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

December 6, 2013

Mr. Joe W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
P.O. Box 2000
Soddy-Daisy, TN 37384

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION – SET 18 (TAC NOS. MF0481 AND MF0482)

Dear Mr. Shea:

By letter dated January 7, 2013, Tennessee Valley Authority submitted an application pursuant to Title 10 of the *Code of Federal Regulations* (CFR) Part 54, to renew the operating license DPR-77 and DPR-79 for Sequoyah Nuclear Plant, Units 1 and 2, for review by the U.S. Nuclear Regulatory Commission 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, outlined in the enclosure were discussed with Henry Lee, 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 301-415-1427 or by e-mail at Richard.Plasse@nrc.gov.

Sincerely,

EMMANUEL SAYOC FOR

Richard A. Plasse, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosure:
Requests for Additional Information

cc w/encl: Listserv

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/RA by Emmanuel Sayoc for/

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ADAMS Accession No.: ML13323A097

OFFICE	LA:RPB1:DLR	PM:RPB1:DLR	BC:RPB1:DLR	PM: RPB1:DLR
NAME	I King	E Sayoc	Y Diaz-Sanabria	R Plasse
DATE	11/19/2013	12/5/2013	12/5/2013	12/6/2013

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SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION
REQUESTS FOR ADDITIONAL INFORMATION

RAI B.1.34-9

Background:

The applicant's Reactor Vessel Internals Program implements the guidance of Materials Reliability Program (MRP)-227-A to manage the aging effects of reactor vessel internals (RVI) components.

Applicant/Licensee Action Item No.1 of MRP-227-A states that each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the failure modes, effects and criticality analysis and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. The applicant provided its response to Applicant/Licensee Action Item No.1 in license renewal application Appendix C.

Issue:

The staff noted that the applicant's response to Applicant/Licensee Action Item No.1 did not adequately address the three key variables at the applicant's site that feed into the screening process for aging degradation (stress, neutron fluence, and temperature) nor determine how these variations, if any, would ultimately affect the aging management recommendations.

The staff's concern was addressed generically with the industry as documented in the following documents: Meeting Summary EPRI-Westinghouse January 22-23, 2013 (ADAMS Accession No. ML13042A048) and Summary of Telecom with EPRI and Westinghouse Electric Company on February 25, 2013 (ADAMS Accession No. ML13067A262).

The staff also noted that by letter dated October 14, 2013, the Materials Reliability Program issued EPRI Letter: MRP 2013-025. The staff noted that the purpose of this letter was to provide an MRP-227-A related guidance document for MRP members to use in developing reactor internals related information for plant-specific inspection programs. Specifically, the enclosure was developed to provide utilities with the basis for a plant to respond to the NRC's request for additional information to demonstrate compliance with the basic technical applicability assumptions in MRP-227-A for originally licensed and uprated conditions.

Request:

1. Cold-worked Materials - Does the plant have non-weld or bolting austenitic stainless steel (SS) components with 20 percent cold work or greater, and if so, do the affected components have operating stresses greater than 30 ksi? (If both conditions are true, additional components may need to be screened in for stress corrosion cracking.)

2. Fuel Design or Fuel Management - Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative for that plant?

RAI B.1.23-2e

Background:

By letter dated November 15, 2013, the applicant responded to RAI B.1.23-2d which addressed the need for an inspection program to manage loss of material and cracking for control rod drive mechanism (CRDM) nozzle thermal sleeves. In its response, the applicant identified the Inservice Inspection Program to manage these aging effects. The applicant also stated that the CRDM thermal sleeve inspections are performed at the same frequency as the reactor vessel head volumetric examinations, in accordance with ASME Code Case N-729-1.

In addition, the applicant revised the Update Final Safety Analysis Report (UFSAR) supplement for the Inservice Inspection Program by adding the following:

> Revise the Inservice Inspection Program procedures to perform an augmented visual inspection of the Unit 1 and Unit 2 CRDM thermal sleeves and a wall thickness measurement of the six thermal sleeves exhibiting the greatest amount of wear. The results of the augmented inspection should be used to project if there is sufficient wall thickness for the period of extended operation, or until the next inspection.

Issue:

The applicant identified an augmented visual inspection and a wall thickness measurement (i.e., volumetric examination) to manage loss of material and cracking for the CRDM nozzle thermal sleeves. However, the applicant's response does not clearly describe whether the augmented visual inspection is periodic inspections at the same frequency as the volumetric examination of ASME Code Case N-729-1 or a one-time inspection. In addition, the applicant's response does not clearly describe whether thickness measurements will be performed on the six thermal sleeves exhibiting the greatest wear at each unit (i.e., thickness measurements of six thermal sleeves in each unit).

Request:

1. Clarify whether the augmented visual inspection is periodic inspections at the same frequency as the volumetric examination of ASME Code Case N-729-1 or a one-time inspection. If the augmented visual inspection is a one-time inspection, provide additional information which demonstrates the adequacy of a one-time visual inspection to manage loss of material and cracking for these thermal sleeves.

2. Clarify whether thickness measurements will be performed on the six thermal sleeves exhibiting the greatest wear in each unit. If thickness measurements are performed on a total of six thermal sleeves for Units 1 and 2, provide additional information which demonstrates the adequacy of the inspection scope (i.e., total six thermal sleeves for Units 1 and 2) to manage loss of material and cracking for these thermal sleeves.

Letter to J. Shea from R. Plasse dated December 6, 2013

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