June 17, 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 (TAC NOS ME1834, ME 1836)

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

June 17, 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 (TAC NOS ME1834, ME 1836)

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

DISTRIBUTION: See next page

ADAMS Accession No. ML101390537

OFFICE	PM:DLR:RPB1	LA:DLR	BC:DLR:RPB1	PM:DLR:RPB1
NAME	B. Brady	S. Figueroa	B. Pham	B. Brady
DATE	06/14/10	06/07/10	06/16/10	06/17/10

OFFICIAL RECORD COPY

Letter to T. Joyce from B. Brady dated June 17, 2010

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR SALEM NUCLEAR GENERATING STATION UNITS 1 AND 2 LICENSE RENEWAL APPLICATION (TAC NO ME1834 / ME 1836)

DISTRIBUTION:

HARD COPY:
DLR RF

E-MAIL:
PUBLIC
RidsNrrDlr Resource
RidsNrrDlrRpb1 Resource
RidsNrrDlrRpb2 Resource
RdsNrrDlrRarb Resource
RidsNrrDlrRasb Resource
RidsNrrDlrRapb Resource
RidsOgcMailCenter Resource

BPham
BBrady
ACunanan
SCuadrado
CEccleston
REnnis
CSanders
BHarris, OGC
ABurritt, RI
RConte, RI
MModes, RI
DTifft, RI
NMcNamara, RI

Salem Nuclear Generating Station,
Units 1 and 2

cc:

Mr. Robert Braun
Senior Vice President Nuclear
PSEG Nuclear LLC
One Alloway Creek Neck Road
Hancocks Bridge, NJ 08038

Mr. Carl Fricker
Station Vice President - Salem
PSEG Nuclear LLC
One Alloway Creek Neck Road
Hancocks Bridge, NJ 08038

Mr. Michael Gallagher
Vice President – License Renewal Projects
Exelon Nuclear LLC
200 Exelon Way
Kennett Square, PA 19348

Mr. Ed Eilola
Plant Manager – Salem
PSEG Nuclear LLC
One Alloway Creek Neck Road
Hancocks Bridge, NJ 08038

Mr. Brian Booth
Director Nuclear Oversight
PSEG Nuclear
P.O. Box 236
Hancocks Bridge, NJ 08038

Mr. Jeffrie J. Keenan, Esquire
Manager - Licensing
PSEG Nuclear LLC
One Alloway Creek Neck Road
Hancocks Bridge, NJ 08038

Senior Resident Inspector
Salem Nuclear Generating Station
U.S. Nuclear Regulatory Commission
Drawer 0509
Hancocks Bridge, NJ 08038

Mr. Ali Fakhar
Manager, License Renewal
PSEG Nuclear LLC
One Alloway Creek Neck Road
Hancocks Bridge, NJ 08038

Mr. William Mattingly
Manager – Salem Regulatory Assurance
PSEG Nuclear LLC
One Alloway Creek Neck Road
Hancocks Bridge, NJ 08038

Township Clerk
Lower Alloways Creek Township
Municipal Building, P.O. Box 157
Hancocks Bridge, NJ 08038

Mr. Paul Bauldauf, P.E., Asst. Director
Radiation Protection Programs
NJ Department of Environmental
Protection and Energy, CN 415
Trenton, NJ 08625-0415

Mr. Brian Beam
Board of Public Utilities
2 Gateway Center, Tenth Floor
Newark, NJ 07102

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Gregory Sosson
Director Corporate Engineering
PSEG Nuclear LLC
One Alloway Creek Neck Road
Hancocks Bridge, NJ 08038

Mr. Paul Davison
Vice President, Operations Support
PSEG Nuclear LLC
One Alloway Creek Neck Road
Hancocks Bridge, NJ 08038

cc:

Ms. Christine Neely
Director – Regulator Affairs
PSEG Nuclear LLC
One Alloway Creek Neck Road
Hancocks Bridge, NJ 08038

Mr. Earl R. Gage
Salem County Administrator
Administration Building
94 Market Street
Salem, NJ 08079

REQUEST FOR ADDITIONAL INFORMATION
FOR SALEM NUCLEAR GENERATING STATION UNITS 1 AND 2
LICENSE RENEWAL APPLICATION (TAC NO ME1834 / ME 1836)

RAI 3.1.2.2.12-01

Background:

10 CFR § 54.21(a)(3) requires that applicants for license renewal demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. Thus, the component and the aging effect must be clearly identified and related to the aging management program that the applicant is proposing to manage the aging effect.

Issue:

License renewal application (LRA) Table 3.1.2-3, "Reactor Vessel Internals," does not distinguish the aging effect discussed in LRA Section 3.1.2.2.12, "Cracking Due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (IASCC)," from that in LRA Section 3.1.2.2.17, "Cracking Due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking." Specifically, Table 3.1.2-3 does not reference the aging effect due to primary water stress corrosion cracking at all. This could cause problems in the future in identifying the right components under the aging effect of either LRA Section 3.1.2.2.12 or LRA Section 3.1.2.2.17 if the industry programs that the applicant is committed to will give different inspection and evaluation guidelines for managing the aging effects of LRA Sections 3.1.2.2.12 and 3.1.2.2.17.

Request:

Provide a revised LRA Table 3.1.2-3 to identify the aging effect discussed in LRA Section 3.1.2.2.17, or justify combining LRA Sections 3.1.2.2.12 and 3.1.2.2.17 under the table column title "Aging Effect Requiring Management" in LRA Table 3.1.2-3.

RAI 3.3.2.2-01

Background:

SRP-LR paragraph 3.3.2.2.4.1 addresses cracking due to stress corrosion cracking and cyclic loading for non-regenerative heat exchanger components exposed to treated borated water greater than 60°C. The Generic Aging Lessons Learned (GALL) Report recommends the Water Chemistry Program to manage cracking and a plant-specific program to verify the effectiveness of the Water Chemistry Program where an acceptable program would include radioactivity monitoring and eddy current testing of tubes.

ENCLOSURE

Issue:

The staff noted the applicant relies on continuous monitoring for radioactivity on the shell side of the non-regenerative stainless steel heat exchangers to detect cracking due to stress corrosion cracking and cyclic loading. The staff noted that monitoring of radioactivity on the shell side will not detect cracking until cracking has progressed through the wall of the component whereas the GALL Report recommends eddy current examination which would detect cracking before leakage.

Request:

Justify how not using the recommended eddy current testing technique would detect cracking before leakage.

RAI 3.3.2.2-02

Background:

SRP-LR paragraph 3.3.2.2.4.1 addresses cracking due to stress corrosion cracking and cyclic loading for regenerative heat exchanger components exposed to treated borated water greater than 60°C. The GALL Report recommends the Water Chemistry Program to manage cracking and a plant-specific program to verify the effectiveness of the Water Chemistry Program.

Issue:

The staff noted that the applicant is using the One-Time Inspection Program in lieu of a plant-specific program where periodic inspections are to be scheduled and note A is assigned to the One-Time Inspection for regenerative heat exchangers.

Request:

Provide justification for using one-time inspection in lieu of periodic inspections of a plant-specific program and the use of Note A instead of Note E when applying the One-Time Inspection Program to verify the effectiveness of the Water Chemistry Program.

RAI 3.3.2.12-01

Background:

LRA Table 3.3.2-12, pages 3.3-241 and 242, states that gray cast iron (retarding chamber) tanks exposed to raw water have an aging effect of loss of material due to general, pitting, crevice, galvanic, and microbiologically influenced corrosion, and fouling; and that the aging effects will be managed by the Fire Water System Program. The aging management review (AMR) line items reference Table 1 line item 3.3.1-68 and GALL Report item VII.G-24, which is for piping, piping components, and piping elements, and cite generic note C, indicating that the component is different, but consistent with NUREG-1801 item for material, environment, and aging effect.

Issue:

The staff reviewed GALL AMP XI.M27, Fire Water System and noted that the aging management program (AMP) recommends wall thickness evaluations of fire protection piping be performed using a non-intrusive technique (e.g., volumetric testing) to identify loss of material due to corrosion. Based on a telephone conversation with the applicant on May 25, 2010, the applicant clarified that the retarding chamber is a vertical pipe installed in horizontal piping system and acts like a tank. The staff determines that this vertical pipe is a low point in the system and will have stagnant water and will be susceptible to loss of material due to general, pitting and crevice corrosion. It is not clear from a review of LRA Section B.2.1.16, Fire Water System Program, whether volumetric inspection to detect loss of material due to corrosion will be performed on the internal surfaces (specifically the bottom) of the fire water and the retarding chamber tanks.

Request:

Clarify if the (retarding chamber) tanks are included in the sample of fire protection system components that will be volumetrically inspected for wall thickness evaluation to detect loss of material prior to loss of intended function. If not included, please justify how loss of material due to general, pitting, crevice, galvanic, and microbiologically influenced corrosion, and fouling will be detected for these tanks by the Fire Water System Program.

RAI 3.4.1- 01

Background:

The SRP-LR, Table 3.4-1, item 16, states that stainless steel piping, piping components, piping elements, tanks, and heat exchanger components exposed to treated water are subject to loss of material due to pitting and crevice corrosion. The GALL Report, under item VIII.B1-4, recommends managing the aging effect using the Water Chemistry and One-Time Inspection Programs.

Issue:

The Salem Nuclear Generating Station LRA Table 3.1.2-4 indicates that stainless steel steam generator tube support plates exposed to treated water greater than 140°F can undergo loss of material due to pitting and crevice corrosion, and the Steam Generator Tube Integrity and Water Chemistry Programs will be used to manage this aging effect. The AMR line items cite generic note E, indicating that they are consistent with the GALL Report item for material, environment and aging effect, but a different aging management program is credited. The LRA references Table 3.4.1, item 3.4.1-16 and GALL Report item VIII.B1-4 for consistency with the GALL Report. However, GALL Report item VIII.B1-4 recommends managing the aging effect using the Water Chemistry Program augmented by the One-Time Inspection Program to verify the effectiveness of the chemistry control program.

Request:

Clarify how the AMR line items in LRA Table 3.1.2-4 for tube support plates loss of material due to pitting and crevice corrosion are consistent with the GALL Report item VIII.B1-4 for piping, piping components, and piping elements and provide a basis for the adequacy of the Steam Generator Tube Integrity Program to verify the effectiveness of the Water Chemistry Program for these components.

RAI 3.5.2.2.1.7-01

Background:

The GALL Report, under items II.A3-2 and II.B4-2, indicates that stress corrosion cracking can occur for stainless steel penetration sleeves, penetration bellows, and associated welds exposed to air – indoor uncontrolled or air - outdoor. For stress corrosion cracking to occur, a susceptible material must be subjected to a tensile stress and a corrosive environment. The GALL Report indicates that these three conditions can exist and cause stress corrosion cracking for stainless steel penetration sleeves, bellows and welds.

Issue:

In terms of the aging management of stainless steel penetration sleeves, penetration bellows and associated welds, LRA Table 3.5.1, item 3.5.1-10 states “stress corrosion cracking will not occur at these components, within the scope of the license renewal, because the normal stress and environmental exposure conditions are not conducive to the development of stress corrosion cracking.”

However, LRA Table 3.5.2-3 on page 3.5-179 addresses loss of material due to pitting and crevice corrosion in the stainless steel penetration sleeves (cap plates) exposed to air with steam or water leakage. The applicant credited the 10 CFR Part 50, Appendix J Program and the ASME Section XI, Subsection IWE Program to manage loss of material. LRA Note 3 on page 3.5-187 states “air with steam or water leakage environment is applicable to local areas inside the containment that are exposed to potential service water leakage or spray.” LRA Note 3 on page 3.5-187 also states “plant operating experience showed that metal components in this environment exhibit aging effects observed in Air-Outdoor environment.” Therefore, the aging management review results of the applicant are in potential conflict with the applicant’s claim that the normal environmental conditions are not conducive to the development of stress corrosion cracking.

LRA Section 3.5.2.2.1.7 further states “the containment pressure boundary welds between stainless steel piping and penetration sleeves, with normal operating temperatures above 140°F, are not highly stressed.” However, the LRA does not provide a detailed technical basis for the applicant’s claim that the penetration sleeves are not highly stressed so that the normal stress conditions are not conducive to the development of stress corrosion cracking. The LRA does not provide the information on how the applicant evaluated the residual stresses in the welds and adjacent material.

<u>Request:</u>

1. Describe detailed operating experience in terms of the observation of pitting and crevice corrosion and stress corrosion cracking in the penetrations (penetration sleeves, bellows and welds) and determine whether the operating experience supports the applicant's claim that the normal stress and environmental exposure conditions are not conducive to the development of stress corrosion cracking in these components.

2. Clarify why the applicant claims that the environmental condition is not conducive to the development of stress corrosion cracking in the containment penetrations although (1) the penetration sleeves are exposed to air with steam or water leakage due to potential service water leakage or spray, (2) the applicant manages loss of material due to pitting and crevice corrosion in the penetration sleeves exposed to the environment, and (3) plant operating experience showed that metal components in this environment exhibit aging effects observed in Air-Outdoor environment (LRA Page 3.5-187, Note 3).

3. Describe how the applicant determined that the welds between the stainless steel piping and penetration sleeves are not highly stressed. In addition, clarify whether the stress evaluation includes residual stresses and clarify whether the condition including residual stresses is not conducive to the development of stress corrosion cracking in the containment penetrations.

4. Based on the information provided for Requests 1 through 3, justify why stress corrosion cracking is not applicable to the stainless steel penetrations (penetration sleeves, penetration bellows and associated welds) that are addressed in LRA Table 3.5.1, item 3.5.1-10.