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

January 31, 2012

Mr. Michael P. Gallagher
Vice President License Renewal Projects
Exelon Generation Company, LLC
200 Exelon Way
Kennett Square, PA 19348

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE LIMERICK GENERATING STATION, UNITS 1 AND 2 LICENSE RENEWAL APPLICATION (TAC NOS. ME6555 AND ME6556)

Dear Mr. Gallagher:

By letter dated June 22, 2011, Exelon Generation Company, LLC submitted an application pursuant to Title 10 of the Code of Federal Regulations Part 54, to renew the operating licenses for Limerick Generating Station, Units 1 and 2, for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review.

These requests for additional information were discussed with Christopher Wilson, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-3733 or e-mail Robert.Kuntz@nrc.gov.

Sincerely,

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

Docket Nos. 50-352, 50-353

Enclosure:
Requests for Additional Information

cc w/encl: Listserv

LIMERICK GENERATING STATION
LICENSE RENEWAL APPLICATION
REQUESTS FOR ADDITIONAL INFORMATION

RAI 4.1-1

Background

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.

Issue

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.

Request

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.

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.

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.

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.

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.

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.

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.

RAI 4.3-7

Background

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.

Issue

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.

Request

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.

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.

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.

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.

RAI 4.3-11

Background

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.

Issue

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.

Request

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.

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.

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.

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.

Letter to M. Gallagher from R. Kuntz dated January 31, 2012

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE LIMERICK GENERATING STATION, UNITS 1 AND 2 LICENSE RENEWAL APPLICATION (TAC NOS. ME6555 AND ME6556)

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Kennett Square, PA 19348

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE LIMERICK GENERATING STATION, UNITS 1 AND 2 LICENSE RENEWAL APPLICATION (TAC NOS. ME6555 AND ME6556)

Dear Mr. Gallagher:

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These requests for additional information were discussed with Christopher Wilson, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-3733 or e-mail Robert.Kuntz@nrc.gov.

Sincerely,

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

Docket Nos. 50-352, 50-353

Enclosure:
Requests for Additional Information

cc w/encl: Listserv

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