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

March 27, 2013

Mr. William R. Gideon, Vice President
Carolina Power & Light Company
H.B. Robinson Steam Electric Plant, Unit 2
3581 West Entrance Road
Hartsville, South Carolina 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE PRESSURIZED WATER REACTORS INTERNALS PROGRAM PLAN FOR AGING MANAGEMENT OF REACTOR INTERNALS (TAC NO. ME9633)

Dear Mr. Gideon:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated September 26, 2012 (Agencywide Documents Access and Management System Accession No. ML12278A398), Carolina Power & Light Company, submitted an aging management program for the reactor vessel internals for H. B. Robinson Steam Electric Plant, Unit No. 2.

The NRC staff is reviewing your submittal and has determined that additional information is required to complete the review. The specific information requested is addressed in the enclosure to this letter. During a discussion with your staff on March 19, 2013, it was agreed that you would provide a response 60 days from the date of this letter.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources.

Please contact me at (301) 415-3302 if you have any questions.

Sincerely,

Fredrick E. Saltz

for Araceli Billoch Colón, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosure:
Request for Additional Information

cc w/encl: Distribution via ListServ

REQUEST FOR ADDITIONAL INFORMATION

REGARDING H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2

PRESSURIZED WATER REACTORS INTERNALS PROGRAM PLAN

FOR AGING MANAGEMENT OF REACTOR INTERNALS

DOCKET NO. 50-261

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated September 26, 2012 (Agencywide Documents Access and Management System Accession No. ML12278A398), Carolina Power & Light Company (the licensee), submitted an aging management program (AMP) for the reactor vessel internals (RVI) for H. B. Robinson Steam Electric Plant, Unit No. 2 (Robinson).

The Materials Reliability Program (MRP)-227-A report, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," and its supporting reports were used as technical bases for developing Robinson's AMP. The staff reviewed this report and issued a final safety evaluation (SE) on December 16, 2011. Based on the review of Robinson's AMP conducted thus far, the staff has developed a first request for additional information (RAI) as addressed below. The staff may however, issue additional RAIs based on the resolution of Action Items 1 and 2 addressed in the staff's SE for the MRP-227-A report.

RAI-1: Historically, the following materials used in the PWR RVI components were known to be susceptible to some of the aging degradation mechanisms that are identified in the MRP-227-A report. In this context, the NRC staff requests that the licensee confirm that these materials are not currently used in the RVI components at Robinson:

- Nickel base alloys—Inconel 600; Weld Metals—Alloy 82 and 182 and Alloy X-750 (excluding control rod guide tube split pins)
- Alloy A-286 ASTM A 453 Grade 660, Condition A or B
- Stainless steel type 347 material (excluding baffle-former bolts)
- Precipitation hardened stainless steel materials: 17-4 and 15-5
- Type 431 stainless steel material

RAI-2: Condition 7 of the NRC staff's SE Revision 1, dated December 16, 2011, stipulates that the licensee shall include a summary of the operating experience related to the aging degradation in the RVI components. The NRC staff requests that the licensee provide information regarding the extent of aging degradation (if any) that occurred thus far in all of the

RVI components specifically, include the operating history of the following components at Robinson:

- Baffle-former bolts, baffle-edge bolts, baffle-former assembly, clevis insert bolts, core barrel bolting, and thermal shields.

Provide a summary that includes a list of RVI components that have been inspected thus far, under the American Society of Mechanical Engineers Code, Section XI Inservice Inspection program and the inspection results. This list need not include any RVI component categorized under the "Existing" inspection category in the MRP-227-A report.

RAI-3: According to Section A.1.4 in MRP-175, "Materials Reliability Program: PWR Internal Aging Degradation Mechanism Screening Threshold Values," susceptibility to stress corrosion cracking (SCC) in nickel-based Alloy X-750 PWR RVI components depends on the type of heat treatment that is performed on the alloy. High temperature heat treatment (HTH) processes that are used on Alloy X-750 components offer better resistance to SCC than the other age hardened heat treatment processes. Licensee determination of the heat treatment applied to its Alloy X-750 PWR RVI components would appear to be a critical parameter in ensuring the licensee's AMP will adequately manage the potential effects of aging. Additionally, Appendix A of the MRP-227-A report, addressed as a part of its operational experience, that Alloy X-750 used for the clevis insert bolt assembly in one unit failed due to pressurized water SCC (PWSCC). Therefore, the staff requests that the licensee provide information related to the type of heat treatment process that was used for the Alloy X-750 clevis insert bolting at Robinson. If the existing clevis insert bolts at Robinson did not undergo an HTH process, the staff recommends that the licensee inspect these bolts (in addition to the inspections to monitor aging due to wear) for identifying PWSCC.

RAI-4: In Appendix C, Table C-1 of the licensee's submittal dated September 26, 2012; the licensee indicated that the control rod guide tube cards are to be inspected no later than two refueling outages from the beginning of the period of extended operation. This submittal stated that during the spring 2010 refueling outage the licensee performed inspections on control rod guide cards to assess the wear of these cards. The staff requests that the licensee provide the following information—(1) the number of cards inspected; (2) the inspection results; (3) how the criteria for maximum allowed wear was established; (4) licensee's corrective actions, if any; and (5) the licensee's plan for subsequent inspections of this component during the extended period of operation.

RAI-5 (a): As discussed in Section 3.3.7 of the staff's SE for the MRP-227-A, Action Item 7 states that licensees of Westinghouse-designed reactors are required to develop plant-specific analyses to be applied to its facilities to demonstrate that lower support column cast austenitic stainless steel bodies will maintain their function during the extended period of operation. This component is subject to neutron embrittlement and irradiation-assisted SCC (IASCC). Supplemental inspection would be recommended for those components that are potentially susceptible to neutron embrittlement, and are subject to significant tensile loadings under any normal operating or design basis condition. Neutron embrittlement and IASCC become active aging degradation mechanisms when the fluence values exceed the threshold limit addressed in Table 4-6 of MRP-191 Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs." The fluence value for the lower support column bodies in Westinghouse-designed reactors is

significantly greater than the $1x10_{17}$ n/cm_2 threshold value provided in the Generic Aging Lessons Learned report. Therefore, the staff requests that the licensee submit an analysis taking into account the aforementioned aging effects. This analysis should consider the effects of transient loading on the structural integrity of the lower support column bodies.

RAI-5 (b): In Section 6.2.7 of the September 26, 2012, submittal, the licensee stated that the following components were evaluated for their susceptibility to thermal embrittlement based on the ferrite and Molybdenum contents and the casting process used. The staff requests that the licensee provide the methodology that was used to perform this evaluation (e.g., usage of an existing certified material test report). The components include: flow mixer devices; upper support column bases, with and without flow mixers; lower support columns; and bottom mounted instrumentation cruciforms, butt, and special columns.

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Araceli Billoch Colón, Project Manager
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ADAMS Accession No.: ML13079A293 ***Memo Dated March 15, 2013**

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