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

May 16, 2016

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
P.O. Box 1295
Bin 038
Birmingham, AL 35201-1295

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2 – INSERVICE INSPECTION PROGRAM ALTERNATIVE FOR SAFETY RELIEF VALVES (CAC NO. MF7692)

Dear Mr. Pierce:

By letter dated May 6, 2016, Southern Nuclear Operating Company, Inc. (the licensee) submitted inservice testing alternative HNP-ISI-ALT-05-02 for the Edwin I. Hatch Nuclear Plant (HNP), Unit No. 2. The alternative requested U.S. Nuclear Regulatory Commission (NRC) approval to perform the VT-2 visual examination during a system leakage test of Class 1 components with mechanical joint connections at a pressure lower than the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, required pressure following the replacement of safety relief valves.

The NRC staff has determined that the proposed alternative provides reasonable assurance that the safety relief valves are operationally ready. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(z)(2), for the proposed alternative. Therefore, pursuant to 10 CFR 50.55a(z)(2), the NRC staff authorizes the use of the alternative for HNP, Unit No. 2, until the next refueling outage, currently scheduled to begin in February 2018.

All other ASME Code requirements for which relief or an alternative was not specifically requested and approved in the subject request remains applicable.

If you have any questions, please contact the Project Manager, Michael Orenak, at 301-415-3229 or Michael.Orenak@nrc.gov.

Sincerely,

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-366

Enclosure:
Safety Evaluation

cc w/enclosure: Distribution via Listserv

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSERVICE INSPECTION PROGRAM ALTERNATIVE HNP-ISI-ALT-05-02

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-366

1.0 INTRODUCTION

By letter dated May 6, 2016 (Agencywide Documents Access and Management System Accession No. ML16127A191), Southern Nuclear Operating Company, Inc. (the licensee) proposed alternative HNP-ISI-ALT-05-02, Version 1.0, to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the Edwin I. Hatch Nuclear Plant (HNP), Unit No. 2.

Specifically, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(ii), the licensee proposed an alternative to the pressure test requirements of ASME Code, Section XI, Subsection IWB-5221(a), for repair/replacement activities of mechanical joints made in the installation of pressure-retaining items. The licensee requested implementation of this alternative for activities that will be performed during a mid-cycle maintenance outage scheduled for May 2016 in the fifth 10-year inservice inspection (ISI) interval.

2.0 REGULATORY EVALUATION

The licensee requested authorization of an alternative to the requirements of Section XI, Subsection IWB-5221(a) of the ASME Code, pursuant to 10 CFR 50.55a(a)(3)(ii). On December 5, 2015, the U.S. Nuclear Regulatory Commission (NRC) reorganized 10 CFR 50.55a (79 FR 65776, November 5, 2014) such that relief requests that had been previously covered by 10 CFR 50.55a(a)(3)(i) are now covered under the equivalent 10 CFR 50.55a(z)(1), and relief requests previously covered by 10 CFR 50.55a(a)(3)(ii) are now covered under the equivalent 10 CFR 50.55a(z)(2).

The regulation at 10 CFR 50.55a(g)(4) states that ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of components.

The regulation at 10 CFR 50.55a(z) states, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if the licensee demonstrates that (1) the proposed alternatives would provide an acceptable level of quality and safety, or

(2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request the use of the alternative and the NRC to authorize the proposed alternative.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Components Affected

Class 1 pressure-retaining mechanical joint connections that require a system leakage test with associated VT-2 visual examination subsequent to repair/replacement activities.

3.2 Applicable ASME Code

ASME Section XI Code, 2001 Edition through 2003 Addenda.

3.3 ASME Code Requirement Affected

ASME Code, Section XI, Article IWA-4540, contains requirements for hydrostatic or system leakage tests for items that have undergone repair replacement activities.

The regulation at 10 CFR 50.55a(b)(2)(xxvi), "Section XI condition: Pressure testing Class 1, 2, and 3 mechanical joints," states, in part:

> The repair and replacement activity provisions in IWA-4540(c) of the 1998 Edition of Section XI for pressure testing Class 1, 2, and 3 mechanical joints must be applied when using the 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section.

ASME Code, Section XI, Article IWA-4540(c),1998 Edition states:

> Mechanical joints made in installation of pressure retaining items shall be pressure tested in accordance with IWA-5211(a). Mechanical joints for component connections, piping, tubing (except heat exchanger tubing), valves and fittings, NPS-1 and smaller, are exempt from the pressure test.

IWA-5211 (both 1998 and 2001 Editions) describes system leakage, hydrostatic, and pneumatic tests.

IWA-5212 (both 1998 and 2001 Editions) describes pressures and temperatures at which system leakage, hydrostatic, and pneumatic tests must be conducted. IWA-5212(a) refers to IWB-5000.

IWB-5221(a) (2001 edition) states, "The system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power."

3.4 Reason for Request

The licensee requested relief from the test pressure requirement of IWB-5221(a) (i.e., 1045 pounds per square inch gauge (psig)) of the 1998 Edition of ASME Code, Section XI, on the basis of hardship as cited below.

The licensee plans to replace some components installed via mechanical joints (e.g., safety relief valves (SRVs)) during a maintenance shutdown that is scheduled to begin May 20, 2016. The SRV replacement activities will require a system leakage test and associated VT-2 visual examination of the mechanical joint connections during HNP, Unit No. 2, startup.

The licensee stated that nominal operation pressure (i.e., 1045 psig) will not be reached for more than 24 hours after reaching 920 psig during the startup sequence because of the following requirements 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
- Warming requirements for the main steam line piping, turbine control, and stop valve and main turbine
- Small increases in pressure over time to provide better seating characteristics of the SRVs.

The licensee stated that the VT-2 leakage examination inside the drywell represents a hardship at the nominal operating pressure of 1,045 psig during startup because of high ambient and component temperatures. During the two most recent startups in September 2013 and March 2015, using instrumentation approximately 8 feet higher in elevation than the SRVs, the licensee measured the ambient temperature, which was approximately 144 degrees Fahrenheit (°F) once pressure reached 920 psig, and ambient temperature increased to approximately 150 °F over a 6-hour period, while holding pressure steady at 920 psig. The licensee stated that the local drywell temperature increased to approximately 170 °F, or higher, when nominal reactor pressure of 1,045 psig was reached.

The licensee noted that the nominal operating pressure of the reactor coolant system (RCS) results in drywell ambient temperatures that require special safety precautions such as ice vests and cool air supply lines for personnel performing the VT-2 examinations. These adverse conditions could compromise the quality of the leakage examination due to the hardship imposed on examination personnel.

The licensee noted that performance of a cold leakage test (i.e., a non-nuclear heat-up such as that required following a normal refueling outage) subsequent to a maintenance shutdown is judged to be an imprudent course of action for the reasons described below:

- Main steam lines are flooded with main steam isolation valves closed.
- The reactor pressure vessel (RPV) is required to be virtually water solid.
- 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 this test will impose an additional challenge to the affected systems. A normal plant startup would then occur after completion and subsequent recovery from the cold leakage test.
- Performing a cold leakage test would add approximately 2 days to the shutdown duration.
- Performance of an additional cold leakage test places the unit in a position of reduced margin, unnecessarily approaching the fracture toughness limits defined in the Technical Specification pressure-temperature curves.
- The scope of the VT-2 visual examination does not include the RPV.

3.5 Proposed Alternative

The licensee proposed to perform the required system leakage test and associated VT-2 visual examination for any repair/replacement activities of mechanical joint connections during the May 2016 maintenance shutdown at a reactor pressure of approximately 920 psig. The licensee proposed to implement a 1-hour hold time for non-insulated components and a 6-hour hold time for insulated components prior to performing the VT-2 visual examination. In addition, if there are unplanned shutdowns with drywell entries before the next refueling outage (currently scheduled to begin in February 2018), the licensee proposed to inspect the affected mechanical joint connections at reactor system pressure of greater than 920 psig to look for leakage.

The licensee stated that it will disposition any observed leakage as the marginal increase in leakage rates that might occur at the nominal operating pressure associated with 100 percent rated reactor power (i.e., 1,045 psig) and the actual reactor pressure when the examination was performed. The licensee stated that drywell monitoring systems would detect leakage that might occur in 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.

The licensee concluded that because the RCS pressure boundary is subjected to a leakage test and visual examination at nominal operating pressure (i.e., 1,045 psig) near the end of every refueling outage, and monitoring systems detect leakage inside the drywell, a leakage test and visual examination performed at 920 psig for the repair/replacement of mechanical joint connections provide adequate assurance of structural and pressure boundary integrity.

3.6 Duration of Proposed Alternative

The HNP, Unit No. 2, maintenance shutdown is currently scheduled to begin on May 20, 2016, with the alternative continuing in effect through the start of the next refueling outage, currently scheduled to begin in February 2018.

3.7 NRC Staff Evaluation

The NRC staff evaluated the licensee's technical basis for the system leakage testing, hold time, additional monitoring, and hardship justification to determine the acceptability of the proposed alternative.

3.7.1 System Leakage Testing

During the HNP, Unit No. 1, spring 2016 outage, all Unit No. 1 SRVs were replaced, and the removed SRVs were pressure tested. Upon the completion of the pressure testing, the licensee determined that several of the removed SRVs exhibited unsatisfactory inspection results, and the licensee decided to proactively replace the SRV main valve bodies in HNP, Unit No. 2, with components that underwent augmented testing. ASME Code, Section XI, IWA-4540(a), of the 2001 Edition through 2003 Addenda, which is the code of record at HNP, Unit No. 2, requires a hydrostatic or system leakage test, in accordance with IWA-5000, for repair/replacement activities of Class 1, 2, and 3 components performed by welding or brazing on a pressure-retaining boundary prior to, or as part of, returning to service. However, IWA-4540(b) exempts mechanical joints from system leakage testing. The NRC staff notes that replacement of SRV valve bodies involves mechanical joint connections. Therefore, the licensee is not required to perform a system leakage test after replacing SRVs in accordance with the code of record for the fifth inservice inspection interval at HNP, Unit No. 2.

However, 10 CFR 50.55a(b)(2)(xxvi) limits the use of ASME Code, Section XI, 2001 Edition, and later editions and addenda, with regard to system leakage tests. This limitation requires that the repair and replacement activity provisions of ASME Code, Section XI, IWA-4540(c), of the 1998 Edition of ASME Code, Section XI, for pressure testing Class 1, 2, and 3 mechanical joints be applied. Article IWA-4540(c) of the 1998 Edition requires that mechanical joints made in the installation of pressure-retaining items be pressure tested in accordance with ASME Code, Section XI, IWA-5211(a).

The 1998 Edition of ASME Code, Section XI, IWA-5211(a), provides the description of a system leakage test performed 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 1998 Edition of ASME Code, Section XI, IWB-5220, provides requirements of the system leakage test specifically for ASME Class 1 components, such as SRVs. ASME Code, Section XI, IWB-5221(a), requires a system leakage test to be performed at a pressure not less than the pressure corresponding to 100 percent rated reactor power (i.e., 1,045 psig). The licensee requested relief from IWB-5221(a) of the 2001 Edition through 2003 Addenda of ASME Code, Section XI, and proposed to deviate from the test pressure of 1,045 psig as required by IWB-5221(a).

The *Federal Register* Notice (69 FR 58804), dated October 1, 2004, which incorporated paragraph 50.55a(b)(2)(xxvi), provided the basis for the system leakage test requirement. The NRC staff noted that it was not clear why the requirements to pressure test Class 1, 2, and 3 mechanical joints undergoing repair and replacement activities were deleted in the 1999 Addenda of ASME Code, Section XI. The NRC staff believed the system pressure testing was necessary to ensure and verify the integrity of the pressure boundary. The *Federal Register* Notice did not discuss specific test pressure values but focused on the need to perform a pressure test and VT-2 examination to verify the integrity of the pressure boundary after repair and replacement activities.

As an alternative to IWB-5221(a), which requires the use of test pressure of 1,045 psig, the licensee proposed to perform the system leakage test at a pressure of at least 920 psig. A test pressure of 920 psig is approximately 88 percent of the required pressure of 1,045 psig, and

the piping system with the associated mechanical joint connections will experience 88 percent of the pressure loading at normal operating conditions. Due to the high pressure of the proposed test, leakage through the mechanical joint connections would be detectable at 920 psig with a slightly lower leakage rate than that at 1,045 psig. The NRC staff finds that the proposed test pressure is sufficiently high to cause detectable leakage from any mechanical joint connections following disassembly and reassembly of the affected SRVs, if a leak-tight connection has not been established.

The licensee stated that it will disposition any observed leakage as the marginal increase in leakage rates that might occur at the nominal operating pressure associated with 100 percent rated reactor power (i.e., 1,045 psig) and the actual reactor pressure when the examination was performed. The NRC staff finds this approach acceptable because the proposed test pressure is close to the normal operating pressure (i.e., 88 percent). Due to this minimal pressure difference, an extrapolation of the test leakage rate, if any leakage occurs, should be close to the leakage rates from a nominal operating pressure test.

The NRC staff notes that the licensee will perform system leakage tests and VT-2 visual examinations of the RCS using a test pressure at 100 percent rate of reactor power (1,045 psig) during scheduled refueling outages. The proposed alternative is only a one time application specifically for the outage to replace the SRVs. Based on the above evaluation, the NRC staff finds that a test pressure of 920 psig for the system leakage test would have a negligible effect on plant safety and, therefore, is acceptable.

3.7.2 Hold Time

The licensee also requested to deviate from Article IWA-5213(a) of ASME Code, Section XI, which requires a 10-minute hold time for non-insulated components and 4-hour hold time for insulated components prior to performing the VT-2 visual examination as part of the system leakage test per IWA-4540. As an alternative, the licensee proposed to implement a 1-hour hold time for non-insulated components and a 6-hour hold time for insulated components prior to performing the VT-2 visual examination. The NRC staff finds that the increased hold time is more conservative than the requirements in IWA-5213(a) because longer hold times increase the possibility of observing leakage, if it occurs. Therefore, the NRC concludes that the proposed hold time is acceptable.

3.7.3 Additional Monitoring

The NRC staff notes that if there are unplanned shutdowns with drywell entries before the next refueling outage (currently scheduled to begin in February 2018), the licensee will inspect the affected mechanical joint connections at reactor system pressure of greater than 920 psig to look for any evidence of leakage. The NRC staff finds that this additional monitoring will provide increased chances to discover SRVs leakage and, therefore, is acceptable.

The NRC staff notes that HNP, Unit No. 2, has drywell air temperature and pressure monitors and the drywell floor and equipment drain sumps to detect leakage that might occur in mechanical joint connections. The NRC staff finds that these detection systems will detect potential leakage and thereby enhance the monitoring of the structural integrity and leak-tightness of the replacement SRVs.

3.7.4 Hardship and Unusual Difficulty Justification

To perform the system leakage test with the associated VT-2 visual examination in accordance with ASME Code, Section XI, IWB-5221(a) requirements, plant personnel would face extreme temperatures inside the drywell (i.e., 170 °F or higher), possibly causing serious heat stress and burn concerns and be forced to take safety precautions such as ice vests and cool air supply lines. Managing this extra safety equipment and working in the high temperatures could affect the quality of the VT-2 visual examination. Based on the above, the NRC staff finds that performing the VT-2 visual examination in accordance with the ASME Code is an unusual difficulty for plant personnel.

As an alternative to satisfy ASME Code, Section XI, IWB-5221(a), the licensee could perform a cold leakage test based on a non-nuclear heatup. Performing this type of test would require filling the main steam lines and the reactor vessel solid with water, which is undesirable for system operation. To establish the necessary test conditions, the licensee would have to perform temporary hanger modifications, valve lineups, and system manipulations, all of which activities would require additional personnel radiation exposure beyond that received in normal startup activities. Additionally, these activities may introduce human errors. The cold leakage test places the unit in a position of reduced margin, unnecessarily approaching the fracture toughness limits defined in the Technical Specification pressure-temperature curves. Based on the above, the NRC staff finds that performance of a cold leakage test would cause a hardship for HNP, Unit No. 2.

4.0 CONCLUSION

As set forth above, the NRC staff has determined 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, the NRC staff has determined that the proposed alternative described in HNP-ISI-ALT-05-02, Version 1.0, provides reasonable assurance of structural integrity and leak tightness of the subject components. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2) and, therefore, authorizes the proposed alternative in accordance with 10 CFR 50.55a(z)(2) for the HNP, Unit No. 2, May 2016 maintenance shutdown through the start of the next refueling outage, currently scheduled to begin in February 2018.

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.

Principal Contributor: John Tsao

Date: May 16, 2016

If you have any questions, please contact the Project Manager, Michael Orenak, at 301-415-3229 or Michael.Orenak@nrc.gov.

Sincerely,

/RA/

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-366

Enclosure:
Safety Evaluation

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