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10 CFR 50
10 CFR 51
10 CFR 54

February 29, 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 January 31, 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, Units1 and 2, License Renewal Application (TAC Nos. ME6555, ME6556)", dated January 31, 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 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 02-29-2012

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 LGS 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-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 LGS License Renewal Application (LRA)

RAI 4.1-1
RAI 4.3-1
RAI 4.3-2
RAI 4.3-3
RAI 4.3-4
RAI 4.3-5
RAI 4.3-6
RAI 4.3-7
RAI 4.3-8
RAI 4.3-9
RAI 4.3-10
RAI 4.3-11
RAI 4.3-12
RAI 4.6.8-1
RAI 4.6.8-2

<u>RAI 4.1-1</u>

<u>Background</u>

The Updated Final Safety Analysis Report (UFSAR), Section 5.4.5.2, states that the design objective for the main steam isolation valve (MSIV) is a minimum of 40 years' service at the specified operating conditions. Operating cycles (excluding exercise cycles) are estimated to be 50 cycles per year during the first year and 20 cycles per year thereafter. License Renewal Application (LRA) Section 2.3.1.1 states that the Reactor Coolant Pressure Boundary includes the main steam piping and components from the piping attached to the reactor pressure vessel (RPV) nozzles to the outboard containment isolation valves.

<u>Issue</u>

The staff noted that the MSIV analysis performed was based on number of operating cycles. However, the applicant did not identify this analysis as a time-limited aging analysis (TLAA) in the LRA.

<u>Request</u>

Justify why the MSIV analysis performed based on operating cycles need not be identified as a TLAA in accordance with 10 CFR 54.21(c)(1). If the analysis needs to be identified as a TLAA, provide necessary information and LRA revision to support the TLAA disposition.

<u>Exelon Response</u>

The MSIV fatigue analysis is a TLAA identified and evaluated in LRA Section 4.3.1. The operating cycles described in UFSAR Section 5.4.5.2 are included in the thermal and pressure cycles evaluated in the MSIV Class 1 fatigue analysis.

UFSAR Table 3.9-6 (h), "Main Steam Isolation Valve" contains a summary of the fatigue analysis, which consists of two parts. The first part, "Normal Duty Valve Fatigue Requirements", addresses startup and shutdown cycles at a 100ºF per hour. For normal duty fatigue, the table shows that the valve is qualified for 55,000 full startup and shutdown cycles, which satisfies the Code minimum criterion of 2,000 cycles. UFSAR Table 3.9-6 (h) also summarizes a second part of the MSIV fatigue analysis, "Cyclic Loading Requirements at Valve Crotch," which shows that the fatigue usage factor is satisfied for transients not excluded by Code. The MSIV fatigue analysis is based on the transients listed in UFSAR Section 3.9.1.1.8, "Main Steam Isolation Valve Transients."

The MSIV fatigue analysis is typical of the analyses performed for ASME Section III Class 1 valves identified as TLAAs in LRA Section 4.3.1. No LRA change is required.

RAI 4.3-1

Background

LRA Table 4.3.1-1 indicates that the cumulative cycles of “Startup” (Transient No. 3) and “Shutdown” (Transient No. 10) are 52 and 50, respectively, for Limerick Generating Station (LGS), Unit 1 as of January 2011. Furthermore, for LGS, Unit 1 as of January 2011, the table indicates that there were 14 occurrences of “Scram – Turbine-Generator Trip, Feedwater Stays ON, Isolation Valves Stay OPEN” (Transient No. 9a) and 47 occurrences of “Scram-all other Scrams” (Transient No. 9b).

LRA Table 4.3.1-2 indicates that the cumulative cycles of “Startup” (Transient No. 3) and “Shutdown” (Transient No. 10) are 35 and 33, respectively, for LGS, Unit 2 as of January 2011. Furthermore, for LGS, Unit 2 as of January 2011, the table indicates that there were 14 occurrences of “Scram – Turbine-Generator Trip, Feedwater Stays ON, Isolation Valves Stay OPEN” (Transient No. 9a) and 35 occurrences of “Scram-all other Scrams” (Transient No. 9b).

Issue

It is not clear to the staff why there are more occurrences of the “Startup” transient than the “Shutdown” transient for each unit. The staff also noted that, for LGS, Unit 1, there are 61 scrams (Transients 9a and 9b) compare to 50 occurrences of the “Shutdown” transient and that, for LGS, Unit 2, there are 39 scrams (Transients 9a and 9b) as compared to 33 occurrences of the “Shutdown” transient. It is not apparent to the staff the relationship between Transients 9a, 9b, and 10 for both units.

The staff noted that the applicant projected the 60-years of occurrences based on the baseline cumulative cycles in LRA Tables 4.3.1-1 and 4.3.1-2. The projections are used to support the TLAA disposition in LRA Section 4.

Request

1) For each unit, provide clarification for why the cumulative cycle count of the “Startup” transient is greater than that of the “Shutdown” transient.

2) For each unit, provide clarification for why the sum of the cumulative cycle counts for Transients 9a and 9b is greater than that of the “Shutdown” transient.

Exelon Response

1) There are two reasons that, as of January 2011, there had been two more Startup transients than Shutdown transients in each unit, as shown in LRA Table 4.3.1-1 for Unit 1 and Table 4.3.1-2 for Unit 2. The first reason is that, as of the date the data was obtained in January 2011, each unit was operating, which accounts for one additional Startup as compared to Shutdowns. The other reason is that each unit has experienced one Emergency Scram – Single Relief Valve or Safety Valve Blowdown transient (Transient No. 14d), which is a more severe transient that transitions the reactor from operating temperature to cold shutdown. In each of these cases, one Emergency Scram – Single Relief Valve or Safety Valve Blowdown transient was recorded, as required, since the event was more severe than a Shutdown transient, and no Shutdown transient was required to be recorded. Therefore, for each unit, the cumulative cycle counts for these events correlate properly.

2) The reason the sum of the Scram transients reported in LRA Tables 4.3.1-1 and 4.3.1-2 for events 9a and 9b, Scrams, exceeds the number of event 10, Shutdowns, is that these events do not necessarily correlate with each other. Scram events transition the plant from power operation to hot standby, which is generally a short period during which preparations are made for further transitioning to cold shutdown. If a transition to cold shutdown is performed after a Scram, then one Scram transient is recorded and one Shutdown transient is recorded. There are occasions, however, when the reactor is placed in hot standby with the intention that it returns to normal power operation after a short period of time rather than to cold shutdown. In cases where the reactor is returned to power operation directly from hot standby, one Scram transient is recorded but no Shutdown transient is recorded. Also, for planned outages, the plant may transition from power operations to cold shutdown via manual control rod insertion without initiating a scram. In this case, one Shutdown transient is recorded and no Scram transient is recorded. The fatigue monitoring results reflect the actual transients that occurred for each unit.

RAI 4.3-2

Background

UFSAR Table 3.9-1, Section T “General Electric Criteria for NSSS Piping” indicates that the “Turbine Stop Valve Closure” transient is an upset transient with a design of 120 cycles. UFSAR Table 3.9-1, Section T also indicates that the “Relief Valve Lift Cycles” transient is an upset transient with a design of 34,200 cycles. UFSAR Figure 3A-394 indicates that “Chugging” is a transient used as an input into the fatigue analysis of main steam relief valve (MSRV) downcomers with a design of 3000 cycles.

Issue

The staff noted that these transients were not included in LRA Tables 4.3.1-1 and 4.3.1-2; therefore, it is not clear whether these transients have been used as inputs into the TLAAs discussed in LRA Section 4. The staff noted that if these transients were input into the TLAAs that are dispositioned in accordance with 10 CFR 54.21(c)(1)(i), the accumulated number of cycles and the 60-years projected number of cycles are needed to verify the adequacy of the TLAA disposition. However, if these transients were inputs into the TLAAs that are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii) and the Fatigue Monitoring Program is credited, the applicant needs to include these transients in the Fatigue Monitoring Program and the cycle-counting procedures, consistent with GALL Report AMP X.M1 to monitor all plant design transients that cause cyclic strains, which are significant contributors to the fatigue usage factor.

Request

1) Identify the TLAAs in LRA Section 4 that used these transients (Turbine Stop Valve Closure, Relief Valve Lift Cycles and Chugging) as an input. Confirm that these transients were monitored since initial plant startup for each unit. If not, justify how the accumulated cycles to date were reconciled.

2) If the identified TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), clarify whether these transients are currently included in the Fatigue Monitoring Program and the cycle-counting procedures. If not, justify why these transients do not need to be monitored by the Fatigue Monitoring Program.

3) If the identified TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i), provide the accumulated number of occurrences for each transient up to January 2011 in LRA Tables 4.3.1-1 and 4.3.1-2. Provide the 60-year projected number of occurrence for these transients in LRA Tables 4.3.1-1 and 4.3.1-2 and justify that these projections are conservative.

Exelon Response

1) The TLAAs that use these transients are the fatigue analyses for the Reactor Pressure Vessel (RPV), RPV internals and supports, and Class 1 piping. The fatigue analyses for the RPV and Class 1 piping are evaluated in LRA Section 4.3.1 that are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), crediting the Fatigue Monitoring program. The RPV internals fatigue TLAAs are evaluated in LRA Section 4.3.4. Refer to the Exelon response to RAI 4.3-10 that provides a revised LRA Section 4.3.4 and changes the disposition for the RPV internals fatigue TLAAs to 10 CFR 54.21(c)(1)(iii), crediting the Fatigue Monitoring program.

   The transient characterized in UFSAR Table 3.9-1 as "Turbine Stop Valve Closure," is an event included within the definition of LGS Transient No. 9a – "Scram, Turbine Generator Trip, Feedwater Stays ON, Isolation Valves Stay OPEN," which has a design limit of 40 events. Event 9a is used as an input in each ASME Section III, Class 1 fatigue TLAA evaluated in LRA Section 4.3.1.

   The fatigue analysis for the main steam piping assumes three stress cycles for each turbine stop valve closure, resulting in 120 total stress cycles from the 40 events. The LGS Fatigue Monitoring program has monitored these events since initial plant startup for each unit. The LGS Fatigue Monitoring program limits the number of turbine stop valve closure events to 40 and LRA Tables 4.3.1-1 and 4.3.1-2 show that the 60-year projection for Event 9a does not exceed the 40-year design limit. Therefore, assuming no more than three stress cycles per event and no more than 40 events, the 120 stress cycles analyzed will not be exceeded during the period of extended operation.

   UFSAR Table 3.9-1, Section T, “General Electric Criteria for NSSS Piping,” also includes Relief Valve Lift Cycles (at 3 cycles per actuation) - 34,200 cycles. This equates to 11,400 MSRV actuations. The downcomers and MSRV discharge piping were analyzed for 1,100 MSRV actuations, which is more limiting and is the basis for the limit being imposed in the Fatigue Monitoring program for MSRV actuations. The Fatigue Monitoring program has an existing commitment to monitor additional transients that are significant contributors to fatigue usage. The MSRV actuations are to be monitored, with a limit of 1,100 MSRV actuations, as discussed in the Exelon response to RAI 4.6.8-1. Therefore, the MSRVs will not exceed 11,400 actuations through the period of extended operation. Refer to the Exelon responses to RAI 4.6.8-1 and RAI 4.6.8-2 for additional detail and for changes to LRA Tables 4.3.1-1 and 4.3.1-2 regarding MSRV actuations.

   Chugging cycles are loads that result from various modes of steam condensation at the downcomer vent ends following a LOCA. The Fatigue Monitoring program includes Event 18, Faulted Condition – Pipe Rupture and Blowdown, which corresponds to a LOCA and the monitoring results show no LOCA event has occurred in either unit. Therefore, no chugging events have occurred to-date in either unit. Refer to the Exelon responses to RAI 4.6.8-1 and RAI 4.6.8-2 for additional details regarding chugging cycles and for changes to LRA Table 4.3.1-1, LRA Table 4.3.1-2, LRA Section 4.6.8, and UFSAR Supplement Section A.4.6.8.

2) The Main Steam system piping TLAA is dispositioned in LRA Section 4.3.1-1 in accordance with 10 CFR 54.21(c)(1)(iii), and the Fatigue Monitoring Program is credited. The Exelon responses to RAI 4.6.8-1 and RAI 4.6.8-2 revise the TLAA disposition for the MSRV discharge piping and downcomer piping from 10 CFR 54.21(c)(1)(i) to 10 CFR 54.21(c)(1)(iii).

3) This question is not applicable based upon the TLAA disposition provided in response 2.

RAI 4.3-3

Background

LRA Table 4.3.1-1 indicates that the “Adjusted 60-year Projected Cycles” are two cycles for the “Core Spray” and the “Low pressure Coolant Injection” transients for LGS, Unit 1. LRA Tables 4.3.1-2 also provides the same information about these transients for LGS, Unit 2.

Issue

The staff noted that the “Design Cycle Limits” for these two transients are not provided in LRA Tables 4.3.1-1 and 4.3.1-2. In addition, the applicant did not explain or justify why the design cycle limits are not needed. The staff noted that if these transients were used as an input into the TLAAs that are dispositioned in accordance with 10 CFR 54.21(c)(1)(i), the design cycle limits for each of the transients are needed to verify the adequacy of the applicant’s TLAA disposition. The staff also noted that if the transients were used as inputs into the TLAAs that are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii) and the Fatigue Monitoring Program is credited, the design cycle limits for each transient are needed because LRA Section B.3.1.1 states that in order to prevent exceeding a cycle limit, corrective actions are triggered.

Request

1) Identify the TLAAs in LRA Section 4 that used these transients as an input and identify the associated design cycle limits.

2) If the design cycle limits are not provided for the identified TLAAs that are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), justify how the Fatigue Monitoring Program can be used to monitor these two transients and ensure that corrective action will be triggered when the cycle limit is approached.

3) If the design cycle limits are not provided for the identified TLAAs that are dispositioned in accordance with 10 CFR 54.21(c)(1)(i), justify that the “Adjusted 60-year Projection Cycles” for both transients are bounded by the design cycle limits.

Exelon Response

1) LRA Tables 4.3.1-1 and 4.3.1-2 include a section for ECCS / RCIC and SLC Injections that includes cycles for "Core Spray" and "Low Pressure Coolant Injections." These are not thermal transients that are used as inputs in fatigue TLAAs. Rather, these are events that result from the numbered thermal transients provided in these tables. The ECCS / RCIC injections are monitored separately to ensure that NRC reporting requirements are met regarding fatigue usage of RPV nozzles resulting from ECCS and RCIC injections. Since the ECCS / RCIC injections are not transient events used as inputs in the fatigue analyses,

and since this information is not used in the evaluation of the fatigue TLAAs, LRA Tables 4.3.1-1 and 4.3.1-2 are revised to remove this information for clarity, as shown in Enclosure B.

2) Not applicable due to the response to question 1.

3) Not applicable due to the response to question 1.

RAI 4.3-4

Background

LRA Section 4.3.2 indicates that the TLAA for American Society of Mechanical Engineers (ASME) Code, Section III, Class 2 and 3 and B31.1 allowable stress calculations is dispositioned in accordance with 10 CFR 54.21(c)(1)(i), that the calculations remain valid for the period of extended operation. LRA Section 4.3.2 and UFSAR Section A.4.3.2 also state that those systems that are not connected to ASME Code, Section III, Class 1 piping are affected by "different thermal and pressure cycles" and an operational review was performed to conclude that the total number of transient cycles for these systems will not exceed 7000. These systems include the Fire Protection, Emergency Diesel Generator and Auxiliary Steam systems.

Issue

LRA Section 4.3.2 did not provide information regarding the accumulated number of occurrences and the 60-year projected number of occurrences for these "different thermal and pressure cycles." Therefore, the staff cannot verify the adequacy of the disposition in accordance with 10 CFR 54.21(c)(1)(i). Furthermore, UFSAR Section A.4.3.2 indicates that the TLAA will be adequately managed for the period of extended operation by the Fatigue Monitoring Program in accordance with 10 CFR 54.21(c)(1)(iii), which is different from the disposition identified in LRA Section 4.3.2.

Request

1) Identify the "different thermal and pressure cycles" that were considered in the allowable stress calculations for the systems that are not connected to ASME Code Section III Class 1 piping.

2) Identify the accumulated number of occurrences for each transient used in these allowable stress calculations up to January 2011. Confirm that these transients were monitored since initial plant startup for each unit. If not, justify how the accumulated cycles to date were reconciled. Provide the 60-year projected number of occurrences for each transient identified above and justify that these projections are conservative.

3) Clarify the TLAA disposition for "ASME Code Section III, Class 2 and 3 and B31.1 allowable stress calculations."

4) Revise LRA Sections 4.3.2 and A.4.3.2 to be consistent with the changes discussed in the response.

Exelon Response

1) The ASME Code, Section III, Class 2 and 3, and ANSI B31.1 piping systems not connected to Class 1 piping that were evaluated include:

   - The diesel engine exhaust piping from the diesel-driven fire pump and backup diesel-driven fire pump in the Fire Protection System. The engine exhaust piping from the diesel-driven fire pumps experiences a thermal cycle each time the engine starts and is later shutdown.

   - The emergency diesel generators (EDGs) exhaust piping in the Emergency Diesel Generator System. The diesel generator engine exhaust piping experiences a thermal cycle each time the engine starts and is later shutdown.

   - The piping from the auxiliary boilers used for plant heating steam in the Auxiliary Steam System. The portions of the plant heating steam piping within the scope of license renewal experience a thermal cycle each time the plant heating steam system is placed in service and is later shutdown.

2) Each of these systems have 60-year cycle projections based upon estimated operational cycles that show the components will experience much less than 7000 cycles through the period of extended operation, as described below for each system. Therefore, these TLAAs will remain valid for the period of extended operation.

   The diesel-driven fire pump is common for both units. The diesel-driven fire pump engine runs were not monitored since initial plant startup, but the number of cycles has been estimated based on operational history. The diesel-driven fire pump engine is normally not in service and is only operated during surveillance testing and in response to a fire when the motor-driven pump does not maintain fire water header pressure. It is estimated that the engine runs no more than two times per year to maintain fire water header pressure. It is also operated an average of 13 times per year during surveillance testing. Based on an average of 15 cycles per year, the diesel-driven fire pump exhaust piping has experienced approximately 405 thermal cycles during 27 years of operation of Unit 1 prior to January 2011 and will experience approximately 975 thermal cycles through the period of extended operation for Unit 2 (65 years total).

   The backup diesel-driven fire pump is also common for both units. The backup diesel-driven fire pump engine runs have not been monitored since initial plant startup, but the number of cycles has been estimated based on operational history. The backup diesel-driven fire pump engine is normally not in service and is only operated during surveillance testing and in response to a fire when both the motor-driven pump and diesel-driven pump do not maintain fire water header pressure. It is estimated that the engine runs no more than two times per year to maintain fire water header pressure. It is also operated an average of 13 times per year during surveillance testing. Based on an average of 15 cycles per year, the diesel-driven fire pump engine exhaust piping has experienced approximately 405 thermal cycles during 27 years of operation of Unit 1 operation prior to January 2011 and will experience approximately 975 thermal cycles through the period of extended operation for Unit 2 (65 years total).

EDG engine runs have not been monitored since initial plant startup, but the number of cycles has been estimated based on operational history. The EDGs are normally not in service but are operated in response to a loss of coolant accident or a loss of off-site power. Since these events occur very infrequently, it is estimated that each EDG is run no more than one time per year in response to these events. A review of surveillance test frequencies determined that each EDG is run an estimated 22 times per year for testing. A review of the system manager's informal log of actual EDG runs revealed that the average number of runs per EDG in 2010 and 2011 was 19 per year, confirming that 22 cycles per year is a conservative estimate for annual EDG surveillance test runs. Based on an average of 23 cycles per year, the exhaust piping for Unit 1 EDGs have experienced approximately 621 thermal cycles during 27 years of operation prior to January 2011. The exhaust piping for Unit 2 EDGs have experienced approximately 552 thermal cycles during 24 years of operation prior to January 2011. Based on an average of 23 cycles per year, each EDG will experience approximately 1380 thermal cycles though 60 years of operation.

The Auxiliary Steam System is common for both units. Auxiliary Steam System thermal cycles have not been monitored since initial plant startup, but the number of cycles has been estimated based on operational history. The Auxiliary Steam System is typically maintained in service for the entire heating system from Fall until Spring each year. However, the system may be removed from service during each heating season to perform maintenance or during periods of warm weather when the plant heating system is not needed. Based on an estimated average of five cycles per year, the Auxiliary Steam System piping has experienced approximately 135 thermal cycles during 27 years of operation of Unit 1 operation prior to January 2011 and will experience approximately 325 thermal cycles through the period of extended operation for Unit 2 (65 years total).

3) The fatigue TLAAs for the ASME Section III, Class 2 and 3 and ANSI B31.1 piping systems that are not connected to Class 1 piping systems have been demonstrated to remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i). This is a separate disposition from the one for the ASME Section III, Class 2 and 3 and B31.1 that are connected to the Class 1 piping, which are managed by the Fatigue Monitoring program in accordance with 10 CFR 54.21(c)(1)(iii).

4) LRA Section 4.3.2 and UFSAR Supplement Section A.4.3.2, are revised as shown in Enclosure B to include two TLAA dispositions; one for the ASME Code, Section III, Class 2 and 3 and B31.1 piping systems that are connected to ASME Code, Section III, Class 1 piping, and one for ASME Code, Section III, Class 2 and 3 and B31.1 piping systems with that are not connected to Class 1 piping. LRA Table 4.1-2 is revised to show the change in TLAA disposition as shown in Enclosure B.

RAI 4.3-5

Background

LRA Table 4.3.3-1 indicates that, for the incore housing penetration, the ASME Code based 60-year cumulative usage factor (CUF), the NUREG/CR-6909 based 60-year CUF, and the 60-year environmentally assisted fatigue cumulative usage factor (CUF_{en}) are 0.108, 0.140, and 0.83, respectively.

Issue

During its audit, the staff noted that the CUF and CUF_{en} values for this component in the basis documents are different from those in LRA Table 4.3.3-1.

Request

1) Clarify the correct values associated with the incore housing penetration and revise LRA Table 4.3.3-1 as necessary to provide correct CUF and CUF_{en} values.

2) Confirm that the remaining information in LRA Table 4.3.3-1 is accurate. Provide appropriate revisions if any inaccurate information is discovered.

Exelon Response

1) The corrected values for incore housing penetration for both units are as follows:
The node location is 152, the ASME CUF = 0.181, the NUREG/CR-6909 CUF = 0.240, the F_{en} =2.42, and the CUF_{en} = 0.581. LRA Table 4.3.3-1 is revised as shown in Enclosure B.

2) Refer to the response to RAI 4.3-9 that provides the results from a revised environmental fatigue analysis for the Core Spray nozzle forging in each unit. This resulted in changes to LRA Table 4.3.3-1 that are also shown in Enclosure B. The remaining values in Table 4.3.3-1 are accurate.

RAI 4.3-6

Background

LRA Section 4.3.3 states that in order to ensure that any other locations that may not be bounded by the NUREG/CR-6260 locations were evaluated, environmental fatigue calculations were performed for each RPV component location that has a reported CUF value in the stress report and for each ASME Code, Section III, Class 1 reactor coolant pressure boundary (RCPB) piping system in each unit. These calculations were performed for the limiting location for each material within the component or system that contacts reactor coolant.

LRA Section 4.3.1 states that the ASME Code, Section III, Class 1 fatigue analyses include the stress reports for the RPV, RCPB piping and components, including ASME Code, Section III, Class 1 valves.

Issue

The methodology and criteria used by the applicant to select the “limiting locations” provided in LRA Tables 4.3.3-1 and 4.3.3-2 for environmentally assisted fatigue (EAF) is not clear. Therefore, it is not apparent to the staff whether other locations should have been considered and were not provided in these tables. The staff needs to understand the methodology and criteria used when selecting these additional “limiting locations” to address the effects of reactor water environment on fatigue life.

The staff noted that only selecting the locations with the highest cumulative usage without considering the specific RPV component or RCPB system, including the associated thermal transients, water chemistry conditions, material and effects of the connected piping (e.g. plant-specific configuration), may not provide the most critical locations to consider environmentally assisted fatigue.

It is also not apparent to the staff whether ASME Code, Section III, Class 1 valves were considered when determining the "limiting locations" to address the effects of reactor water environment. The staff noted that LRA Tables 4.3.3-1 and 4.3.3-2 do not provide CUF_{en} for any ASME Code, Section III, Class 1 valves.

Request

1) Describe and justify the adequacy of the methodology used to select the limiting RPV and RCPB locations in LRA Tables 4.3.3-1 and 4.3.3-2.

2) Discuss whether the variation of thermal transient loadings, water chemistry conditions, material, and (where relevant) the effects of the attached piping and their effects on different portions of a component were considered when selecting additional limiting locations in the RPV and each RCPB ASME Code, Section III, Class 1 system. If not, for each factor justify that it was not relevant when selecting the "limiting locations" to address the effects of reactor water environment.

3) Clarify whether ASME Code, Section III, Class 1 valves were considered when selecting "limiting locations." Discuss and justify that the effects of reactor water environment on ASME Code, Section III, Class 1 valves have been evaluated.

Exelon Response

1) NUREG/CR-6260 RPV Component Locations: The initial environmental fatigue analyses were performed for the RPV component locations identified in NUREG/CR-6260 for the newer-vintage BWR plant design. This included the RPV shell and lower head (including CRD housing, stub tube, and vessel shell adjacent to stabilizer bracket), feedwater nozzle and safe end, recirculation inlet and outlet nozzle and safe end, core spray nozzle and safe end, and LPCI nozzle and safe end. All of the environmental fatigue analyses performed for LGS are based upon NUREG/CR-6909 methodology, as described in LRA Section 4.3.3.

   Other RPV Component Locations: In order to determine if any other RPV component locations beyond those identified in NUREG/CR-6260 are limiting, additional environmental fatigue analyses were performed. For the RPV, the remaining RPV locations that have a reported CUF value in the design stress report were evaluated. This does not include locations identified in the stress report as having met the exemption requirements of ASME Section III and therefore had no CUF value reported. For each RPV component with a reported CUF value, a further review was performed to determine each material type within the component that contacts reactor coolant (disregarding cladding) and for each material type, the location with the highest CUF value was evaluated for environmental fatigue. Therefore, the limiting location was evaluated for each material type within each component with a CUF value reported in the RPV stress report.

   NUREG/CR-6260 RCPB Piping Locations: The RCPB locations identified in NUREG/CR-6260 for LGS include the RHR Return piping, LPCI piping, Feedwater piping, Core Spray piping, and Reactor Recirculation piping. Each of these locations was then analyzed for reactor water environmental effects using environmental fatigue multipliers determined for the location, based upon the appropriate dissolved oxygen content values determined for the system.

Other RCPB Locations: The stress report for each Class 1 RCPB piping system was reviewed to determine the limiting location to be evaluated. The stress reports for systems that only have dry steam inside were included to demonstrate that no Class 1 piping system was overlooked. In some cases, a single stress report includes coupled piping from more than one system. The location within each Class 1 stress report with the highest CUF value was selected for environmental fatigue evaluation. This approach is valid because the entire system is analyzed for the same thermal and pressure transients, so the stress and fatigue analysis determines the limiting location. If multiple materials were included within the analysis, calculations were performed for the limiting location for each wetted material type. Therefore, the limiting location was evaluated for each material type within each component analyzed in each Class 1 system stress report.

2) Each Class 1 stress report evaluates the components within its scope for the same thermal and pressure transients. The applied loadings from interfacing systems are evaluated within each stress analysis, which is used to determine the fatigue usage values. For the environmental fatigue analyses, dissolved oxygen values were determined for each region within the RPV based upon reactor vessel water chemistry data for each of three operating regimes, including Normal Water Chemistry (NWC), Hydrogen Water Chemistry (HWC), and HWC and Noble Metal Chemical Addition (NMCA). Separate F_{en} multipliers were determined for each of these regimes, and a weighted average was determined based on operational dates applicable for each regime. Dissolved oxygen values were also determined for each piping system, and F_{en} multipliers were determined in a similar manner. Therefore the environmental fatigue analyses properly considered the effects of thermal and pressure transient loadings, material types, water chemistry conditions throughout the history of each unit, and the effects of attached piping.

3) ASME Section III, Class 1 valve design reports were reviewed and representative valves were evaluated for environmental fatigue effects. These analyses are based upon NUREG/CR-6909 methodology, as described in LRA Section 4.3.3. The analyzed valves were demonstrated to have an environmentally-adjusted usage value that does not exceed the design limit of 1.0. The results from these valves are representative of the remaining valves.

   The ASME Code design requirements specified in NB-3500 for the Class 1 valves are different than the requirements for the Class 1 piping systems. NB-3552 allows the following cycles to be excluded from valve cyclic loading analysis.

   1) Cycles with pressure variations less than $p_d/3$ for carbon and low alloy steels and less than $p_d/2$ for austenitic stainless steels;
   2) Cycles with temperature variations less than 30°F;
   3) Accident or mal-operation cycles expected to occur less than five times (total) during the expected valve life;
   4) Startup, shutdown cycles with temperature change rates of 100°F / hour or less not in excess of 2000.

   The LGS Class 1 valve fatigue analyses did not always exclude these transients and therefore have conservative fatigue usage results in many cases. The environmental fatigue analyses performed for the representative valves removed conservatism from the original analyses by removing events that meet the exclusion criteria described above. The delta-T values for the revised transient sets were updated, and a revised usage value was computed. Then an environmental fatigue multiplier was developed, based upon temperatures associated with the evaluated transients and environmental conditions appropriate for the system. The resulting environmentally-adjusted CUF values were determined to be less than the design limit of 1.0.

The table shown below identifies the valves that were analyzed for environmental fatigue and provides the results. The table also lists similar valves that are represented by the analyzed valves, along with some of the characteristics used to show their similarity to the evaluated valves. The comparisons are explained further after the table. Based upon the environmental fatigue evaluations shown below, the limiting Class 1 valves have been satisfactorily evaluated for environmental fatigue.

ASME Section III, Class 1 Valve Environmental Fatigue Evaluation									
						ASME	NUREG/CR-6909		
Valve No.[1]	System	Size (in.)	Type	Material	Class	U_{orig}	U_{adj}	F_{en}	CUF_{en}
F031	Recirculation	28	Gate	SS	900#	0.0039	0.039	11.07	0.428
F023	Recirculation	28	Gate	SS	900#	Represented by F031 valve			
F011	Feedwater	24	Gate	CS	900#	0.879	0.194	2.23	0.432
F010	Feedwater	24	Check	CS	900#	0.499	0.194	2.23	0.432
F074	Feedwater	24	Check	CS	900#	0.499[2]	0.194	2.23	0.432[2]
F077	RHR SDC	20	Gate	SS	900#	0.212	0.048	11.07	0.533
F008	RHR SDC	20	Gate	SS	900#	Represented by F077 valve			
F009	RHR SDC	20	Gate	SS	900#	Represented by F077 valve			
F050	RHR SDC	12	Check	SS	900#	0.213	0.209	3.02	0.629
F015	RHR SDC	12	Globe	SS	900#	Represented by F050 valve			
F060	RHR SDC	12	Gate	SS	900#	Represented by F050 valve			
F065	RHR LPCI	12	Gate	SS	900#	Represented by F050 valve			
F007	Core Spray	12	Gate	SS	900#	Represented by F050 valve			
108, 208	Core Spray	12	Check	SS	900#	Represented by F050 valve			
F005	Core Spray	12	Gate	CS	900#	No [3] usage	No [4] usage	N/A	N/A
F006	Core Spray	12	Check	CS	900#	Represented by F005 valve [5]			
F017	RHR LPCI	12	Gate	CS	900#	No [3] usage	No [4] usage	N/A	N/A
F041	RHR LPCI	12	Check	CS	900#	Represented by F017 valve [5]			
F001	RWCU	6	Gate	SS	900#	0.018	0.031	11.07	0.343
F004	RWCU	6	Gate	SS	900#	0.018[6]	0.031	11.07	0.343
F027	RWCU	6	Gate	SS	900#	Represented by F004 valve			
F105	RWCU	6	Globe	SS	900#	Represented by F004 valve			
F100 (U/1)	RWCU	6	Gate	SS	900#	Represented by F004 valve			

Note 1: These valve numbers represent multiple units and multiple trains, such as 1F031A/B and 2F031A/B.
Note 2: The original design analysis for the F074 valve is identical to the valve analysis for valve F010.
Note 3: The original design analysis demonstrated that all transients meet the exclusion criteria.
Note 4: The environmental fatigue analysis concluded that all transients meet the exclusion criteria.
Note 5: The transients were reviewed for this application and all transients meet the exclusion criteria.
Note 6: The original design analysis for the F001 valve is also applicable for the F004 valve.

Valve F031 is a 28-inch stainless steel recirculation pump discharge valve. The environmental fatigue results for the F031 valve are representative for the F023 28-inch recirculation pump suction valve since they are similar in material, pressure class, and are exposed to similar transients and environment.

Valve F011 is a 24-inch carbon steel feedwater inboard manual isolation valve. Valve F010 and F074 are 24-inch feedwater inboard and outboard isolation check valves. The environmental fatigue analysis performed for these valves used bounding input values for all three locations to determine the bounding alternating stress used to find the number of cycles. Therefore, this analysis is applicable to all three valves.

Valve F077 is a 20-inch stainless steel RHR shutdown cooling supply manual valve. The environmental fatigue results for this valve are representative of the F008 and F009 RHR shutdown cooling supply isolation gate valves since they are all 20-inch, stainless steel, 900 lb. pressure class gate valves that are affected by similar transients and environmental conditions.

Valve F050 is a 12-inch stainless steel RHR shutdown cooling return manual valve. The environmental fatigue results for this valve are representative of the F015 RHR shutdown cooling return check valve and F060 RHR shutdown cooling return isolation valve because they are all 12-inch, stainless steel, 900 lb. pressure class valves affected by similar transients and environmental conditions.

Valve F050 is also representative of the F065 RHR LPCI injection manual isolation valve and the F007 and 108/208 core spray injection isolation valves because they are all 12-inch, stainless steel, 900 lb. pressure class valves. The RHR shutdown cooling system valves are exposed to transients associated with shutdown cooling operations that are not experienced by the RHR LPCI and core spray injection valves. The RHR LPCI and core spray injection valves are only exposed to transients that are also experienced by the RHR shutdown cooling return valves. The environmental conditions are similar.

Valve F005 is a 12-inch carbon steel core spray injection isolation valve. The environmental fatigue results for this valve are considered representative of the F006 core spray isolation valve because they are both 12-inch, carbon steel, 900 lb pressure class gate valves that are affected by similar transients and environmental conditions.

Valve F017 is a 12-inch carbon steel RHR LPCI injection isolation valve. The environmental fatigue results for this valve are considered representative of the F041 RHR LPCI injection isolation valve because they are both 12-inch, carbon steel, 900 lb pressure class valves that are affected by similar transients and environmental conditions.

Valves F001 and F004 are 6-inch stainless steel reactor water cleanup (RWCU) supply isolation valves. The environmental fatigue results for these valves are representative of the

F027, F100 and F105 valves that are also in the RWCU supply piping because they are all 6-inch, stainless steel, 900 lb pressure class valves that are affected by similar transients and environmental conditions.

<u>RAI 4.3-7</u>

<u>Background</u>

LRA Table 4.3.3-2 provided the CUF values for LGS, Units 1 and 2, ASME Code, Section III, Class 1 piping system environment fatigue analysis results. LRA Table 4.3.3-2 states that for the reactor recirculation piping, the CUFs calculated using the fatigue design curve in NUREG/CR-6909 for LGS, Units 1 and 2 are 0.3505 and 0.1056 respectively.

LRA Tables 4.3.3-2 states that for the MSIV drains, the CUFs calculated using the fatigue design curve in ASME Code, Section III for LGS, Units 1 and 2 are 0.0211 and 0.0798 respectively.

<u>Issue</u>

The staff noted that there is approximately a factor of three difference in the CUFs that are reported for these components between LGS, Unit 1 and LGS, Unit 2. The LRA did not include any justification to explain why the CUF values would be different between the units.

The staff noted that for core spray piping, MSIV drains, MSIV drain and test, reactor core isolation cooling (RCIC) steam supply, head vent, and safeguard piping fill systems, the nodes are different between LGS, Unit 1 and LGS, Unit 2. It is not apparent to the staff whether the different nodes between LGS, Unit 1 and LGS, Unit 2 indicated different locations between the units. No explanation for the difference was provided in the LRA.

<u>Request</u>

1) For the reactor recirculation piping and MSIV drains, explain why there is approximately a factor of three differences between the CUFs reported for LGS, Unit 1 and LGS, Unit 2. Also, justify why the differences in CUF values reported between LGS, Unit 1 and LGS, Unit 2 are acceptable.

2) For each component in LRA Table 4.3.3-2 that indicated different nodes between LGS, Unit 1 and LGS, Unit 2, describe the configuration of these locations in the system piping that is being referred to for each unit. Justify why the locations are different between LGS, Unit 1 and LGS, Unit 2.

<u>Exelon Response</u>

1) Differences between the CUFs reported for Units 1 and 2 piping systems are due to differences in piping configuration or other inputs to stress analyses. The main reason for the difference in CUF reported for the reactor recirculation piping is that different S_m (allowable stress) values were used in the original stress reports for the two units. The Unit 1 Recirculation system piping stress report used a S_m value of 13,725 psi at 575ºF for ASME SA-358, TP316L material. This is conservative because it is lower than the S_m value of 17,250 psi at 575ºF that is applicable for the SA-358, TP316 material used for the piping. The original Unit 2 stress report used the S_m value of 17,250 psi applicable for SA-358, TP

316 material at 575ºF. This difference in S_m values leads to a difference in the elastic-plastic correction factor (K_e) that leads to a higher alternating stress and higher CUF value for the Unit 1 analysis. The Unit 1 analysis is conservative, but it meets applicable requirements. The Unit 2 fatigue analysis was subsequently revised to account for increased piping loads associated with a locked snubber condition on the recirculation piping line, which did not apply to Unit 1, contributing to the difference in CUF values between the two units.

The difference in CUF reported for the MSIV drains piping is due to a difference in the piping support configuration downstream of the outboard isolation valves HV-041-1(2)F016. On Unit 1, the limiting nodes 41 and 261 are inside containment on the inboard MSIV drains header upstream of valves 041-1029C and 041-1029B. On Unit 2 the limiting node 145 is outside containment immediately upstream of HV-041-2F016. The piping immediately downstream of HV-041-1F016 on Unit 1 includes a vertical restraint that is not present on Unit 2. This results in a calculated CUF of 0.0798 at node 145 on Unit 2 which is higher than the CUF value of 0.0211 computed for nodes 41 and 261 on Unit 1.

2) The piping configurations for systems which have different Unit 1 and Unit 2 limiting node locations shown in LRA Table 4.3.3-2 are described below:

   a. Core Spray – Unit 1 had a locked snubber condition on one core spray piping line for a period of time before it was discovered and repaired, which required a revision of the fatigue analysis, resulting in increased fatigue usage and a change in the limiting node as compared to Unit 2.

   b. MSIV Drain and Test (MSIV Leakage Control System) – On Unit 2, node 517 is located at the tee where the ¾" DBA-211 piping connects to 2" DBA-211 piping upstream of valve 041-1025A off the 'A' Main Steam Line. It has a current CUF value of .0204. On Unit 1, node 265 was located on 1" DBA-111 piping connected to 'D' Main Steam Line that was removed from the system by a modification. The Class 1 stress reports for this piping on both Units concluded that the since the modification changed the system function and reduced the operating conditions and thermal transients; revised CUF values were not calculated as the prior CUF usage was low and the original CUF values were conservative. The current CUF value of .0478 for node 265 is conservative relative to the current piping configuration. The piping configuration for the MSIV Leakage Control System was substantially different between Units 1 and 2, prior to and after the modifications, resulting in the different node locations and values for limiting CUF.

   c. Reactor Core Isolation Cooling (RCIC) Steam Supply – On Unit 1, node 391 is located at the connection where the 3" RCIC steam supply piping connects to the 26" 'B' Main Steam Line and is the limiting CUF location. On Unit 2, node 148 is located immediately upstream of outboard isolation valve HV-049-2F008 and is the limiting CUF location. The stress analyses performed for power rerate in 1993 for Unit 2 and in 1994 for Unit 1 determined the different bounding locations and CUF values. The most likely cause of the difference in bounding values and locations is that the piping support configuration outside containment is different between the units.

   d. Head Vent - On Unit 1, node 28 is located at the 4"x2" reducer to instrumentation condensing chamber XY-1D002 and is the limiting CUF location. It has a current CUF value of 0.0891. On Unit 2, node 292 is located several feet downstream of the 4"x2" reducer in the common 2" DBA-210 piping to 'C' Main Steam Line and the Clean Radwaste System (CRW) and is the limiting CUF location. It has a current CUF value of 0.5044. The current CUF values are obtained from the ASME Section III Class 1 Stress Analysis Reports. The most likely cause of the difference in bounding values and locations is that the piping support configurations for the attached Class 2 HBB piping downstream of valves HV-041-1(2)F002 are different between the units.

e. Safeguard Piping Fill Systems - On Unit 1, node 276 is located at an elbow in the 1½" DBA-116 piping between valves HV-041-130B and HV-041-133B and is the limiting CUF location. It has a current CUF value of 0.0047. On Unit 2, node 890 is located in the 1½" DBA-216 piping between valves HV-041-230A and HV-041-233B, at HV-041-233B, and is the limiting CUF location. It has a current CUF value of 0.0021. The current CUF values are obtained from the ASME Section III Class 1 Stress Analysis Reports. The cause of the difference in bounding values and locations is that the piping and piping support configuration is substantially different between the units.

RAI 4.3-8

Background

UFSAR Table 3A-27 identifies fatigue usage factors for components in the MSRV discharge lines in the wetwell air space. The flush weld for the pipe anchor has a CUF value of 0.870 and the tapered transition (thin end) has a CUF value of 0.868. UFSAR Table 3.6-12 also identifies cumulative usage factors for different locations of the reactor vessel drain piping. Furthermore, in its submittal, dated March 25, 2010, “Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate” (ADAMS Accession No. ML100850379), the applicant provided non-proprietary report NEDO-33484, Rev. 0 “Safety Analysis Report for Limerick Generating Station Units 1 & 2 thermal Power Optimization” (ADAMS Accession No. ML100850403). Table 3-7 of the non-proprietary report provided 40-year CUF values for the core differential-pressure and liquid control nozzle, closure bolts, and stabilizer bracket.

Issue

LRA Section 4.3.3 states that in order to ensure that any other locations that may not be bounded by the NUREG/CR-6260 locations were evaluated, environmental fatigue calculations were performed for each RPV component location that has a reported CUF value in the stress report and for each ASME Code, Section III, Class 1 RCPB piping system in each unit. The staff noted that LRA Table 4.3.3-1 and 4.3.3-2 did not identify any components in MSRV discharge lines, reactor vessel drain piping, Core Differential-Pressure & Liquid Control nozzle, closure bolts, and stabilizer bracket in the environment fatigue analysis. It is not apparent to the staff whether the effects of reactor coolant environment on component fatigue life have been evaluated for these components consistent with the aforementioned statement in LRA Section 4.3.3.

Request

1) Justify why the effects of reactor coolant environment need not be considered for the components in MSRV discharge lines, reactor vessel drain piping, core differential-pressure and liquid control nozzle, closure bolts, and stabilizer bracket.

2) Identify and provide justifications for other RPV components and ASME Code, Section III, Class 1 RCPB piping system that had reported CUF values but have not been considered for the effects of reactor coolant environment.

Exelon Response

1) The following paragraphs explain why each component was evaluated for environmental fatigue or does not require evaluation:

   a. The MSRV discharge lines do not require evaluation for environmental fatigue because they are ASME Section III, Class 3 components. They are not within the reactor coolant pressure boundary and do not contact reactor coolant.

   b. Reactor vessel drain piping was evaluated for environmental fatigue, but since this piping is included within the Class 1 stress report for the Reactor Water Cleanup System, this piping was included within the environmental fatigue analysis for the Reactor Water Cleanup system.

   c. The core differential pressure and liquid control nozzle was analyzed for environmental fatigue. However, the environmental fatigue analysis determined that the alternating stress value is below the endurance limit for the material, resulting in a CUF value of 0.000. The SLC injection transients assumed in the original nozzle analysis were not included in the environmental fatigue analysis because SLC is not injected through the core differential pressure and liquid control nozzle at LGS. SLC is injected through the Core Spray system. Since the CUF value is 0.000, the CUF_{en} value is also 0.000.

   d. RPV closure bolts do not require evaluation for environmental fatigue because they do not contact reactor coolant since they are located within the containment air space outboard of the RPV main flange seals.

   e. The RPV stabilizer bracket was evaluated for environmental fatigue, as shown on LRA Table 4.3.3-1 for RPV Shell at Stabilizer Bracket. The stabilizer bracket is welded to the outside surface of the RPV shell. The RPV stress report includes a fatigue analysis of the stabilizer bracket along with a segment of the RPV shell where the bracket is attached. The analysis provides a CUF value for a location on the inside surface of the RPV shell. Since this location contacts reactor coolant, it was evaluated for environmental fatigue.

2) Components that are not in contact with reactor coolant are not required to be evaluated for environmental fatigue, as described above. Piping systems with a dry steam internal environment were evaluated and shown to have a F_{en} value of 1.0 just to demonstrate that all Class 1 piping systems were evaluated. RPV components and RCPB piping systems that have reported CUF values and contact reactor coolant have been evaluated for environmental fatigue, as reported in LRA Tables 4.3.3-1 and 4.3.3-2.

RAI 4.3-9

Background

In its submittal, dated March 25, 2010, “Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate” the applicant provided non-proprietary report NEDO-33484, Rev. 0 “Safety Analysis Report for Limerick Generating Station Units 1 & 2 Thermal Power Optimization.” Table 3-7 of the non-proprietary report provided 40-year CUF values for the core spray nozzle (low alloy steel), low pressure core injection (LPCI) nozzle, support skirt, feedwater nozzles.

Issue

LRA Section 4.3.3 states that for carbon and low-alloy steel components, the CUF value from the current ASME Code, Section III, Class 1 fatigue analysis, derived from the ASME Code fatigue curve was initially used in conjunction with a bounding F_{en} multiplier. The staff noted that the ASME Code CUF values in LRA Table 4.3.3-1 for these components are different from those in Table 3-7 of the non-proprietary report. The LRA did not provide any justification why these CUF values are different between the tables.

Request

Reconcile and justify the differences of the CUF values between LRA Table 4.3.3-1 and Table 3-7 of NEDO-33484, Rev. 0 for the core spray nozzle, LPCI nozzle, support skirt, and feedwater nozzles.

Exelon Response

1) Core Spray Nozzle – The environmental fatigue analysis for the core spray nozzle forging was based upon the CUF value provided in the original RPV stress report, but did not address changes to the nozzle forging CUF value provided in subsequent reanalyses. Therefore, the environmental fatigue analysis has been revised to address the changes introduced in the later analyses, including new loads. However, since the previously analyzed location is on the outside surface of the nozzle forging that does not contact reactor coolant, the revised environmental fatigue analysis evaluates the inside surface location at the clad / base metal interface directly below the limiting outside surface location. This location was selected to represent the wetted internal surface of the forging but takes no credit for the presence of the cladding. The alternating stresses at this location are lower than those for the external surface location, resulting in a lower CUF value. Since this location was not originally analyzed for fatigue, no ASME CUF value is reported. The NUREG/CR-6909 CUF value is 0.0022. The F_{en} multiplier is 7.53 and the CUF_{en} value is 0.017. LRA Table 4.3.3-1 is revised as shown in Enclosure B to include the revised analysis results.

2) LPCI Nozzle – The CUF value of 0.79 shown in NEDO-33484, Table 3-7, is for the LPCI nozzle safe end (nickel alloy) that was provided in the MUR uprate evaluation. The CUF value of 0.504 shown in LRA Table 4.3.3-1 is the revised CUF value from the environmental fatigue analysis that resulted from the removal of conservatism in the MUR analysis. The CUF value of 0.420 was determined using the fatigue curve in NUREG/CR-6909. The F_{en} value of 1.94 resulted in a CUF_{en} value of 0.815. No change to LRA Table 4.3.3-1 is required.

3) Support Skirt – The location reported in LRA Table 4.3.3-1 as Support Skirt (RPV ID Location Adjoining Skirt) is for a wetted surface location inside the vessel shell at the location where the support skirt attaches to the vessel. NEDO-33484 reports the highest fatigue location on the Support Skirt, which is at the bottom of the skirt where it rests on the concrete. Therefore, the values are not comparable, and no LRA change to LRA Table 4.3.3-1 is required.

4) Feedwater Nozzle – The environmental fatigue analysis for the feedwater nozzle includes a new ASME Section III, NB-3200 fatigue analysis that includes a finite element model and an updated loads analysis that reduces the number of feedwater injections assumed during shutdown events from five cycles to one cycle. This more closely correlates with the LGS

feedwater system design because LGS has a low-flow control valve installed in parallel with the full-flow control valve that permits continuous low-flow makeup rather than cycling feedwater on and off multiple times at a high flow rate during operations. The environmental fatigue analysis still conservatively assumes one full flow injection per shutdown cycle. Therefore, the CUF value determined in the environmental fatigue analysis of the feedwater nozzle is lower than the CUF value determined in the original design report. No change to LRA Table 4.3.3-1 is required.

RAI 4.3-10

Background

In the applicant's submittal, dated March 25, 2010, "Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate," the applicant provided non-proprietary report NEDO-33484, Rev. 0 "Safety Analysis Report for Limerick Generating Station Units 1 & 2 Thermal Power Optimization." Section 3.3.2 of the non-proprietary report stated that the loads considered in the evaluation of the RPV internals include safety relief valve (SRV) transients. LRA Sections 4.3.4 and A.4.3.4 state that the fatigue analyses performed for the reactor vessel internals (RVI) components are based upon the same set of design transients as those used in the fatigue analyses for the reactor pressure vessel.

Issue

The staff noted that transients related to SRV are not included in LRA Tables 4.3.1-1 and 4.3.1-2. It is not apparent to the staff whether transients related to the SRVs were used in the fatigue analyses for reactor vessel internal components or not. Furthermore, the staff noted that LRA Section A.4.3.4, but not LRA Section 4.3.4, explicitly identifies several reactor vessel internal components that have been analyzed for fatigue including the top guides, core support plate, and core shroud. The staff also noted that the LRA does not provide any CUF values for the reactor vessel internals components. Without these values, the staff cannot ascertain whether the CUF for any location exceeded the allowable limit or evaluate the dispositions of these TLAAs in accordance with 10 CFR 54.21(c).

Request

1) Clarify whether transients related to the SRVs were used in the fatigue analyses for reactor vessel internal components.

2) Identify the RVI components for which fatigue analyses were performed and provide the 40-year CUF values for the RVI components.

Exelon Response

1) SRV transients were considered in the fatigue evaluation of the RPV internals, as stated in NEDO-33484, Rev. 0. UFSAR Section 3.9.1.1.9, Main Steam Relief Valves (MSRV) Transients, indicates that for the RPV and RPV internals, at least 7,700 SRV cycles are considered to account for the pool dynamic loads. It further explains that these 7,700 cycles are based on 1,100 actuations of all MSRVs times seven stress cycles per actuation. Refer to the Exelon response to RAI 4.6.8-1 for details regarding 60-year projections of the MSRV cycles, the addition of MSRV actuation cycles to the Fatigue Monitoring program, and related changes to LRA Tables 4.3.1-1 and 4.3.1-2.

2) The following reactor vessel internals components have been analyzed for fatigue. The CUF value is listed alongside each component.

   Shroud Support - 0.22
   Shroud – 0.71
   Core Support Plate - 0.257
   Top Guide – 0.901
   Control Rod Drive Housing – 0.75
   Control Rod Drive – 0.30
   Control Rod Drive Penetration – 0.15
   Control Rod Guide Tube – exempt
   Orificed Fuel Support – 0.13
   Feedwater Sparger – Bounded by FW Nozzles
   Jet Pump (Riser Brace) - 0.839
   Core Spray Line (in-vessel) – 0.57
   Core Spray Sparger – 0.20
   Shroud Head and Steam Separator Assembly – Not analyzed for fatigue
   Incore Housing and Guide Tube –Bounded by vessel penetration analyses
   Core Differential Pressure and Liquid Control Line – less than 1.0
   LPCI Coupling – 0.50
   Steam Dryer – exempt
   Steam Dryer Support Brackets – exempt

   In order to ensure that the transient cycle projections used in evaluating the RPV internals fatigue TLAAs remain bounding through the period of extended operation, the disposition for these TLAAs is revised from 10 CFR 54.21(c)(1)(i) to 10 CFR 54.21(c)(1)(iii), crediting the Fatigue Monitoring Program with managing fatigue through the period of extended operation. LRA Table 4.1-2 is revised to show the change in TLAA disposition as shown in Enclosure B.

   LRA Section 4.3.4, Reactor Vessel Internals Fatigue Analyses, UFSAR Supplement Section A.4.3.4, Reactor Vessel Internals Fatigue and LRA Section 4.3.1, ASME Section III, Class 1 Fatigue Analyses are revised as shown in Enclosure B to clarify that transients associated with MSRVs and LOCA-induced loads were used in the fatigue analyses.

   LRA Table 3.1.2-3, Reactor Vessel Internals Summary of Aging Management Evaluation is revised as shown in Enclosure B to add aging management review line items for managing fatigue of reactor vessel internal components listed above that did not previously have such line items.

   LRA Appendix C is revised to identify that fatigue usage has been identified as a TLAA issue for the LPCI couplings and core spray piping and spargers as shown in Enclosure B. Note that LRA Appendix C was provided as a new LRA Appendix within Enclosure B of the Exelon response to RAI BWRVIP-1 dated February 15, 2012.

<u>RAI 4.3-11</u>

<u>Background</u>

UFSAR Table 3.6-8 indicates that the feedwater piping of LGS, Unit 1 has the highest CUF value of 0.6192 at node 75, which is a tapered transition joint. Node 100 of the feedwater piping of LGS, Unit 1 has a CUF value of 0.3651. UFSAR Table 3.6-8 also indicates that the feedwater piping of LGS, Unit 2 has the highest CUF value of 0.8011 at node 100, which is a butt-welding tee.

<u>Issue</u>

LRA Table 4.3.3-2 indicates that the 40-year CUF values for feedwater piping of LGS, Units 1 and 2 are 0.8011 at node 100. It is not apparent to the staff whether the node 100 for LGS, Unit 1 in LRA Table 4.3.3-2 refers to the same node as in UFSAR Table 3.6-8 with a CUF value of 0.3651 or the LRA is intended to show that node 100 of LGS, Unit 2 bounded that of LGS, Unit 1.

<u>Request</u>

1) Clarify and justify the entry of the CUF value of the feedwater piping for LGS, Unit 1 in LRA Table 4.3.3-2. If applicable, revise LRA Table 4.3.3-2 indicating that the LGS, Unit 2 CUF value bounded that of LGS, Unit 1.

2) Identify and justify for any other components or locations in LRA Tables 4.3.3-1 and 4.3.3-2 for which a CUF value of one unit was used to bound the same component/location in another unit.

<u>Exelon Response</u>

1) The locations and CUF values shown in LRA Table 4.3.2-2 for Unit 1 and 2 feedwater piping are applicable to both Units. Stress analysis documentation shows that the locations and CUF values for the feedwater piping system are the same between the Units because the piping configurations are essentially the same. Since a bounding value was not used, LRA Table 4.3.3-2 does not require revision.

2) The following Class 1 piping systems in LRA Table 4.3.3-2 used the same CUF value for the same component/location on both Units:

   a. RHR Supply and Return
   b. Recirculation Drain
   c. Main Steam (Line C)
   d. Instrumentation
   e. HPCI Steam Supply

   The same locations and CUF values are used because the values from the current stress analyses are the same for both Units for the piping systems listed above. The locations and CUF values are the same because the piping configurations are essentially the same. Therefore, bounding values were not used.

   In LRA Table 4.3.3-1 the only cases where a bounding value for CUF is used is for the Core Spray Nozzle (Safe End) and Core Spray Nozzle (Forging) where the bounding location and

CUF values for Unit 1 are used for Unit 2. Unit 1 bounds Unit 2 because of a historical locked snubber condition on the Unit 1 core spray piping which resulted in increased applied loads from the piping system to the safe end and nozzle. For all other reactor pressure vessel component/locations shown on LRA Table 4.3.3-1, the same CUF values are used because the current values from the stress analyses are the same for both Units. Therefore, bounding values were not used. LRA Table 4.3.3-1 is revised as shown in Enclosure B to include a note to indicate that the bounding Unit 1 ASME CUF values for the Core Spray Nozzle (Safe End) and Core Spray Nozzle (Forging) are used for Unit 2.

RAI 4.3-12

Background

LRA Table 4.3.3-2 provides the environmental fatigue analysis results for the LGS, Units 1 and 2, ASME Code, Section III, Class 1, piping system. For the RCIC steam supply system, head vent system and high pressure core injection (HPCI) steam supply system the F_{en} factor is 1.0 for both units. The LRA included footnote 8 which states that the F_{en} multiplier of 1.00 is used because the internal environment is dry steam.

Issue

It is not clear to the staff why the LRA considered locations that are exposed to dry steam when addressing the effects of reactor water environment on component fatigue life and whether this was appropriate.

Request

1) Clarify why these piping systems that are exposed to dry steam were selected for addressing the effects of reactor water environment.

2) Clarify whether there are other locations, which are not exposed to dry steam, within these systems that would be more appropriate when addressing the effects of reactor water environment.

Exelon Response

1) Each LGS plant system with ASME Code, Section III, Class 1 piping was evaluated for reactor water environmental effects to ensure the limiting locations were evaluated. Since the RCIC Steam Supply system, Head Vent system and HPCI Steam Supply system only include a dry steam environment, the F_{en} value of 1.0 was applied, indicating there is no increase in fatigue usage due to environmental effects.

2) There are no other locations within the ASME Code, Section III, Class 1 boundary for these systems that are not exposed to dry steam that would be more appropriate to evaluate for environmental fatigue effects.

RAI 4.6.8-1

Background

LRA Section 4.6.8 states that TLAAs for downcomers and MSRV discharge piping are dispositioned in accordance with 10 CFR 54.21(c)(1)(i), that the fatigue analyses remain valid for the period of extended operation. LRA Section 4.6.8 also states that a minimum of 7,700 MSRV cycles were considered to account for the pool dynamic loads. In addition, for the most frequently actuated MSRVs, the analysis was based on 4,700 actuations times three stress cycles per actuation (14,100 total cycles). The LRA states the MSRV lift cycle has been projected and it will not exceed the number analyzed for 40 years.

Issue

LRA Section 4.6.8 did not provide the current accumulated number of occurrences of the MSRV cycles and the 60-year projected number of occurrences of the MSRV cycles; therefore, the staff cannot verify the adequacy of the TLAA disposition in accordance with 10 CFR 54.21(c)(1)(i). The staff reviewed LRA Tables 4.3.1-1 and 4.3.1-2 and could not determine the transient that is associated with the MSRV lift cycle that is being projected to 60-years. Furthermore, UFSAR Figure 3A-394 indicates that the cycles associated with chugging, operational basis earthquakes, and safe-shutdown earthquakes are included in the fatigue analysis for the downcomers.

Request

1) Identify the transients that were used in the fatigue analysis for the downcomers.

2) Identify the accumulated number of occurrences for each transient used in the fatigue analysis up to January 2011. Confirm that these transients were monitored since initial plant startup for each unit. If not, justify how the accumulated cycles to date were reconciled.

3) Provide the 60-year projected number of occurrences for each transient identified above and justify that these projections are conservative.

4) Revise LRA Sections 4.6.8 and A.4.6.8 consistent with the changes discussed in the response.

Exelon Response

1) A fatigue analysis of the downcomers was performed in accordance with ASME Section III, Class 1 requirements. Only that portion of the downcomer in the air space of the suppression chamber was evaluated for fatigue. The downcomers are subject to numerous dynamic and hydrodynamic loads from LOCA-related plant operating conditions. For purposes of fatigue evaluation, the following loads were considered: 1) cyclic loads due to hydrodynamic effects, including MSRV actuations and LOCA-related condensation oscillation and chugging cycles, and 2) seismic effects.

   The downcomers and bracing were analyzed for a minimum of 7,700 MSRV stress cycles that were considered to account for the pool hydrodynamic loads, based on 1,100 actuations of all MSRVs times seven stress cycles per actuation. For the most frequently actuated MSRVs, the analysis was based on 4,700 actuations times 3 stress cycles per actuation (14,100 total cycles). (Note: One MSRV actuation is one opening and closing of one MSRV, leading to one steam discharge and up to seven stress cycles). Reactor operating transients that can cause MSRV actuations are listed in UFSAR Section 3A.3.1.1.

The downcomers and bracing were also evaluated for 3,000 chugging cycles that can occur following a LOCA event, 50 OBE cycles (five events with ten stress cycles per event), and 10 SSE cycles (one event with ten cycles).

2) MSRV actuation cycles were not monitored as a separate event within the Fatigue Monitoring program since original plant startup. However, during development of the LRA, an operational review was performed to determine when each MSRV was actuated during preoperational startup testing and during all past plant operations for each unit. This was accomplished by reviewing surveillance tests that actuated MSRVs, monthly operating reports that reported operational MSRV actuations, and fatigue monitoring data for other events that resulted in MSRV actuations. These reviews determined the specific MSRVs that were actuated on specific dates, with the exception of a Unit 1 scram and MSIV actuation event in December 1985 where the data was not available. For that event, it was assumed that each MSRV actuated three times, for a total of 42 actuations. A similar event on Unit 2 in 1989 resulted in 12 MSRV actuations; therefore this assumption is reasonable.

   Based upon this operational review, the cumulative number of MSRV actuations for Unit 1 through January 2011 was determined to be 90. This includes 56 actuations during startup testing prior to commercial operation and 30 actuations due to surveillance testing performed following the first six refueling outages. The surveillance test required each of the five ADS valves to be opened once during each startup after a refueling outage. This surveillance test requirement was eliminated in 1997. The total also includes three momentary MSRV actuations that occurred in 1991. In 1995, MSRV "M" stuck open, and this event was also counted as an Emergency Scram – Single Relief Valve Blowdown (event 14d). No MSRV actuations have occurred on Unit 1 since 1995. However, since each MSRV actuation can result in up to seven stress cycles, the cumulative number of stress cycles through January 2011 resulting from the 90 actuations is assumed to be 630. This is appropriate since MSRV stress cycles were used in the fatigue analyses, not MSRV actuations. 60-year projections for MSRV actuations are described in response 3 below.

   The cumulative number of MSRV actuations for Unit 2 through January 2011 is 49. This includes 32 actuations during startup testing prior to commercial operation and 15 due to surveillance testing following the first three refueling outages. In 1996, MSRV "E" opened due to a Scram/ Generator Lockout event. In 2001, MSRV "N" stuck open, and this event was counted as an Emergency Scram – Single Relief Valve Blowdown (event 14d). No MSRV actuations have occurred on Unit 2 since 2001. The 49 actuations are assumed to have resulted in 343 stress cycles through January 2011.

   The LOCA-related loads include condensation oscillation loads and 3,000 chugging cycles. Condensation oscillations result from mixed flow (air-steam) passing through the downcomers into the wetwell, followed by pure steam flow passing through the downcomers into the wetwell. Once the pure steam flow is established, chugging occurs, which is the pulsating condensation of the steam flow exiting the downcomers. The significance of these loading cycles with respect to fatigue monitoring is that the Fatigue Monitoring program monitors LOCA events, not the various loading cycles that result from them, as described in the Exelon response to RAI 4.6.8-2. The Fatigue Monitoring program includes event 18, Faulted Condition – Pipe Rupture and Blowdown, that corresponds with a large-break LOCA, which shows no event has occurred in either unit. No small-break LOCA events have occurred on either unit. Therefore, no chugging events have occurred to-date in either unit.

LRA Tables 4.3.1-1and 4.3.1-2 show that no OBE event has occurred at LGS. Fatigue Monitoring program records also show that no OBE or SSE event has occurred at LGS to-date.

3) The 60-year projection for Unit 1 MSRV actuations is 211, based upon a linear extrapolation of the MSRV actuation data using the average rate of occurrence through January 2011. This results in a 60-year projection of 1477 stress cycles, since each actuation is assumed to cause up to seven stress cycles. The 60-year projection for Unit 2 MSRV actuations is 138, which results in 966 stress cycles. These values are conservative because they are based upon the average rates of MSRV actuation during the baseline period that includes startup tests and surveillance tests that are no longer performed.

   LRA Tables 4.3.1-1 and 4.3.1-2 show that the design limit of one OBE event (with up to 10 cycles per event) is not projected to be exceeded in 60 years for Unit 1 or Unit 2. Since the SSE event is more severe than an OBE event, the SSE event design limit of one cycle is not projected to be exceeded in 60 years for Unit 1 or Unit 2.

4) LRA Tables 4.3.1-1and 4.3.1-2 are revised as shown in Enclosure B to include Event 20, MSRV Actuations, based upon the operational review described above and the corresponding 60-year projections. Prior to the period of extended operation, the Fatigue Monitoring program procedures will also be revised to add Event 20, MSRV Actuations, as part of the program enhancement that was previously described in LRA Section B.3.1.1. The limit to be imposed for MSRV actuations is 1,100, which corresponds to 7700 stress cycles used in the analysis of the downcomers and MSRV piping, which is more limiting than the minimum of 7,000 MSRV actuations used for MSRV discharge piping quenchers.

   LRA Tables 4.3.1-1 and 4.3.1-2 are also revised as shown in Enclosure B to include the monitoring results for the faulted condition events that are already monitored by the Fatigue Monitoring program. This includes Event 18, Pipe Rupture and Blowdown, and Event 19, Safe Shutdown Earthquake (SSE) at Rated Operating Conditions. Since these faulted events were already included within the Fatigue Monitoring program, this is not part of the program enhancement, but is a clarification in the LRA.

   Since the Fatigue Monitoring program monitors each of the transients analyzed for the downcomers, or in the case of chugging, monitors the LOCA events that cause these cycles, the disposition of the downcomer fatigue TLAA is revised from 10 CFR 54.21(c)(1)(i) to 10 CFR 54.21(c)(1)(iii), crediting the Fatigue Monitoring program for managing fatigue of the downcomers. LRA Table 4.1-2 is revised to show the change in TLAA disposition as shown in Enclosure B.

   LRA Section 4.6.8 and UFSAR Supplement Section A.4.6.8 are revised as shown in Enclosure B to incorporate the clarifications described in this RAI.

RAI 4.6.8-2

Background

LRA Section 4.6.8 states that TLAAs for downcomers and MSRV discharge piping are dispositioned in accordance with 10 CFR 54.21(c)(1)(i), that the fatigue analyses remain valid for the period of extended operation. LRA Section 4.6.8 also states the quenchers were analyzed for 7,000 SRV opening and closing cycles and 1,000,000 irregular condensation load cycles.

Issue

LRA Section 4.6.8 did not provide current accumulated number of occurrences of the SRV opening and closing cycles as well as the irregular condensation load cycles. In addition, the 60-year projected number of occurrences of these cycles (SRV opening/closing cycles and irregular condensation load cycles); therefore, the staff cannot verify the adequacy of the applicant's TLAA disposition in accordance with 10 CFR 54.21(c)(1)(i).

The staff reviewed LRA Tables 4.3.1-1 and 4.3.1-2 and could not determine the transient that is associated with the SRV opening/closing cycles or the irregular condensation load cycles that is being projected to 60-years. Furthermore, UFSAR Section 3A.7.1.5.1.1 indicates that the following loads are included for the purpose of the fatigue evaluation: (1) significant thermal and pressure transients, (2) cyclic loads due to hydrodynamic effects including MSRV actuations, condensation oscillation and chugging, and (3) seismic effects.

Request

1) Identify the transients that were used in the fatigue analysis for the MSRV discharge piping and quenchers.

2) Identify the accumulated number of occurrences for each transient used in the fatigue analysis up to January 2011. Confirm that these transients were monitored since initial plant startup for each unit. If not, justify how the accumulated cycles to date were reconciled.

3) Provide the 60-year projected number of occurrence for each transient identified above and justify that these projections are conservative.

4) Revise LRA Sections 4.6.8 and A.4.6.8 to be consistent with the changes discussed in the response

Exelon Response

1) Fatigue evaluation of the MSRV discharge piping was performed in accordance with ASME Section III, Class 1 fatigue rules. The MSRV discharge lines are subject to numerous dynamic and hydrodynamic loads from normal, upset, and LOCA-related plant operating conditions. For purposes of fatigue evaluation, the following loads are included: 1) significant thermal and pressure transients, and 2) cyclic loads due to hydrodynamic effects, including MSRV actuations, LOCA-related condensation oscillation and chugging, and seismic effects.

   The thermal transients used in the fatigue analysis include those specified in UFSAR Section 3.9.1.1.9, including 150 preoperational and inservice testing cycles, 120 Startups, 120 Shutdowns, 180 Scrams, 10 system temperature changes greater than 30ºF, 8 cycles where system temperature changes from 546ºF to 375ºF in ten minutes (Emergency/Faulted event), and three other Emergency/Faulted events with a limit of 1 cycle each.

The MSRV discharge loads include a minimum of 7,700 MSRV stress cycles that account for the pool dynamic loads, based on 1,100 actuations of all MSRVs times seven stress cycles per actuation. For the most frequently actuated MSRVs, the analysis was based on 4,700 actuations times 3 stress cycles per actuation (14,100 total cycles).

The LOCA-related loads include condensation oscillation loads and 3,000 chugging cycles. Condensation oscillations result from mixed flow (air-steam) passing through the downcomers into the wetwell, followed by pure steam flow passing through the downcomers into the wetwell. Once the pure steam flow is established, chugging occurs, which is the pulsating condensation of the steam flow exiting the downcomers. The significance of these loading cycles with respect to fatigue monitoring is that the Fatigue Monitoring program monitors LOCA events, not the various loading cycles that result from them, as described in the response to question 2 below.

The seismic loads include 50 OBE cycles (5 events with ten stress cycles per event), and 10 SSE cycles (one event with 10 cycles).

The quenchers at the ends of the MSRV discharge lines were analyzed for 7000 MSRV actuations and for one million irregular condensation cycles. During a steam discharge from an MSRV, steam condensation is not uniform at the quencher throughout the entire range of steam mass flow rates and pool temperatures, which results in irregular condensation cycles that create these hydrodynamic and thermal fatigue loads.

2) Thermal transients other than MSRV actuations were monitored since original plant startup as shown in LRA Tables 4.3.1-1 and 4.3.1-2. MSRV actuation cycles were not monitored since initial plant startup, but an operational review was performed to determine MSRV cycles that occurred during past plant operations. See the Exelon response to RAI 4.6.8-1, response 2, for further details. LRA Tables 4.3.1-1 and 4.3.1-2 are revised as shown in Enclosure B to include Event 20, MSRV Actuations, as discussed in the Exelon response to RAI 4.6.8-1, response 4.

   The Fatigue Monitoring program does not directly monitor chugging cycles, but monitors Event 18, Faulted Condition – Pipe Rupture and Blowdown, that corresponds with a LOCA. Monitoring data shows that no LOCA event has occurred in either unit. Therefore, no chugging events have occurred to-date in either unit.

   LRA Tables 4.3.1-1 and 4.3.1-2 show that no OBE event has occurred at LGS. Fatigue Monitoring program records currently show that no SSE events have occurred at LGS to-date. LRA Tables 4.3.1-1 and 4.3.1-2 are revised as shown in Enclosure B to include Event 19, Safe Shutdown Earthquake (SSE) at Rated Operating Conditions, as discussed in the Exelon response to RAI 4.6.8-1, response 4.

3) LRA Tables 4.3.1-1 and 4.3.1-2 show the 60-year projections for the thermal transients used in the MSRV discharge piping analyses for Unit 1 and 2. The 60-year projected numbers of thermal transients are less than the 40-year design limits and are therefore conservative. LRA Tables 4.3.1-1 and 4.3.1-2 show that the design limit of one OBE event (with up to 10 cycles per event) is not projected to be exceeded in 60 years for Unit 1 or Unit 2. LRA Table 4.3.1-1 and 4.3.1-2 are revised as shown in Enclosure B to also include SSE monitoring

data (Event 19) and chugging data (Event 18) that indicates the design value of one SSE event (with up to 10 cycles per event) and one chugging event will not be exceeded in 60 years for Unit 1 or Unit 2.

The 60-year projection for MSRV actuations is 211 for Unit 1 and 138 for Unit 2, as described in the Exelon response to RAI 4.6.8-1, response 2. The conservatism of these projections is justified in the Exelon response to RAI 4.6.8-1, response 3.

4) LRA Tables 4.3.1-1 and 4.3.1-2 are revised as shown in Enclosure B to include the monitoring results for the faulted condition events that are monitored by the Fatigue Monitoring program. This includes Event 18, Pipe Rupture and Blowdown, and Event 19, Safe Shutdown Earthquake (SSE) at Rated Operating Conditions. Since these faulted events were already included within the Fatigue Monitoring program, this is not a program enhancement, but is a clarification in the LRA. LRA Tables 4.3.1-1 and 4.3.1-2 are also being revised to include Event 20, MSRV Actuations, described above and within RAI 4.6.8-1. LRA Section 4.6.8 and UFSAR Supplement Section A.4.6.8 are revised as shown in Enclosure B to provide separate discussions of the TLAA evaluation and disposition for MSRV discharge piping and for the downcomers, and to include clarifications provided above and in the Exelon response to RAI 4.8.6-1. The TLAA disposition is revised from 10 CFR 54.21 (c)(1)(i) to 10 CFR 54.21 (c)(1)(iii) to credit the Fatigue Monitoring program for managing fatigue of the MSRV discharge piping. LRA Table 4.1-2 is revised to show the change in TLAA disposition as shown in Enclosure B.

Enclosure B
LGS License Renewal Application Updates

Notes:

- Updated LRA Sections and Tables are provided in the same order as the RAI responses contained in Enclosure A.
- To facilitate understanding, portions of the original LRA have been repeated in this Enclosure, with revisions indicated.
- Existing LRA text, and updates provided in previous RAI responses, are shown in normal font. Changes are highlighted with ***bold italics*** for inserted text and strikethroughs for deleted text.

As a result of the responses to RAIs 4.3-3, 4.6.8-1 and 4.6.8-2 provided in Enclosure A of this letter, Table 4.3.1-1 on page 4-57 of the LRA is revised as shown below:

Table 4.3.1-1 – 60-Year Transient Cycle Projections for Unit 1						
Transient Number	**Transient Description**	**Cumulative Cycles to-date (Jan. 2011)**	**60-Year Projected Cycles**	**Roundup or Added Margin**	**Adjusted 60-Year Projected Cycles**	**Design Cycle Limits**
18.	***Faulted Condition – Pipe Rupture and Blowdown (1000 psig to 35 psig in 15 seconds)***	***0***	***0***	***10***	***10***	***10 [5]***
19.	***Faulted Condition – Safe Shutdown Earthquake (SSE) at Rated Operating Conditions***	***0***	***0***	***10***	***10***	***10 [6]***
20.	***Main Steam Relief Valve (MSRV) Actuations***	***90***	***211***	***0***	***211***	***1100 [7]***
	~~ECCS / RCIC and~~ SLC Injections					
~~N/A~~	~~Reactor Core Isolation Cooling (RCIC)~~	~~6~~	~~14~~	~~0~~	~~14~~	~~15~~
~~N/A~~	~~High Pressure Coolant Injection (HPCI)~~	~~6~~	~~9~~	~~0~~	~~9~~	~~15~~
~~N/A~~	~~Core Spray (CS)~~	~~0~~	~~0~~	~~2~~	~~2~~	~~N/A~~
~~N/A~~	~~Low Pressure Coolant Injection (LPCI) I~~	~~0~~	~~0~~	~~2~~	~~2~~	~~N/A~~
N/A	Standby Liquid Control (SLC)	0	0	2	2	10

[1] The design cycle limit for transient 8a, OBE for NSSS piping components, is 5 events with 10 cycles per event, for a total of 50 cycles.
[2] The design cycle limit for transient 8b, OBE for Other NSSS components, is 1 event with 10 cycles per event.
[3] For the RPV, 5 Loss of A/ C Power transients (no. 13) and 5 Loss of Feedpump transients (no. 14b) were analyzed.
[4] For RCPB piping, 10 cycles of the Loss of Feedpump transient (no. 14b) were analyzed and 0 Loss of A/C Power transients (no. 13) were analyzed.
[5] The design cycle limit for transient 18, Faulted Condition – Pipe Rupture and Blowdown, is 1 event with 10 cycles per event.
[6] The design cycle limit for transient 19, Faulted Condition – Safe Shutdown Earthquake (SSE) at Rated Conditions, is 1 event with 10 cycles per event.
[7] The design cycle limit for transient 20, MSRV Actuations, is 1100 actuation events with 7 stress cycles assumed per actuation, for a total of 7700 cycles.

As a result of the response to RAIs 4.3-3, 4.6.8-1 and 4.6.8-2 provided in Enclosure A of this letter, Table 4.3.1-2 on page 4-60 of the LRA is revised as shown below:

Table 4.3.1-2 – 60-Year Transient Cycle Projections for LGS Unit 2						
Transient Number	**Transient Description**	**Cumulative Cycles to-date (Jan. 2011)**	**60-Year Projected Cycles**	**Roundup or Added Margin**	**Adjusted 60-Year Projected Cycles**	**Design Cycle Limits**
18.	***Faulted Condition – Pipe Rupture and Blowdown (1000 psig to 35 psig in 15 seconds)***	***0***	***0***	***10***	***10***	***10 [5]***
19.	***Faulted Condition – Safe Shutdown Earthquake (SSE) at Rated Operating Conditions***	***0***	***0***	***10***	***10***	***10 [6]***
20.	***Main Steam Relief Valve (MSRV) Actuations***	***49***	***138***	***0***	***138***	***1100[7]***
	~~ECCS / RCIC and~~ SLC Injections					
~~N/A~~	~~Reactor Core Isolation Cooling (RCIC)~~	~~4~~	~~6~~	~~0~~	~~6~~	~~15~~
~~N/A~~	~~High Pressure Coolant Injection (HPCI)~~	~~3~~	~~5~~	~~0~~	~~5~~	~~15~~
~~N/A~~	~~Core Spray (CS)~~	~~0~~	~~0~~	~~2~~	~~2~~	~~N/A~~
~~N/A~~	~~Low Pressure Coolant Injection (LPCI) I~~	~~0~~	~~0~~	~~2~~	~~2~~	~~N/A~~
N/A	Standby Liquid Control (SLC)	0	0	2	2	10

[1] The design cycle limit for transient 8a, OBE for NSSS piping components, is 5 events with 10 cycles per event, for a total of 50 cycles.
[2] The design cycle limit for transient 8b, OBE for Other NSSS components, is 1 event with 10 cycles per event.
[3] For the RPV, 5 Loss of A/ C Power transients (no. 13) and 5 Loss of Feedpump transients (no. 14b) were analyzed.
[4] For RCPB piping, 10 cycles of the Loss of Feedpump transient (no. 14b) were analyzed and 0 Loss of A/C Power transients (no. 13) were analyzed.
[5] The design cycle limit for transient 18, Faulted Condition – Pipe Rupture and Blowdown, is 1 event with 10 cycles per event.
[6] The design cycle limit for transient 19, Faulted Condition – Safe Shutdown Earthquake (SSE) at Rated Conditions, is 1 event with 10 cycles per event.
[7] The design cycle limit for transient 20, MSRV Actuations, is 1100 actuation events with 7 stress cycles assumed per actuation, for a total of 7700 cycles.

As a result of the responses to RAIs 4.3-4, 4.3-10, 4.6.8-1 and 4.6.8-2, provided in Enclosure A of this letter, Table 4.1-2, page 4-5 of the LRA (previously revised by Exelon responses to RAIs 4.6.6-1 and 4.6.9-1 dated 1/24/2012, and RAI B.2.1.35-1 dated 2/28/2012), 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)(iii) and*** §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)~~***(iii)***	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)(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)~~***(iii)***	4.6.8
Jet Pump Slip Joint Repair Clamps	§54.21(c)(1)(i)	4.6.9
Fuel Pool Girder Loss of Prestress	§54.21(c)(1)(i)	4.6.10

As a result of the response to RAI 4.3-4 provided in Enclosure A of this letter, sections 4.3.2 and A.4.3.2 of the LRA, on pages 4-61, 4-62 and A-40, are revised as shown below to include two TLAA dispositions; one for the ASME Code, Section III, Class 2 and 3 and B31.1 piping systems that are connected to ASME Code, Section III, Class 1 piping; and one for ASME Code, Section III, Class 2 and 3 and B31.1 piping systems with that are not connected to Class 1 piping.

TLAA Evaluation:

For the ASME Section III, Class 2 and 3 and ANSI B31.1 systems that are connected to ASME Section III, Class 1 piping, and are affected by the same operational transients, the 60-year cycle projections demonstrate that the total number of thermal and pressure cycles of all of the transient types added together will not exceed 7,000 cycles during the period of extended operation. Therefore, the stress range reduction factor will not change and the TLAAs will remain valid for the period of extended operation. This includes the applicable portions of the following systems: Residual Heat Removal, Core Spray, Reactor Core Isolation Cooling, High Pressure Coolant Injection, Reactor Water Cleanup, Control Rod Drive, Main Steam, Main Turbine, Extraction Steam, Feedwater, Condenser and Air Removal, and Radwaste.

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 by the Fatigue Monitoring program in accordance with 10 CFR 54.21(c)(1)(iii).

TLAA Evaluation:

For the remaining systems that are affected by different thermal and pressure cycles, an operational review was performed that ~~also~~ concluded that the total number of cycles, projected for ***the period of extended operation***~~60 years~~, will not exceed 7,000 cycles for these systems. This includes ***portions of*** the Fire Protection, Emergency Diesel Generator, and Auxiliary Steam systems. Systems with operating temperatures below specified thresholds were determined to have low numbers of equivalent full temperature cycles since the fluid temperature changes are small. ~~Therefore, since the stress range reduction factors originally selected for the components in all of these systems remain applicable, the TLAAs remain valid for the period of extended operation.~~

TLAA Disposition: 10 CFR 54.21(c)(1)(i) – ~~The ASME Section III, Class 2 and 3 and ANSI B31.1 allowable stress calculations remain valid for the period of extended operation.~~ ***The maximum allowable stress range values for the existing fatigue analysis remain valid because the allowable limit for the number of full thermal range transient cycles will not be exceeded during the period of extended operation.***

A.4.3.2 ASME Section III, Class 2 and 3 and ANSI B31.1 Allowable Stress Calculations

Piping designed in accordance with ASME Section III, Class 2 or 3 design rules or ANSI B31.1 Piping Code design rules is not required to have an explicit analysis of cumulative fatigue usage, but cyclic loading is considered in the design process. If the numbers of anticipated thermal cycles exceed specified limits, these codes require the application of a stress range reduction factor to the allowable stress to prevent damage from cyclic loading. This is considered to be an implicit fatigue analysis since it is based upon cycles anticipated for the life of the component.

These codes first require the overall number of thermal and pressure cycles expected during the 40-year lifetime of these components to be determined. A stress range reduction factor is then determined for that number of cycles using the applicable design code. If the total number of cycles is 7,000 or less, the stress range reduction factor of 1.0 is applied which would not reduce the allowable stress values. For higher numbers of cycles, the stress range reduction factor limits the allowable stresses that can be applied to the piping.

For the piping and components that are affected by the reactor vessel operational transients, including the Class 2 and 3 piping extending from Class 1 systems, the 60-year cycle projections demonstrate that the total number of thermal and pressure cycles will not exceed 7,000 cycles during the period of extended operation. ***The effects of aging on the intended function(s) of components will be adequately managed for the period of extended operation by the Fatigue Monitoring program in accordance with 10 CFR 54.21(c)(1)(iii).***

For the remaining systems that are affected by different thermal and pressure cycles, an operational review was performed that ~~also~~ concluded that the total number of cycles, projected for ***the period of extended operation***~~60 years~~, will not exceed 7,000 cycles for these systems. This includes ***portions of*** the Fire Protection, Emergency Diesel Generator, and Auxiliary Steam systems. Systems with operating temperatures below specified thresholds were determined to have low numbers of equivalent full temperature cycles since the fluid temperature changes are small. ***For these systems, the maximum allowable stress range values for the existing fatigue analysis remain valid in accordance with 10 CFR 54.21(c)(1)(i) because the allowable limit for the number of full thermal range transient cycles will not be exceeded during the period of extended operation.*** ~~Therefore, since the stress range reduction factor originally selected for the components in all of these systems remain applicable, the TLAAs remain valid for the period of extended operation.~~

~~The effects of aging on the intended function(s) of components analyzed in accordance with ASME Section III, Class 1 requirements will be adequately managed for the period of extended operation by the Fatigue Monitoring program in accordance with 10 CFR 54.21(c)(1)(iii).~~

As a result of the response to RAIs 4.3-5, 4.3-9, and 4.3-11 provided in Enclosure A of this letter, Table 4.3.3-1, pages 4-66 and 4-67 of the LRA is revised as shown below:

Table 4.3.3-1
Unit 1 and Unit 2 Reactor Pressure Vessel (RPV) Environmental Fatigue Analysis Results

RPV Component and Data			**Unit 1 CUF_{en} Results**				**Unit 2 CUF_{en} Results**				
RPV Component	**Node**	**Material**	**ASME CUF**	**NUREG 6909 CUF**	**NUREG 6909 F_{en}**	**CUF_{en}**	**Node**	**ASME CUF**	**NUREG 6909 CUF**	**NUREG 6909 F_{en}**	**CUF_{en}**
NUREG/CR-6260 Components Ranked Highest CUF_{en} to Lowest											
LPCI Nozzle (Forging)	19	LAS	0.186	0.186	4.61	0.858	19	0.186	0.186	4.61	0.858
LPCI Nozzle (Safe End / Thermal Sleeve)	4	A600	0.504	0.420	1.94	0.815	4	0.504	0.420	1.94	0.815
Recirculation Outlet Nozzle (Forging)	4	LAS	0.475	0.140	5.31	0.746 [1]	4	0.475	0.140	5.31	0.746 [1]
Recirculation Outlet Nozzle (Safe End)	5	SS	0.012	0.048	6.21	0.297 [1]	5	0.012	0.048	6.21	0.297 [1]
RPV Shell at Stabilizer Bracket	5	LAS	0.228	0.099	6.97	0.687	5	0.228	0.099	6.97	0.687
Feedwater Nozzle (Safe End Inlay)	N/A	SS	0.473	0.276	2.34	0.645	N/A	0.473	0.276	2.34	0.645
Feedwater Nozzle (Forging)	4	LAS	0.82	0.030	2.17	0.064 [2]	4	0.82	0.030	2.17	0.064 [2]
Core Spray Nozzle (Safe End)	3	A600	0.073	0.085	3.59	0.304	3	0.073 ***[4]***	0.085	3.59	0.304 ***[4]***
Core Spray Nozzle (Forging)	~~22~~ ***17***	LAS	~~0.097~~ ***.0016***	~~0.040~~ ***0.0022***	~~6.97~~ ***7.53***	~~0.277~~ ***0.017***	~~22~~ ***17***	~~0.097~~ ***.0016 [4]***	~~0.040~~ ***0.0022***	~~6.97~~ ***7.53***	~~0.277~~ ***0.017 [4]***

Table 4.3.3-1 Unit 1 and Unit 2 Reactor Pressure Vessel (RPV) Environmental Fatigue Analysis Results											
RPV Component and Data			Unit 1 CUF_{en} Results				Unit 2 CUF_{en} Results				
RPV Component	Node	Material	ASME CUF	NUREG 6909 CUF	NUREG 6909 F_{en}	CUF_{en}	Node	ASME CUF	NUREG 6909 CUF	NUREG 6909 F_{en}	CUF_{en}
Other Locations Ranked Highest CUF_{en} to Lowest											
Shroud Support	6	LAS	N/A [3]	0.278	3.42	0.949	6	N/A [3]	0.278	3.22	0.895
Incore ~~Monitor~~ ***Housing*** Penetration	~~163~~ ***152***	SS	~~0.108~~ ***0.181***	~~0.140~~ ***0.240***	~~5.94~~ ***2.42***	~~0.83~~ ***0.581***	~~163~~ ***152***	~~0.108~~ ***0.181***	~~0.140~~ ***0.240***	~~5.94~~ ***2.42***	~~0.83~~ ***0.581***
Main Steam Outlet Nozzle (Forging)	4	LAS	0.85	0.330	2.77	0.914	4	0.85	0.330	2.74	0.903
Main Steam Outlet Nozzle (Safe End)	1	CS	0.085	N/A	2.77	0.235	1	0.085	N/A	2.74	0.233
CRD Housing Penetration (Stub Tube / Shell Weld)	12	A600	0.153	0.2690	3.61	0.970	12	0.153	0.2690	3.61	0.970
Support Skirt (RPV Shell ID Location Adjoining Skirt)	29	LAS	N/A [3]	0.022	5.31	0.116	29	N/A [3]	0.022	4.80	0.105
Main Closure Flange (ID)	7	LAS	N/A [3]	0.0044	7.68	0.034	7	N/A [3]	0.0044	7.47	0.033
CRD Housing Penetration	4	SS	0.0120	0.0417	4.34	0.181	4	0.0120	0.0417	4.34	0.181

[1] These environmentally-adjusted fatigue usage values for the Recirculation Outlet nozzles are bounding for the Recirculation Inlet nozzles.
[2] The fatigue usage value of 0.064 is the 60-year environmentally-adjusted fatigue usage value for the feedwater nozzle forging blend radius resulting from the thermal and pressure design transients for the feedwater nozzles. There is an additional contribution to fatigue in the analysis that results from rapid cycling due to assumed bypass leakage around the seals between the thermal sleeve and the feedwater nozzle. The rapid cycling fatigue usage through 40 years of operation is 0.035 but is projected to increase up to 0.772 by 51 years of operation. The feedwater nozzle is qualified for 60 years operation with a total fatigue usage of 0.064 + 0.772 = 0.836 based upon assumed seal refurbishment or reanalysis prior to exceeding 51 years of operation.
[3] The original ASME Code fatigue analysis provided a bounding CUF value for a location that does not contact reactor coolant. Therefore, a new analysis was prepared for this evaluation which was performed using the fatigue curve from NUREG/CR-6909. There was no CUF value computed using the ASME Code curve.
[4] The bounding Unit 1 ASME CUF value is used for Unit 2 due to a historical locked snubber condition on Unit 1.

As a result of the response to RAI 4.3-10 provided in Enclosure A of this letter, sections 4.3.1, 4.3.4 and A.4.3.4, pages 4-53, 4-54, 4-70 and A-42 of the LRA are revised as shown below:

4.3.1 ASME SECTION III, CLASS 1 FATIGUE ANALYSES

TLAA Evaluation:

The ASME Section III, Class 1 fatigue analyses for LGS include the stress reports for the Reactor Pressure Vessel (RPV), reactor coolant pressure boundary (RCPB) piping and components, including Class 1 valves. The current Class 1 fatigue analyses are based upon the same 40-year design transients as the original analyses, which are those listed in UFSAR Table 3.9-2, "Plant Events," for the RPV and UFSAR Table 5.2-9 "RCPB Operating Thermal Cycles" for the RCPB components (Reference 4.7.18). ***The RPV was also evaluated for MSRV discharge loads and LOCA-induced pool dynamic loads, including condensation oscillation and chugging cycles.*** Each Class 1 fatigue analysis demonstrates that the component has a CUF value that does not exceed the design Code limit of 1.0.

4.3.4 REACTOR VESSEL INTERNALS FATIGUE ANALYSES

TLAA Description:

LGS reactor internals were designed and procured prior to the issuance of ASME Section III, Subsection NG. However, an earlier draft of the ASME Code was used as a guide in the design of the reactor internals. Subsequent to the issuance of Subsection NG, comparisons were made that ensure the pre-NG design meets the equivalent level of safety as presented by Subsection NG. These fatigue analyses have been identified as TLAAs that require evaluation for the period of extended operation.

TLAA Evaluation:

The RPV and RPV internal components were included in the NSSS New Loads Design Adequacy Evaluations performed for each unit to address the effects of plant-specific seismic loadings and suppression pool hydrodynamic structural loadings on NSSS equipment. These evaluations included fatigue analyses of components if the applied loadings exceed certain thresholds. The fatigue analyses performed for the reactor internals components are based upon the same set of design transients as those used in the fatigue analyses for the reactor pressure vessel. ***This includes MSRV discharge loads and LOCA-induced loads, including condensation oscillation and chugging cycles.*** As ~~previously~~ shown on Tables 4.3.1-1 and 4.3.1-2, transient cycle projections were prepared that demonstrate these design transient cycle limits will not be exceeded in 60 years. ~~Therefore, these analyses will remain valid through the period of extended operation.~~ ***In order to ensure that these cycle projections will remain valid, the Fatigue Monitoring program will be used to manage fatigue of these components through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).***

TLAA Disposition: 10 CFR 54.21(c)(1)~~(i)~~*(iii)* – The reactor vessel internals fatigue analyses ~~remain valid for the period of extended operation.~~ ***will be managed by the Fatigue Monitoring program through the period of extended operation.***

A.4.3.4 Reactor Vessel Internals Fatigue Analyses

The RPV and RPV internal components were included in the NSSS New Loads Design Adequacy Evaluations performed for each unit to address the effects of plant-specific seismic loadings and suppression pool hydrodynamic structural loadings on NSSS equipment. These evaluations included fatigue analyses of components if the applied loadings exceed certain thresholds. ~~Several reactor vessel internal components have been analyzed for fatigue.~~ ~~including the top guide, core support plate, and core shroud~~. These fatigue analyses have been identified as TLAAs that require evaluation for the period of extended operation.

The fatigue analyses performed for the reactor internals components are based upon the same set of design transients as those used in the fatigue analyses for the reactor pressure vessel. Transient cycle projections were prepared that demonstrate the design transient cycle limits will not be exceeded in 60 years. ~~Therefore, these analyses remain valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).~~ ***In order to ensure that these cycle projections will remain valid, the Fatigue Monitoring program will be used to manage fatigue of these components through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).***

As a result of the response to RAI 4.3-10 provided in Enclosure A of this letter, Table 3.1.2-3 of the LRA (previously revised by Exelon response to RAI 4.6.9-1, Exelon letter dated January 24, 2012) pages 3.1-61, 63, 64, 66 and 70 are revised as shown below:

Table 3.1.2-3

Reactor Vessel Internals

Summary of Aging Management Evaluation

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
Core Shroud (including repairs) and Core Plate: Core Shroud (upper, central, lower)	Structural Support to maintain core configuration and flow distribution	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.9)	IV.B1.R-92	3.1.1-103	A
					Water Chemistry (B.2.1.2)	IV.B1.R-92	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
				Cumulative Fatigue Damage	***TLAA***	***IV.B1.R-53***	***3.1.1-3***	***A,1***
Core Shroud (including repairs) and Core Plate: Shroud support structure (shroud support cylinder, shroud support plate, shroud support legs)	Structural Support to maintain core configuration and flow distribution	Nickel Alloy	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.9)	IV.B1.R-96	3.1.1-103	A
					Water Chemistry (B.2.1.2)	IV.B1.R-96	3.1.1-103	A
				Cumulative Fatigue Damage	TLAA	IV.B1.R-53	3.1.1-3	A, 1
				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

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
Core Shroud and Core Plate: LPCI coupling	Direct Flow	Cast Austenitic Stainless Steel (CASS)	Reactor Coolant and Neutron Flux	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-97	3.1.1-103	A
					Water Chemistry (B.2.1.2)	IV.B1.R-97	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
				Cumulative Fatigue Damage	***TLAA***	***IV.B1.R-53***	***3.1.1-3***	***A,1***
		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
				Loss of Fracture Toughness	BWR Vessel Internals (B.2.1.9)	IV.B1.RP-200	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
Core Spray Lines and Spargers: Core spray lines (headers), Spray rings, Spray nozzles, Thermal sleeves	Direct Flow	Cast Austenitic Stainless Steel (CASS)	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.9)	IV.B1.R-99	3.1.1-103	A
					Water Chemistry (B.2.1.2)	IV.B1.R-99	3.1.1-103	A
				Loss of Fracture Toughness	BWR Vessel Internals (B.2.1.9)	IV.B1.RP-219	3.1.1-99	C
				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

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
Core Spray Lines and Spargers: Core spray lines (headers), Spray rings, Spray nozzles, Thermal sleeves	Direct Flow	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.9)	IV.B1.R-99	3.1.1-103	A
					Water Chemistry (B.2.1.2)	IV.B1.R-99	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
				Cumulative Fatigue Damage	***TLAA***	***IV.B1.R-53***	***3.1.1-3***	***A,1***
Fuel Supports and Control Rod Drive Assemblies: Orificed fuel support	Structural Support to maintain core configuration and flow distribution	Cast Austenitic Stainless Steel (CASS)	Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.9)	IV.B1.R-104	3.1.1-102	A
					Water Chemistry (B.2.1.2)	IV.B1.R-104	3.1.1-102	A
				Loss of Fracture Toughness	BWR Vessel Internals (B.2.1.9)	IV.B1.RP-220	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
				Cumulative Fatigue Damage	***TLAA***	***IV.B1.R-53***	***3.1.1-3***	***A,1***
Instrumentation: Intermediate range monitor (IRM) dry tubes, Source range monitor (SRM) dry tubes, Incore neutron flux monitor guide tubes	Structural Support to maintain core configuration and flow distribution	Stainless Steel	Air/Gas - Dry (Internal)	None	None	IV.E.RP-07	3.1.1-107	C
			Reactor Coolant and Neutron Flux	Cracking	BWR Vessel Internals (B.2.1.9)	IV.B1.R-105	3.1.1-103	A
					Water Chemistry (B.2.1.2)	IV.B1.R-105	3.1.1-103	A

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
				Cumulative Fatigue Damage	***TLAA***	***IV.B1.R-53***	***3.1.1-3***	***A,9***
		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

Page 3.1-70 of the LRA is revised as follows:

8. The TLAA designation in the aging management program column indicates that fatigue of the jet pump auxiliary spring wedge assemblies is evaluated in Section 4.6.

9. The TLAA designation in the aging management program column indicates that fatigue of the jet pump riser brace is evaluated in Section 4.3.

As a result of the response to RAI 4.3-10 provided in Enclosure A of this letter, LRA Appendix C, provided within Enclosure B of the Exelon response to RAI BWRVIP-1 dated February 15, 2012, is revised as shown below:

Action Item Description	**LGS Response**
BWRVIP-All (3) 10 CFR 54.22 requires that each application for license renewal include any technical specification changes (and the justification for the changes) or additions necessary to manage the effects of aging during the period of extended operation as part of the renewal application. The applicable BWRVIP reports may state that there are no generic changes or additions to technical specifications associated with the report as a result of its aging management review and that the applicant will provide the justification for plant-specific changes or additions. Those applicants for license renewal referencing the applicable BWRVIP report shall ensure that the inspection strategy described in the reports does not conflict with or result in any changes to their technical specifications. If technical specification changes or additions do result, then the applicant must ensure that those changes are included in its application for license renewal.	There are no technical specification changes identified that are required to meet the requirements of the BWRVIP reports during the period of extended operation. Reference LRA Appendix D.
Additional Action Items	
BWRVIP-18 Core Spray Internals Inspection and Flaw Evaluation Guidelines	
Action Item Description	**LGS Response**
BWRVIP-18 (4) Applicants referencing the BWRVIP-18 report for license renewal should identify and evaluate any potential TLAA issues which may impact the structural integrity of the subject RPV core spray internal components.	~~There were no TLAA Issues identified for the core spray components that are internal to the reactor vessel.~~ ***Fatigue usage is considered a TLAA for reactor vessel internals, including core spray piping and spargers. This is addressed in LRA Section 4.3.4.***

BWRVIP-42-A BWR LPCI Coupling Inspection and Flaw Evaluation Guidelines.	
Action Item Description	**LGS Response**
BWRVIP-42-A (4) Applicants referencing the BWRVIP-42 report for license renewal should identify and evaluate any potential TLAA issues which may impact the structural integrity of the subject RPV internal components	~~There were no TLAA Issues identified for the LPCI coupling.~~ ***Fatigue usage is considered a TLAA for reactor vessel internals, including the LPCI coupling. This is addressed in LRA Section 4.3.4.***
BWRVIP-42-A (5) The BWRVIP committed to address development of the technology to inspect inaccessible welds and to have the individual LR applicant notify the NRC of actions planned. Applicants referencing BWRVIP-42 report for license renewal should identify the action as open and to be addressed once the BWRVIP's response to this issue has been reviewed and accepted by the staff.	Inspection of the LPCI coupling is performed in accordance with guidelines described in BWRVIP-42. There are no inaccessible welds associated with the LPCI Couplings.

BWRVIP-47-A, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines	
Action Item Description	**LGS Response**
BWRVIP-47-A (4) Due to fatigue of the subject safety-related components, applicants referencing the BWRVIP-47 report for LR should identify and evaluate the projected CUF as a potential TLAA issue.	Fatigue usage is considered a TLAA for reactor vessel internals, including lower plenum components. This is addressed in LRA Section 4.3.4.

As a result of the responses to RAI 4.6.8-1 and RAI 4.6.8-2 provided in Enclosure A of this letter, sections 4.6.8 (TLAA Evaluation and TLAA Disposition sections) and A.4.6.8, pages 4-84, A-47 and A-48 of the LRA, are revised as shown below:

4.6.8 DOWNCOMERS AND MSRV DISCHARGE PIPING FATIGUE ANALYSES

TLAA Description:

Downcomer vents and Main Steam Relief Valve (MSRV) discharge piping penetrate the drywell and suppression pool diaphragm slab with the purpose of transporting steam and noncondensible gases to the suppression pool from the reactor and from the drywell during MSRV lifts and under accident conditions. MSRV quenchers are located at the bottom end of the MSRV discharge piping and are mounted to the floor of the suppression pool. Holes in the arms of the quenchers provide the dispersion of the steam into the suppression pool.

A Class 1 fatigue analysis was performed for the portion of the downcomers in the air space of the suppression chamber. The significant area analyzed for the downcomers in the suppression pool air space was the downcomer penetration through the diaphragm slab. Structural analyses were performed for of all the MSRV discharge lines from the diaphragm slab penetration to the quencher, including flued-head penetrations, elbows, tees, taper transitions, and three-way restraint attachments.

The downcomers and MSRV discharge lines were analyzed for the appropriate load combinations and their associated number of cycles. The combined stresses and corresponding equivalent stress cycles were computed to obtain the fatigue usage factors in accordance with the equations of Subsection NB-3600 of the ASME Code. Therefore, these Class 1 fatigue analyses have been identified as TLAAs that require evaluation for 60 years.

TLAA Evaluation *of Downcomers*:

The ~~MSRV~~ downcomers and bracing inside the suppression chamber ~~the MSRV discharge piping, and the main steam piping~~ have been evaluated for transient cycles predicted to occur in 40 years. A minimum of 7,700 MSRV ***stress*** cycles were considered to account for the pool dynamic loads, based on 1,100 actuations of all MSRVs times seven stress cycles per actuation. For the most frequently actuated MSRVs, the analysis was based on 4,700 actuations times three stress cycles per actuation (14,100 total cycles). ***They were also evaluated for 3,000 chugging cycles that could occur following a LOCA event, 50 OBE cycles (five events with ten cycles per event), and ten SSE cycles (one event with ten cycles).*** ~~The quenchers were analyzed for 7,000 SRV opening and closing cycles and 1,000,000 irregular condensation load cycles.~~

An operational review was performed for ~~the~~ MSRV ***actuations*** ~~lift cycles~~ that concluded the total number of ***stress*** cycles, projected for 60 years, will not exceed the number analyzed for 40 years. ***For Unit 1, the cumulative number of MSRV actuations that occurred through January 2011 is 90, but since each MSRV actuation can result in up to seven stress cycles, the cumulative number of stress cycles is assumed to be 630. This includes 56 actuations during startup testing prior to commercial operation and 30 actuations due to surveillance testing performed following six refueling outages. The surveillance testing requirement was eliminated in 1997 and no actuations have occurred on Unit 1 since 1995. The 60-year projection based on the average rate of occurrence through January 2011 is 211 actuations, assumed to result in 1,477 stress cycles, which is well below the 7,700 cycles analyzed. For Unit 2, the cumulative number of MSRV actuations that occurred through January 2011 is 49. This includes 32 actuations during startup testing prior to commercial operation and 15 actuations due to surveillance testing. No actuations have occurred on Unit 2 since 2001. The 60-year projection is 138 actuations, assumed to result in 966 stress cycles. These***

60-year projected values are conservative because they are based upon the average rates of occurrence over the time period that includes the startup test cycles and surveillance test cycles that are no longer performed.

Chugging cycles are loads that result from various modes of steam condensation at the downcomer vent ends following a LOCA. The Fatigue Monitoring program monitors Event 18, Faulted Condition – Pipe Rupture and Blowdown, which corresponds with a LOCA and has a design limit of one cycle. The Fatigue Monitoring program records show that no event has occurred in either unit. Therefore, it is concluded that no chugging events have occurred in either unit since they follow LOCA events.

LRA Tables 4.3.1-1 and 4.3.1-2 show that Operating Basis Earthquake (OBE) events are tracked in the Fatigue Monitoring program and that no OBE event has occurred in either unit to-date. The 60-year projection of one cycle does not exceed the design limit of one cycle. The Fatigue Monitoring program also monitors Faulted Condition-Safe Shutdown Earthquake (SSE) at Rated Operating Conditions. No event has occurred in either unit to-date. Since the SSE event is more severe than OBE event, the SSE event design limit will not be exceeded in 60 years for either unit.

Since there are monitored transients associated with each of the cycles analyzed for the downcomers, as described above, the Fatigue Monitoring program will be credited for managing fatigue of the downcomers and supports.

~~Therefore, the fatigue analyses for the downcomers and MSRV discharge piping will remain valid for the period of extended operation.~~

TLAA Disposition *of Downcomers*: ~~**10 CFR 54.21(c)(1)(i) –** The analyses remain valid for the period of extended operation.~~ ***10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended functions of the downcomers and supports will be managed by the Fatigue Monitoring (B.3.1.1) program for the period of extended operation.***

TLAA Evaluation of MSRV Discharge Piping:

A Class 1 fatigue analysis was performed for the MSRV discharge lines located in the suppression chamber wetwell. The MSRV discharge lines are subject to dynamic and hydrodynamic loads from normal, upset, and LOCA-related plant operating conditions. For purposes of fatigue evaluation, the following loads are included: (1) significant thermal and pressure transients, (2) cyclic loads due to hydrodynamic effects, including MSRV actuations, condensation oscillation and chugging, and (3) seismic effects. The quenchers were analyzed for 7,000 MSRV actuations (opening and closing cycles) and 1,000,000 irregular condensation load cycles.

Fatigue evaluation of the unsubmerged portion of the MSRV discharge piping was performed in accordance with ASME Section III, Class 1 fatigue rules. The transients used in the fatigue analysis include those specified for the MSRVs in UFSAR Section 3.9.1.1.9, including 150 preoperational and inservice testing cycles, 120 Startups, 120 Shutdowns, 180 Scrams, ten system temperature changes greater than 30ºF, eight cycles where system temperature changes from 546ºF to 375ºF in ten minutes (Emergency/Faulted event), and three other Emergency/Faulted events with a limit of one cycle each. LRA Tables 4.3.1-1 and 4.3.1-2 provide 60-year projections for these transients that demonstrate the numbers of cycles analyzed will not be exceeded during the period of extended operation.

The MSRV discharge piping was also evaluated for 4700 MSRV actuations that correspond to 14,100 stress cycles (submerged structure load), 1100 MSRV actuations that correspond to 7,700 stress cycles (inertia), 3,000 chugging cycles, 50 OBE cycles, and ten SSE cycles.

An operational review was performed that concluded the total number of MSRV actuations projected for 60 years, will not exceed the limiting number of 1100 analyzed for 40 years. Chugging and condensation oscillation cycles are caused by LOCA events that are monitored and have 60-year projections that do not exceed the number of cycles analyzed for 40 years. LRA Tables 4.3.1-1 and 4.3.1-2 show that no OBE event has occurred at LGS to-date and the 60-year projections do not exceed the number of cycles analyzed for 40 years. The two faulted events, Pipe Rupture and Blowdown, and Safe Shutdown Earthquake (SSE) at Rated Operating Conditions are also monitored, have had no occurrences to-date, and have a 60-year projected value of one cycle that does not exceed the 40-year design limit of one cycle. The Fatigue Monitoring program will be credited for managing fatigue of the MSRV discharge piping.

TLAA Disposition of MSRV Discharge Piping: 10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended functions of the downcomers and supports will be managed by the Fatigue Monitoring (B.3.1.1) program for the period of extended operation.

A.4.6.8 Downcomers and MSRV Discharge Piping Fatigue Analyses

The ~~MSRV~~ downcomers and bracing inside the suppression chamber~~,~~ ***and*** the MSRV discharge piping~~, and the~~ ***quenchers*** ~~main steam piping~~ have been evaluated for transient cycles predicted to occur in 40 years. Therefore, these fatigue analyses have been identified as TLAAs.

The downcomers and bracing were analyzed for a~~A~~ minimum of 7,700 MSRV ***stress*** cycles ***that*** were considered to account for the pool dynamic loads, based on 1,100 actuations of all MSRVs times seven stress cycles per actuation. For the most frequently actuated MSRVs, the analysis was based on 4,700 actuations times three stress cycles per actuation (14,100 total cycles). ***They were also analyzed for 3,000 chugging cycles that could occur following a LOCA event, 50 OBE cycles (five events with ten cycles per event), and ten SSE cycles (one event with ten cycles).*** The quenchers were analyzed for 7,000 ***M***SRV ***actuations (***opening and closing cycles***)*** and 1,000,000 irregular condensation load cycles.

An operational review was performed for ~~the~~ MSRV ***actuations***~~lift cycles~~ that concluded the total number of ***MSRV actuations***~~cycles~~, projected for 60 years, will not exceed the number analyzed for 40 years ***in either unit. The 60-year projection for Unit 1 is 211 actuations and for Unit 2 is 138 actuations, which are well below the 1100 actuations analyzed. No OBE or SSE event has occurred to-date and neither has a 60-year projection that exceeds the design limit of one event. The Fatigue Monitoring program is credited for managing fatigue of these components in accordance with 10 CFR 54.21 (c)(1)(iii) by ensuring that the analyzed numbers of cycles are not exceeded during the period of extended operation.*** ~~Therefore, the fatigue analyses for the downcomers and MSRV discharge piping will remain valid for the period of extended operation in accordance with 10CFR54.21(1)(c)(i).~~

The MSRV discharge lines were analyzed for significant thermal and pressure transients defined for the Main Steam system and for the same MSRV, chugging, OBE and SSE cycles described above for the downcomers. Each of these transients has been evaluated and shown to have 60-year projections that do not exceed the numbers of cycles analyzed. The Fatigue Monitoring program is credited for managing fatigue of these components in accordance with 10 CFR 54.21 (c)(1)(iii) by ensuring that the analyzed numbers of cycles are not exceeded during the period of extended operation.