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SOUTHERN NUCLEAR
A SOUTHERN COMPANY

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Docket Nos.: 50-366

NL-16-0717

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

Edwin I. Hatch Nuclear Plant
<u>ISI Program Alternative HNP-ISI-ALT-05-02</u>

Ladies and Gentlemen:

It has recently been determined that several safety relief valves (SRVs) removed from Plant Hatch Unit 1 exhibited unsatisfactory post-installation inspection results. Based on these results, Plant Hatch has conservatively decided to replace the Unit 2 SRV main valve bodies with components which have undergone augmented testing following the 2015 Target Rock Part 21 recommendations. This replacement will be performed during a Plant Hatch Unit 2 maintenance shutdown that has been scheduled to begin on May 20, 2016 and will result in the need to perform system leakage testing in accordance with certain provisions of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).

Southern Nuclear Operating Company (SNC) had previously requested approval of an alternative to the requirements of the ASME Code Section XI, 2001 Edition through 2003 Addenda, Subsection IWB-5221(a) in our letter NL-11-2498 dated December 13, 2011. Specifically, SNC requested NRC approval of proposed Alternative HNP-ISI-ALT-15, Version 2, 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 Code-required pressure following repair and replacement activities on several SRVs installed in Plant Hatch Unit 2.

The 2001 Edition through 2003 Addenda of the ASME Code, Section XI requires the performance of a system pressure test in accordance with Section IWB-5220 for ASME Code Class 1 pressure boundary components prior to plant startup following each reactor outage. Paragraph IWB-5221(a) requires that the system leakage test be conducted at a test pressure not less than the nominal operating pressure associated with 100% rated reactor power, which for Plant Hatch is 1045 psig. These pressure test requirements are supplemented by 10 CFR50.55a(b)(2)(xxvi) which invokes the requirements of IWA-4540(c) of the 1998 Edition of the ASME Section XI Code for repair/replacement activities of Class 1, 2, and 3 mechanical joint connections.

Pursuant to 10 CFR 50.55a(a)(3)(ii), SNC hereby requests approval of an alternative to the requirements of the ASME Code Section XI, 2001 Edition through 2003 Addenda, Subsection IWB-5221(a). Specifically, SNC requests NRC approval of proposed Alternative HNP-ISI-ALT-05-02, 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 Code-required pressure following repair and replacement activities. This proposed alternative would allow the performance of the VT-2 visual leakage examination following the SRV repair and replacement activities at the lower pressure of greater or equal to 920 psig while employing a one-hour hold-time for non-insulated components and a six hour hold time for insulated components.

10 CFR 50.55a(a)(3)(ii) provides for the use of such alternatives to the requirements of paragraph 50.55a(g) when so authorized by the NRC. Such authorization is contingent upon a demonstration by the applicant that the proposed alternative would provide an acceptable level of quality and safety or that the specified requirements would result in hardship or unusual difficult without a commensurate increase in the level of quality and safety. It can be noted that this reduced reactor system pressure will result in more tolerable ambient temperature conditions for examination personnel and therefore should facilitate a higher quality examination. This alternative is therefore justified since compliance with the cited requirements of the subject code would result in a plant hardship without a commensurate increase in the level of quality and safety of the associated maintenance activity. Further details of the demonstration required by 10 CFR 50.55a(a)(3)(ii) are provided in Attachment 1 to this letter.

The NRC has previously approved several similar relief requests for performing pressure tests at less than nominal operating pressure including SNC's request for approval of ISI Program Alternative HNP-ISI-ALT-15, version 2 transmitted by letter NL-11-2498 dated December 13, 2011.

Expedited approval of the proposed alternative is requested on or before May 20, 2016, to allow the use of this proposed alternative for the planned SRV replacements scheduled to be performed during the Plant Hatch Unit 2 May 2016 maintenance shutdown.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at (205) 992-7369.

Respectfully submitted,

C. R. Pierce

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Regulatory Affairs Director

CRP/cdp

Enclosure 1: Request for ISI Program Alternative HNP-ISI-ALT-05-02

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<u>Edwin I. Hatch Nuclear Plant</u>

Enclosure 1 to NL-16-0717

HNP-ISI-ALT-05-02

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(ii)

SOUTHERN NUCLEAR OPERATING COMPANY
HNP-ISI-ALT-05-02, VERSION 1.0
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(ii)

Plant Site-Unit:	Edwin I. Hatch Nuclear Plant - Unit 2
Interval-Interval Dates:	5th ISI Interval, January 1, 2016 through December 31, 2025
Requested Date for Approval:	Approval is requested by May 20, 2016 to support testing during a Unit 2 maintenance shutdown that is currently scheduled to begin on May 20, 2016.
ASME Code Components Affected:	Class 1 pressure-retaining mechanical joint connections which require a VT-2 examination for leakage subsequent to repair/replacement activities.
Applicable Code Edition and Addenda:	ASME Section XI Code, 2001 Edition through the 2003 Addenda
Applicable Code Requirements:	1. IWA-4540(a) requires a hydrostatic or system leakage test, in accordance with IWA-5000, for repair/replacement activities performed by welding or brazing on a pressure-retaining boundary prior to, or as part of, returning to service. 2. IWA-5213(b) requires a 10 minute hold time for non-insulated components and 4 hour hold time for insulated components prior to performing the VT-2 leakage test. 3. IWB-5221(a) requires the system leakage test to be conducted at a pressure not less than the nominal pressure associated with 100% rated reactor power.

Reason for Request:	10CFR50.55a(b)(2)(xxvi) *Pressure Testing Class 1, 2, and 3 Mechanical Joints* provides supplemental code requirements to those of IWA-4540(a) stated above. 10CFR50.55a(b)(2)(xxvi) invokes the IWA-4540(c) repair/replacement activity provisions of the 1998 Edition of Section XI for pressure testing of Class 1, 2, and 3 mechanical joints when using the 2001 Edition through the latest edition and addenda of ASME Section XI. Therefore, even though the ISI Code of Record applicable at Plant Hatch does not require pressure testing and VT-2 examination of mechanical joint connections, the 1998 Edition of Section XI does.
	
	Relief is requested from the test pressure requirement of IWB-5221(a) (i.e., 1045 psig) on the basis of hardship as cited below.
	• Replacement of some components installed via mechanical joints (e.g., Safety Relief Valves (SRVs)) is planned during a maintenance shutdown which is scheduled to begin May 20, 2016. These repair/replacement activities will require a VT-2 leakage examination of the mechanical joint connections during unit startup.
	• Nominal operation pressure (i.e., 1045 psig) will not be reached for more than 24 hours after reaching 920 psig during the startup sequence:
	  o Control Rod Drive withdrawal limitations and the associated gradual increases in reactor power, pressure and temperature.
	  o Technical Specification-required Pressure versus Temperature limitations.
	  o Main Steam line piping, turbine control and stop valve warming requirements.
	  o Main turbine warming requirements.
	  o Small increases in pressure over time to provide better seating

	characteristics of the SRVs. • VT-2 leakage examination inside the drywell (primary containment) represents a hardship at the nominal operating pressure of 1045 psig during start-up because of high ambient and component temperatures. o Data was retrieved for the two most recent startups (9/2013, 3/2015) using instrumentation approximately 8 ft higher in elevation than the SRVs. • Ambient temperature was approximately 144 degrees Fahrenheit once reaching 920 psig. • Data shows ambient temperature increases to approximately 150 degrees Fahrenheit over a 6 hour period while holding pressure steady at 920 psig. o The local drywell temperature seen once nominal reactor pressure of 1045 psig is attained is approximately 170 degrees Fahrenheit or higher. • Reactor coolant system nominal operating pressure 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 also compromise the quality of the leakage examination due to the hardship imposed on examination personnel. • Performance of a cold leakage test (that is, a non-nuclear heat-up such as that required following a refueling outage) subsequent to a maintenance shutdown is judged to be an imprudent course of action for the reasons described below: o Main Steam Lines are flooded with Main Steam Isolation Valves closed. o The reactor pressure vessel (RPV) is required to be virtually water solid. o Extensive valve manipulations, system lineups, and procedural controls are required in order to heat up and pressurize the reactor coolant system to establish the necessary test pressure. o 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. o Performing a cold leakage test would add approximately 2 days to the shutdown duration. o 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 (P–T) curves. o The scope of the VT-2 leakage examination does not include the reactor pressure vessel.
Proposed Alternative and Basis for Use:	Plant Hatch will perform the required VT-2 leakage examination for any repair/ replacement activities of mechanical joint connections performed during the May 2016 maintenance shutdown at a reactor pressure of ≥ 920 psig. Similar to previous similar submittals, SNC agrees 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 Leakage Test. In addition, if there are unplanned shutdowns with drywell entries before the next refueling outage (currently scheduled to begin in February 2018), inspections of the affected mechanical joint connections will be performed at reactor system pressure of ≥ 920 psig to look for any evidence of

	leakage. Disposition of any observed leakage will consider the marginal increase in leakage rates that might occur at the nominal operating pressure associated with 100% rated reactor power (i.e., 1045 psig) and the actual reactor pressure when the examination was performed. In addition, 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. Since the reactor coolant system pressure boundary is subjected to a leakage test and visual examination at nominal operating pressure (i.e., 1045 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. Therefore, this proposed alternative should be granted pursuant to 10 CFR 50.55a(a)(3)(ii). This alternative is essentially the same as Alternatives HNP-ISI-ALT-09, Version 2.0, and HNP-ISI-ALT-15, Version 2.0, both previously approved by the NRC for Hatch Unit 2 (References 5 and 6, respectively).
Duration of Proposed Alternative:	The Hatch Unit 2 maintenance shutdown 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.
References:	1. Entergy Nuclear Northeast, Pilgrim Nuclear Power Station, 4th 10-Year Interval ISI Program Relief Request PRR-2, NRC TAC NO. MC8286 dated June 29, 2006. 2. Nuclear Management Company, Monticello Nuclear Generating Plant, 3rd 10-Year Interval ISI Program Relief Request RR-17, NRC TAC NO. MC0593 dated March 25, 2004. 3. PSEG Nuclear, LLC, Hope Creek Nuclear Generating Station, 2nd 10-Year Interval ISI Program Relief Request HC-RR-12-023, NRC TAC No. MC2396 dated August 27, 2004. 4. Nebraska Public Power District, Cooper Nuclear Station, 3rd 10-Year Interval ISI Program Relief Request PR-10, NRC TAC No. MA0677 dated February 26, 1998. 5. Southern Nuclear Operating Company, Inc., Edwin I. Hatch Plant Unit 2, ISI Program Alternative HNP-ISI-ALT-09 Version 2.0, March 29, 2010 (NRC ADAMS Accession No. ML100890051). 6. Southern Nuclear Operating Company, Inc., Edwin I. Hatch Plant Unit 2, ISI Program Alternative HNP-ISI-ALT-15 Version 2.0, December 13, 2011 (NRC ADAMS Accession No. ML113480294). 7. NextEra Energy Duane Arnold, LLC, Duane Arnold Energy Center, Request for Authorization of Alternative Regarding Pressure Test Requirements, NRC TAC No. ME5143 dated September 6, 2011.

Status:	Awaiting NRC approval.