Mark J. Ajluni, P.E.
Nuclear Licensing Director

Southern Nuclear Operating Company, Inc.
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.7673
Fax 205.992.7885

SOUTHERN COMPANY

December 9, 2011

Docket No.: 50-366 NL-11-2464

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

Edwin I. Hatch Nuclear Plant
ISI Program Alternative HNP-ISI-ALT-15, Version 1

Ladies and Gentlemen:

It was recently determined that several safety relief valves (SRVs) installed in Plant Hatch Unit 2 are experiencing internal leakage. To correct this unforeseen condition, Southern Nuclear Operating Company (SNC) has conservatively decided to replace the main valve bodies and their associated three-stage pilot valves of the SRVs experiencing internal leakage with SRVs with two-stage pilot valves equipped with platinum-coated seating surfaces. This replacement will be performed during a Plant Hatch Unit 2 maintenance shutdown that has been scheduled to begin on December 14, 2011, 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).

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-15, Version 1, 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 at the lower

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pressure of ≥ 920 psig following the SRV repair and replacement activities which are not associated with a refueling outage.

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 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.

10CFR50.55a(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 requirement would result in hardship or unusual difficulty without a commensurate increase in the level of quality and safety. That demonstration is 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-09 Version 2 transmitted by letter NL-10-0618 dated March 29, 2010. A similar relief request submitted by the Duane Arnold Energy Center was approved as recently as earlier this fall as documented in the NRC letter of September 6, 2011 (TAC NO. MC5143).

Expedited approval of the proposed alternative is requested on or before December 16, 2011, to allow the use of this proposed alternative for the planned SRV replacements scheduled to be performed during the Plant Hatch Unit 2 December 2011 maintenance shutdown.

This letter contains no NRC commitments. If you have any questions regarding this request, please contact Doug McKinney at (205) 992-5982.

Respectfully submitted,

Mark J. Ajluni

M. J. Ajluni
Nuclear Licensing Director

MJA/WEB

Enclosure: Request for ISI Program Alternative HNP-ISI-ALT-15, Version 1

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Chief Nuclear Officer
Mr. D. R. Madison, Vice President – Hatch
Ms. P. M. Marino, Vice President – Engineering
RTYPE: CHA02.004

<u>U. S. Nuclear Regulatory Commission</u>
Mr. V. M. McCree, Regional Administrator
Mr. P. G. Boyle, NRR Senior Project Manager - Hatch
Mr. E. D. Morris, Senior Resident Inspector – Hatch

ATTACHMENT 1 to NL-11-2464

HNP-ISI-ALT-15, Version 1

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:	4th ISI Interval, January 1, 2006 through December 31, 2015
Requested Date for Approval:	Approval is requested by December 16, 2011 to support testing during a Unit 2 maintenance shutdown that is currently scheduled to begin on December 14, 2011.
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. 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 December 14, 2011. 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 achieved until approximately 14 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.

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<li style="list-style:circle">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.

Data was retrieved for a previous shutdown (5/2008) using instrumentation approximately 8 ft higher in elevation than the SRVs.

Ambient temperature was approximately 143 degrees Fahrenheit once reaching 920 psig.
Data shows ambient temperature increases to approximately 156 degrees Fahrenheit over a 14 hour period while holding pressure steady at 920 psig.

The expected time during a maintenance shutdown to increase pressure from 920 psig to 1045 psig is approximately 14 hours, therefore an ambient temperature increase of 13 degrees would be expected.

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.

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 reactor coolant system 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.

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<tr><td>Proposed Alternative and Basis for Use:</td><td>Plant Hatch will perform the required VT-2 leakage examination for any repair/ replacement activities of mechanical joint connections performed during the December 2011 maintenance shutdown at a reactor pressure of ≥ 920 psig. In addition, if there are unplanned shutdowns with drywell entries before the next refueling outage (currently scheduled to begin in February 2013), 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</td></tr>
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	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 Alternative HNP-ISI-ALT-09, Version 2.0, previously approved by the NRC for Hatch Unit 2 (Reference 5). The commitment cited herein relative to inspection activities of mechanical joint connections in the event of unplanned shutdowns with drywell entries before the next refueling outage is similar to that approved by the NRC for Duane Arnold Energy Center (Reference 6).
Duration of Proposed Alternative:	The Hatch Unit 2 maintenance shutdown currently scheduled to begin on December 14, 2011, with the alternative continuing in effect through the start of the next refueling outage currently scheduled to begin in February 2013.
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. 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.