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

April 8, 2010

Mr. Mark J. Ajluni
Manager, Nuclear Licensing
Southern Nuclear Operating Company, Inc
40 Inverness Center Parkway
Birmingham, Alabama 35201

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2, SAFETY EVALUATION OF RELIEF REQUEST HNP-ISI-ALT-09, VERSION 2.0, FOR THE FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL (TAC NOS. ME3327 AND ME3328)

Dear Mr. Ajluni:

By letter to the U.S. Nuclear Regulatory Commission (NRC), dated February 16, 2010, as supplemented March 29, 2010 (References 1 and 2, respectively), Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted request for relief HNP-ISI-ALT-09 from certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) at the Edwin I. Hatch Nuclear Plant, Unit 2. Specifically, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii), the licensee proposed an alternative to the pressure test requirements of ASME Code, Section XI, IWA-4540(c), for repair and replacement activities for mechanical joints made in the installation of pressure-retaining items. The licensee requested implementation of this alternative for activities during a mid-cycle maintenance outage scheduled for April 2010, during the fourth 10-year interval.

Based on the review of the information the licensee provided, the NRC staff concludes that the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety and the alternatives provide reasonable assurance of structural integrity. Therefore, the licensee's proposed alternative is authorized in accordance with 10 CFR 10 50.55a(a)(3)(ii) for the licensee's fourth 10-year inservice inspection interval. All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Sincerely,

Gloria Kulesa, Branch Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosure: Safety Evaluation

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO RELIEF REQUEST HNP-ISI-ALT-09

REGARDING PRESSURE TESTING OF MECHANICAL JOINTS

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NUMBERS 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated February 16, 2010 (Reference 1), Southern Nuclear Operating Company (SNC, the licensee), submitted Relief Request (RR) HNP-ISI-ALT-09 for Edwin I. Hatch, Unit 2 (HNP2) to use an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. By letter dated March 29, 2010 (Reference 2), SNC revised RR HNP-ISI-ALT-09 as Version 2 (HNP-ISI-ALT-09, Version 2) in response to the U.S. Nuclear Regulatory Commission's (NRC) request for additional information dated March 23, 2010. Specifically, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii), the licensee proposed an alternative to the pressure test requirements of ASME Code, Section XI, IWA-4540(c), for repair/replacement activities of mechanical joints made in the installation of pressure-retaining items. The licensee requested implementation of this alternative for activities during a mid-cycle maintenance outage scheduled for April 2010 during the fourth 10-year interval.

2.0 REGULATORY EVALUATION

NRC regulations in 10 CFR 50.55a(g) specify that inservice inspection (ISI) of nuclear power plant components shall be performed in accordance with the requirements of the ASME Code, Section XI, except where specific written relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(g)(6)(i) states that the NRC may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest, given the consideration of the burden upon the licensee. 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. 10 CFR 50.55a(g)(5)(iii) states that if the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the NRC and submit, as specified in §50.4, information to support the determinations.

Enclosure

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements of the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed in paragraph 50.55a(b)(2). The applicable code of record for the fourth ISI interval for HNP2 is the ASME Code, Section XI, 2001 Edition through the 2003 Addenda. The proposed alternative is sought for the April 2010, maintenance shutdown which is in the fourth 10-year ISI interval which ends on December 31, 2015.

Paragraph 55a(b)(2)(xxvi), *Pressure Testing Class 1, 2 and 3 Mechanical Joints,* places a limitation on licensees using the ASME Code, Section XI, 2001 Edition and later editions and addenda. This limitation requires that the repair replacement activity provisions of ASME Code, Section XI, IWA-4540(c) of the 1998 Edition of Section XI for pressure testing Class 1, 2, and 3 mechanical joints be applied. ASME Code, Section XI, IWA-4540(c), of the 1998 Edition requires that mechanical joints made in the installation of pressure-retaining items shall be pressure tested in accordance with ASME Code, Section XI, IWA-5211(a).

ASME Code, Section XI, IWA-5211(a), provides the description of a system leakage test to be done while the system is in operation, during a system operability test, or while the system is at test conditions using an external pressurization source. The test conditions are further described in ASME Code, Section XI, IWB-5221(a), as being conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Request for Alternative

ASME Code Components Affected

Class 1 pressure retaining mechanical joints.

ASME Code Requirement Affected

The ASME Code, Section XI, subarticle IWB-5221(a) requirement that a Class 1 system leakage test be conducted at a pressure not less than the pressure corresponding to 100 percent of rated reactor power.

Proposed Alternative

The licensee will perform a visual examination for leakage (VT-2) of any Class 1 mechanical joints, installed as repair/replacement activities during the April 2010, maintenance shutdown, at an reactor coolant system (RCS) pressure of greater than or equal to 920 psig during reactor startup at approximately 5 percent reactor power versus 1045 psig which is the pressure associated with 100 percent rated reactor power.

Basis for Hardship

HNP2 has scheduled a mid-cycle maintenance shutdown for April 2010. The activities scheduled to take place during this maintenance outage include replacement of two safety relief valve (SRV) pilots and an SRV main body and its associated pilot assembly. These activities would make it necessary to perform the system leakage test required by the provisions of ASME Code, Section XI, IWA-4540(c), in accordance with Paragraph 50.55a(b)(2)(xxvi). The licensee believes performing the system leakage test at 1045 psig would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety over performing the system leakage test at the proposed pressure of greater than or equal to 920 psig.

In References 1 and 2, the licensee describes the hardships related with performing the test at 1045 psig. These hardships would include personnel safety hardships related to the elevated dry well temperatures caused by the additional 14 hours at RCS temperature required to achieve the pressure increase from 920 psig to 1045 psig. During these 14 hours, the ambient temperature in the drywell could be expected to increase 13 degrees. These elevated temperatures would cause personnel hazards associated with heat stress as well as burn dangers from elevated component temperatures. The 14 hours to increase pressure from 920 psig to 1045 psig are required by the following operational activities and limitations:

- Control Rod Drive withdrawal limitations and the associated gradual increases in reactor power, pressure and temperature.
- Technical Specification required Pressure versus Temperature limitations.
- Main steam line piping, turbine control and stop valve warming.
- Main turbine warming.
- Small increases in pressure over time to provide better seating performance of the SRV's

Performance of a cold leakage test (non-nuclear heat-up), such as that done following a refueling outage to satisfy the ASME Code, Section XI, IWB-5221, requirements is not desirable following a maintenance outage for the following reasons:

- The Reactor Pressure Vessel and Reactor Coolant System Pressure Boundary are required to be virtually water solid.
- Main Steam Lines are flooded with the Main Steam Isolation Valves closed.
- Extensive valve manipulations, system lineups, and procedural controls are required in order to heat up and pressurize the RCS to establish the necessary test pressure.
- The additional valve lineups and system reconfigurations necessary to support the test would impose an additional challenge to the affected systems.
- These additional valve lineups and system manipulations would add extra radiation exposure to plant personnel.
- Performing a cold leakage test would add approximately two days to the shutdown duration.

Additionally, any leakage discovered during the 920 psig leakage test will be adjusted to account for the marginal increase in leakage rates that might occur at the higher 1045 psig pressure associated with 100 percent rated reactor power. This will ensure the leakage is properly dispositioned in accordance with ASME Code, Section XI, requirements.

The licensee states that drywell monitoring systems are available that would detect leakage that might occur in the mechanical joint connections at higher pressures associated with nominal reactor operation. These systems include drywell air temperature and pressure monitoring and the drywell floor and equipment drain sumps.

3.2 NRC Staff Evaluation

ASME Code, Section XI, IWA-4540, provides the pressure testing requirements for repair/replacement activities of Class 1, 2 and 3 items. The code of record for HNP2 specifies leakage or hydrostatic testing is required for repair/replacement activities performed by welding or brazing on a pressure-retaining boundary. This would not require mechanical joints to be leakage tested. However, paragraph 55a(b)(2)(xxvi) was incorporated into the regulations when incorporating the 2001 Edition through the 2003 Addenda of ASME Section XI because the NRC staff found that ASME Code pressure testing of mechanical joints after repair and replacement activities is still warranted.

In accordance with paragraph 50.55a(b)(2)(xxvi), ASME Code, Section XI, IWA-4540(c), of the 1998 Edition of Section XI, requires that mechanical joints made in the installation of pressure-retaining items be pressure tested during a system leakage test at nominal operating pressure. In the 1998 Edition of ASME Code, Section XI, IWB-5221(a) the test conditions are further described as being conducted at a pressure not less than nominal operating pressure associated with normal system operation. The 2001 Edition through the 2003 Addenda of ASME Code, Section XI, IWB-5221(a), describes a system leakage test as being conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power (i.e. 1045 psig).

In the Federal Register Notice (FRN)(Reference 3) which incorporated paragraph 50.55a(b)(2)(xxvi), the NRC staff explained the reasoning for this requirement. The requirements to pressure test Class 1, 2 and 3 mechanical joints undergoing repair and replacement activities were deleted in the 1999 Addenda of Section XI. There was no justification for eliminating the requirements for pressure testing Class 1, 2 and 3 mechanical joints and the NRC staff believed the pressure testing was necessary to ensure and verify the integrity of the pressure boundary. The FRN never discussed specific pressures of the test to be performed but focused on the need to perform a post repair and replacement pressure test and VT-2 examination to verify the integrity of the pressure boundary. The NRC staff believes the licensee's proposed alternative will verify the leak tightness and structural integrity of the mechanical joints involved.

As an alternative to the system pressure test at operating pressure (1045 psig), the licensee proposed to perform the system leakage test at a pressure of at least 920 psig at approximately 5 percent reactor power. The NRC staff believes that the proposed test pressure (approximately 90 percent of the pressure corresponding to 100 percent rated reactor power) will be sufficiently high to cause leakage from any mechanical connections following opening and reassembly of the item if the leak-tight connection has not been established. Leakage through the mechanical joint would be detectable at 920 psig with a slightly lower leakage rate than that at 1045 psig which is the ASME Code-required test pressure for the system leakage test. The licensee stated that disposition of any observed leakage at 920 psig will account for the marginal increase in leakage rates that might occur at 1045 psig.

The NRC staff agrees that performance of the VT-2 examination at the ASME Code-required pressure of 1045 psig low reactor power levels versus 920 psig would involve hardship from a personnel safety standpoint. The environmental conditions would require consideration of serious heat stress and valid burn hazard concerns. These conditions would also require additional special safety precautions such as ice vests and cool air supply lines. These adverse conditions and the additional burden of the safety precautions could impact the quality of the leakage examination due to the hardship imposed on the examination personnel. Performing the leakage test at 1045 psig during low power operations would also present operational challenges such as altering normal steam pressure controls, possible SRV seating issues and Control Rod Drive withdrawal limitations.

The NRC staff also agrees that performance of a cold leakage test (non-nuclear heatup), such as that required following a refueling outage would involve hardship. Performing this type test would require filling the main steam lines and the reactor vessel solid with water. In order to establish the necessary test conditions would require performing extensive temporary hanger modifications, valve lineups and system manipulations. All of these activities would require personnel radiation exposure in addition to normal startup activities.

Based on the above evaluation, the NRC staff finds that performing the alternative, a visual examination for leakage during a system leakage test at equal to or greater than 920 psig, provides reasonable assurance of the leakage and structural integrity of the items, and that compliance with the ASME Code-specified requirements would result in hardship without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff concludes that complying with the ASME Code requirement would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Furthermore, based on the above, the NRC staff determines that the proposed alternative described in RR HNP-ISI-ALT-09, Version 2, provides reasonable assurance of structural integrity and leak tightness of the subject components. Therefore, the NRC authorizes the proposed alternative in accordance with 10 CFR 50.55a(a)(3)(ii) for the April 2010 maintenance shutdown in the fourth 10-year inspection interval at HNP2.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

REFERENCES

1. Letter, M.J. Ajluni (SNC) To U.S. Nuclear Regulatory Commission containing ISI

Program Alternative HNP-ISI-ALT-09, Edwin I. Hatch Nuclear Plant, Units 1&2, February 16, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100480010).

2. Letter, M.J. Ajluni (SNC) To U.S. Nuclear Regulatory Commission containing ISI Program Alternative HNP-ISI-ALT-09, Edwin I. Hatch Nuclear Plant, Unit 2, March 29, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100890051).

3. 69 Fed Reg 58804, 58820 (October 1, 2004) Industry Codes and Standards; Amended Requirements.

Principal Contributor: Keith M. Hoffman, NRR/DCI

Date of issuance: April 8, 2010

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Sincerely,

/RA/

Gloria Kulesa, Branch Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosure: Safety Evaluation

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