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

February 8, 2018

Mr. James J. Hutto
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 NOS. 1 AND 2 – PROPOSED INSERVICE INSPECTION ALTERNATIVE HNP-ISI-ALT-05-06 (CAC NOS. MF9851 AND MF9852; EPID L-2017-LLR-0054)

Dear Mr. Hutto:

By letter dated June 5, 2017 (Agencywide Documents Access and Management System Accession No. ML17156A831), Southern Nuclear Operating Company (the licensee), requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, regarding the use of alternative test pressures for certain Class 1 pressure tests using non-nuclear heat at Edwin I. Hatch Nuclear Plant (Hatch), Units 1 and 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee requested to use the proposed alternative, HNP-ISI-ALT-05-06, on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality of safety.

The U.S. Nuclear Regulatory Commission (NRC) staff has completed its review of request for alternative HNP-ISI-ALT-05-06. As set forth in the enclosed safety evaluation, the NRC staff has determined that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Accordingly, the NRC staff concludes that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the NRC staff authorizes the use of Alternative HNP-ISI-ALT-05-06 at Hatch, Units 1 and 2, until December 31, 2025.

All other requirements of ASME Code, Section XI, for which relief has not been specifically requested and approved in this request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

The staff's review of proposed alternative HNP-ISI-ALT-05-05, also submitted in the licensee's letter of June 5, 2017, will be addressed in separate correspondence.

If you have any questions, please contact the Project Manager, Randy Hall, at 301-415-4032 or by e-mail at Randy.Hall@nrc.gov.

Sincerely,

Michael T. Markley

Michael T. Markley, 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

cc: Listserv

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVE HNP-ISI-ALT-05-06

REGARDING CERTAIN CLASS 1 PRESSURE TESTS

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

SOUTHERN NUCLEAR OPERATING COMPANY

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated June 5, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17156A831), Southern Nuclear Operating Company (the licensee), requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code) Section XI, regarding the alternative test pressures for certain Class 1 pressure tests using non-nuclear heat at Edwin I. Hatch Nuclear Plant (Hatch), Units 1 and 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee requested to use the proposed alternative, HNP-ISI-ALT-05-06, on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality of safety.

The staff's review of proposed alternative HNP-ISI-ALT-05-05, also submitted in the licensee's letter of June 5, 2017, will be addressed in separate correspondence.

2.0 REGULATORY EVALUATION

Adherence to Section XI of the ASME Code is mandated by 10 CFR 50.55a(g)(4), which states, in part, that ASME Code Class 1, 2, and 3 components will meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in Section XI of the ASME Code.

Paragraph 10 CFR 50.55a(z) states that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The licensee must demonstrate that: (1) the proposed alternatives 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 an alternative, and for the NRC to authorize the proposed alternative.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Component Affected

Code Class:	1
Reference:	IWB-5221(a)
Examination Category:	B-P
Item Number:	B15.10
Component Number:	Various

3.2 Applicable Code Edition and Addenda

The Code of Record for Hatch, Units 1 and 2, Inservice Inspection Fifth Ten-Year Interval is ASME Code, Section XI, 2007 Edition with 2008 Addenda.

3.3 Applicable Code Requirements

Paragraph 10 CFR 50.55a(b)(2)(xxvi) requires the use of the 1998 Edition, IWA-4540(c) for pressure testing of Class 1, 2, & 3 mechanical joints. The 1998 Edition of ASME Code, Section XI, IWA-4540(c) 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.

The 1998 Edition of ASME Code, Section XI, IWA-5211(a) refers to "...a system leakage test conducted during operation at nominal operating pressure, or when pressurized to nominal operating pressure and temperature."

ASME Code, Section XI, 2007 Edition with the 2008 Addenda, IWA-4540(a) states:

> Unless exempted by IWA-4540(b), repair/replacement activities performed by welding or brazing on a pressure-retaining boundary shall include a hydrostatic or system leakage test in accordance with IWA-5000, prior to, or as part of, returning to service. Only brazed joints and welds made in the course of a repair/replacement activity require pressurization and VT-2 visual examination during the test.

ASME Code, Section XI, 2007 Edition with the 2008 Addenda, IWB-5221(a) states:

The system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power.

3.4 Reason for Alternative Request

The licensee summarized the reason for the alternative request in its submittal. At Hatch, Units 1 and 2, the performance of Class 1 pressure tests for repair/replacement activities (in accordance with IWA-4540), when performed after Table IWB-2500-1, Category B-P testing has been completed, requires abnormal plant conditions/alignments. Testing at these abnormal plant conditions and alignments results in additional risks and delays while providing little added benefit beyond tests which could be performed at slightly reduced pressures under normal plant conditions.

During the typical startup sequence, the normal system operating pressure of 1045 pounds per square inch gauge (psig) is not attained until 24 hours after a pressure of 920 psig is reached. The delay in time is caused by the following: control rod drive withdrawal limitations and the associated gradual increases in reactor power, pressure, and temperature; required pressure versus temperature limitations listed in the Technical Specifications; main steam line piping, turbine control and stop valve warming requirements; and main turbine warming requirements.

The high ambient and component temperatures existing inside the drywell (primary containment) during start-up at the nominal operating pressure present a hardship in performing the VT-2 leakage examination. This hardship has been documented in data from two startups, in September 2013 and March 2015, using instrumentation approximately eight feet higher than the main steam safety relief valves. The ambient temperature was approximately 144 degrees Fahrenheit once a system pressure of 920 psig was reached. The ambient temperature increased to approximately 150 degrees Fahrenheit over a 6-hour period, while holding pressure steady at 920 psig. The local drywell temperature reached approximately 170 degrees Fahrenheit or higher once the nominal reactor pressure of 1045 psig was attained. The reactor coolant system nominal operating pressure results in extreme drywell ambient temperatures that require special safety precautions, such as ice vests and cool air supply lines for the personnel performing the VT-2 examinations.

ASME Code Case N-795 is intended to provide alternative test pressures for certain Class 1 pressure tests using non-nuclear heat. The code case would be used by the licensee following those repair/replacement activities (excluding those on the reactor vessel) which occur subsequent to the periodic Class 1 pressure test required by Table IWB-2500-1, Category B-P and prior to the next refueling outage on those components that cannot be isolated. Components that can be isolated will be pressure tested at a pressure in accordance with IWB-5221(a).

The performance of the Class 1 pressure test required by Table IWB-2500-1, Category B-P each refueling outage places the Hatch units in a position of significantly reduced margin, approaching the fracture toughness limits defined in the Technical Specification Pressure Temperature curves. Violating these curves would place the reactor vessel in a Low Temperature Overpressure (LTOP) condition. With strict operational control procedures, specific component alignment, and operations staff training regarding LTOP, it may be considered acceptable to be at this reduced margin condition for the purpose of verifying the leakage status and integrity of the primary system in order to meet the ASME Code, Section XI, Category B-P requirements prior to startup from a refueling outage. However, to perform this evolution more frequently would increase the overall risk to the plant.

3.5 Licensee's Proposed Alternative

In its letter dated June 5, 2017, the licensee described its proposal to perform the system leakage test and associated VT-2 examination following the repair/replacement activities on those components that cannot be isolated in accordance with ASME Code Case N-795, while using longer hold times than specified in the code case. The system leakage test will be performed during the normal operational start-up sequence at a minimum of 920 psig following a one-hour hold time (for uninsulated components) and a six-hour hold time (for insulated components), in lieu of the nominal operating pressure of approximately 1045 psig associated with 100 percent reactor power. Note that this code case is not applicable to Class 1 pressure tests performed to satisfy the periodic requirement of Table IWB-2500-1, Category B-P and is not applicable to pressure tests required following the repair/replacement activities on the reactor vessel. The licensee will continue to conduct the periodic system leakage tests required by IWB-2500-1, Category B-P at the end of each refueling outage at a pressure corresponding to 100 percent rated reactor power.

3.6 Basis for Use

In its letter dated June 5, 2017, the licensee provided the basis for use of the alternative, as summarized in this section. By the end of a normal refueling outage, the core decay heat has had time to decrease and some spent fuel has been removed and some new fuel has been added. The result is a much lower decay heat load and much lower heatup rates. At the end of a normal refueling outage, the rate of temperature increase is able to be tolerated during the system leakage test. During normal performance of this system leakage test, the pressurization phase of the test is taken at a slow and very controlled pace. The pressurization phase normally takes several hours to reach test conditions.

However, following a maintenance or forced outage, there is a much larger decay heat load from the reactor core. That heat load is difficult to control once Shutdown Cooling (SDC) has been removed from service. Once SDC is removed from service, heatup starts immediately. During a short term mid-cycle shutdown, the projected heatup rate could be in the order of 0.5 °F per minute. Under those conditions, the time available to pressurize up to test conditions, perform the VT-2 exam and return to SDC will be greatly reduced. These limited time frames may result in a greater likelihood for errors.

During short mid-cycle outages, the reactor core has a large decay heat load. There is some inherent risk in isolating SDC from the vessel under high decay heat loads. Once isolated, mechanical, control or operational problems could occur that could delay return to SDC.

The required VT-2 examinations performed following the repair/replacement activities are limited to the areas affected by the work, thereby allowing for a focused exam. The VT-2 exams, therefore, have a much smaller examination boundary than the periodic test.

The ability to identify leakage through the VT-2 examinations during a test will not be significantly different for tests conducted at pressures associated with the 100 percent rated reactor power level or at 88 percent of that value. A higher pressure under otherwise same conditions will produce a higher flow rate, but the difference is not significant for the purpose of the test. ASME Code Case N-795 proposes increased hold times, as compared to a test performed at normal operating pressure, to allow for more leakage from the pressure boundary if a through-wall or mechanical joint leakage condition exists. Further, the licensee proposes to implement longer hold times than specified by ASME Code Case N-795. The licensee believes these longer hold times will allow for additional leakage to occur at the area of interest so as to be more evident during the VT-2 examination, should a through-wall or mechanical joint leakage condition exist. The alternate test pressure, when combined with longer hold times, will provide adequate evidence of leakage, should a leak exist.

While the licensee does not expect that leakage will occur, any leakage observed will be related to the differential pressure at the point of leakage, or across the connection. A 12 percent reduction in the test pressure in the range under consideration is not expected to result in the arrest of a leak that would occur at nominal operating pressure. In the unlikely event that leakage would occur at nominal operating pressure subsequent to the alternative VT-2 examination at 920 psig, such leakage would be detected by the drywell monitoring systems, which include drywell pressure monitoring, the containment atmosphere monitoring system, and the drywell floor drain sumps. These leakage monitoring methods are required by the Hatch Technical Specifications.

The combination of the methods of ASME Code Case N-795 and the licensee's proposed longer hold times allow for an adequate pressure test to be performed; ensuring the safety margin is not reduced due to VT-2 examination being performed at the slightly reduced pressure. The affected pressure boundary will be tested and will be otherwise fully capable of performing its intended safety function as part of the reactor coolant pressure boundary.

The use of ASME Code Case N-795 will only be applied if the System Leakage Test required by IWB-2500-1, Category B-P has been completed for the cycle on components that cannot be isolated. The Code Case will not be implemented for any repair/replacement activity performed on the reactor pressure vessel.

In summary, the proposed alternative is to perform the required system leakage test and VT-2 examination in accordance with ASME Code Case N-795 at 920 psig, with a minimum hold time of one hour for uninsulated components and a six-hour hold time for insulated components during maintenance, forced outages, or following the performance of the periodic pressure test required by Table IWB-2500-1, Category B-P during refueling outages. The provisions of this alternative are not applicable to the Examination Category B-P pressure test performed during refueling outages, or to pressure tests of repair/replacement activities of the reactor pressure vessel or components that can be isolated. Considering the discussion above, the licensee believes that this alternative will provide an acceptable verification of the leak integrity of the locations having repair/replacement activities performed without putting the plant in a non-conservative operational condition, and without unnecessary radiation exposure and safety challenges to personnel.

3.7 Duration of Alternative

The licensee stated that the alternative is applicable upon approval, until the end of the fifth 10-year ISI interval for both Hatch units, December 31, 2025.

4.0 NRC STAFF EVALUATION

Performance of a system leakage test of pressure retaining boundaries usually occurs at the end of a refueling outage when the core decay heat has had time to decrease, some spent fuel has been removed, and some new fuel has been added, which results in a relatively low decay heat load. As previously discussed above by the licensee, the isolation of SDC under high decay heat loads leads to inherent risk and hurried time frames, which could create a more error-likely environment. The nominal operating pressure of 1045 psig will not be reached for more than 24 hours after reaching 920 psig during the normal startup sequence. At the nominal operating pressure, there are high ambient and component temperatures inside the drywell and they increase over time while holding the pressure constant at 920 psig. If the licensee's personnel performed the VT-2 visual examination at nominal operating pressure, higher ambient temperatures would exist, and the use of ice vests and cool air supply lines would be necessary. The NRC staff finds that performance of a system leakage test at these conditions presents unusual difficulty and a risk to plant personnel and, therefore, would present a hardship.

At the end of every outage, the licensee performs the required Category B-P pressure test at the nominal operating pressure. The performance of this test places the licensee in a position of reduced margin by approaching the fracture toughness limits defined in their Technical Specification Pressure Temperature curves. If these curves are violated, the vessel will be placed in a LTOP condition. In order to remain in this reduced margin for the required amount of time for the VT-2 examination, the licensee's staff will have to adhere to strict control procedures and have specific component alignment and operations training regarding LTOP. If these system leakage tests need to be performed at additional times, there is an increased risk to the plant. The NRC staff finds that performance of a system leakage test at these conditions presents unusual difficulty and a risk to plant operation and, therefore, would present a hardship.

Based on the above, the NRC staff concludes that after repair/replacement activities, the performance of a VT-2 visual examination during a system leakage test at normal operating pressure would present a hardship. The NRC staff also concludes that performance of a system leakage test at the proposed reduced pressure, in combination with compliance with ASME Code requirements for design, fabrication, and volumetric NDE, provides reasonable assurance of structural integrity. Furthermore, the NRC staff finds that performing the VT-2 examination at a pressure of at least 88 percent of the pressure at 100 percent rated reactor power, with hold times of one hour for non-insulated components and six hours for insulated components, provides reasonable assurance of leak tightness.

5.0 CONCLUSION

As set forth above, the NRC staff has determined that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Accordingly, the NRC staff concludes that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the NRC staff authorizes the use of Alternative HNP-ISI-ALT-05-06 for the duration of the fifth 10-year ISI interval at Hatch, Units 1 and 2, which ends on December 31, 2025.

All other requirements of ASME Code, Section XI, for which relief has not been specifically requested and approved in this request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: D. Render, NRR

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2 – PROPOSED INSERVICE INSPECTION ALTERNATIVE HNP-ISI-ALT-05-06 (CAC NOS. MF9851 AND MF9852; EPID L-2017-LLR-0054) DATED FEBRUARY 8, 2018

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