June 14, 2010

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear LLC
P.O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2 LICENSE RENEWAL APPLICATION REGARDING SECTIONS 4.3 AND 4.4 (TAC NOS. ME1834 AND ME1836)

Dear Mr. Joyce:

By letter dated August 18, 2009, as supplemented by letter dated January 23, 2009, Public Service Enterprise Group Nuclear, LLC, submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 for renewal of Operating License Nos. DPR-70 and DPR-75 for Salem Nuclear Generating Station, Units 1 and 2, respectively. The staff of the U.S. Nuclear Regulatory Commission (NRC or the staff) is reviewing this application in accordance with the guidance in NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants." During its review, the staff has identified areas where additional information is needed to complete the review. The staff's request for additional information is included in the Enclosure. Further requests for additional information may be issued in the future.

Items in the enclosure were provided to John Hufnagel and other members of your staff, 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 by telephone at 301-415-2981 or by e-mail at bennett.brady@nrc.gov.

Sincerely,

/RA/

Bennett M. Brady, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosure:
As stated

cc w/encl: See next page

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ADAMS Accession No. ML101480189

OFFICE	PM:DLR:RPB1	LA:DLR	BC:DLR:RPB1	PM:DLR:RPB1
NAME	B. Brady	I. King	B. Pham	B. Brady
DATE	06/09/10	06/08/10	06/11/10	06/14/10

OFFICIAL RECORD COPY

Letter to T. Joyce from B. Brady dated June 14, 2010

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2 LICENSE RENEWAL APPLICATION REGARDING SECTIONS 4.3 AND 4.4 (TAC NOS. ME1834 AND ME1836)

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Salem Nuclear Generating Station,
Units 1 and 2

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Hancocks Bridge, NJ 08038

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Salem, NJ 08079

REQUEST FOR ADDITIONAL INFORMATION FOR
SALEM NUCLEAR GENERATING STATION UNITS 1 AND 2
LICENSE RENEWAL APPLICATION REGARDING
SECTIONS 4.3 AND 4.4 (TAC NO. ME1834 / ME1836)

RAI 4.3-01

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue:

License renewal application (LRA) Section 4.3 states that, as of December 31, 2007, Salem Nuclear Generating Station (SNGS), Units 1 and 2 have been operational for 31.4 and 27.8 calendar years, respectively. This included non-operational periods of 2.9 and 2.2 years for SNGS, Units 1 and 2, respectively. LRA Section 4.3 further states that the average rate of cycle occurrences was determined from the cumulative number of cycle occurrences and 28.5 and 25.6 years of past operation for SNGS, Units 1 and 2, respectively. However, LRA Section 4.3 does not provide sufficient information for the staff to confirm that cycle counting has been performed from the plant start-up and during the entire period of past operation prior to December 31, 2007.

Request:

Clarify whether the cycle counting for the design basis transients at SNGS Units 1 and 2 has been performed during the entire period of past operation (i.e., over the entire time of operation since the initial startups of the units, including times during heatup and cooldown conditions but not including hot or cold shutdown conditions or hot standby conditions), or whether any unmonitored periods exist during plant operation.

RAI 4.3-02

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue:

LRA Table 4.3.1-2 lists the limiting design basis number of occurrences for 40 years for nuclear steam supply system (NSSS) Class A and Class 1 components at SNGS, Units 1 and 2. However, LRA Section 4.3.1 does not reference the design basis documents that confirm the limiting design basis number of occurrences provided in LRA Table 4.3.1-2.

Request:

Clarify which current licensing basis (CLB) documents or design basis documents provide CLB or design basis transient cycle limits for the transients that are listed in LRA Table 4.3.1-2 for NSSS Class A and Class 1 components at SNGS.

RAI 4.3-03

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue:

LRA Section 4.3.4.4 states that the 60-year projected cycles applicable to the SNGS Unit 1 steam generator (SG) manway studs are bounded by the cycles used in 40-year fatigue analysis. However, LRA does not provide sufficient information to confirm this assertion.

Request:

Identify the transients that were used in 40-year fatigue analyses of the SG manway studs. Provide the 60-year cycle projections for any transients that are applicable to fatigue analyses of the SG manway studs but that are not within the scope of the transients that are listed in LRA Table 4.3.1-3 and 4.3.1-4 (which list the 60-year transient cycle projections for SNGS, Units 1 and 2, respectively).

RAI 4.3-04

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue:

LRA Section 4.3.4.3 states that the thermal stratification loads are managed by LRA aging management program (AMP) B.3.1.1 "Metal Fatigue of Reactor Coolant Pressure Boundary Program," where the number of auxiliary feedwater flow operational hours are tracked and compared to the design limit of 12,000 hours. However LRA AMP B3.1.1 is described in the LRA as consistent with Generic Aging Lessons Learned (GALL) AMP X.M1, "Metal Fatigue of the Reactor Coolant Pressure Boundary," which pertains to cycle counting of occurrence of design basis transients, not to the tracking of amassed time of operation (in seconds, minutes, hours, etc.). The LRA does not provide sufficient information for the staff to determine how LRA

AMP B.3.1.1 is designed to track amassed hours against an hourly design limit of 12,000 hours when the program is designed to track cumulative number of design basis transient occurrences that occur at the Salem unit facilities (i.e., to perform cycle tracking) in order to comply with technical specification 5.7.1 and other design basis requirements.

Request:

Justify why the tracking of hours for the auxiliary feedwater nozzle components has not been identified as an enhancement to the "detection of aging effects" program element in GALL AMP X.M1 and why it is valid to use the program as a basis for tracking the cumulative number of hours for which the auxiliary feedwater nozzles have been in operation, when the program is designed to track the number of design basis transient cycle occurrences. If it is valid to track the auxiliary feedwater nozzles in this manner, the LRA description of the AMP should be amended to identify this as an enhancement to the following program elements in GALL AMP X.M1, with explanations and justifications on the enhancements: (1) "detection of aging effects" program element, with an explanation/justification on how the LRA AMP B.3.1.1 is different from GALL AMP X.M1 to permit tracking the cumulative number of hours that these components have been in operation as opposed to tracking discrete occurrences of transients; (2) "acceptance criteria" program element, with an explanation/justification of the action limit that will be used to take appropriate corrective actions if the number of tracked hours is determined to encroach on the 12,000 hour design limit; (3) "monitoring and trending" program element, with an explanation/justification on how the tracking of the cumulative hours of operation will be trended against the defined action limit on hourly tracking; and (4) the "corrective actions" program element, with an explanation/justification on the corrective actions that will be applied to the auxiliary feedwater nozzle components if the monitoring and trending of cumulative hours in operation reaches the established action limit on cumulative hour tracking.

RAI 4.3-5

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue (1):

LRA Section 4.3.7 states that, using plant specific design fatigue results, the applicant identified the plant-specific components for the NUREG/CR-6260 sample locations and the SNGS locations that bound those of NUREG/CR-6260 sample locations (LRA Tables 4.3.7-1 and 4.3.7-2, for SNGS Units 1 and 2, respectively). Further, the applicant performed environmentally-assisted fatigue (EAF) calculations for these SNGS locations to evaluate the effects of the reactor coolant system environment on fatigue life. However, the LRA does not provide sufficient information on the methodology used in determining those SNGS locations that bound those of NUREG/CR-6260 sample locations and the basis for performing EAF calculations for these locations in place of EAF calculations for identified NUREG/CR-6260 plant-specific components.

Request (1):

Explain the methodology used in determining the plant-specific, limiting locations within the boundary of the applicable NUREG/CR-6260 component locations.

Issue (2):

LRA Section 4.3.7 does not provide sufficient information on the assumptions and the basis for assumptions used in the 60-year cumulative usage factor (CUF) calculations for the NUREG/CR-6260 sample locations.

Request (2):

Identify all assumptions used in the 60-year CUF calculations for the NUREG/CR-6260 sample locations. Provide the basis why the assumptions applied to the 60-year CUF calculations are considered to be capable of yielding sufficiently conservative CUFs for application to SNGS EAF calculations.

Issue (3):

LRA Section 4.3.7 does not provide sufficient information on the basis for assumptions used in the environmental fatigue multipliers (F_{en}) calculations for the NUREG/CR-6260 sample locations.

Request (3):

Identify all assumptions used (e.g., sulfur content, dissolved oxygen, temperature, strain rate) in the F_{en} calculations for the NUREG/CR-6260 sample locations. Provide your basis why the assumptions applied to the F_{en} calculations are considered to be capable of yielding sufficiently conservative F_{en} factors for application to SNGS EAF calculations.

Issue (4):

LRA Section 4.3.7 does not indicate the material at each of the critical fatigue locations identified in the section.

Request (4):

Clarify whether any of the critical fatigue locations include nickel alloys. If so, identify the source and justification of the F_{en} formula used for the nickel alloy calculations.

Issue (5):

The locations identified and analyzed in NUREG/CR-6260 include typical limiting locations but do not consider all plant-specific components and configurations.

Request (5):

Clarify whether any other plant-specific locations at Salem are more limiting than those identified in NUREG/CR-6260. If other Salem plant-specific locations exceed those from NUREG/CR-6260, provide EAF calculations for those locations.

RAI 4.3-6

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue:

LRA Section 4.3.7 states that the fatigue analyses for the NUREG/CR-6260 sample locations have been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii). However, LRA Section B.3.1.1 states that the Metal Fatigue of Reactor Coolant Pressure Boundary Program will be enhanced to address the effects of the reactor coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant, as identified in NUREG/CR-6260. Therefore, it is not clear whether the applicant has chosen to use a more conservative approach and manage the effects of aging on the NUREG/CR-6260 sample locations intended functions in accordance with 10 CFR 54.21(c)(1)(iii) for the period of extended operation using the Metal Fatigue of Reactor Coolant Pressure Boundary Program.

Request:

Clarify how LRA AMP B.3.1.1 “Metal Fatigue of Reactor Coolant Pressure Boundary Program,” will address the effects of the reactor coolant environment on the critical components identified in NUREG/CR-6260, and whether the effects of aging on all/any NUREG/CR-6260 sample locations intended functions will be managed in accordance with 10 CFR 54.21(c)(1)(iii) for the period of extended operation using LRA AMP B3.1.1. If so, describe plans to clarify LRA Section 4.3.7 to indicate for which NUREG/CR-6260 sample locations the effects of aging will be managed in accordance with 10 CFR 54.21(c)(1)(iii).

RAI 4.4.2-1

Background:

LRA Section 4.4.2 discusses reactor coolant pump (RCP) flywheel fatigue crack growth analyses and states that Westinghouse Topical Report WCAP-14535A, “RCP Flywheel Inspection Elimination” includes a fatigue flaw growth analyses that has been identified as a time-limited aging analysis (TLAA). The LRA also states that the purpose of the report was to provide an engineering basis for elimination of RCP flywheel inservice inspection requirements for all operating Westinghouse plants and certain Babcock and Wilcox plants. The LRA concludes that RCP flywheels will maintain their structural integrity during the period of the extended operation because the maximum number of start-stop cycles projected for 60 years (e.g., 661 start-stop cycles for Unit 1 and 703 start-stop cycles for Unit 2) have been demonstrated to be bounded by the 6,000 start-stop cycles limit assumed in the WCAP-14535A fatigue flaw growth analysis.

Issue:

The U.S. Nuclear Regulatory Commission (NRC or the staff) endorsed WCAP-14535 in a safety evaluation (SE) dated September 12, 1996 (ADAMS Legacy Library Accession #9609230010). In the conclusion section of the SE (Section 4.0), the staff concluded that that the inspections of the flywheels should be performed even if all of the recommendations of Regulatory Guide (RG) 1.14, Revision 1, "Reactor Coolant Pump Flywheel Integrity" [August 1976], were met, and that the inspections of the RCP flywheels should not be completely eliminated. To be consistent with this SE position, the staff concluded that licensees should conduct either qualified ultrasonic testing or surfaces examinations of their flywheels once every ten years. It is not evident from the TLAA discussion whether the applicant intends to continue the inservice inspection (ISI) examinations of the RPC flywheels during the period of extended operation consistent with position taken in the staff's SE of September 12, 1996, or whether the applicant is proposing to discontinue the ISI examinations of the RCP flywheels during the period of extended operation.

Request:

Clarify whether the safety basis in the TLAA for the RCP flywheels is being used to justify elimination of the RCP flywheel examinations altogether, or whether the applicant intends to continue the ISI examinations of the RCP flywheels consistent the NRC's SE on WCAP-14535, dated September 12, 1996. If ISI examinations will be performed during the period of extended operation, clarify what type of examinations will be performed on the RCP flywheels during the period of extended operation and the frequency that will be used for the examinations. Justify the examination method and frequency that will be used for the ISI examinations during the period of extended operation. Otherwise, justify your basis for discontinuing the ISI examinations of the RCP flywheels if ISI examinations will be discontinued during the period of extended operation.

RAI 4.4.5-1

Background:

LRA Section 4.4.5 discusses Salem Unit 1 volume control tank (VCT) flaw growth analysis performed to address flaws that were identified in the circumferential lower head-to-shell weld of the Salem Unit 1 VCT during refueling outage 1RF13 (1999). The LRA also states that the analyses concluded that an initial flaw would grow an insignificant amount of only 1.1×10^{-5} inches, based on 1,000 pressurization cycles. The LRA further states that the major pressurization cycles (transients) experienced by the VCT would be Inadvertent Safety Injection events and Operating Basis Earthquake cycles, and to a lesser extent, Plant Heatups and Cooldowns. The LRA concludes that the VCT flaw growth analysis will remain valid during the period of extended operation because the maximum number of pressurization cycles projected for 60 years (e.g., 312 cycles) has been demonstrated to be bounded by the 1,000 pressurization cycles limit assumed in the Salem Unit 1 VCT flaw growth analysis.

Issue:

The LRA does not identify the methodology used to perform the analyses. It is not evident from the TLAA discussion whether ASME Code, Appendix A methodology or other similar industry standard was used to perform the analyses.

LRA Table 4.3.1-3 provides the design transients and 60-year projections for NSSS Class A and Class 1 components at Salem Unit 1. The staff noted that one of the upset condition transient is reactor trip from full power, which would SCRAM the reactor from full power and cause a full depressurization of the reactor coolant system. This design transient is not considered in the projected number of pressurization cycles that were used to conclude that the VCT crack growth analyses remained valid during the period of extended operation.

Request:

(a) Clarify which methodology was used to perform the Salem 1 VCT flaw growth analysis and whether the methodology has been approved for use by the NRC. Clarify which NRC document provides the approval of methodology. If the methodology has not been approved by the NRC, justify the bases for use of the analysis methodology and the rationale for choosing an acceptance criterion of 1000 pressurization cycles.

(b) Justify why the upset condition transient of "Reactor Trip from Full Power," which would SCRAM the reactor from full power and cause a full depressurization of the reactor coolant system, was not considered in the 60-year projection of pressurization cycles that were used to conclude that the VCT crack growth analyses remained valid during the period of extended operation.

RAI 4.4.5-2

Background:

LRA Section 4.4.5 states that flaws were identified in the shell to lower head weld of the Salem Unit 1 VCT during 1RF13 (1999). The LRA states further that the flaws found during the inspection were subsurface and not in contact with the environment, therefore, only fatigue would be the contributing mechanism to flaw growth.

Issue:

The applicant has identified that the flaw growth analysis for the Salem 1 VCT is a TLAA for the LRA. Two of the six criteria for defining an analysis as a TLAA in 10 CFR 54.3, is that the analysis must include the effects of aging of the intended function of the component, and that the analysis must be used in a safety basis decision. Since this analysis is TLAA, presumably there should be an applicable aging management review (AMR) item that is specific to management of fatigue flaw growth (i.e., crack growth due to fatigue) for the flaws in the VCT.

The staff has noted that LRA Table 3.3.2-2, "Chemical and Volume," only includes an applicable line item of management of cumulative fatigue damage, and do not include any AMR items of management of crack growth due to fatigue on the Salem Unit 1 VCT.

Request:

Justify why LRA Table 3.3.2-2 does not include any AMR line item for the Salem 1 VCT in a borated treated water environment with an aging effect of crack growth due to fatigue.