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10 CFR 50
10 CFR 51
10 CFR 54

RS-15-232

September 15, 2015

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

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Response to NRC Requests for Additional Information, Set 9, dated August 18, 2015 related to the LaSalle County Station, Units 1 and 2, License Renewal Application (TAC Nos. MF5347 and MF5346)

References: 1. Letter from Michael P. Gallagher, Exelon Generation Company LLC (Exelon), to NRC Document Control Desk, dated December 9, 2014, "Application for Renewed Operating Licenses"

2. Letter from Jeffrey S. Mitchell, US NRC to Michael P. Gallagher, Exelon, dated August 18, 2015, "Requests for Additional Information for the Review of the LaSalle County Station, Units 1 and 2 License Renewal Application – Set 9 (TAC Nos. MF5347 and MF5346)"

In Reference 1, Exelon Generation Company, LLC (Exelon) submitted the License Renewal Application (LRA) for the LaSalle County Station (LSCS), Units 1 and 2. In Reference 2, the NRC requested additional information to support staff review of the LRA.

Enclosure A contains the response to this request for additional information.

Enclosure B contains updates to sections of the LRA affected by the response.

There are no new or revised regulatory commitments contained in this letter.

If you have any questions, please contact Mr. John Hufnagel, Licensing Lead, LaSalle License Renewal Project, at 610-765-5829.

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

Executed on 09-15-2015

Respectfully,

Michael P. Gallagher

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

Enclosures: A: Responses to Set 9 Requests for Additional Information
B: LSCS License Renewal Application Updates

cc: Regional Administrator – NRC Region III
NRC Project Manager (Safety Review), NRR-DLR
NRC Project Manager (Environmental Review), NRR-DLR
NRC Project Manager, NRR-DORL- LaSalle County Station
NRC Senior Resident Inspector, LaSalle County Station
Illinois Emergency Management Agency - Division of Nuclear Safety

Enclosure A

Responses to Set 9 Requests for Additional Information Related to various sections of the LaSalle County Station (LSCS) License Renewal Application (LRA)

RAI B.2.1.7-3a
RAI 4.2.1-1
RAI 4.2.8-1
RAI 4.2.10-1

RAI B.2.1.7-3a

Background:

By letter dated June 25, 2015, the applicant responded to request for additional information (RAI) B.2.1.7-3 that addressed limited examination coverage for Generic Letter (GL) 88-01 welds. In its response, the applicant assessed a time period (February 2008 through March 2015) to evaluate the limited examination coverage of the welds within the scope of the BWR Stress Corrosion Cracking program. The applicant indicated that during the assessed period (February 2008 through March 2015) none of the Unit 1 Category A welds and Unit 2 Category C welds were inspected.

The applicant also indicated that there are only two Category D welds at Unit 2. The applicant further indicated that the examination coverage for each of the welds was limited to 50 percent because each weld is between pipe and a cast valve body where the current approved Performance Demonstration Initiative (PDI) method does not provide qualified and reliable results for the side of the weld adjacent to the cast valve body due to the specific contour of the valve body.

Issue:

As described in the background section, the response did not provide information regarding the examination coverage of Unit 1 Category A welds and Unit 2 Category C welds. The staff needs information on examination coverage of these welds based on previous inspection results in order to resolve the concern related to limited examination coverage.

The staff also needs additional information to evaluate applicant's justification for the limited examination coverage of the Unit 2 Category D welds. It is unclear whether the cast valve bodies associated with the Unit 2 Category D welds are made of cast materials resistant to intergranular stress corrosion cracking (IGSCC). In addition, the staff needs to confirm whether inspection results for the cast valve bodies indicate occurrence of IGSCC.

Request:

1. Provide information on the average examination coverage of Unit 1 Category A welds and Unit 2 Category C welds to characterize the overall degree of limited examination coverage. If limited examination coverage is identified for these welds, provide justification for why additional inspections are not necessary to compensate for the limited examination coverage of these welds.
2. Provide the following information regarding the cast valve bodies associated with the two Category D welds at Unit 2:
    a. Cast material grade (e.g., CF3M), carbon content, and ferrite content (including the method for determining ferrite content) of the cast valve bodies to clarify whether the materials are resistant to IGSCC in accordance with the staff positions in GL 88-01
    b. American Society of Mechanical Engineers (ASME) Code Class and, if available, examination results of the cast valve bodies such as internal visual or surface examination results per ASME Code Section XI requirements

Exelon Response:

1. In order to provide examination results to characterize the overall degree of limited examination coverage for Unit 1 Category A welds and Unit 2 Category C welds, the period from February 2006 through March 2015 was reviewed. The average examination coverage for welds examined for the requested IGSCC Category of welds is as follows:

LSCS Unit	IGSCC Category	Date Examined	Number of Welds Inspected	Average Exam Coverage
1	A	March 2006	5	100%
2	C	March 2007	2	50%

Note that since there are only eight welds in the population of Unit 2 Category C welds, only two welds (25 percent) are examined in each 10-year period, per BWRVIP-75-A, “BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules.”

All volumetric weld examinations performed under the scope of the LSCS commitment to NRC GL 88-01 were to the maximum examination volume coverage practical using current approved Performance Demonstration Initiative (PDI) methods and procedures. For the two Unit 2 Category C welds, where 50 percent exam coverage is reported, the welds are located between 24 inch piping and 4 inch weld-o-lets to the reactor water cleanup system and to a decontamination connection. For these weld configurations, due to the specific contour of the weld-o-let to pipe weld, the PDI method can only provide qualified and reliable results for the side of the weld adjacent to the piping (i.e., 50 percent of the weld). For one-sided weld examinations, PDI examination procedures require volumetric interrogation of the opposite side weld and base material to the extent possible with unqualified techniques. PDI is currently making efforts to qualify examination techniques for detection of IGSCC on the opposite side of the weld for austenitic welds. Exelon intends to continue to utilize the latest commercially available examination techniques that provide coverage to the maximum extent practical.

Justification for why additional inspections are not necessary to compensate for the limited examination coverage of the Unit 2 Category C welds is as follows:

- Use of approved PDI methods for examination of welds being managed by the BWR Stress Corrosion Cracking program results in a best-effort approach using the best available approved volumetric examination techniques. For one-sided weld examinations, PDI examination procedures require volumetric interrogation of the opposite side weld and base material to the extent possible.
- More than 90 percent examination coverage is achieved for the majority of the examined welds. For the 85 weld examinations reported in the Exelon response to RAI B.2.1.7-3 and in this response, the average examination coverage is 89.4 percent. Therefore, the use of the PDI examination method results in a high

percentage of qualified and reliable examination coverage for welds within the scope of the BWR Stress Corrosion Cracking program.

- The eight Unit 2 Category C welds all received mechanical stress improvement during the second refueling outage in October 1988, after less than five years of operation.
- During the 31-year Unit 2 operating history, there have been no indications of IGSCC for welds within the scope of GL 88-01.
- Additional examinations will not result in increased examination coverage for welds where the PDI method can only provide qualified and reliable results for one side of the weld. The use of unqualified one-sided examination techniques required by PDI examination procedures remains the best available method to detect degradation on the opposite side of austenitic stainless steel welds.

Therefore, performance of additional inspections to compensate for limited examination coverage is not considered necessary to manage cracking due to IGSCC during the period of extended operation.

2. The two Unit 2 Category D welds connect valve 2E12-F090A to the upstream pipe 2RH03DA-12" and to the downstream elbow on pipe 2RR07AA-12". This valve is shown on drawing LR-LAS-M-142, Sheet 1 (D-8).

   a. The following information regarding the body of valve 2E12-F090A is provided:
      - Cast Material Grade - SA-351 Grade CF8M
      - Carbon Content - 0.04%
      - Ferrite Content - 19.3%
      - Method for determining ferrite content - Hull's Equivalent Factors method

      The staff position on materials within NRC GL 88-01 defines cast austenitic stainless steel with a maximum of 0.035 percent carbon and a minimum of 7.5 percent ferrite content as resistant to IGSCC within BWR piping systems. The carbon content for valve 2E12-F090A body is slightly above the maximum stated in GL 88-01 to be considered resistant to IGSCC; but the ferrite content is significantly above the minimum stated in GL 88-01 to be considered resistant to IGSCC. Therefore, the cast 2E12-F090A valve body is expected to have good resistance to IGSCC. The piping welded to valve 2E12-F090A is not resistant to IGSCC, therefore if IGSCC were to initiate in these welds, it would be expected to initiate in the piping side of the weld. The piping side of these welds is subject to qualified and reliable examination using the PDI method every six years in accordance with the examination schedule of BWRVIP-75-A, and no indications of cracking have been identified. As described in the response to Request 1, PDI examination procedures require interrogation of the opposite side weld and base material to the extent possible.

   b. Valve 2E12-F090A and the welds to attached piping are classified as ASME Code Class 1. ASME Code Section XI requires a Visual VT-3 examination of the internal surface of the valve body only when the valve is disassembled for maintenance or repair in accordance with Table IWB-2500-1, Examination Category B-M-2, Item No. B12.50 and associated notes 2 and 3. There have been no maintenance activities during the Unit 2 operating history, where valve

2E12-F090A was disassembled such that the internal surface of the valve was made accessible for examination per ASME Section XI, Table IWB-2500-1, Exam Category B-M-2, Item B12.50. Therefore, there are no examination results of the cast valve body such as internal visual or surface examination results per ASME Code Section XI requirements.

RAI 4.2.1-1

Background:

License Renewal Application (LRA) Section 4.2.1 describes the applicant's time-limited aging analysis (TLAA) on reactor vessel fluence calculations. LRA Section 4.2.1 also indicates that the 54-effective full power year (EFPY) fluence projections for 60 years of operation were calculated by using the NRC-approved Radiation Analysis Modeling Application (RAMA) methodology. The LRA further states that the 54-EFPY fluence projections compile the cumulative fluence resulting from each past operating cycle and add the predicted fluence estimate for future operating cycles through the period of extended operation. As discussed below, the NRC staff noted that these 54-EFPY fluence projections are independent from the fluence projections used in the current P-T limits.

LRA Section 4.2.1 states that the neutron fluence projections used as inputs to the current 40-year neutron embrittlement analyses for LSCS, Units 1 and 2 were developed in accordance with GE Licensing Topical Report NEDO-32983, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," (ADAMS Accession No. ML072480121). LRA Section 4.2.1 also indicated that this GE methodology is in compliance with NRC Regulatory Guide 1.190, as approved by the NRC staff in the safety evaluation dated September 14, 2001.

Under LRA Section 4.2.1, the TLAA evaluation states that the 60-year RAMA neutron fluence projections compile the cumulative neutron fluence resulting from each past operating cycle and add the predicted neutron fluence estimated for future operating cycles through the period of extended operation. The RAMA neutron fluence projections are prepared using the BWRVIP-126 methodology. These 60-year neutron fluence projections are independent from the neutron fluence projections used with the current P-T curve submittals.

In comparison, the following reference indicates that the current 40-year (32-EFPY) P-T limits for LSCS Unit 1 use neutron fluence values that are calculated via the RAMA fluence methodology.

- NRC Safety Evaluation, "LaSalle County Station, Unit 1, Issuance of Amendment Revising Pressure and Temperature Limits (TAC No. MF3270)," dated November 25, 2014 (ADAMS Accession No. ML14220A517).

Section 3.1, Neutron Fluence Calculation, of the above reference states that the safety evaluation shall not be constructed as endorsement, agreement with, or approval of, the position regarding the combined use of neutron fluence methods to determine a total neutron fluence.

Issue:

The LRA does not address the RAMA methodology that the NRC staff evaluated in the safety evaluation regarding the current P-T limits for LSCS, Unit 1. Therefore, the NRC staff needs additional clarification as to the specific neutron fluence methodology that is used in the license renewal TLAAs. It is also unclear to the NRC staff how the applicant evaluates potential effects of updated neutron fluence calculation on existing neutron embrittlement analysis.

Request:

1. Clarify whether the 54-EFPY neutron fluences described in the LRA are based on the GE methodology, the RAMA methodology, or the combination of the two methodologies.
2. Clarify how the applicant will ensure that the actual fluence levels are bounded by the fluence levels analyzed in LRA Section 4.2.1.

Exelon Response:

Background

Section 4.2.1 of the LRA does not describe the fluence methodology used to support the current 40-year (32 EFPY) pressure-temperature (P-T) limits for Unit 1 since the current P-T limits were approved (November 25, 2014) just prior to the LSCS LRA submittal (December 9, 2014).

The Unit 1 P-T limit submittal dated December 20, 2013 (ADAMS Accession No. ML13358A363) used a fluence methodology that combined the fluence results from the General Electric-Hitachi (GEH) methodology with the fluence results from the Radiation Analysis Modeling Application (RAMA) methodology. The staff determined that combining the two methods is not consistent with NRC Regulatory Guide 1.190. A supplemental letter from LSCS to the NRC dated February 26, 2014 provided neutron fluence projections that were developed using just the RAMA fluence methodology alone. These Unit 1 RAMA fluence projections were evaluated by the staff in the NRC Safety Evaluation dated November 25, 2014 (ADAMS Accession No. ML14220A517) and were determined to be consistent with the guidance set forth in NRC Regulatory Guide 1.190, including analytic uncertainty and benchmarking. These fluence projections were then used by the staff to determine that the P-T limits based on the fluence values using the combination of GEH and RAMA methods are acceptable since the fluence values are higher and more conservative than the RAMA-only values.

Consistent with the above discussion, the TLAA Description in LRA Section 4.2.1 is updated to provide a description of the fluence methodology associated with the current licensing basis P-T limits for Unit 1, as shown in Enclosure B.

For Unit 2, the current 40-year (32 EFPY) P-T limits are based upon a GEH fluence analysis used in support of the Measurement Uncertainty Recapture Power Uprate approved September 16, 2010 (ADAMS Accession No. ML101830361). This GEH fluence analysis was previously identified as a TLAA in LRA Section 4.2.1 and is unchanged; however the TLAA Description is updated for clarity as shown in Enclosure B.

Response to Request 1

The 54-EFPY neutron fluence projections for Units 1 and 2 described in LRA Section 4.2 to account for vessel exposure from 0 to 54 EFPY are based on the RAMA fluence methodology. It should be noted that the RAMA 1/4T neutron fluence values submitted to support the Unit 1 P-T Limits discussed above were based on reactor vessel nominal wall thickness whereas the RAMA 1/4T neutron fluence values for Unit 1 and 2 provided in LRA Section 4.2 are conservatively based on reactor vessel minimum wall thickness. Using minimum wall thickness rather than nominal wall thickness results in slightly higher fluence values.

Response to Request 2

To ensure the Unit 1 and 2 actual fluence levels are bounded through the period of extended operation, the fluence values projected in LRA Section 4.2.1 are based on conservative estimates of future unit operating capacity factor and previously withdrawn and tested flux wires and capsule dosimetry. EFPY and cumulative fluence are tracked to ensure analyzed conditions are not exceeded. Prior to each scheduled reactor refueling outage, actual EFPY and estimated EFPY at the end of the fuel cycle are determined and reconciled against the current licensing basis. Prior to exceeding the EFPY value qualified in the current P-T limits, fluence projections will be updated and revised P-T limits will be calculated and incorporated into the current licensing basis.

Furthermore, LSCS Unit 1 is a BWRVIP integrated surveillance program (ISP) host plant. As a BWRVIP ISP host plant, one remaining surveillance capsule will be withdrawn per the BWRVIP ISP withdrawal schedule that will provide future dosimetry data for use as an additional benchmark for the LSCS Unit 1 fluence projections.

LRA Section 4.2.1 and 4.8 are revised as shown in Enclosure B to clarify the fluence projection inputs to the current 40-year neutron embrittlement analyses.

RAI 4.2.8-1

Background:

LRA Section 4.2.8 describes the TLAA for the loss of preload of the reactor pressure vessel (RPV) core plate rim hold-down bolts resulting from irradiation effects. The LRA states that a fluence evaluation based on 54 EFPY has identified the bolts with the highest fluence values after 60 years of operation for Unit 1 (3.60×10^{19} n/cm^2) and Unit 2 (3.85×10^{19} n/cm^2). The LRA also states that an average fluence value of 8.0×10^{19} n/cm^2 was evaluated for 40 years of operation which resulted in a maximum relaxation of 19 percent in preload for the RPV core plate rim hold-down bolts. The TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i) to remain valid through the period of extended operation.

The loss of preload analysis was performed as part of Boiling Water Reactor Vessel and Internals Project Technical Report 25 (BWRVIP-25), "BWR Core Plate Inspection and Flaw Evaluation Guidelines." Section 2.1.3 of BWRVIP-25 states that it has been determined that a 5 percent to 19 percent reduction in preload is expected over a 40-year operating experience and that the bolts will maintain some amount of preload throughout the life of the plant. Section B.4 of Appendix B states that the loss of preload was recalculated based on 60 years of plant operation and that the reduction in preload remains 5 percent to 19 percent.

Issue:

The staff is unable to determine if the amount of relaxation experienced after 60 years of operation is bounded by the 40-year analyses because the fluence values used to determine the range in percent reduction of preload are not provided in BWRVIP-25 or the appendices of the report.

Request:

Provide the analysis used to determine that an average fluence value of 8.0×10^{19} n/cm^2 produces a maximum relaxation of 19 percent in preload for the RPV core plate rim hold-down bolts. Justify that the analysis adequately considers plant-specific configurations of reactor vessel internals and is bounded by the relaxation evaluations in BWRVIP-25.

Exelon Response:

The analysis used to determine that an average fluence value of 8.0×10^{19} n/cm^2 produces a maximum relaxation of 19 percent in preload for the RPV core plate rim hold-down bolts is summarized in a letter from Brian D. Frew, GE Nuclear, to Randy Stark, EPRI, Subject: Relaxation of Core Plate Rim Hold-down Bolts, dated June 29, 2006 (GE-EN-0000-0055-6793).

This letter states that "the original fluence estimates that were used to derive the amount of relaxation [of the core plate bolts] [within BWRVIP-25] were not bounding for the plant-specific conditions, including the effects of power uprate and the additional exposure associated with the period of extended operation. As a result, GE has re-evaluated the maximum relaxation value of 19%, and determined that this value is applicable to an average fluence level of 8E19 n/cm^2 over the entire length of the bolt, determined at the peak azimuthal fluence location."

The letter concludes that: “Evaluation of core plate bolt relaxation has determined that the maximum reported relaxation value of 19% remains valid to an average fluence value of 8E19 n/cm^2 or less. This fluence is an average fluence over the entire length of the core plate bolt. Plant-specific evaluations required for license renewal, or other structural evaluation, should incorporate this limitation.”

As described in LRA Section 4.2.8, plant-specific 54 EFPY RAMA fluence projections were prepared for all 34 core plate rim hold-down bolts that consider the plant-specific configuration of reactor vessel internals to identify the bolts at the peak azimuthal fluence location (limiting bolts) on each LSCS unit. 54 EFPY fluence projections were made at 30 points equally spaced along the length of the limiting bolts at their centerline. These values were averaged to determine the average fluence over the entire length of the bolt and compared to the maximum acceptable average fluence value of 8.0×10^{19} n/cm^2 that corresponds to a relaxation value of 19 percent. The average 54 EFPY fluence values for the limiting bolts are 3.60×10^{19} n/cm^2 on LSCS Unit 1 and 3.85×10^{19} n/cm^2 on LSCS Unit 2 which are less than 8.0×10^{19} n/cm^2. Therefore, the analysis as described above adequately considers plant-specific projections for neutron fluence and plant-specific configurations of reactor vessel internals.

BWRVIP-25, Appendix B, Section B-4 states that the analysis described in BWRVIP-25, Section 2.1.3 “was recalculated for 60-year plant life, and it was determined that all but two BWR/3s would show a 5-19% reduction in stress (e.g., loss of preload).” Since LSCS Units 1 and 2 are BWR/5 units and the average 54 EFPY fluence values for the limiting bolts were determined to be less than 8.0×10^{19} n/cm^2, these results are consistent with, and are bounded by the relaxation evaluations in BWRVIP-25. It should be noted that the referenced letter from GE Nuclear, dated June 29, 2006 states that “GE’s limited results [as of June 2006] indicate that no plant will exceed the 8E19 n/cm^2 value for the period of extended operation, even including the effects of power uprate.”

RAI 4.2.10-1

Background:

Section 4.7.3.1.3 of the Standard Review Plan for License Renewal (SRP-LR) (NUREG-1800, Rev. 2) provides the NRC's review procedures for plant-specific TLAAs which will be managed in accordance with 10 CFR 54.21(c)(1)(iii). SRP-LR Section 4.7.3.1.3 states that the staff is to review the aging management program proposed by the applicant to verify the program is adequate to manage the aging effects associated with the TLAA.

LRA Section 4.2.10 describes the TLAA evaluation for the loss of preload of the jet pump slip joint repair clamp. LRA Section 4.2.10 dispositions the TLAA in accordance with 10 CFR 54.21(c)(1)(iii), to be managed by Commitment No. 47 in LRA Section A.5. This commitment states:

> Prior to exceeding the limiting fluence value of 1.17E+20 n/cm^2 at the Unit 1 jet pump slip joint clamp location, estimated to be at 50.7 EFPY, revise the analysis for the slip joint clamps for a higher acceptable fluence value or take other corrective action such as repair or replacement of the clamps to ensure acceptable clamp preload.

The implementation schedule for Commitment No. 47 is "Prior to the period of extended operation."

Issue:

The LRA does not clearly identify how the applicant will ensure that the limiting fluence value is not exceeded. The staff is unable to determine if the applicant's program or activities are adequate to ensure that the limiting fluence value is not exceeded.

Request:

Identify the program or a set of activities that will be used for the jet pump slip joint repair clamp to ensure that the limiting fluence value is not exceeded. Justify that the program or activities are adequate for managing the aging effect of loss of preload of the jet pump slip joint repair clamp.

Exelon Response:

The program or set of activities that will be used for the jet pump slip joint repair clamps, to ensure that the limiting fluence value is not exceeded, will be integrated into the Exelon commitment management program. In accordance with the Exelon commitment management program and procedures, license renewal commitments will have action tracking items assigned to the responsible commitment owners to assure that the commitments are met prior to their due date. The commitment tracking assignment for Commitment No. 47 will include an action tracking assignment with a due date approximately five years prior to the estimated calendar date that 50.7 EFPY will be reached for Unit 1. This assignment will ensure that appropriate corrective actions are taken to repair, replace, or reanalyze the clamps well in advance of reaching the fluence limit.

LSCS procedure LTS-1200-4, Reactor Engineer's Core Monitoring Surveillance, requires the Reactor Engineer to update the cumulative EFPY value for LSCS Units 1 and 2 each month and provide the updated EFPY values to the Reactor Internals Engineer. As part of implementation of Commitment No. 47, this procedure will be revised prior to the period of extended operation to require the Reactor Internals Engineer, who is the BWR Vessel Internals (B.2.1.9) aging management program owner, to ensure corrective action has been taken prior to exceeding 50.7 EFPY on Unit 1 relative to License Renewal Commitment No. 47 that manages limiting neutron fluence at the Unit 1 jet pump slip joint clamps.

The assignment within the commitment tracking program and the revision to procedure LTS-1200-4 are the set of activities that will ensure that the aging effect of loss of preload for the jet pump slip joint repair clamps is adequately managed so the limiting fluence value of 1.17E+20 n/cm^2 is not exceeded and minimum slip joint repair clamp preload is maintained.

Enclosure B

LSCS License Renewal Application Updates
Resulting from the Response to the following RAI:

RAI 4.2.1-1

Note: To facilitate understanding, portions of the LRA have been repeated in this Enclosure, with revisions indicated. Previously submitted information is shown in normal font. Changes are highlighted with ***bolded italics*** for inserted text and ~~strikethroughs~~ for deleted text.

As a result of the response to RAI 4.2.1-1 provided in Enclosure A of this letter, the TLAA Description subsection of LRA Section 4.2.1 on page 4-9 of the LRA is revised as shown below:

4.2.1 NEUTRON FLUENCE ANALYSES

TLAA Description:

Neutron fluence is the term used to represent the cumulative number of neutrons per square centimeter that contact the reactor vessel shell and its internal components over a given period of time. The fluence projections that quantify the number of neutrons that contact these surfaces have been used as inputs to the neutron embrittlement analyses that evaluate the loss of fracture toughness aging effect resulting from neutron fluence.

The fluence projections used as inputs to the current 40-year neutron embrittlement analyses for LSCS Units 1 and 2 were developed ~~in accordance with GE Licensing Topical Report NEDC-32983P, which was approved by the NRC in a SER dated September 14, 2001 and which is in compliance with~~ ***based on the guidance contained in*** NRC Regulatory Guide 1.190 (Reference 4.8.1). The projections predict the neutron fluence expected for 32 Effective Full Power Years (EFPY) of plant operation. At the time the projections were prepared, 32 EFPY was considered to represent the amount of power to be generated over 40 years of plant operation, assuming a 40-year average capacity factor of 80 percent.

~~A 1.7 percent power uprate was implemented at the beginning of Cycle 14 for each unit, and as part of the evaluation of the uprates, the initial 32 EFPY fluence values described above were conservatively scaled up by 2 percent. The resulting peak inside-surface fluence value (at the clad-base metal interface) for Unit 1 is 1.04E+18 n/cm^2 and is 1.112E+18 n/cm^2 for Unit 2. These adjusted fluence values were used as inputs in the current (40-year) neutron embrittlement analyses, including Adjusted Reference Temperature (ART) calculations, Upper-Shelf Energy (USE) calculations, Pressure-Temperature (P-T) Limits, Axial Weld Failure Probability calculations, and Circumferential Weld Failure Probability calculations that are evaluated in LRA Sections 4.2.2, 4.2.3, 4.2.4, 4.2.5, and 4.2.6, respectively. These adjusted fluence projections have been identified as TLAAs requiring evaluation for the period of extended operation.~~

For Unit 1, the fluence projection values used as inputs to the current 40-year neutron embrittlement analyses were determined using a combination of NRC-approved methods and considered a 1.7 percent power uprate that was implemented during Cycle 14 operation. The first 13 cycles of fluence were calculated using the Radiation Analysis Modeling Application (RAMA) methodology in accordance with BWRVIP-126: BWR Vessel and Internals Project, RAMA Fluence Methodology Software (Reference 4.8.3). The subsequent cycle fluence values were calculated using the General Electric-Hitachi (GEH) methodology in accordance with the NEDC-32983P-A, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation (Reference 4.8.47). In addition, separate fluence projections were performed using the RAMA fluence methodology alone to account for the vessel exposure from 0 to 54 EFPY. This single method RAMA fluence methodology calculation adhered to the guidance in NRC RG 1.190 and was determined to be acceptable by the NRC in SER dated November 25, 2014 (Reference 4.8.48). The projected fluence values determined using the combination of methodologies (RAMA/GEH) were shown to be conservative as compared to the

projected fluence values determined using the single method (RAMA) and considered acceptable for use (Reference 4.8.48). The projected peak inside-surface fluence value (at the clad-base metal interface) for Unit 1 at 32 EFPY is 8.34E+17 n/cm^2.

For Unit 2, the fluence projections used as inputs to the current 40-year neutron embrittlement analyses were developed in accordance with GE Licensing Topical Report NEDC-32983P, which was approved by the NRC in a SER dated September 14, 2001 (Reference 4.8.49) and is in compliance with NRC Regulatory Guide 1.190 (Reference 4.8.1). A 1.7 percent power uprate was implemented at the beginning of Cycle 14 and, as part of the evaluation of the uprate, the initial 32 EFPY fluence values (pre-uprate) were conservatively scaled up by 2 percent (Reference 4.8.50). The projected peak inside-surface fluence value (at the clad-base metal interface) for Unit 2 at 32 EFPY is 1.11E+18 n/cm^2.

The fluence values determined above were used as inputs in the current (40-year) neutron embrittlement analyses, including Upper-Shelf Energy (USE) calculations, Adjusted Reference Temperature (ART) calculations, Pressure-Temperature (P-T) Limits, Axial Weld Failure Probability calculations, and Circumferential Weld Failure Probability calculations that are evaluated in LRA Sections 4.2.2, 4.2.3, 4.2.4, 4.2.5, and 4.2.6, respectively. These fluence projections have been identified as TLAAs requiring evaluation for the period of extended operation.

As a result of the response to RAI 4.2.1-1 provided in Enclosure A of this letter, LRA Section 4.8 on page 4-114 of the LRA is revised to add new references as shown below:

4.8.47 NEDC-32983P-A, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation, Revision 2, January 2006

4.8.48 Letter from Blake Purnell (U. S. NRC) to Michael Pacilio (Exelon), LaSalle County Station, Unit 1, Issuance of Amendment Revising Pressure and Tempurature (sic) Limits (TAC NO. MF3270), dated November 25, 2014 (ADAMS Accession No. ML14220A517)

4.8.49 NRC Safety Evaluation for NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation" (TAC No. MA9891), dated September 14, 2001 (ADAMS Accession No. ML012400381)

4.8.50 Letter from Christopher Gratton (U. S. NRC) to Michael Pacilio (Exelon), LaSalle County Station, Units 1 and 2 – Issuance of Amendments RE: Measurement Uncertainty Recapture Power Uprate (TAC NOS. ME3288 and ME3289), dated September 16, 2010 (ADAMS Accession No. ML101830361)