Southern Nuclear

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant – Units 1&2
Response to NRC Request for Additional Information for Alternative HNP-ISI-ALT-05-05

Ladies and Gentlemen:

By letter dated June 5, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17156A831), Southern Nuclear Operating Company (SNC) submitted alternative request number HNP-ISI-ALT-05-05, in accordance with Paragraph 50.55a(z)(1) of Title 10 of the Code of Federal Regulations (10 CFR) for a proposed alternative to the requirements of 10 CFR 50.55a, "Codes and standards," for Edwin I. Hatch Nuclear Plant (HNP), Units 1 and 2. Specifically, the licensee proposes an alternative to the inservice inspection requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code applicable to the reactor pressure vessel nozzle-to-vessel shell welds and nozzle inner radius, based on ASME Code Case N-702, "Alternative Requirements for Boiling Water Reactor Nozzle Inner Radius and Nozzle-to-Shell Welds." On September 9, 2017, the Nuclear Regulatory Commission (NRC) staff notified SNC that additional information is needed for the staff to complete their review. The Enclosure provides the SNC response to the NRC requests for additional information.

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

Respectfully submitted,

J. J. Hutto
Regulatory Affairs Director

A047
NRR

JJH/PDB/CBG

Enclosure: SNC Response to NRC Request for Additional Information (RAIs)

Cc: Regional Administrator, Region II
NRR Project Manager – Hatch
Senior Resident Inspector – Hatch
RTYPE: CHA02.004

Edwin I. Hatch Nuclear Plant Unit 1 and 2

Enclosure

SNC Response to NRC Request for Additional Information (RAIs)

NRC RAI 1

One of the technical basis report for ASME Code Case N-702 is BWR Vessel and Internals Project (BWRVIP)-108, "Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," EPRI Technical Report 1003557, October 2002. In its safety evaluation of BWRVIP-108, dated December 19, 2007 (ADAMS Accession No. ML073600374), the NRC staff documented the BWRVIP's supplemental probabilistic fracture mechanics (PFM) evaluation that showed that the limiting probability of failure (PoF) is 1.98×10^{-6} per year for normal operation, compared to 1.19×10^{-7} per year for a low temperature over-pressure (LTOP) event. However, the PFM analysis in Attachment 1 to Enclosure 1 of the licensee's submittal reported PoF values only for the LTOP event. The NRC staff requests the licensee to report the PoF values for normal operation or discuss how the PoF values for LTOP are more limiting than those for normal operation.

SNC Response to RAI 1

Only the PoF for the LTOP event was calculated by SIA as reported in Attachment 1 to Enclosure 1. Hatch Unit 1 and 2 satisfied the criteria for demonstrating compliance to the conditions imposed by the safety evaluation report for BWRVIP-108. As stated in the License Renewal Appendix for BWRVP-241, the effect of neutron fluence on material fracture toughness is the only aging effect for the nozzles that requires aging management review for the license renewal period. Fast (>1MeV) neutron fluence may exceed 1 X 10^{17} n/cm^2 in the upper portion of the recirculation nozzle-to-shell welds near the end of the 60-year period of extended operation. SNC therefore instructed SIA to evaluate the potential degradation effect from neutron fluence. The LTOP scenario is the appropriate event to assess the possibility of RPV brittle fracture failures and therefore was the only event considered in the SIA calculations. The BWRVIP-05 and BWRVIP-108 reports only evaluate PoF due to LTOP events to address the possibility of RPV brittle fracture failures. The PoF limit of 5E-06 was taken from NUREG-1806 for pressurized thermal shocks and therefore used as the acceptance criteria to demonstrate that the Hatch Unit 1 and 2 inlet and outlet recirculation nozzles meet the acceptable failure probability for reduced examination percentage of 25% even with the elevated fluence values assumed.

NRC RAI 2

Section 3.1 "Fatigue Cycles" of Attachment 1 to Enclosure 1 of the licensee's submittal described the fatigue cycles used in the PFM analysis. The NRC staff requests the licensee to confirm the following: (1) the fatigue cycles of all other transients applicable to the HNP, Units 1 and 2 recirculation inlet and outlet nozzles were lumped into the selected bounding transients, and (2) fatigue crack growth in the PFM analysis was performed using the fatigue cycles projected to 60 years of operation.

SNC Response to RAI 2

(1) All transients applicable to HNP, Units 1 and 2 recirculation inlet and outlet nozzles were grouped into the selected bounding transients.

(2) The fatigue crack growth (FCG) cycle counts in Table 5 of Attachment 1 to Enclosure 1 are actual recorded cycles from SIA FatiguePro monitoring calculations for Hatch Units 1

and 2 (Reference 18 in Attachment 1 of Enclosure 1), which are then extrapolated to 60 years.

NRC RAI 3

Section 4.2.2 "Fluence" of Attachment 1 to Enclosure 1 of the licensee's submittal indicated that since fluence values were not available for the HNP, Units 1 and 2 recirculation inlet and outlet nozzles, the peak fluence at the inside RPV surface from the closest RPV circumferential weld (C4) and the lower vessel shell were used in the PFM analysis. The NRC staff requests the licensee to confirm that RPV circumferential weld C4 and the lower vessel shell selected for fluence values are closer to the active core in the RPV axial direction than the HNP, Units 1 and 2 recirculation inlet and outlet nozzles.

SNC Response to RAI 3

The recirculation inlet and outlet nozzles are included in the fluence models for Hatch Unit 1 and 2 but the fast fluence values were not specifically calculated when HNP-ISI-ALT-05-05 was submitted. The lower shell course wholly contains the inlet and outlet nozzles and are at elevations shown below. As can be seen, the use of the peak fast fluence values from the C-4/lower shell course elevations of 253" and 247" are conservative.

Location	Unit 1 Elevation	Unit 2 Elevation
Top of Active Fuel	359"	359"
Bottom of Active Fuel	209"	209"
Circumferential Weld C-4	253"	247"
Lower Vessel Shell Course	103"-253"	97"-247"
Lower "beltline" Elevation at end of PEO	194"	193"
Recirc Inlet (N2) Nozzle Centerline	178.5"	178.5"
Recirc Outlet (N1) Nozzle Centerline	161.5"	161.5"

Note -Elevations given are in relation to the Vessel Elevation, starting at the bottom head where the elevation is zero inches, moving up to the top of the Reactor Vessel Head Vent (Nozzle 2N7) elevation of 852.7 inches.