the Unit 1 auxiliary wedge assemblies will only be exposed to 30 EFPY through the period of extended operation and the Unit 2 assemblies will be exposed to 33.6 EFPY. The fluence values for the auxiliary spring wedge assemblies described in LRA Section 4.6.5 (1.30 E+20 n/cm^2 for Unit 1 and 1.33 E+20 n/cm^2 for Unit 2) were derived from the 57 EFPY fluence values shown above for the RS-9 weld by subtracting the fluence that had occurred prior to the installation dates shown above.

See Enclosure B for revised LRA Section 4.6.5 and UFSAR Supplement Section A.4.6.5 to incorporate clarification that the preload is for the auxiliary spring wedge and that the RS-9 fluence bounds the location of the auxiliary spring wedges, not necessarily the entire jet pump.

RAI 4.6.6-1

Background

LRA Section 4.6.6 describes the TLAA evaluation of the jet pump restrainer bracket pad repair clamps. The LRA states that the original design analysis included an evaluation for the decrease in the clamping bolts' preload due to thermal and radiation-induced relaxation over 40 years. The LRA projected this analysis to the end of the period of extended operation. The projection is based on the assumption that the bolt preload will decrease 5 percent over 20 additional years of operation, the same decrease predicted over 40 years in the original design analysis.

Issue

The projection of this analysis relies on the assumption that the preload will decrease by 5 percent over an additional 20 years of operation; however, the LRA did not justify this assumption.

Request

Justify the assumption that the bolt preload will decrease by 5 percent over an additional 20 years of operation as a result of thermal and radiation-induced relaxation.

Exelon Response

The jet pump restrainer bracket pad repair clamp TLAA was reevaluated and the existing design analysis has been determined to remain valid without an assumption that the bolt preload will decrease by an additional 5 percent over an additional 20 years of operation. 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. Therefore, the disposition is revised from 10 CFR 54.21(c)(1)(ii) to 10 CFR 54.21(c)(1)(i).

LRA Section 4.6.6 and Table 4.1-2 are revised to be consistent with the above response, as shown in Enclosure B.

<u>RAI 4.6.6-2</u>

<u>Background</u>

LRA Section 4.6.6 describes the applicant's TLAA evaluation of the jet pump restrainer bracket pad repair clamps. The LRA states that the original design analysis included an evaluation for fatigue, but it was determined that the fatigue usage would be insignificant because the stress amplitude for cyclic loads was well below the American Society of Mechanical Engineers (ASME) Code stress limit of 13,600 psi for 10^{11} cycles and less than the 10,000 psi lower limit considered for flow-induced vibration stress cycles.

<u>Issue</u>

The staff reviewed LRA Section 4.6.6 and determined that it includes separate analyses involving two aging effects: (1) decrease in preload due to thermal and radiation-induced relaxation, and (2) fatigue. The applicant provided one disposition for LRA Section 4.6.6, stating that it projected the analysis to the end of the period of extended operation. However, for the evaluation of fatigue, it is not clear how the applicant projected the analysis because it did not describe changes to any parameters of the original analysis. Therefore, for its analysis of fatigue for the jet pump restrainer bracket pad repair clamps, the applicant has not provided a disposition in accordance with 10 CFR 54.21(c)(1).

<u>Request</u>

For the fatigue analysis of the jet pump restrainer bracket pad repair clamps, provide one of the demonstrations required by 10 CFR 54.21(c)(1):

- If demonstrating that the analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i), describe and justify why the analysis is bounding during the period of extended operation. Include in this demonstration the applicable materials and their properties, the value of the stress amplitude for cyclic loads, and the specific sources of the 13,600 and 10,000 psi stress limits to which the stress amplitude was compared.

- If demonstrating that the analysis has been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii), describe and justify how the original analysis was revised.

- If demonstrating that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii), provide an aging management program and justify how it adequately manages fatigue.

Exelon Response

The fatigue analysis of the jet pump restrainer bracket pad repair clamps remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i) as described below. Clamping loads do not cycle except for the load changes associated with startup/shutdown temperature changes. The largest stress range resulting from the temperature changes is 14,022 psi, calculated for the restrainer bracket, which results in an alternating stress amplitude of 7,011 psi. After adjustment for the difference between operating temperature of 550°F and the ASME fatigue curve reference temperature of 70°F, the stress amplitude is 7,781 psi. The restrainer bracket is fabricated from Type 304 stainless steel, with a stress intensity limit of 16,000 psi at 550°F and yield strength of 17,750 psi at 550°F. The American Society of Mechanical Engineers (ASME) Code stress limit is 13,600 psi for 10^{11} cycles and the General Electric Hitachi design limit for flow-induced vibration stress cycles is 10,000 psi. Since the cyclic loads of 7,781 psi are less than these limits, fatigue usage will be insignificant for up to 10^{11} startup/shutdown cycles. Therefore the fatigue analysis remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

LRA Section 4.6.6 is revised to clarify the separate dispositions for the loss of preload TLAA and the fatigue TLAA, as shown in Enclosure B.

RAI 4.6.6-3

Background

LRA Section 4.6.6 describes the TLAA evaluation of the jet pump restrainer bracket pad repair clamps, and LRA Section A.4.6.6 provides a summary description of this TLAA for the UFSAR supplement. As discussed in RAI 4.6.6-2, the staff determined that LRA Section 4.6.6 describes separate analyses involving two aging effects: (1) decrease in preload due to thermal and radiation-induced relaxation, and (2) fatigue.

Issue

LRA Section A.4.6.6 describes the analysis for the decrease in preload, but it does not address the analysis for fatigue.

Request

Consistent with the response to RAI 4.6.6-2, provide for the UFSAR supplement a summary description of the TLAA evaluation for fatigue of the jet pump restrainer bracket pad repair clamps.

Exelon Response

The UFSAR Supplement, LRA Section A.4.6.6, as shown in Enclosure B, is revised to provide a summary description of the TLAA evaluation for fatigue of the jet pump restrainer bracket pad repair clamps that is consistent with the responses to RAI 4.6.6-1 and RAI 4.6.6-2.

RAI 4.6.9-1

Background

LRA Section 4.6.9 describes the TLAA evaluation of the jet pump slip joint repair clamps, which are subject to a loss of preload due to neutron fluence. The LRA states that the loss of preload will be managed through periodic inspections under the reactor vessel internals program. LRA Section B.2.1.9 describes the BWR vessel internals program as consistent, with enhancements, with GALL Report AMP XI.M9, "BWR Vessel Internals."

Issue

The staff reviewed GALL Report AMP XI.M9 and found that this aging management program does not manage loss of preload; therefore, it is not clear how the applicant's BWR vessel internals program will manage this aging effect.

Request

Specific to the jet pump slip joint repair clamp components subject to a loss of preload, address the following:

1. Describe the parameters that are monitored or inspected to detect the presence and extent of loss of preload. Justify how monitoring or inspecting these parameters will ensure that this aging effect is adequately managed.

2. Describe how loss of preload is detected or identified. Include in this description the method or technique, frequency, and timing of inspections, and provide justification, including references to any codes or standards, that these measures are adequate to detect loss of preload before loss of the jet pump slip joint repair clamps' intended function.

3. Describe the monitoring and trending activities and justify how they are used to predict the extent and rate of degradation so that corrective or mitigative actions can be taken or so that future inspections will occur before a loss of intended function.

4. Describe and justify the acceptance criteria for the jet pump slip joint repair clamp inspections for loss of preload, so that corrective actions are taken before loss of the components' intended function. In addition, describe and justify the corrective actions taken when the acceptance criteria are not met, such that a future recurrence is prevented.

Exelon Response

The jet pump slip joint repair clamp TLAA was reevaluated and the existing design analysis has been determined to remain valid through the period of extended operation because the fluence value used to determine loss of preload in the design analysis will not be exceeded. Periodic inspections under the BWR Vessel Internals program are not credited to manage loss of preload. The TLAA disposition is revised from 10 CFR 54.21(c)(1)(iii), Aging Management, to 10 CFR 54.21(c)(1)(i). LRA Section 4.6.9, LRA Table 4.1-2, and UFSAR Supplement Section A.4.6.9 are revised as shown in Enclosure B.
LRA Table 3.1.2-3, Reactor Vessel Internals, included an aging management review line item on page 3.1-66 to credit inspection of Jet Pump Slip Joint Repair Clamps by the BWR Vessel

Internals program to support the TLAA disposition described in LRA Section 4.6.9. Since LRA Section 4.6.9 is revised such that inspection is not credited within the TLAA disposition, LRA Table 3.1.2-3, page 3.1-66, is revised to delete the aging management review line item for managing loss of preload of stainless steel jet pump assembly components by the BWR Vessel Internals program. Also, Plant-Specific Note 5, referenced by that line item, is deleted from page 3.1-69. LRA Table 3.1.2-3 is revised as discussed above as shown in Enclosure B.

Enclosure B

Limerick Generating Station Units 1 and 2
License Renewal Application Updates

Note: To facilitate understanding, portions of the original LRA have been repeated in this Enclosure, with revisions indicated. Existing LRA text is shown in normal font. Changes are highlighted with bold italics for inserted text and strikethroughs for deleted text.

As a result of the response to RAI 4.2.6-1 provided in Enclosure A of this letter, LRA Table 4.2.6-1 is revised as shown below:

Table 4.2.6-1
Comparison of NRC 64 EFPY Circumferential Weld Failure Probability Assessment to LGS 57 EFPY Circumferential Weld Failure Probability Assessments

Parameter	**NRC Staff 64 EFPY Circumferential Weld Assessment** [1]	**Unit 1 57 EFPY Circumferential Weld Assessment**	**Unit 2 57 EFPY Circumferential Weld Assessment**
	(CB&I RPV)	(CB&I Vessel)	(CB&I Vessel)
Copper Content (%)	0.1	0.03	0.03
Nickel Content (%)	0.99	0.97	0.97
Chemistry Factor (CF)	134.9	41	41
Fluence at 0T (n/cm^2)	1.02 E+19	***E+17*** 8.35 ~~E+19~~	***E+17*** 8.02 ~~E+19~~
Unirradiated Reference Temperature $RT_{NDT(U)}$ (°F)	-65	-6	-6
Shift in Reference Temperature ΔRT_{NDT} (°F)(without margin) [2]	135.6	16	15
Mean RT_{NDT} (°F)	70.6	10	9
Probability of Failure Event	1.78E-05	Bounded by NRC Probability [3]	Bounded by NRC Probability [3]

NOTES:

[1] The NRC data is obtained from BWRVIP-05 Report, "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report, July 28, 1998.

[2] $\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)}$

[3] Although a conditional failure probability has not been calculated for LGS, the fact that the LGS Mean RT_{NDT} values are significantly less than the NRC values leads to the conclusion that the Unit 1 and Unit 2 conditional failure probability is bounded by the NRC analysis, consistent with the requirements defined in GL 98-05.

As a result of the response to RAI 4.6.5-1 provided in Enclosure A of this letter, LRA Section 4.6.5, Jet Pump Auxiliary Spring Wedge Assembly, is revised as shown below:

4.6.5 JET PUMP AUXILIARY SPRING WEDGE ASSEMBLY

TLAA Description:

The LGS jet pump assemblies have had auxiliary spring wedge assemblies designed and installed to maintain lateral support for the jet pump inlet mixer. The design analysis considered potential aging effects based upon a design life of 40 years, including fatigue usage and relaxation in ~~bolt~~ preload due to neutron fluence. The first auxiliary spring wedge assembly was installed in Unit 1 in March 2004 and will have a design life of 40.7 years by the end of the period of extended operation in October 2044. The first auxiliary spring wedge assembly was installed in Unit 2 in March 2005 and will have a design life of 44.3 years by the end of the period of extended operation in June 2049. Therefore these analyses have been identified as TLAAs.

TLAA Evaluation:

The jet pump auxiliary spring wedge assembly is not an ASME Code component. However, it was evaluated using stress and fatigue limits of the ASME Code as guidelines. The cumulative fatigue usage for applicable Service Level B loads was required to be less than the allowable limit of 1.0. The original design basis load cycles from the reactor vessel thermal cycle diagram were applied. These are the same transients as those from UFSAR Table 3.9-2 (Reference 4.7.18) previously described in LRA Section 4.3.1. The resulting fatigue usage was determined to be 0.77, which is below the allowable limit of 1.0. LRA Tables 4.3.1-1 and ~~4.3.2-2~~ ***4.3.1-2*** show the 60-year transient projections for LGS that demonstrate these transient cycle limits will not be exceeded in 60 years of operation. Therefore, the fatigue TLAA has been demonstrated to remain valid for the period of extended operation.

The auxiliary spring wedge assembly design analysis was also required to evaluate relaxation in ~~the bolt~~ preload due to integrated neutron fluence of 1.4 E+20 n/cm^2 for a 40-year design life. The analysis concluded that all design requirements were met for the 40-year design life of the component.

In order to evaluate relaxation in ~~the bolt~~ preload for the period of extended operation, RAMA fluence projections for the jet pump riser brace weld RS-9 location have been applied because they are bounding for ~~all locations on the jet pump, including~~ the location lower on the jet pump where the auxiliary spring wedge assembly is installed. The RS-9 weld attaches the riser brace to the riser pipe, located at approximately the 304 inch elevation, while the auxiliary spring wedge assembly is located at approximately the 230 inch elevation where the fluence values are lower.

The fluence projection for the Unit 1 jet pump auxiliary spring wedge assemblies through the period of extended operation is 1.30 E+20 n/cm^2 and for Unit 2 is 1.33 E+20 n/cm^2. These values are less than the 1.4 E+20 n/cm^2 fluence value used in the design, which shows that the auxiliary spring wedge assemblies will not reach the previously evaluated fluence value. Therefore, the analysis has been demonstrated to remain valid for the period of extended operation.

TLAA Disposition: 10 CFR 54.21(c)(1)(i) – The analysis remains valid for the period of extended operation.

As a result of the response to RAI 4.6.5-1 provided in Enclosure A of this letter, the UFSAR Supplement LRA Section A.4.6.5, Jet Pump Auxiliary Wedge Assembly, is revised as shown below:

A.4.6.5 Jet Pump Auxiliary Spring Wedge Assembly

Auxiliary spring wedge assemblies have been designed and installed in LGS jet pumps to maintain lateral support for the jet pump inlet mixer. The design analysis considered potential aging effects based upon a design life of 40 years, including fatigue usage and loss of ~~bolt~~ preload due to neutron fluence. These analyses have been identified as TLAAs.

The jet pump auxiliary spring wedge assembly is not an ASME Code component, but it was evaluated using stress and fatigue limits of the ASME Code as guidelines. The cumulative fatigue usage was determined to be 0.77 using the original design basis load cycles from the reactor vessel thermal cycle diagram. The 60-year transient projections for LGS demonstrate that these transient cycle limits will not be exceeded in 60 years of operation. Therefore, the fatigue TLAA remains valid for the period of extended operation.

The auxiliary spring wedge assembly design analysis also evaluated loss of ~~bolt~~ preload due to integrated neutron fluence of 1.4 E+20 n/cm^2 for 40 years. In order to evaluate loss of ~~bolt~~ preload during the period of extended operation, the maximum service life was first determined. The first Unit 1 wedge assembly installed will have a service life of 41 years by the end of the period of extended operation, and the first Unit 2 wedge assembly will have a service life of 45 years.

The RAMA fluence projections were used to determine how long the auxiliary spring wedge assemblies could remain in service before the analyzed fluence value of 1.4 E+20 n/cm^2 would be reached. The fluence projections for the jet pump riser brace location were used since they are bounding for ~~all jet pump~~ ***the*** location~~s~~ ***of the auxiliary spring wedge assembly***. The evaluation determined that the auxiliary spring wedge assemblies ***will not reach the previously evaluated fluence value of*** ~~could be in service for 50 years before they will experience~~ 1.4 E+20 n/cm^2. Therefore, the loss of preload TLAA also remains valid for the period of extended operation in accordance with 10CFR 54.21(c)(1)(i).

As a result of the response to RAI 4.6.6-1 and RAI 4.6.6-2 provided in Enclosure A of this letter, LRA Section 4.6.6 is revised as shown below:

4.6.6 JET PUMP RESTRAINER BRACKET PAD REPAIR CLAMPS

TLAA Description:

Visual inspections at LGS have found wear at the inlet-mixer wedge/restrainer bracket pad interface on several jet pumps. A repair clamp has been designed and installed that replaces the support function of the restrainer bracket pad. The repair clamp design analysis evaluated end-of-life preload relaxation for 40 years. This analysis has been identified as a TLAA.

TLAA Evaluation:

~~The following information is provided from the 40-year TLAA identified above. The jet pump repair clamp uses four clamping bolts that are each loaded to a minimum preload of 1,380 lbs at the minimum installation temperature of 70 degrees F. The bolt preload will decrease by 10 percent to 1,243 lbs at the 550 degrees F operating temperature. Therefore, the four bolts will apply a minimum beginning-of-life preload of 4,972 lbs. This minimum beginning- of-life preload will further decrease by 5 percent due to thermal and radiation-induced relaxation due to 40 years of operation, resulting in an end-of-life preload of 4,724 lbs. This load was used in the clamping evaluation described below.~~

~~For the static friction coefficient of 0.5, the minimum end-of-life bolt preload will provide a resistance to sliding under lateral loads of 4,724 x 0.5 = 2,362 lbs. This is larger than the limiting lateral load of 2,059 lbs. Therefore, clamping will be maintained under all operating conditions.~~

~~In order to evaluate this TLAA for an additional 20 years, the preload values will be assumed to decrease a further 5 percent due to thermal and radiation-induced relaxation, the same decrease originally predicted for the first 40 years of operation. Therefore, the preload from the four bolts will be reduced from 4,724 lbs to 4,476 lbs. When the static friction coefficient is applied, the minimum 60-year end-of-life bolt preload will be 4,476 x 0.5 = 2,238 lbs. This is larger than the limiting lateral load of 2,059 lbs. Therefore, clamping will continue to be maintained under all operating conditions through the period of extended operation.~~

~~The clamp design analysis also evaluated fatigue. However, the stress amplitude for cyclic loads was well below the ASME Code stress limit of 13,600 psi for 1011 cycles and less than the 10,000 psi lower limit considered for flow-induced vibration stress cycles. Therefore, the fatigue usage will be insignificant. Therefore, even if the number of cycles increased 50 percent during the period of extended operation, the fatigue usage would remain insignificant.~~

The fluence value used to determine the loss of preload in the repair clamp design analysis, 1.0 E+19 n/cm^2, is five percent higher than the 9.5 E+18 n/cm^2 fluence value calculated for the repair location for a 40-year service life. Therefore, the repair clamp design analysis includes an allowance for loss of preload that is valid for 105 percent of 40 years, or 42 years.

Since the first clamps were installed in LGS Unit 2 in April 2009 and the period of extended operation for Unit 2 ends June, 22, 2049, the clamps will have a maximum service life of 40.25 years. No clamps have been installed in LGS Unit 1. Therefore,

since no clamps will have a service life exceeding 42 years, the repair clamp design analysis remains valid for the period of extended operation.

TLAA Disposition: ~~10 CFR 54.21(c)(1)(ii) – The analysis has been projected to the end of the period of extended operation.~~***10 CFR 54.21(c)(1)(i) – The repair clamp design analysis for loss of preload remains valid for the period of extended operation.***

TLAA Description:

The restrainer bracket pad repair clamp design analysis also evaluated fatigue, which has been identified as a TLAA.

TLAA Evaluation:

Clamping loads do not cycle except for the load changes associated with startup/shutdown temperature changes. The largest stress range resulting from the temperature changes is 14,022 psi, calculated for the restrainer bracket, which results in an alternating stress amplitude of 7,011 psi. After adjustment for the difference between operating temperature of 550ºF and the ASME fatigue curve reference temperature of 70ºF, the stress amplitude is 7,781 psi. The restrainer bracket is fabricated from Type 304 stainless steel, with a stress intensity limit of 16,000 psi at 550ºF and yield strength of 17,750 psi at 550ºF. The American Society of Mechanical Engineers (ASME) Code stress limit is 13,600 psi for 10^{11} cycles and the General Electric Hitachi design limit for flow-induced vibration stress cycles is 10,000 psi. Since the cyclic loads of 7,781 psi are less than these limits, fatigue usage will be insignificant for up to 10^{11} startup/shutdown cycles. Therefore the fatigue analysis remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

TLAA Disposition: 10 CFR 54.21(c)(1)(i) – The repair clamp design analysis for fatigue remains valid for the period of extended operation.

As a result of the responses to RAI 4.6.6-1, RAI 4.6.6-2, and RAI 4.6.6-3 provided in the Enclosure A of this letter, UFSAR Supplement, Section A.4.6.6 of the LRA is revised as shown below:

A.4.6.6 Jet Pump Restrainer Bracket Pad Repair Clamps

Visual inspections at LGS have found wear at the inlet-mixer wedge/restrainer bracket pad interface on several jet pumps. A repair clamp has been designed and installed that replaces the support function of the restrainer bracket pad. The repair clamp design analysis evaluated end-of-life preload relaxation for 40 years based upon a 5 percent decrease due to thermal and radiation effects. This analysis was identified as a TLAA.

~~The TLAA was evaluated by assuming a further 5 percent decrease in preload during the period of extended operation. The evaluation determined that adequate clamping force will be maintained with this further reduction of preload through the period of extended operation. The analysis has been projected to the end of the period of extended operation in accordance with 10CFR54.21(c)(1)(ii).~~

The fluence value used to determine the loss of preload in the design analysis is five percent higher than the fluence value calculated for the repair location for a 40-year service life. Since the first clamps were installed in Unit 2 in 2009 and no clamps have been installed in Unit 1, these clamps will have a maximum service life of 40.25 years. Therefore, since the repair clamp design analysis includes an allowance for loss of preload that is bounding for 42 years, the analysis remains valid in accordance with 10 CFR 54.21(c)(1)(i).

The repair clamp design analysis also evaluated fatigue for 40 years, which was identified as a TLAA. The clamping loads do not cycle except for the load changes associated with startup/shutdown temperature changes. The maximum calculated stress amplitude is 7,781 psi, which is within the American Society of Mechanical Engineers (ASME) Code stress limit for 304 stainless steel of 13,600 psi for 10^{11} cycles, and the General Electric Hitachi design limit of 10,000 psi for flow-induced vibration stress cycles. Since the calculated cyclic loads of 7,781 psi are less than these limits, fatigue usage will be insignificant for up to 10^{11} startup/shutdown cycles. Therefore the fatigue analysis remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

As a result of the response to RAI 4.6.9-1 provided in Enclosure A of this letter, LRA Table 3.1.2-3 is revised as shown below:

Table 3.1.2-3 Reactor Vessel Internals (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Jet Pump Assemblies: Thermal sleeve inlet header, Riser brace arm, Holddown beams, Inlet elbow, Mixing assembly, Diffuser Castings, Slip joint clamp, Wedge assemblies	Pressure Boundary	Cast Austenitic Stainless Steel (CASS)	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.9)	IV.B1.R-100	3.1.1-103	A
					Water Chemistry (B.2.1.2)	IV.B1.R-100	3.1.1-103	A
				Loss of Fracture Toughness	BWR Vessel Internals (B.2.1.9)	IV.B1.RP-219	3.1.1-99	A
				Loss of Material	BWR Vessel Internals (B.2.1.9)	IV.B1.RP-26	3.1.1-43	E, 2
					Water Chemistry (B.2.1.2)	IV.B1.RP-26	3.1.1-43	A
		Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.9)	IV.B1.R-100	3.1.1-103	A
					Water Chemistry (B.2.1.2)	IV.B1.R-100	3.1.1-103	A
				Loss of Material	BWR Vessel Internals (B.2.1.9)	IV.B1.RP-26	3.1.1-43	E, 2
					Water Chemistry (B.2.1.2)	IV.B1.RP-26	3.1.1-43	A
					BWR Vessel Internals (B.2.1.9)	IV.B1.RP-377	3.1.1-100	A
				Loss of Preload	TLAA			H, 4
					~~BWR Vessel Internals (B.2.1.9)~~			~~H, 5~~
		X-750 alloy	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.9)	IV.B1.RP-381	3.1.1-104	A
					Water Chemistry (B.2.1.2)	IV.B1.RP-381	3.1.1-104	A
				Cumulative Fatigue Damage	TLAA	IV.B1.R-53	3.1.1-3	A, 8

As a result of the response to RAI 4.6.9-1 provided in Enclosure A of this letter, LRA Table 3.1.2-3 is revised as shown below:

Notes	**Definition of Note**
A	Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.
B	Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP.
C	Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-NUREG-1801 AMP.
D	Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP.
E	Consistent with NUREG-1801 item for material, environment and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program.
F	Material not in NUREG-1801 for this component.
G	Environment not in NUREG-1801 for this component and material.
H	Aging effect not in NUREG-1801 for this component, material and environment combination.
I	Aging effect in NUREG-1801 for this component, material and environment combination is not applicable.
J	Neither the component nor the material and environment combination is evaluated in NUREG-1801.

Plant Specific Notes:

1. The TLAA designation in the aging management program column indicates that fatigue of this component is evaluated in Section 4.3.
2. The BWR Vessel Internals (B.2.1.9) program is substituted to manage the aging effect(s) applicable to this component type, material and environment combination.
3. The TLAA designation in the aging management program column indicates that loss of preload of the core plate rim bolts due to high neutron fluence is evaluated in Section 4.6.
4. The TLAA designation in the aging management program column indicates that loss of preload due to neutron fluence of the jet pump auxiliary spring wedge assemblies, jet pump restrainer bracket pad repair clamps, and jet pump slip joint clamps is evaluated in Section 4.6.
5. Not used. ~~The BWR Vessel Internals (B.2.1.9) program is used to manage loss of preload due to thermal effects and self-loosening of screws and bolts associated with the jet pumps.~~
6. Loss of material due to wear is applicable to the steam dryer support seismic blocks as identified by operating experience review.

As a result of the responses to RAI 4.6.6-1 and RAI 4.6.9-1 provided in Enclosure A of this letter, LRA Table 4.1-2 is revised as shown below:

Table 4.1-2
SUMMARY OF RESULTS - LGS TIME-LIMITED AGING ANALYSES

TLAA DESCRIPTION	DISPOSITION	LRA SECTION
IDENTIFICATION OF TIME-LIMITED AGING ANALYSES		4.1
Identification of LGS Time-Limited Aging Analyses		4.1.1
Evaluation of LGS Time-Limited Aging Analyses		4.1.2
Acceptance Criteria		4.1.3
Summary of Results		4.1.4
Identification and Evaluation of LGS Exemptions		4.1.5
REACTOR PRESSURE VESSEL NEUTRON EMBRITTLEMENT ANALYSIS		4.2
Neutron Fluence Projections	§54.21(c)(1)(ii)	4.2.1
Upper-Shelf Energy	§54.21(c)(1)(ii)	4.2.2
Adjusted Reference Temperature	§54.21(c)(1)(ii)	4.2.3
Pressure – Temperature Limits	§54.21(c)(1)(iii)	4.2.4
Axial Weld Inspection	§54.21(c)(1)(ii)	4.2.5
Circumferential Weld Inspection	§54.21(c)(1)(iii)	4.2.6
Reactor Pressure Vessel Reflood Thermal Shock	§54.21(c)(1)(ii)	4.2.7
METAL FATIGUE		4.3
ASME Section III, Class 1 Fatigue Analyses	§54.21(c)(1)(iii)	4.3.1
ASME Section III, Class 2 and 3 and ANSI B31.1 Allowable Stress Calculations	§54.21(c)(1)(i)	4.3.2
Environmental Fatigue Analyses for RPV and Class 1 Piping	§54.21(c)(1)(iii)	4.3.3
Reactor Vessel Internals Fatigue Analyses	§54.21(c)(1)(i)	4.3.4
High-Energy Line Break (HELB) Analyses Based Upon Fatigue	§54.21(c)(1)(i)	4.3.5
ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC COMPONENTS		4.4
Environmental Qualification (EQ) of Electric Components	§54.21(c)(1)(iii)	4.4.1
CONTAINMENT LINER AND PENETRATIONS FATIGUE ANALYSIS		4.5
Containment Liner and Penetrations Fatigue Analysis	§54.21(c)(1)(i)	4.5.1
OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES		4.6
Reactor Enclosure Crane Cyclic Loading Analysis	§54.21(c)(1)(i)	4.6.1
Emergency Diesel Generator Enclosure Cranes Cyclic Loading Analysis	§54.21(c)(1)(i)	4.6.2
RPV Core Plate Rim Hold-Down Bolt Loss of Preload	§54.21(c)(1)(i)	4.6.3
Main Steam Line Flow Restrictors Erosion Analysis	§54.21(c)(1)(i)	4.6.4
Jet Pump Auxiliary Spring Wedge Assembly	§54.21(c)(1)(i)	4.6.5
Jet Pump Restrainer Bracket Pad Repair Clamps	§54.21(c)(1)~~(ii)~~***(i)***	4.6.6
Refueling Bellows and Support Cyclic Loading Analysis	§54.21(c)(1)(i)	4.6.7
Downcomers and MSRV Discharge Piping Fatigue Analyses	§54.21(c)(1)(i)	4.6.8
Jet Pump Slip Joint Repair Clamps	§54.21(c)(1)~~(iii)~~***(i)***	4.6.9

As a result of the response to RAI 4.6.9-1 provided in Enclosure A of this letter, LRA Section 4.6.9, Jet Pump Slip Joint Repair Clamps, is revised as shown below:

4.6.9 Jet Pump Slip Joint Repair Clamps

TLAA Description:

Jet pump slip joint repair clamps have been designed and installed at LGS to minimize vibration and wear of the jet pump assemblies. The clamps apply a lateral preload to the slip joint between the exit end of the inlet-mixer and the entrance end of the diffuser to damp out jet pump vibration. The design specification for the repair clamp states that the peak neutron fluence at the slip joint is 1.115 E+20 n/cm^2 for the 40-year design life of the clamp. The specification requires the design to account for the effect of fluence on the properties of the slip joint materials for the design life of the clamp. The structural evaluation states that the "cold" bolt preload is 550 lbs at 100 degrees F and the initial preload for operating conditions is 500 lbs at 550 degrees F with a minimum, end-of-life preload of 350 lbs at 550 degrees F. Therefore, 150 lbs of preload loss during the life of the component is associated with neutron fluence. The clamps will be in service for more than 40 years by the end of the period of extended operation. Therefore, loss of preload due to neutron fluence has been identified as a TLAA.

TLAA Evaluation:

~~Jet pumps and repair hardware are required to be periodically inspected by the Reactor Vessel Internals program. Aging effects that are included within the scope of these inspections are cracking, wear and bolt loosening. Therefore, this TLAA will be managed for the period of extended operation.~~

The design specification for the slip joint repair clamps requires the analysis to account for a peak neutron fluence of 1.115 E+20 n/cm^2 predicted for 40 years of service. The jet pump slip joint repair clamp structural evaluation includes an allowance for loss of preload during the 40-year life of the component that will result from this fluence exposure.

RAMA fluence projections have been prepared that determine the maximum fluence the slip joint clamps can receive during their service life. Slip joint repair clamps were initially installed in Unit 1 in March, 2006 and in Unit 2 in March, 2005. The period of extended operation ends on October 26, 2044 for Unit 1 and on June 22, 2049 for Unit 2. Therefore, the service life for the Unit 1 slip joint clamps will be less than 40 years. The service life for the Unit 2 slip joint clamps will be less than 45 years.

Since the core shroud is closer to the reactor core than the jet pumps, the peak fluence received on the outer surface of the core shroud is less than the fluence at the same elevation on the jet pumps due to the extra shielding provided by the reactor coolant in the space between the shroud and the jet pumps. Therefore, the peak fluence on the outer surface of the core shroud at the 207.3-inch elevation of the slip joint clamps was used to provide a conservative estimate of the fluence at the highest point on the slip joint clamps. The resulting fluence projection is 1.24 E+18 n/cm^2 for the Unit 1 slip joint clamps and is 1.28 E+18 n/cm^2 for the Unit 2 slip joint clamps. Since these fluence values are less than the 1.115 E+20 n/cm^2 fluence value used to determine the loss of preload analyzed in the structural evaluation of the clamps, the analysis remains valid for the period of extended operation.

~~TLAA Disposition: 10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.~~

TLAA Disposition: 10 CFR 54.21(c)(1)(i): The slip joint clamp design analysis remains valid for the period of extended operation.

As a result of the response to RAI 4.6.9-1 provided in Enclosure A of this letter,
The UFSAR Supplement LRA Section A.4.6.9, Jet Pump Slip Joint Repair Clamps, is revised as shown below:

A.4.6.9 Jet Pump Slip Joint Repair Clamps

Jet pump slip joint repair clamps have been designed and installed at LGS to minimize vibration and wear of the jet pump assemblies. The structural evaluation determined the loss of preload that would result from neutron fluence during the design life of the clamps. This analysis was identified as a TLAA. ~~The jet pumps and repair hardware are required to be periodically inspected by the Reactor Vessel Internals program. Therefore, this TLAA will be managed in accordance with 10 CFR 54.21(c)(1)(iii).~~ ***The maximum service life of the slip joint clamps from initial installation through the period of extended operation will be less than 40 years for the Unit 1 clamps and less than 45 years for the Unit 2 clamps. RAMA fluence projections for each unit have determined the maximum fluence the slip joint clamps can receive during their service life. These fluence values are less than the fluence value used to determine the loss of preload allowance applied in the structural evaluation of the clamps. Therefore, the jet pump slip joint clamp design analysis for loss of preload due to neutron fluence remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).***