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U. S. Nuclear Regulatory Commission
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Washington, D. C. 20555-0001

Vogtle Electric Generating Plant – Unit 1
Response to Request for Additional Information Regarding Reactor Pressure Vessel
Attachments B-N-2 and B-N-3 Weld Examinations

Ladies and Gentlemen:

By application dated August 4, 2016 (ADAMS Accession No. ML16217A428), Southern Nuclear Operating Company (SNC) submitted request for Alternative VEGP-ISI-ALT-12 for the Vogtle Electric Generating Plant (VEGP), Unit 1. Request for Alternative VEGP-ISI-ALT-12 proposes to defer the Unit 1 Category B-N-2 and B-N-3 examinations from the spring 2017 refueling outage until the following fall 2018 refueling outage. By letter dated October 12, 2016, the Nuclear Regulatory Commission (NRC) staff issued a request for additional information. The Enclosure provides the SNC response to the NRC RAI.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at 205.992.7369.

Respectfully submitted,

C. R Pierce

C. R. Pierce
Regulatory Affairs Director

CRP/RMJ

Enclosure: SNC Response to NRC RAI

cc: Southern Nuclear Operating Company
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Vogtle Electric Generating Plant – Unit 1
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Vessel Attachments B-N-2 and B-N-3 Weld Examinations

Enclosure

SNC Response to NRC RAI

RAI-1:

Request for Alternative VEGP-ISI-ALT-12 proposes to defer the Unit 1 Category B-N-2 and B-N-3 examinations from the spring 2017 refueling outage until the following fall 2018 refueling outage. This request is in accordance with the provisions of Title 10 of the *Code of Federal Regulations* 50.55a(z)(2) (i.e., hardship without a compensating increase in quality and safety). Although information provided supports that performing the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI required inspection in 2017 may not have a compensating increase in quality and safety, more information is needed for the NRC staff to make a regulatory decision. Please provide information to support the request based on hardship, including the operational challenges to complete this work during the spring 2017 refueling outage as compared to the following refueling outage in the fall of 2018. Please include details in your response on the potential for schedule impacts and the basis for the as low as reasonable achievable hardship.

SNC Response to RAI-1:

As discussed in request for Alternative VEGP-ISI-ALT-12, flux thimble tube replacement and core barrel removal are currently planned for the spring 2017 refueling outage 1R20. Vogtle Electric Generating Plant (VEGP) will also be performing eddy current examinations of all four steam generators (SGs) during 1R20. In order to perform the SG eddy current examinations, SG nozzle dams must be installed. Nozzle dams are installed at the beginning of the defueled window and will remain in place throughout the duration of SG eddy current testing. Performing outage activities with the SG nozzle dams installed represents an off-normal configuration and results in accrual of incremental risk each day they are installed. SNC strives to maintain outage dose and contamination to as low as reasonably achievable. If the SG nozzle dams were to fail with the reactor core barrel removed, there would be potential for significant personnel exposure and significant contamination. The significant personnel exposure would be from the loss of the core barrel water shielding. The significant contamination would be from the highly contaminated refueling water draining to the lower levels of containment. VEGP strives to minimize the amount of time spent in off-normal configurations. However, attempting to rearrange the outage schedule such that these major outage activities are performed in a "normal" configuration is not practical. Simultaneous performance of flux thimble tube replacement and core barrel removal is not possible due to space limitations in the refueling cavity. Completing all eddy current testing and restoring the SG manways before performing the category B-N-2 and B-N-3 visual examinations would extend outage critical path duration by more than six days.

Of the major outage activities currently planned for 1R20, flux thimble tube replacement and eddy current testing represent a more significant safety impact. Installation of the replacement in-core thimble tubes will ensure movable in-core instrumentation remains operable as required by the Technical Specifications (TS). By performing eddy current examinations on all four SGs during 1R20, no SG eddy current examinations will be required for 1R21. This will allow major outage activities for 1R21 to be performed without SG nozzle dams installed (i.e. in a "normal" configuration).

Splitting these major project activities will provide a safety benefit by allowing the core barrel removal to be performed in a "normal" configuration during 1R21. Performing this activity during 1R20 induces a hardship without a commensurate level of safety benefit. Deferring B-N-2 and B-N-3 examinations helps minimize time the plant must be in an off-normal configuration between the two refueling outages (1R20 and 1R21). Therefore, SNC proposes an extension of

the Third Inspection Interval which would allow the category B-N-2 and B-N-3 visual examinations to be performed during Refueling Outage 1R21 scheduled to start in September of 2018.

RAI-2:

Historically, the following materials used in pressurized water reactor (PWR) vessel attachments were known to be susceptible to primary water stress corrosion cracking (i.e., nickel based alloys such as Inconel 600 and weld metals such as Alloy 82 and Alloy 182). In some PWRs, the vessel attachment welds were made with Alloy 82 and/or Alloy 182 weld metal. Please discuss whether, at the VEGP units, the vessel attachment welds were fabricated with the materials discussed above or with stainless steel weld metal.

In addition, for some PWR reactor vessel internals, the lower support clevis cap screws are made from Alloy X-750 (without high temperature heat treatment), which is susceptible to stress corrosion cracking. Please provide information on the type of material used for lower support clevis cap screws at the VEGP units.

SNC Response to RAI-2:

The VEGP Unit 1 reactor contains six core support lugs fabricated from Alloy 600 base material which are attached to the RPV wall with alloy 82/132/182 welds. The inlay pad is alloy 82 material, the attachment weld is 132/82, and a lug tie-in weld is alloy 182. The finished lugs and attachment welds received post-weld heat treatment with the entire reactor vessel at the fabricator (Combustion Engineering), which relieves the stresses from the welding operation. SNC is unaware of any industry operating experiencing (OE) related to primary water stress corrosion cracking for similar vessel attachment welds.

The VEGP Unit 1 six core support lug locations described above have one-piece Alloy 600 (SB-166) clevis inserts interference fit and fastened to the lugs with eight SB-637 Grade 688, type 2 (alloy X-750) cap screws and two alloy 600 dowel pins.