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SOUTHERN COMPANY

May 4, 2015

Docket Nos.: 50-321
50-366

NL-15-0784

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

Edwin I. Hatch Nuclear Plant – Units 1 & 2
Submittal of the Inservice Testing Program Relief Requests and Alternatives for
Pumps and Valves - Fifth Ten-Year Interval

Ladies and Gentlemen:

Pursuant to 10 CFR 50.55a(z) and 10 CFR 50.55a(f), Southern Nuclear Operating Company (SNC) hereby requests Nuclear Regulatory Commission (NRC) approval of the following relief requests and alternatives for Edwin I. Hatch Nuclear Plant (HNP) Units 1 and 2. These relief requests and alternatives are applicable to the Fifth Ten-Year Interval Inservice Testing Program which will start on January 1, 2016:

RR-P-2	Standby Liquid Control pumps vibration monitoring location
RR-P-3	Residual Heat Removal pumps pressure accuracy
RR-P-4	Residual Heat Removal pumps flow accuracy
RR-P-5	RHR and Plant Service Water pumps vibration monitoring location
RR-P-6	Core Spray pumps pressure accuracy
RR-P-7	High Pressure Coolant Injection pump pressure accuracy
RR-P-8	High Pressure Coolant Injection pump flow accuracy
RR-P-9	Standby EDG Service Water pump vibration monitoring location
RR-P-11	Suppression Pool suction pressure for IST pumps
RR-P-12	Standby EDG Service Water pump flow accuracy
RR-P-13	Establish test flow reference ranges per Code Case OMN-21
RR-V-1	Scram Discharge AOV ganged test timing
RR-V-2	Traversing Incore Probe shear valve leak rate testing
RR-V-3	Cooling Water AOV local test timing and actuation
RR-V-5	High Pressure Coolant Injection rupture disks test interval
RR-V-8	Establish IST interval grace periods per Code case OMN-20
RR-V-9	Excess Flow Check Valve testing per Technical Specifications
RR-V-10	Pressure Isolation Valve performance-based testing

SNC requests that the NRC approve the proposed relief requests and alternatives for HNP Units 1 and 2 by December 1, 2015.

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

Respectfully submitted,

C. R. Pierce

C. R. Pierce
Regulatory Affairs Director

CRP/RMJ

Enclosure: Proposed Relief Requests and Alternatives - Fifth Ten-Year Interval Inservice Testing Program

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Edwin I. Hatch Nuclear Plant – Units 1 & 2
Submittal of the Inservice Testing Program Relief Requests and Alternatives for Pumps and Valves - Fifth Ten-Year Interval

Enclosure

Proposed Relief Requests and Alternatives - Fifth Ten-Year Interval Inservice Testing Program

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(2)
RR-P-2

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED: 1C41-C001A & B (Positive Displacement Pump) - Group B
2C41-C001A & B (Positive Displacement Pump) - Group B

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: ISTB-3510(e) - The frequency response range of the vibration measuring transducers and their readout system shall be from one-third minimum pump shaft rotational speed to at least 1000 Hz.

REASON FOR REQUEST: This alternative is a re-submittal of NRC approved 4th Interval relief request RR-P-2 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-P-2, is based on the ASME OM Code-2004 Edition with Addenda through OM-2006. There have been no substantive changes to this alternative, to the OM Code requirements or to the basis for use, which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-P-2.)

The Standby Liquid Control (SBLC) Pumps operate at 370 RPM (6.2 Hz), which corresponds to a required instrument range of 2.06 Hz to 1000 Hz. The minimum frequency response range of the Plant Hatch IST Program instrumentation is 2.5 Hz, which does not satisfy the code requirement. In lieu of the requirements of ISTB-3510(e), the vibration measuring instrument frequency response range utilized for the Standby Liquid Control Pumps will be as described below.

1. Vibration monitoring equipment with a calibration accuracy of at least ±5% over a frequency response range of 2.5 Hz to 1,000 Hz will be utilized for IST.
2. These lower frequency response limits result from high-pass filters which eliminate low-frequency elements associated with the input signal from the integration process. These filters prevent low frequency electronic noise from distorting vibration readings thus any actual vibration occurring at frequencies < 2.5 Hz is filtered out.

3. The SBLC pumps are Union Pump Company reciprocating pumps. The subject pumps utilize roller bearings instead of sleeve bearings. Sleeve bearings can exhibit vibration at sub synchronous frequencies when a condition of oil whirl is present. However, oil whirl does not occur in roller or ball bearings.
4. Roller and ball bearing degradation symptoms typically occur at 1X (6.2 Hz) shaft rotational frequency and greater. Therefore, vibration measurements at frequencies less than shaft speed would not provide meaningful data relative to degradation of the pump bearings.
5. The SBLC pumps are standby pumps only. They are only operated during Technical Specification Surveillance and Inservice Testing which results in very little run time. In the unlikely event that the system is required to perform its safety function, the pump run time would be from 19 to 74 minutes to exhaust the volume of the sodium pentaborate storage tank.
6. In addition to the IST vibration monitoring program, these pumps are included in the site maintenance department vibration program which has the capability to perform spectral analysis. The maintenance vibration program will also be utilized to analyze any IST vibration data which places the pumps in the ALERT or ACTION Ranges. The need for any corrective actions would be based on evaluation of IST and maintenance testing program data.

PROPOSED ALTERNATIVE AND BASIS: None, use of the existing vibration monitoring equipment which is calibrated to at least ±5% full scale over a frequency response range of 2.5 Hz to 1,000 Hz during Comprehensive and Preservice Testing will provide sufficient data for monitoring the mechanical condition of the SBLC pumps. This equipment will provide accurate vibration measurements over the frequency range in which typical roller bearing or other mechanical degradation conditions would occur.
This monitoring program should meet the intent of the code and will relieve the utility from the burden and expense involved with procurement, calibration, training and administrative control of new testing equipment which seems unjustified for assessing the mechanical condition of the subject pumps.

The above proposed alternative provides reasonable assurance of operational readiness since the SLC pumps have rolling element bearings and the instruments used to measure vibration are accurate at running speeds of < 1 X and greater. Based on the determination that compliance with the Code requirements results in a hardship without a compensating increase in the level of quality and safety, this proposed alternative should be granted pursuant to 10 CFR 50.55a(z)(2).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-P-2 for the Fourth 10 Year IST Interval.

REFERENCES: NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(2)
RR-P-3

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED: 1E11-C002A,B,C,D (Centrifugal Pumps) – Group A
2E11-C002A,B,C,D (Vertical Line Shaft Pumps) – Group A

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OMb-2006

REQUIREMENTS: ISTB-3510(b)(1) requires that the full-scale range for each analog instrument shall not be greater than three times the reference value. The Residual Heat Removal (RHR) system pump discharge pressure indicators 1(2)E11-PI-R003A-D exceed this Code allowable range limit.

REASON FOR REQUEST: This alternative is a re-submittal of NRC approved 4th Interval relief request RR-P-3 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-P-3, is based on the ASME OM Code - 2004 Edition with Addenda through OM-2006. There have been no substantive changes to this alternative or to the basis for use, which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-P-3).

The original installed instrumentation associated with these pumps was not designed with the instrument range limits of OM Code ISTB-3510(b)(1) taken into consideration. The actual instrument ranges are itemized below.

INSTRUMENT	RANGE	REF VALUE	ALLOWED RANGE *	ACCURACY
1E11-PI-R003A-D	0-600 psig	171–185 psig	0-513 psig	± 0.5%
2E11-PI-R003A-D	0-600 psig	180–195 psig	0-540 psig	± 0.5%

* - Allowed Range corresponds to 3 times the lowest reference value

PROPOSED ALTERNATIVE AND BASIS: None, use installed instrumentation during Group A pump testing.
Even though 1(2)E11-PI-R003A-D exceed the Code allowable range limit of three times the reference value, this additional gage range only results in

RR-P-3 (Cont.)

a 1.74 psig maximum variance from the Code allowable in the measured parameter (i.e. .02 x 513 = 10.26 psig versus .02 x 600 = 12.00 psig).

Using other (temporary) instrumentation during Group A testing to account for a 1.74 psig improvement in measurement accuracy is not justifiable considering the difficulty and dose associated with such a requirement. The installed pressure indicators will provide data that is sufficiently accurate to allow assessment of pump condition and to detect degradation during the performance of the Group A IST pump testing. M&TE, which meets all Code requirements, will be installed during Comprehensive and Preservice testing.

The above proposed alternative provides an acceptable means of assessing the condition of an RHR pump; because, if a pump was operating in the required action range, there would be limited difference in the information obtained if a more accurate pressure indicator was utilized. Based on the determination that compliance with the Code requirements results in a hardship without a compensating increase in the level of quality and safety, this proposed alternative should be granted pursuant to 10 CFR 50.55a(z)(2).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-P-3 for the Fourth 10 Year IST Interval

REFERENCES: NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(1)
RR-P-4

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED: 1E11-C002A,B,C,D (Centrifugal Pumps) – Group A
2E11-C002A,B,C,D (Vertical Line Shaft Pumps) – Group A

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: ISTB-3510(b)(1) requires that the full-scale range for each analog instrument shall not be greater than three times the reference value. RHR pump flow indicators 1(2)E11-FI-R608A&B exceed this Code allowable range limit

REASON FOR REQUEST: This alternative is a re-submittal of NRC approved 4th Interval relief request RR-P-4 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-P-4, is based on the ASME OM Code-2004 Edition with Addenda through OM-2006. There have been no substantive changes to this alternative, to the OM Code requirements or to the basis for use, which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-P-4)

The original installed instrumentation associated with these pumps was not designed with the instrument range limits of OM Code ISTB- 3510(b)(1) taken into consideration. The actual instrument ranges and loop accuracies are itemized below

INSTRUMENT	RANGE	TEST RANGE	ALLOWED RANGE	ACCURACY
1E11-FI-R608A&B	0-25000gpm	≈7700gpm	0-23100gpm	± 0.87%
2E11-FI-R608A&B	0-25000gpm	≈7700gpm	0-23100gpm	± 0.87%

COMPONENT/ ACCURACY	COMPONENT/ ACCURACY	COMPONENT/ ACCURACY	LOOP ACCURACY PER ISTA-2000
1E11-FT-N015A,B 0.5%	1E11-K600A,B 0.5%	1E11-FI-R608A,B 0.5%	0.87%
2E11-FT-N015A,B 0.5%	2E11-K600A,B 0.5%	2E11-FI-R608A,B 0.5%	0.87%

RR-P-4 (Cont.)

1(2)E11-FI-R608A(B) exceed the Code allowable full scale range limit of three times the reference value. The design of the indicator range includes consideration for LPCI flow rate (17,000 gpm for two pumps), whereas the minimum IST pump flow rate reference value is 7,700 gpm for Unit 1 and Unit 2. The Code maximum allowable inaccuracy in measured flow rate would be 462 gpm (i.e., .02 x 23,100) for Units 1 and 2, whereas the actual maximum inaccuracy in measured flow is 218 gpm (i.e., .0087 x 25,000) for both Unit 1 and Unit 2. Therefore, the actual accuracy of the installed flow indicators is better than required by the Code, thus the range of the indicator exceeding the Code limit of three times the reference value is of no consequence.

PROPOSED ALTERNATIVE AND BASIS: None, use installed instrumentation for Group A, Comprehensive Pump, and Preservice Testing.

Even though 1(2)E11-FI-R608A&B exceed the Code allowable range limit of three times the reference value, the overall loop accuracy is better than required by the Code. Therefore, the measured parameter is more accurately displayed than the Code requires. The above proposed alternative is acceptable since the variance in the actual test results is more conservative than that allowed by the Code.

Based on the determination that this alternative provides an acceptable level of quality and safety, the proposed alternative should be granted pursuant to 10 CFR 50.55a(z)(1).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-P-4 for the Fourth 10 Year IST Interval

REFERENCES:

1. NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627

2. NUREG-1482 Revision 2 Section 5.5.1 "Range and Accuracy of Analog Instruments"

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(2)
RR-P-5

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED:
1E11-C001A,B,C,D (RHRSW Vertical Line Shaft Pumps) – Group A
2E11-C001A,B,C,D (RHRSW Vertical Line Shaft Pumps) – Group A
1P41-C001A,B,C,D (PSW Vertical Line Shaft Pumps) – Group A
2P41-C001A,B,C,D (PSW Vertical Line Shaft Pumps) – Group A

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: ISTB-3540(b) requires that vibration measurements on vertical line shaft pumps be taken on the upper motor-bearing housing in three approximately orthogonal directions, one of which is the axial direction.

REASON FOR REQUEST: This alternative is a re-submittal of NRC approved 4th Interval relief request RR-P-5 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-P-5, is based on the ASME OM Code-2004 Edition with Addenda through OM-2006. There have been no substantive changes to this alternative, to the OM Code requirements or to the basis for use, which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-P-5)

The Code required vibration measurements on the upper motor bearing housing on these vertical line shaft pumps are impractical because of the following reasons.

1. Plant design did not include permanent scaffolding or ladders which provide access to the top of the motors for the subject pumps.

2. Physical layout of the pumps and interference with adjacent components does not allow for the installation of temporary scaffolding or ladders which are adequate and safe for routine use.

3. There is a thin cover plate bolted to the top-center of each motor which prevents measurements in line with the motor bearing. Measurement on the edge of the motor housing would be influenced by eccentricity and may not be representative of actual axial vibration.

4. Special tools (extension rod) for placing the vibration transducers are not practical because placement would not be sufficiently accurate for trending purposes.

5. Research within the industry has indicated that vibration monitoring of vertical line shaft pumps has been of limited benefit for detecting mechanical degradation due to problems inherent with pump design. The OM Code imposes more stringent hydraulic acceptance criteria on these pumps than for centrifugal or positive displacement pumps. These more stringent hydraulic acceptance criteria place more emphasis on detection of degradation through hydraulic test data than through mechanical test data.

PROPOSED ALTERNATIVE AND BASIS: Vibration measurements will be taken in three orthogonal directions, one of which is in the axial direction in the area of the pump to motor mounting flange when conducting Group A, Comprehensive Pump and Preservice Testing. This is the closest accessible location to a pump bearing housing and this location is easily and safely accessible for test personnel which should ensure repeatable vibration data and should provide readings which are at least as representative of pump mechanical condition as those required by the Code.

The above proposed alternative provides reasonable assurance of operational readiness since vibration measurements will be taken in three orthogonal directions at the pump to motor mounting flange which will provide information as to the mechanical integrity of the pump. Based on the determination that compliance with the Code requirement results in a hardship without a compensating increase in the level of quality and safety, this proposed alternative should be granted pursuant to 10 CFR50.55a(z)(2).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-P-5 for the Fourth 10 Year IST Interval

REFERENCES: NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(1)
RR-P-6

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED: 1E21-C001A&B (Centrifugal Pumps) – Group B
2E21-C001A&B (Vertical Line Shaft Pumps) – Group B

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: Table ISTB-3500-1 requires a total instrument loop accuracy for pressure indicators of ± 2% of full scale for Group B pump tests. This request is only applicable to the Group B pump test.

REASON FOR REQUEST: This alternative is a re-submittal of NRC approved 4th Interval relief request RR-P-6 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-P-6, is based on the ASME OM Code-2004 Edition with Addenda through OM-2006. There have been no substantive changes to this alternative, to the OM Code requirements or to the basis for use, which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-P-6).

Core Spray pump pressure indicators 1(2)E21-PI-R600A(B) exceed the maximum code allowable total loop accuracy of ± 2%. The actual instrument ranges and loop accuracies are itemized below.

INSTRUMENT	RANGE	REFERENCE VALUE	ALLOWED RANGE	ACCURACY
1E21-PI-R600A&B	0-500psi	273-282.6 psi	0-847.8 psi	± 2.06%
2E21-PI-R600A&B	0-500psi	332.3-335 psi	0-1005 psi	± 2.06%

COMPONENT/ ACCURACY	COMPONENT/ ACCURACY	COMPONENT/ ACCURACY	LOOP ACCURACY PER ISTA-2000
1E21-PT-N001A,B 0.5%	1E21-PI-R600A,B 2%	N/A N/A	2.06%
2E21-PT-N001A,B 0.5%	2E21-PI-R600A,B 2%	N/A N/A	2.06%

The indicators used have full scale ranges less than that allowed by the Code. The maximum code allowable variance in measurement is 16.96 psig (.02 x 847.8) for Unit 1 and 20.1 psig for Unit 2 (.02 x 1005). By using an indicator with a range less than the allowed limit, the actual maximum variance is 10.5 psig (.021 x 500) which is more accurate than required by the Code. Therefore, the actual accuracy of the instruments is within the Code allowable as specified in Table ISTB-3500-1 for a Group B pump test.

PROPOSED ALTERNATIVE AND BASIS: None, the installed instruments are more accurate than required by the Code for the range of application when performing a quarterly Group B pump test. Temporary pressure instruments that meet code requirements will be used during Comprehensive Pump and Preservice Testing.

The above proposed alternative provides an acceptable level of quality and safety since the variance in the actual test results is less than the maximum variance allowed by the Code. Based on the determination that the use of installed instrumentation provides an acceptable level of quality and safety, this proposed alternative should be granted pursuant to 10 CFR 50.55a(z)(1).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-P-6 for the Fourth 10 Year IST Interval

REFERENCES:

1. NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627
2. NUREG-1482 Revision 2 Section 5.5.1 "Range and Accuracy of Analog Instruments"

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(1)
RR-P-7

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED: 1E41-C001 (Centrifugal Pump) – Group B
2E41-C001 (Centrifugal Pump) – Group B

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: ISTB-3510(b)(1) requires that the full-scale range for each analog instrument shall not be greater than three times the reference value. Unit 1 and 2 HPCI pump suction pressure indicators 1(2)E41-PI-R004 exceed this Code allowable range limit. This request is only applicable to the Group B pump test.

REASON FOR REQUEST: This alternative is a re-submittal of NRC approved 4th Interval relief request RR-P-7 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-P-7, is based on the ASME OM Code-2004 Edition with Addenda through OM-2006. There have been no substantive changes to this alternative, to the OM Code requirements or to the basis for use, which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-P-7).

HPCI pump suction pressure gauges 1(2)E41-PI-R004 exceed the range limit of three times the reference value. The actual instrument ranges are itemized below.

INSTRUMENT	FULL SCALE	REFERENCE VALUE	ALLOWED RANGE	ACCURACY
1E41-PI-R004	100 psig	32.2 psig	0-96.6 psig	± 1%
2E41-PI-R004	100 psig	26.4 psig	0-79.2 psig	± 1%

The indicators are calibrated to ± 1 % full scale accuracy, resulting in a maximum inaccuracy of +/- 1 psig (100 * 0.01). The Code allowable inaccuracy, based on a gauge with a full scale exactly 3 x Reference value calibrated to +/- 2%, would be +/-1.93 psig for Unit 1 (96.6 * 0.02) and +/- 1.58 psig for Unit 2 (79.2 * 0.02). The better than required accuracy of the indicators overcomes the inaccuracy created by the full scale range being greater than 3 x reference values.

PROPOSED ALTERNATIVE AND BASIS: None, the installed pressure indicators provide measurements which are within the Code allowable accuracy specified in Table ISTB-3500-1 for quarterly Group B pump tests. Pressure instruments that meet the code requirements will be used during Comprehensive Pump and Preservice Testing.

The above proposed alternative provides an acceptable level of quality and safety since the variance in the actual test results is less than the maximum variance allowed by the Code. Based on the determination that the use of installed instrumentation provides an acceptable level of quality and safety, this proposed alternative should be granted pursuant to 10 CFR 50.55a(z)(1).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-P-7 for the Fourth 10 Year IST Interval

REFERENCES: NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(1)
RR-P-8

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED: 1E41-C001 (Centrifugal Pump) – Group B
2E41-C001 (Centrifugal Pump) – Group B

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: Table ISTB-3500-1 requires a total instrument loop accuracy for flow indicators of ± 2% of full scale for pump Inservice Testing. HPCI flow indicators 1(2)E41-FI-R612 do not meet this requirement.

REASON FOR REQUEST: This alternative is a re-submittal of NRC approved 4th Interval relief request RR-P-8 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-P-8, is based on the ASME OM Code-2004 Edition with Addenda through OM-2006. There have been no substantive changes to this alternative, to the OM Code requirements or to the basis for use, which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-P-8).
Flow indicators 1(2)E41-FI -R612 exceed the maximum code allowable total loop accuracy. The actual instrument loop accuracies are itemized below.

INSTRUMENT	RANGE	REFERENCE VALUE	ALLOWED RANGE	ACCURACY
1E41-FI-R612	0-5000gpm	4250 gpm	0-12750 gpm	± 2.12%
2E41-FI-R612	0-5000gpm	4250gpm	0-12750gpm	± 2.12%

COMPONENT/ ACCURACY	COMPONENT/ ACCURACY	COMPONENT/ ACCURACY	LOOP ACCURACY PER ISTA-2000
1E41-FT-N008 0.5%	1E41-FI-R612 2%	1E41-K601 0.5%	2.12%
2E41-FT-N008 0.5%	2E41-FI-R612 2%	2E41-K601 0.5%	2.12%

The indicator used has a full scale range less than that allowed. Therefore, the maximum variance allowable by the Code is 255 gpm (.02 x 12750) whereas the actual maximum variance is 106 gpm (.0212 x 5000). Therefore, the actual accuracy of the instrument loop is better than that allowable by the Code.

PROPOSED ALTERNATIVE AND BASIS: None, the installed flow indicators provide measurements which are within the Code allowable accuracy as specified in Table ISTB-3500-1 for flow testing. These flow indicators will be used during the Group B, Comprehensive Pump, and Preservice Test.

The above proposed alternative provides an acceptable level of quality and safety since the variance in the actual test results is less than the maximum variance allowed by the Code. Based on the determination that the use of installed instrumentation provides an acceptable level of quality and safety, this proposed alternative should be granted pursuant to 10 CFR 50.55a(z)(1).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-P-8 for the Fourth 10 Year IST Interval

REFERENCES:

1. NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627
2. NUREG-1482 Revision 2 Section 5.5.1 "Range and Accuracy of Analog Instruments"

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(2)
RR-P-9

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED: 2P41-C002 (Vertical Line Shaft Pump) – Group B

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: ISTB-3540(b) requires that vibration measurements on vertical line shaft pumps be taken on the upper motor-bearing housing in three approximately orthogonal directions, one of which is the axial direction.

REASON FOR REQUEST: This alternative is a re-submittal of NRC approved 4th Interval relief request RR-P-9 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-P-9, is based on the ASME OM Code-2004 Edition with Addenda through OM-2006. There have been no substantive changes to this alternative, to the OM Code requirements or to the basis for use, which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-P-9).

The Code required vibration measurements on the upper motor-bearing housing on this Unit 2 Standby Diesel Generator Service Water vertical line shaft pump are impractical because of the following reasons.

1. The motor has a cooling fan mounted at the top which is attached to the rotating shaft. The fan is protected by a relatively thin cover plate which prevents access to the motor housing for vibration measurements. Removing the cover does not provide for transducer placement since the rotating fan would still be in the way.

2. Research within the industry has indicated that vibration monitoring of vertical line shaft pumps has been of limited benefit for detecting mechanical degradation due to problems inherent with pump design. The OM Code imposes more stringent hydraulic acceptance criteria on these pumps than for centrifugal or positive displacement pumps. These more stringent hydraulic acceptance criteria place more emphasis on detection of degradation through hydraulic test data than through mechanical test data.

PROPOSED ALTERNATIVE AND BASIS: Vibration measurements will be taken in three orthogonal directions, one of which is in the axial direction in the area of the pump to motor mounting flange. This is the closest accessible location to a pump bearing housing and this location is easily accessible for test personnel which should ensure repeatable vibration data and should provide readings which are at least as representative of pump mechanical condition as those required by the Code.

Therefore, application of the OM Code hydraulic testing criteria along with radial and axial vibration monitoring in the area of the pump to motor mounting flange should provide adequate data for assessing the condition of the subject pumps and for monitoring degradation. This request is only applicable to Comprehensive Pump and Preservice Testing. The above proposed alternative provides reasonable assurance of operational readiness since vibration measurements will be taken in three orthogonal directions at the pump to motor mounting flange which will provide information as to the mechanical integrity of the pump. Based on the determination that compliance with the Code requirements results in a hardship without a compensating increase in the level of quality and safety, this proposed alternative should be granted pursuant to 10 CFR50.55a(z)(2).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-P-9 for the Fourth 10 Year IST Interval

REFERENCES: NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(2)
RR-P-11

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED:
1E11-C002A-D (Centrifugal Pumps) – Group A
1E21-C001A&B (Centrifugal Pumps) – Group B
2E11-C002A-D (Vertical Line Shaft Pumps) – Group A
2E21-C001A&B (Vertical Line Shaft Pumps) – Group B

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: Table ISTB-3520(b) requires that differential pressure be determined by the difference between the pressure at a point in the inlet pipe and the pressure at a point in the discharge pipe if a direct indicating instrument is not provided.

REASON FOR REQUEST: This alternative is a re-submittal of NRC approved 4th Interval relief request RR-P-11 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-P-11, is based on the ASME OM Code-2004 Edition with Addenda through OM-2006. There have been no substantive changes to this alternative, to the OM Code requirements or to the basis for use, which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-P-11).

The RHR and CS pumps are aligned to the suppression pool (torus) during all modes of normal plant operation. The installed suction pressure gauges do not meet Code requirements. Suction pressure to these pumps is primarily a function of suppression pool level, which is controlled within a 4 inch range, and this results in a virtually constant suction pressure. IST is performed utilizing a full flow test line which circulates water to and from the suppression pool.

The Plant's Technical Specifications require that the suppression pool be maintained within a narrow range of level, temperature, and internal pressure during plant operation which results in a suction pressure of approximately 5 psig. The Unit 1 and 2 Technical Specification operability limits for the suppression pool are itemized below:

Level	≥ 146" & ≤ 150"
Internal Pressure	≤ 1.75 psig
Water Temperature	≤ 100°F

These Technical Specification operability limits for the suppression pool result in a maximum difference in calculated pump suction pressure of < 2 psig.

This 2 psig variance (<u>ΔPi</u>) is insignificant in relation to nominal discharge pressure and the calculation of differential pressure (ΔP = Po-Pi) when considering the Group A pump test acceptable operating range (i.e., 95-110% for vertical line shaft pumps from Table ISTB-5200-1 and 90-110% for centrifugal pumps from Table ISTB-5100-1) and the allowable ± 2% instrument accuracy from Table ISTB-3500-1; or when considering the Group B pump test acceptable operating range (i.e., 90-110% for centrifugal and vertical line shaft pumps from Table ISTB-5100-1 and Table ISTB-5200-1) and the allowable ± 2% instrument accuracy from Table ISTB-3500-1. Therefore, direct suction pressure measurement for differential pressure derivation provides no added benefit for determining pump operational readiness or for monitoring pump degradation.

PUMP	**LOWEST REFERENCE DISCHARGE PRESSURE (Po)**	**MAXIMUM VARIANCE (ΔPi/Po)**
Unit 1 RHR	171 psig	1.17% max
Unit 1 CS	273 psig	0.73% max
Unit 2 RHR	180 psig	1.11% max
Unit 2 CS	332.3 psig	0.60% max

The following table summarizes several years' worth of IST pump suction pressure data. This summary confirms that the RHR and Core Spray pump's suction pressures are consistent and are relatively insignificant in comparison with the pumps' discharge pressure. Applying an average suction pressure of 5 psig, when calculating differential pressure, will provide data that is meaningful for assessing operational readiness and for monitoring pump degradation.

PUMP MPL No.	MIN. PRESS.	MAX. PRESS.	AVG. PRESS.	REFERENCE VALUES
1E11-C002A	3.9	6.8	5.1 (52)	Qr = 8000 gpm, ΔPr = 166 psid
1E11-C002B	3.2	6.25	4.8 (47)	Qr = 7700 gpm, ΔPr = 175 psid
1E11-C002C	3.0	6.2	4.8 (46)	Qr = 7700 gpm, ΔPr = 176 psid
1E11-C002D	3.4	6.0	4.6 (40)	Qr = 7700 gpm, ΔPr = 180 psid
1E21-C001A	2.5	5.8	4.1 (68)	Qr = 4620 gpm, ΔPr = 277.6 psid
1E21-C001B	1.7*	5.9	3.7 (47)	Qr = 4300 gpm, ΔPr = 268 psid

RR-P-11
(Cont.)

2E11-C002A	3.0	6.8	5.2 (50)	Qr = 7700 gpm, ΔPr = 184.6 psid
2E11-C002B	4.3	7.1	5.3 (48)	Qr = 7800 gpm, ΔPr = 190 psid
2E11-C002C	3.0	6.9	5.3 (55)	Qr = 7700 gpm, ΔPr = 184.9 psid
2E11-C002D	3.8	6.2	4.9 (47)	Qr = 7700 gpm, ΔPr = 175 psid
2E21-C001A	4.15	6.9	5.1 (43)	Qr = 4250 gpm, ΔPr = 327.3 psid
2E21-C001B	3.3	6.4	5.0 (53)	Qr = 4250 gpm, ΔPr = 330 psid
AVERAGE	3.3	6.4	4.9	N/A

Number in parenthesis "( )" indicates the number of test values averaged to get indicated value.
* One time occurrence only.

The permanently installed pump suction pressure gages encompass a wider range of pressures than does IST and thus exceed the OM Code allowable range limit (3 times the reference value). The installed RHR pump gages must account for the pressure experienced with the RHR loop in the shutdown cooling mode of operation. The installed CS pump gages must account for the pressure experienced with the CS suction aligned to the Condensate Storage Tank. Therefore, a temporary test gage which satisfies the Code range limits would have to be installed each time that IST is required.

Applying a constant pump suction pressure, when calculating differential pressure, will allow the Group A and B testing to be performed with the installed pressure gages, thus lessening the burden on operations personnel responsible for the testing. Since temporary test gages are required to be calibrated both prior to and after usage, it also eliminates the possibility of invalidating test data due to a gage being damaged during transportation, installation or removal. Mechanical degradation of centrifugal pumps which experience significant differences in suction (inlet) pressure would be indicated by changes in the differential pressure. However, for these pumps, the suction pressure variance is insignificant in comparison to the developed head (pressure).

Therefore, monitoring discharge pressure and calculating differential pressure assuming a constant 5 psig suction pressure provides an adequate method to determine operational readiness and detect potential degradation.

PROPOSED ALTERNATIVE AND BASIS: Pump suction pressure will be assumed to be 5 psig based on a review of several years of IST data which support suction pressure being virtually constant when performing Group A and Group B testing. During these tests pump differential pressure will be calculated by measuring pump discharge pressure and subtracting 5 psig. This value will then be compared to the corresponding reference value. The acceptance criteria of Tables ISTB-5100-I and ISTB-5200-1 will be applied for assessing pump operational readiness and for monitoring potential pump degradation during the applicable Group A or Group B pump test. This testing method meets the intent of the Code for monitoring pump operational readiness and degradation, and relieves the Licensee of the burden associated with the use of temporary test gages.

This request is not applicable to Comprehensive Pump or Preservice Testing.
The above proposed alternative provides an acceptable means of evaluating pump performance without a substantial decrease in the ability to monitor operational readiness. Based on the determination that compliance with the Code requirements, results in a hardship or unusual difficulty without a compensating increase in the level of quality and safety, this proposed alternative should be granted pursuant to 10 CFR 50.55a(z)(2).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-P-11 for the Fourth 10 Year IST Interval

REFERENCES: NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(1)
RR-P-12

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED: 2P41-C002 (Vertical Line Shaft Pump) - Group B

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: ISTB-3510(b)(1) requires that the full-scale range of analog instruments shall not be greater than three times the reference value, and Table ISTB-3500-1 requires an accuracy of ± 2% full scale.

REASON FOR REQUEST: This alternative is a re-submittal of NRC approved 4th Interval relief request RR-P-12 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-P-12, is based on the ASME OM Code-2004 Edition with Addenda through OM-2006. There have been no substantive changes to this alternative, to the OM Code requirements or to the basis for use, which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-P-12).

The flowrate for the Unit 2 Standby Diesel Generator Service Water (SDSW) pump is determined by measuring the differential pressure (dp), in inches of water, across a flow element and then using the vendor correlation chart to convert dp to flowrate in gallons-per-minute (gpm). The dp indicator (2P41-R383) has a full-scale range of -178 inches of water to + 178 inches (356 inches total range) of water, which is greater than three times the reference value, and is calibrated to ± 4 inches of water (i.e., ± 1.125% of full-scale). The indicator has a range which allows measurement of the flowrate in either direction across the flow element, thus the negative and positive scale ranges. The vendor supplied dp to flow correlation chart has a range of 50 ÷ 145 inches of water which corresponds to a flowrate range of 500 – 850 gpm.

The reference flow for this pump is 707 gpm which corresponds to 100 inches of water. The OM Code would allow a full-scale range of 0 - 300 inches of water (i.e., 3 X 100) and a calibration accuracy of ± 6.0 inches of water (i.e., 0.02 X 300).

The combined range and accuracy of the installed instruments is within the maximum allowable of ISTB-3510(b)(1) and Table-3500-1. The maximum Code

allowable dp variance would be ± 6.0 inches of water whereas the actual dp variance is ± 4.0 inches of water. Therefore, use of the existing dp indicators and the vendor correlation chart provides flowrate measurements for IST that are at least as accurate as required by the OM Code.

PROPOSED ALTERNATIVE AND BASIS: None, the installed instrumentation will be utilized to determine flowrate for the SDSW pump test. The use of this instrumentation is supported by the guidance contained in NRC NUREG-1482, Revision 2 Section 5.5.1, since the combined range and accuracy variance of the installed instrumentation is within the maximum allowable variance of the OM Code. This request applies to flowrate measurements for Group B, Comprehensive Pump, and Preservice Testing.

The above proposed alternative is acceptable since the accuracy of the instrumentation is better than the absolute accuracy required by the Code. Based on the determination that this alternative provides an acceptable level of quality and safety, the proposed alternative should be granted pursuant to 10 CFR 50.55a(z)(1).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-P-12 for the Fourth 10 Year IST Interval

REFERENCES: NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(1)
RR-P-13

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED: 1(2)C41-C001A&B, 1(2)E11-C001A-D, 1(2)E11-C002A-D, 1(2)E21-C002A&B, 1(2)E41-C001, 1(2)P41-C001A-D, 2P41-C002.
Augmented Components:
1(2)E51-C001, 1Y52-C001A-C, 1Y52-C101A&B, 2Y52-C001A&C, 2Y52-C101A&C.

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: Applicable Code Requirements:

ISTB-5121, "Group A Test Procedure" ISTB-5121(b) states that "The resistance of the system shall be varied until the flow rate equals the reference point".

ISTB-5122, "Group B Test Procedure" ISTB-5122(c) states that "System resistance may be varied as necessary to achieve the reference point".

ISTB-5123, "Comprehensive Test Procedure" ISTB-5123(b) states that "For centrifugal and vertical line shaft pumps, the resistance of the system shall be varied until the flow rate equals the reference point".

ISTB-5221, "Group A Test Procedure" ISTB-5221(b) states that "The resistance of the system shall be varied until the flow rate equals the reference point".

ISTB-5222, "Group B Test Procedure" ISTB-5222(c) states that "System resistance may be varied as necessary to achieve the reference point".

ISTB-5223, "Comprehensive Test Procedure" ISTB-5123(b) states that "For centrifugal and vertical line shaft pumps, the resistance of the system shall be varied until the flow rate equals the reference point"

REASON FOR REQUEST: Pursuant to 10 CFR 50.55a, "Codes and standards," paragraph (z)(1), an alternative is proposed to the pump testing reference value requirements of the ASME OM Code. The basis of the request is that the proposed alternative would provide an acceptable level of quality and safety. Specifically, this alternative is requested for all inservice testing of IST

Program pumps listed in this Relief Request.

For pump testing, there is difficulty adjusting system throttle valves with sufficient precision to achieve exact flow reference values during subsequent IST exams. Section ISTB of the ASME OM Code does not allow for variance from a fixed reference value for pump testing.

However, NUREG-1482, Revision 2, Section 5.3, acknowledges that certain pump system designs do not allow for the licensee to set the flow or pressure at an exact value because of limitations in the instruments and controls for maintaining steady flow.

ASME OM Code Case OMN-21 provides guidance for adjusting reference flow/pressure to within a specified tolerance during Inservice Testing. The Code Case states "It is the opinion of the Committee that when it is impractical to operate a pump at a specified reference point and adjust the resistance of the system to a specified reference point for either flow rate, differential pressure or discharge pressure, the pump may be operated as close as practical to the specified reference point with the following requirements. The Owner shall adjust the system resistance to as close as practical to the specified reference point where the variance from the reference point does not exceed + 2% or - 1% of the reference point when the reference point is flow rate, or + 1% or – 2% of the reference point when the reference point is differential pressure or discharge pressure. The NRC also discusses this ASME Code change in NUREG-1482, Revision 2, Section 5.3.

PROPOSED ALTERNATIVE AND BASIS:

Hatch seeks to perform future Pump Inservice Testing in a manner consistent with the requirements as stated in ASME OM Code Case OMN-21. Specifically, testing of all centrifugal pumps identified in this Relief Request will be performed such that flow rate is adjusted as close as practical to the reference value and within proceduralized limits of +2% / -1% of the reference value. For positive displacement pumps the discharge pressure will be adjusted as close as practical to the reference value and within proceduralized limits of +1% / -2% of the reference value.

Hatch plant operators will still strive to achieve the exact test reference values during testing. Typical test guidance will be to adjust flow/pressure to the specific reference value with additional guidance that if the reference value cannot be achieved with reasonable effort the test will be considered valid if the steady state flow rate is within the proceduralized limits of +2% / -1% of the reference value or discharge pressure within proceduralized limits of +1% / -2% of the reference value.

Using the provisions of this request as an alternative to the specific requirements of ISTB-5121, ISTB-5122, ISTB-5123, ISTB-5221, ISTB-

5222 and ISTB-5223 as described above will provide adequate indication of pump performance and continue to provide an acceptable level of quality and safety. Based on the determination that the use of controlled reference value ranges provides an acceptable level of quality and safety, this proposed alternative should be granted pursuant to 10 CFR 50.55a(z)(1).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: Callaway Relief Request PR-06, approved by the NRC via letter dated July 15, 2014 (ML14178A769)

REFERENCES:

1. ASME Code Case OMN-21, "Alternate Requirements for Adjusting Hydraulic Parameters to Specified Reference Points"

2. NUREG-1482, Revision 2, Section 5.3 "Allowable Variance from Reference Points and Fixed-Resistance Systems"

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED RELIEF IN ACCORDANCE WITH 10 CFR 50.55a(f)(6)(i)
RR-V-1

PLANT/UNIT:	Edwin I Hatch Nuclear Plant/Unit 1 and 2.
INTERVAL:	5th Interval beginning January 1, 2016 and ending December 31, 2025.
COMPONENTS AFFECTED:	1C11-F010A&B, 1C11-F011, 1C11-F035A&B, 1C11-F037
CODE EDITION AND ADDENDA:	ASME OM Code-2004 Edition with Addenda through OM-2006
REQUIREMENTS:	Establish limiting values of valve stroke time and measure individual valve stroke time per ISTC-5131 (b).
REASON FOR REQUEST:	This relief request is a re-submittal of NRC approved 4th Interval relief request RR-V-1 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-V-1, is based on the ASME OM Code-2004 Edition with Addenda through OM-2006. There have been no substantive changes to the OM Code requirements or to the basis for use, which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-V-1).

A limiting value of stroke time cannot be specified for the air operated scram discharge volume vent and drain valves and they cannot be individually stroked and timed. In order to prevent water hammer induced damage to the system during a full CRD scram, plant Technical Specifications require that system valve operation is adjusted so that the outboard vent and drain valves (F035A&B, F037) fully close at least five seconds after each respective inboard vent and drain valve (F010A&B, F011). All valves must be fully closed in less than forty-five (45) seconds.

Additionally, the system is adjusted so that the inboard vent and drain valves (F010A&B, F011) start to open at least five seconds after each respective outboard vent and drain valve (F035A&B, F037) upon reset of a full core scram. The valves are not equipped with individual valve control switches and cannot be individually stroke timed. Because of the adjustable nature of the valve control system, individual valve stroke timing would not provide any meaningful information for monitoring valve degradation. System design prevents stroke timing these valves during normal operation without disabling the Reactor Protection System Scram Signal to the valves. Disabling this signal requires the installation of electrical jumpers and the opening of links in energized control circuits which increase the potential for a Reactor Scram.

PROPOSED RELIEF AND BASIS: The valves will be exercised quarterly but not timed. Additionally, the total valve sequence response time will be verified to be less than Technical Specifications requirements during each refueling outage when a complete stroke time test is performed. The above proposed relief provides a reasonable assurance of operational readiness since the valves will be exercised quarterly and total valve response time will be tested each refueling outage. Based on the impracticality of performing testing in accordance with the Code requirements, and in consideration of the burden on SNC if the Code requirements were imposed, this proposed relief should be granted pursuant to 10 CFR 50.55a(f)(6)(i).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-V-1 for the Fourth 10 Year IST Interval

REFERENCES: NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED RELIEF IN ACCORDANCE WITH 10 CFR 50.55a(f)(6)(i)
RR-V-2

PLANT/UNIT:	Edwin I Hatch Nuclear Plant/Unit 1 and 2.
INTERVAL:	5th Interval beginning January 1, 2016 and ending December 31, 2025.
COMPONENTS AFFECTED:	1C51-Shear A,B,C,D and 2C51-Shear A,B,C,D
CODE EDITION AND ADDENDA:	ASME OM Code-2004 Edition with Addenda through OM-2006
REQUIREMENTS:	ISTC-3620 requires Category A containment isolation valves to be periodically leak tested per the 10 CFR 50 Appendix J Program.
REASON FOR REQUEST:	This relief request is a re-submittal of NRC approved 4th Interval relief request RR-V-2 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-V-2, is based on the ASME OM Code-2004 Edition with Addenda through OM-2006. There have been no substantive changes to the OM Code requirements or to the basis for use which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-V-2). These valves are explosive actuated shear valves. The shear valve isolates the TIP tubing by shearing the tube and TIP drive cable, and by jamming the sheared ends of the tubing into a teflon coating on the shear valve disc. Thus the shear valves cannot be local leak rate tested without destroying the drive tube.
PROPOSED RELIEF AND BASIS:	Each lot of shear valves is sample leakage tested by the manufacturer prior to delivery. This sample leakrate testing satisfies the requirements of the Plant Hatch 10 CFR 50, Appendix J Leakrate Program. These valves are also tested in accordance with ISTC-5260 as explosive actuated valves. The above proposed relief provides a reasonable assurance of operational readiness since the manufacturer's testing is conducted on each shear valve and this testing meets the requirements of Appendix J and the testing requirements of Appendix J provide an adequate assessment of leaktightness for containment isolation valves. Based on the impracticality of performing testing in accordance with the Code requirements, and in consideration of the burden on SNC if the Code requirements were imposed, this proposed relief should be granted pursuant to 10 CFR 50.55a(f)(6)(i).

RR-V-2 (cont.)

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-V-2 for the Fourth 10 Year IST Interval

REFERENCES: NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED RELIEF IN ACCORDANCE WITH 10 CFR 50.55a(f)(6)(i)
RR-V-3

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED: 1P41-F035A&B, 1P41-F036A&B, 1P41-F037A-D, 1P41-F039A&B, 2P41-F035A&B, 2P41-F036A&B, 2P41-F037A-D, 2P41-F039A&B, 2P41-F340

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: ISTC-3530 requires verification of valve obturator movement by observing an appropriate indicator, such as indicator lights, or by observing other evidence, such as changes in system pressure, flow rate, level, or temperature, that reflects changes in obturator movement.

REASON FOR REQUEST: This relief request is a re-submittal of NRC approved 4th Interval relief request RR-V-3 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-V-3, is based on the ASME OM Code-2004 Edition with Addenda through OM-2006. There have been no substantive changes to the OM Code requirements or to the basis for use which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-V-3.

These valves are normally closed, fail open air operated valves which have a safety function to open and provide cooling water flow to the associated safety related equipment. System design did not provide indicating lights, instrumentation or direct valve control switches. All valves receive an open signal upon initiation of the associated equipment and a close signal upon termination of operation of the associated equipment. Verification of obturator movement and stroke time measurement can only be performed by observation of actual stem movement for all valves.

PROPOSED RELIEF AND BASIS: Verification of obturator movement and measurement of valve stroke time will be performed by observing actual valve stem movement. Stroke time will be considered to be the time from start to stop of valve stem movement. Each valve is equipped with either a stem mounted pointer and a yoke mounted position indicating scale, or a percent open/closed indicator. This position indicating device will be observed during stroke timing to determine full open/full close operation. The requirements of ISTC-5130 will be applied to monitor valve degradation.

RR-V-3 (Cont.)

The above proposed relief provides a reasonable assurance of operational readiness since the actual stroke time of the valve movement is being measured in a repeatable manner and full stroke exercise of the valves is verified. Based on the impracticality of performing testing in accordance with the Code requirements, and in consideration of the burden on SNC if the Code requirements were imposed, this proposed relief should be granted pursuant to 10 CFR 50.55a(f)(6)(i).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-V-3 for the Fourth 10 Year IST Interval

REFERENCES: NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(1)
RR-V-5

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED: 1E41-D003, 1E41-D004, 2E41-D003, 2E41-D004

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: ASME OM Code, 2004 Edition, Appendix I, paragraph 1-1360 requires Class 2 and 3 nonreclosing pressure relief devices (rupture discs) to be replaced every 5 years, unless historical data indicates a requirement for more frequent replacement.

REASON FOR REQUEST: This alternative is a re-submittal of NRC approved 4th Interval relief request RR-V-5 that was based on the ASME OM Code-2001 Edition. This 5th Interval request for relief, RR-V-5, is based on the ASME OM Code-2004 Edition with Addenda through OM-2006. There have been no substantive changes to this alternative, to the OM Code requirements or to the basis for use, which would alter the previous NRC Safety Evaluation conclusions. (See References for SER date and TAC numbers associated with RR-V-5).

The subject rupture discs are supplied by Continental Disc Corporation. Southern Nuclear Operating Company requested the supplier to perform cyclic testing, to destruction, of a disc that had previously been installed in the HPCI system at Plant Hatch. The test disc was installed in an appropriate disc holder and flange assembly which simulated the installed configuration. The rupture disc assembly was cycled from full vacuum to 70% of the ambient burst pressure (219 psig). The cycle testing was conducted at ambient room temperature. Since a rupture disc is a differential pressure relief device, cycling conditions were achieved by placing a constant 15 psig pressure on the downstream side of the rupture disc and cycling the upstream pressure from zero to 70% of the ambient burst pressure plus 15 psig. The 15 psig added to the upstream cycling pressure compensates for the constant 15 psig pressure on the downstream side. An electronic counter recorded each cycle. The test disc completed 2,788 cycles before failure occurred. The rupture disc burst in the normal fashion as with disc of this design.

The HPCI system is typically tested every 3 months, but for conservatism a test frequency of each month will be assumed. Monthly testing would result in approximately 72 tests during 3 operating cycles (i.e., 72 months). To meet the Code 5-year replacement frequency, the disc must be replaced every 2nd refueling outage (48 months) or after approximately 48 HPCI system tests. Therefore, a change from

replacement every 48 months to every 72 months is insignificant when compared to the expected life of the disc as proven by the number of cycles required for disc rupture by vendor testing.

Plant Hatch operates on a 24-month fuel cycle. Replacement every 6 years results in replacement every 3rd refueling outage whereas a 5-year replacement results in replacement every 2nd refueling outage. Extension of the replacement frequency by 1-year will coincide with the fuel cycle for Plant Hatch.

PROPOSED ALTERNATIVE AND BASIS: The subject rupture discs will be replaced at least once every 3rd refueling outage, corresponding to once every 6 years.

As proven by the vendor testing, the subject rupture discs have adequate margin for operation well beyond the requested 6-year replacement frequency. Therefore, the proposed alternative provides an acceptable level of quality and safety and should be granted pursuant to 10CFR50.55a(z)(1).

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS: This Relief Request was approved as RR-V-5 for the Fourth 10 Year IST Interval

REFERENCES: NRC Safety Evaluation dated February 14, 2006 - TAC Nos. MC6837, MC6838, MC7626 and MC7627

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(2)
RR-V-8

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED: Pumps and Valves contained within the Inservice Testing Program scope

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OMb-2006

REQUIREMENTS: This request applies to the following exam frequency requirements of the ASME OM Code.

ISTA-3120(a)- "The frequency for the inservice testing shall be in accordance with the requirements of Section IST."

ISTB-3400-	Frequency of Inservice Tests
ISTC-3510-	Exercising Test Frequency
ISTC-3540-	Manual Valves
ISTC-3630(a)-	Frequency
ISTC-3700-	Position Verification Testing
ISTC-5221(c)(3)-	"At least one valve from each group shall be disassembled and examined at each refueling outage; all valves in a group shall be disassembled and examined at least once every 8 years."

Appendix I, 1-1320- Test Frequencies, Class 1 Pressure Relief Valves
Appendix I, 1-1330- Test Frequencies, Class 1 Nonreclosing Pressure Relief Devices
Appendix I, 1-1340- Test Frequencies- Class 1 Pressure Relief Devices That Are Used for Thermal Relief Application
Appendix I, 1-1350- Test Frequencies- Class 2 and 3 Pressure Relief Valves
Appendix I, 1-1360- Test Frequencies- Class 2 and 3 Nonreclosing Pressure Relief Devices
Appendix I, 1-1370- Test Frequencies- Class 2 and 3 Primary Containment Vacuum Relief Valves
Appendix I, 1-1380- Test Frequencies- Class 2 and 3 Vacuum Relief Valves Except for Primary Containment Vacuum Relief Valves
Appendix I, 1-1390- Test Frequencies- Class 1 Pressure Relief Devices That Are Used for Thermal Relief Application
Appendix II, II-4000(a)(1)- Performance Improvement Activities Interval
Appendix II, II-4000(b)(1)(e)- Optimization of Condition Monitoring Activities Interval

REASON FOR REQUEST:

Pursuant to 10 CFR 50.55a, "Codes and Standards," paragraph (z)(2), relief is requested from the frequency specifications of the ASME OM Code. The basis of the relief request is that the Code requirement presents an undue hardship without a compensating increase in the level of quality or safety.

ASME OM Code Section IST establishes the inservice test frequency for all components within the scope of the Code. The frequencies (e.g., quarterly) have always been interpreted as "nominal" frequencies (generally as defined in the Table 3.2 of NUREG 1482, Revision 1) and Owners routinely applied the surveillance extension time period (i.e., grace period) contained in the plant Technical Specifications (TS) Surveillance Requirements (SRs). The TS typically allow for a less than or equal to 25% extension of the surveillance test interval to accommodate plant conditions that may not be suitable for conducting the surveillance (SR 3.0.2). However, regulatory issues have been raised concerning the applicability of the TS "Grace Period" to ASME OM Code required inservice test frequencies irrespective of allowances provided under TS Administrative Controls (i.e., TS 5.5.6, "Inservice Testing Program," invokes SR 3.0.2 for various OM Code frequencies).

The lack of a tolerance band on the ASME OM Code inservice test frequency restricts operational flexibility. There may be a conflict where a surveillance test could be required but where it is not possible or not desired that it be performed until sometime after a certain restricted plant condition is cleared . Therefore, to avoid this conflict, the surveillance test should be performed as soon as it is practicable. The NRC recognized this potential issue in the TS by allowing a frequency tolerance as described in TS SR 3.0.2. The lack of a similar tolerance applied to OM Code testing places an unusual hardship on the plant to adequately schedule work tasks without operational flexibility.

Thus, just as with TS required surveillance testing, some tolerance is needed to allow extending OM Code testing intervals. Interval extension is to facilitate test scheduling and considers plant operating conditions that may not be suitable for performance of the required testing (e.g., performance of the test would cause an unacceptable increase in the plant risk profile due to transient conditions or other ongoing surveillance, test or maintenance activities). Such extensions are not intended to be used repeatedly merely as an operational convenience to extend test intervals beyond those specified.

PROPOSED ALTERNATIVE AND BASIS:

ASME OM Code establishes component test frequencies that are based either on elapsed time periods (e.g., quarterly, 2 years, etc.) or on the occurrence of plant conditions or events (e.g., cold shutdown, refueling outage, upon detection of a sample failure, following maintenance, etc.).

a. Components whose test frequencies are based on elapsed time periods shall undergo Inservice Testing at frequencies as specified in the Hatch Technical Specifications (TS 5.5.6) and shown in the following table:

Frequency	Specified Time Period Between Tests
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly	At least once per 92 days
Semiannually	At least once per 184 days
Yearly or Annually	At least once per 366 days

b. The specified time period between tests may be extended as follows:

  i. For periods specified as less than 2 years, the period may be extended by up to 25% for any given test.

  ii. For periods specified as greater than or equal to 2 years, the period may be extended by up to 6 months for any given test.

c. Components whose test frequencies are based on the occurrence of plant conditions or events (e.g., cold shutdown, refueling outage, upon detection of a sample failure, following maintenance, etc.) may not have their period between tests extended except as allowed by the ASME OM Code.

d. Period extensions may not be applied to the test frequency requirements specified in Subsection ISTD, Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-water Reactor Nuclear Power Plants, as Subsection ISTD contains its own rules for period extensions.

e. Period extensions of 25% may also be applied to accelerated test frequencies (e.g., pumps in Alert Range) and other less than two year test frequencies not specified in the table above.

This relief is requested citing the guidance found in ASME approved Code Case OMN-20. Based on the determination that compliance with the Code requirement results in a hardship without a compensating increase in the level of quality and safety, this proposed alternative should be granted pursuant to 10 CFR50.55a(z)(2).

RR-V-08 (Cont.)

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS:

1. Quad Cities Relief Request RV-01 - SER dated 2/14 2013 (ML 13042A348)
2. Callaway Relief Request PR-04, SER dated 7/15/2014 (ML14178A769)
3. Calvert Cliffs Relief Request IST-RR-01 - approved in NRC Safety Evaluation dated 9/29/2014 (ML14247A555)
4. TMI Relief Request VR-02 – SER dated 8/15/2013 (ML13227A024)
5. Dresden Relief Request RV-01 – SER dated 10/31/2013 (ML13297A515)

REFERENCES:

1. NRC Regulatory Issue Summary 2012-10- "NRC STAFF POSITION ON APPLYING SURVEILLANCE REQUIREMENTS 3.0.2 AND 3.0.3 TO ADMINISTRATIVE CONTROLS PROGRAM TESTS
2. ASME OM Code Case OMN-20- "Inservice Test Frequency"

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(1)
RR-V-9

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED:

Comp ID	Unit	Code Class	Cat	Comp ID	Unit	Code Class	Cat
1B21-F015A	1	1	A/C	1B21-F051D	1	1	A/C
1B21-F015B	1	1	A/C	1B21-F053A	1	1	A/C
1B21-F015C	1	1	A/C	1B21-F053B	1	1	A/C
1B21-F015D	1	1	A/C	1B21-F053C	1	1	A/C
1B21-F015E	1	1	A/C	1B21-F053D	1	1	A/C
1B21-F015F	1	1	A/C	1B21-F055	1	1	A/C
1B21-F015G	1	1	A/C	1B21-F057	1	1	A/C
1B21-F015H	1	1	A/C	1B21-F059A	1	1	A/C
1B21-F015J	1	1	A/C	1B21-F059B	1	1	A/C
1B21-F015K	1	1	A/C	1B21-F059C	1	1	A/C
1B21-F015L	1	1	A/C	1B21-F059D	1	1	A/C
1B21-F015M	1	1	A/C	1B21-F059E	1	1	A/C
1B21-F015N	1	1	A/C	1B21-F059F	1	1	A/C
1B21-F015P	1	1	A/C	1B21-F059G	1	1	A/C
1B21-F015R	1	1	A/C	1B21-F059H	1	1	A/C
1B21-F015S	1	1	A/C	1B21-F059L	1	1	A/C
1B21-F041	1	1	A/C	1B21-F059M	1	1	A/C
1B21-F043A	1	1	A/C	1B21-F059N	1	1	A/C
1B21-F043B	1	1	A/C	1B21-F059P	1	1	A/C
1B21-F045A	1	1	A/C	1B21-F059R	1	1	A/C
1B21-F045B	1	1	A/C	1B21-F059S	1	1	A/C
1B21-F047A	1	1	A/C	1B21-F059T	1	1	A/C
1B21-F047B	1	1	A/C	1B21-F059U	1	1	A/C
1B21-F049A	1	1	A/C	1B21-F061	1	1	A/C
1B21-F049B	1	1	A/C	1B31-F003A	1	1	A/C
1B21-F051A	1	1	A/C	1B31-F003B	1	1	A/C
1B21-F051B	1	1	A/C	1B31-F004A	1	1	A/C
1B21-F051C	1	1	A/C	1B31-F004B	1	1	A/C

RR-V-9 (Cont.)

COMPONENTS AFFECTED:

Comp ID	Unit	Code Class	Cat
1B31-F009A	1	1	A/C
1B31-F009B	1	1	A/C
1B31-F009C	1	1	A/C
1B31-F009D	1	1	A/C
1B31-F010A	1	1	A/C
1B31-F010B	1	1	A/C
1B31-F010C	1	1	A/C
1B31-F010D	1	1	A/C
1B31-F011A	1	1	A/C
1B31-F011B	1	1	A/C
1B31-F011C	1	1	A/C
1B31-F011D	1	1	A/C
1B31-F012A	1	1	A/C
1B31-F012B	1	1	A/C
1B31-F012C	1	1	A/C
1B31-F012D	1	1	A/C
1B31-F040A	1	1	A/C
1B31-F040B	1	1	A/C
1B31-F040C	1	1	A/C
1B31-F040D	1	1	A/C
1E21-F018A	1	1	A/C
1E21-F018B	1	1	A/C
1E21-F018C	1	1	A/C
1E41-F024A	1	1	A/C
1E41-F024B	1	1	A/C
1E41-F024C	1	1	A/C
1E41-F024D	1	1	A/C
1E51-F044A	1	1	A/C
1E51-F044B	1	1	A/C
1E51-F044C	1	1	A/C
1E51-F044D	1	1	A/C
2B21-F041	2	1	A/C
2B21-F043A	2	1	A/C
2B21-F043B	2	1	A/C
2B21-F045A	2	1	A/C
2B21-F045B	2	1	A/C

Comp ID	Unit	Code Class	Cat
2B21-F047A	2	1	A/C
2B21-F047B	2	1	A/C
2B21-F049A	2	1	A/C
2B21-F049B	2	1	A/C
2B21-F051A	2	1	A/C
2B21-F051B	2	1	A/C
2B21-F051C	2	1	A/C
2B21-F051D	2	1	A/C
2B21-F053A	2	1	A/C
2B21-F053B	2	1	A/C
2B21-F053C	2	1	A/C
2B21-F053D	2	1	A/C
2B21-F055	2	1	A/C
2B21-F057	2	1	A/C
2B21-F059A	2	1	A/C
2B21-F059B	2	1	A/C
2B21-F059C	2	1	A/C
2B21-F059D	2	1	A/C
2B21-F059E	2	1	A/C
2B21-F059F	2	1	A/C
2B21-F059G	2	1	A/C
2B21-F059H	2	1	A/C
2B21-F059L	2	1	A/C
2B21-F059M	2	1	A/C
2B21-F059N	2	1	A/C
2B21-F059P	2	1	A/C
2B21-F059R	2	1	A/C
2B21-F059S	2	1	A/C
2B21-F059T	2	1	A/C
2B21-F059U	2	1	A/C
2B21-F061	2	1	A/C
2B21-F070A	2	1	A/C
2B21-F070B	2	1	A/C
2B21-F070C	2	1	A/C
2B21-F070D	2	1	A/C
2B21-F071A	2	1	A/C

RR-V-9 (Cont.)

COMPONENTS AFFECTED:

Comp ID	Unit	Code Class	Cat	Comp ID	Unit	Code Class	Cat
2B21-F071B	2	1	A/C	2B31-F011A	2	1	A/C
2B21-F071C	2	1	A/C	2B31-F011B	2	1	A/C
2B21-F071D	2	1	A/C	2B31-F011C	2	1	A/C
2B21-F072A	2	1	A/C	2B31-F011D	2	1	A/C
2B21-F072B	2	1	A/C	2B31-F012A	2	1	A/C
2B21-F072C	2	1	A/C	2B31-F012B	2	1	A/C
2B21-F072D	2	1	A/C	2B31-F012C	2	1	A/C
2B21-F073A	2	1	A/C	2B31-F012D	2	1	A/C
2B21-F073B	2	1	A/C	2B31-F040A	2	1	A/C
2B21-F073C	2	1	A/C	2B31-F040B	2	1	A/C
2B21-F073D	2	1	A/C	2B31-F040C	2	1	A/C
2B31-F003A	2	1	A/C	2B31-F040D	2	1	A/C
2B31-F003B	2	1	A/C	2E21-F018A	2	1	A/C
2B31-F004A	2	1	A/C	2E21-F018B	2	1	A/C
2B31-F004B	2	1	A/C	2E21-F018C	2	1	A/C
2B31-F009A	2	1	A/C	2E41-F024A	2	1	A/C
2B31-F009B	2	1	A/C	2E41-F024B	2	1	A/C
2B31-F009C	2	1	A/C	2E41-F024C	2	1	A/C
2B31-F009D	2	1	A/C	2E41-F024D	2	1	A/C
2B31-F010A	2	1	A/C	2E51-F044A	2	1	A/C
2B31-F010B	2	1	A/C	2E51-F044B	2	1	A/C
2B31-F010C	2	1	A/C	2E51-F044C	2	1	A/C
2B31-F010D	2	1	A/C	2E51-F044D	2	1	A/C

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: ISTC-3522, "Category C Check Valves"

(a) "During operation at power, each check valve shall be exercised or examined in a manner that verifies obturator travel by using the methods in ISTC-5221."

(c) "If exercising is not practicable during operation at power and cold shutdown, it shall be performed during refueling outages."

ISTC- 3700, "Position Verification Testing"

> "Valves with remote position indicators shall be observed locally at least once every 2 years to verify that valve operation is accurately indicated."

REASON FOR REQUEST: Pursuant to 10 CFR 50.55a, "Codes and Standards", paragraph (z)(1), relief is requested from the requirements of ASME OM Code ISTC-3522 and ISTC-3700 for the subject valves. The basis of the Relief Request is that the proposed alternative would provide an acceptable level of quality and safety.

PROPOSED ALTERNATIVE AND BASIS: Excess flow check valves will be tested on a representative sample basis at the frequency specified in Technical Specifications Surveillance Requirement 3.6.1.3.8.

Functional testing with verification that flow is checked will be performed per Technical Specification 3.6.1.3.8 during refueling outages. Surveillance Requirement 3.6.1.3.8 allows a "representative sample" of EFCVs to be tested every refueling outage, such that each EFCV will be individually tested approximately every ten years. The sample groups are representative of the various plant configurations, models, sizes, and operating environments.

The EFCVs have position indication in the control room. Check valve remote position indication is excluded from Regulatory Guide 1.97 as a required parameter for evaluating containment isolation. The remote position indication will be verified accurate at the same frequency as the functional test prescribed in Technical Specification Surveillance Requirement 3.6.1.3.8. Although inadvertent actuation of an EFCV during operation is highly unlikely due to the spring poppet design, Plant Hatch monitors the EFCVs indications on a daily basis as part of the Operations routine. Corrective Action documents are initiated for any EFCVs with abnormal position indication displays and repairs are scheduled for the next refueling outage.

Excess flow check valves are provided in each instrument process line that is part of the reactor coolant pressure boundary. The excess flow check valve is designed so that it will not close accidentally during normal operation, will close if a rupture of the instrument line occurs downstream of the valve, and can be reopened, when appropriate, after a closure.

As detailed in FSAR 5.2.2.5.4, Plant Hatch has incorporated into the design of each instrument source line a 0.25-inch restricting orifice as close to the RPV as possible. This is a redundant design feature which, along with the EFCV, will limit leakage to a level where the integrity and functional performance of the secondary containment and its associated air treatment systems (e.g., filters and the standby gas treatment system) are maintained. The coolant loss is well within the capabilities of the reactor coolant makeup system, and the potential offsite exposure is substantially below the guidelines of 10CFR100.

PROPOSED ALTERNATIVE AND BASIS (Cont.):

Additionally, the design and installation of the excess flow check valves at Plant Hatch follow the guidance of Regulatory Guide 1.11.

Testing the subject valves quarterly or during cold shutdown is not practicable, based on plant conditions. These valves have been successfully tested throughout the life of Plant Hatch and they have shown no degradation or other signs of aging.

The technology for testing these valves is simple and has been demonstrated effectively during the operating history of Plant Hatch. The basis for this alternative is that testing a sample of EFCVs each refueling outage provides a level of safety and quality equivalent to that of the Code-required testing.

Excess flow check valves are required to be tested in accordance with ISTC-3522, which requires exercising check valves nominally every three months to the positions required to perform their safety functions. ISTC-3522(c) permits deferral of this requirement to every reactor refueling outage. Excess flow check valves are also required to be tested in accordance with ISTC-3700, which requires remote position indication verification at least once every 2 years.

10CFR50 Appendix J testing is only applicable to EFCVs if they perform a containment isolation function. EFCVs are not required to close in response to a containment isolation signal and are not required to operate under post-LOCA conditions. As discussed in Reference 2, the functioning of EFCVs is not necessary to remain within 10CFR100 limits. Consequently, for purpose of 10CFR50, Appendix J, CIV testing, EFCVs do not provide a containment isolation function and are exempt from consideration under Appendix J.

The testing described above requires removal of the associated instrument or instruments from service. Since these instruments are in use during plant operation, removal of any of these instruments from service may cause a spurious signal, which could result in a plant trip or an unnecessary challenge to safety systems. Additionally, process liquid will be contaminated to some degree, requiring special measures to collect flow from the vented instrument side and also will contribute to an increase in personnel radiation exposure.

Testing on a cold shutdown frequency is impractical considering the large number of valves to be tested and the locations in which the test fixtures must be located. Considering the number of valves to be tested and the conditions required for testing, it is also a hardship to test all of these valves during refueling outages. Improvements in refueling outage schedules have minimized the time that is planned for refueling and testing activities during the outages.

The excess flow check valve is a simple and reliable device. The major components are a poppet and spring. The spring holds the poppet open under static conditions. The valve will close upon sufficient differential pressure across the poppet.

PROPOSED ALTERNATIVE AND BASIS (Cont.):

Functional testing of the valve is accomplished by venting the instrument side of the valve. The resultant increase in flow imposes a differential pressure across the poppet, which compresses the spring and decreases flow through the valve.

Industry experience as documented in GE Nuclear Energy topical report NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," indicates the EFCVs have a very low failure rate. The report indicates similarly that many reported test failures at other plants were related to test methodologies and not actual EFCV failures. In addition, the SER for that report assumed a 5 fold increase in failure rate to account for any potential aging influence and the resultant failure potential over 10 years was still found to not be significant. Test history at Plant Hatch shows a very low failure rate and no evidence of common mode failure, which is consistent with the findings of the NEDO report. The EFCVs at Plant Hatch, consistent with the industry, have exhibited a high degree of reliability, availability, and provide an acceptable level of quality and safety.

The Plant Hatch 2 Technical Specifications detail what frequency is required to maintain a high degree of reliability and availability and as an alternative will provide an acceptable level of quality and safety. Therefore, Southern Nuclear Co. requests relief pursuant to 10CFR50.55a (z)(1) to test excess flow check valves on a representative sample basis and at the frequency specified in Plant Hatch Technical Specifications Surveillance Requirements (SR) 3.6.1.3.8.

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS:

1. Fermi 2 Relief Request VRR-011 (SER Sept. 2010 ML102360570)
2. Susquehanna Steam Electric Station Unit 2 – SER dated 4/11/2001 (ML010960041)
3. Nine Mile Point Nuclear Station – SER dated 9/17/2001 (ML012340462).

REFERENCES:

1. NRC Regulatory Guide 1.11, "INSTRUMENT LINES PENETRATING THE PRIMARY REACTOR CONTAINMENT"
2. GE Nuclear Energy topical report NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation" as evaluated in SER dated 3/14/2000 (ML003729011)

SOUTHERN NUCLEAR OPERATING COMPANY
IST PROGRAM
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(z)(1)
RR-V-10

PLANT/UNIT: Edwin I Hatch Nuclear Plant/Unit 1 and 2.

INTERVAL: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

COMPONENTS AFFECTED:

Valve No.	Description	ASME Class
1E11-F008	RHR SDC Suction Outtboard Isol. Valve	1
1E11-F009	RHR SDC Suction Inboard Isol. Valve	1
1E11-F015A	LPCI Inboard Isol. Valve	1
1E11-F015B	LPCI Inboard Isol. Valve	1
1E11-F050A	LPCI Injection Check Valve	1
1E11-F050B	LPCI Injection Check Valve	1
1E11-F122A	RHR F050A Bypass Valve	1
1E11-F122B	RHR F050B Bypass Valve	1
1E21-F005A	CS Injection Inboard Valve	1
1E21-F005B	CS Injection Inboard Valve	1
1E21-F006A	CS Injection Check Valve	1
1E21-F006B	CS Injection Check Valve	1
1E21-F037A	CS F006A Bypass Valve	1
1E21-F037B	CS F006B Bypass Valve	1
1E41-F005	HPCI Injection Check Valve	2
1E41-F006	HPCI Injection Outboard Isol. Valve	2
1E51-F013	RCIC Injection Outboard Isol. Valve	2
1E51-F014	RCIC Injection Check Valve	2
2E11-F008	RHR SDC Suction Outboard Isol. Valve	1
2E11-F009	RHR SDC Suction Inboard Isol. Valve	1
2E11-F015A	LPCI Inboard Isol. Valve	1
2E11-F015B	LPCI Inboard Isol. Valve	1
2E11-F050A	LPCI Injection Check Valve	1
2E11-F050B	LPCI Injection Check Valve	1
2E11-F122A	RHR F050A Bypass Valve	1
2E11-F122B	RHR F050B Bypass Valve	1
2E21-F005A	CS Injection Inboard Valve	1
2E21-F005B	CS Injection Inboard Valve	1
2E21-F006A	CS Injection Check Valve	1
2E21-F006B	CS Injection Check Valve	1
2E21-F037A	CS F006A Bypass Valve	1
2E21-F037B	CS F006B Bypass Valve	1
2E41-F005	HPCI Injection Check Valve	2
2E41-F006	HPCI Injection Outboard Isol. Valve	2
2E51-F013	RCIC Injection Outboard Isol. Valve	2
2E51-F014	RCIC Injection Check Valve	2

CODE EDITION AND ADDENDA: ASME OM Code-2004 Edition with Addenda through OM-2006

REQUIREMENTS: ASME OM Code, 2004 Edition, ISTB Table ISTC-3630 - Leakage Rate for other than Containment Isolation Valves

REASON FOR REQUEST: Pursuant to 10 CFR 50.55a, "Codes and Standards", paragraph (a)(3), relief is requested from the requirement of ASME OM Code ISTC-3630(a). The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

ISTC-3630 requires that leakage rate testing for pressure isolation valves be performed at least once every 2 years. Pressure Isolation Valves (PIVs) are not specifically included in the scope for performance-based testing as provided for in 10CFR50 Appendix J Option B. The concept behind the Option B alternative for containment isolation valves is that licensees should be allowed to adopt cost effective methods for complying with regulatory requirements. Additionally, NEI 94-01 describes the risk-informed basis for the extended test intervals under Option B. That justification shows that for valves which have demonstrated good performance by passing their leak rate tests for two consecutive cycles, further failures appear to be governed by the random failure rate of the component. NEI 94-01 also presents the results of a comprehensive risk analysis, including the statement that "the risk impact associated with increasing [leakrate] test intervals is negligible (less than 0.1% of total risk)." The valves identified in this relief request are all in water applications, CIV valves are tested in accordance with Appendix J Requirements using air. PIV testing is typically performed at lower pressures, such as for Appendix J Requirements, are acceptable provided the results are extrapolated to system functional differential pressure. Plant Hatch applies the extrapolated values to both PIV and CIV values. This relief request is intended to provide for a performance-based scheduling of PIV tests at Hatch. The reason for requesting this relief is dose reduction / ALARA. Recent historical data was used to identify that PIV testing alone each refuel outage incurs a total dose of approximately 400 milliRem. Assuming all of the PIVs remain classified as good performers the extended test intervals would provide for a savings of 800 mR over a 4-1/2 year period.

NUREG 0933 Issue 105 (Interfacing Systems LOCA at LWRs) discussed the need for PIV leak rate testing based primarily on 3 pre-1980 historical failures of applicable valves industry-wide. These failures all involved human errors in either operations or maintenance. None of these failures involved inservice equipment degradation. The performance of PIV leak rate testing provides assurance of acceptable seat leakage with the valve in a closed condition. Typical PIV testing does not identify functional problems which may inhibit the valves ability to re-position from open to close. For check valves, such functional testing is accomplished per ASME OM Code ISTC-3522 and ISTC-3520. Power-operated valves are routinely full stroke tested per ASME OM Code to ensure their functional

capabilities. At Hatch, these functional tests for PIVs are performed only at a Cold Shutdown or Refuel Outage frequency. Such testing is not performed online in order to prevent any possibility of an inadvertent ISLOCA condition. The 24 month functional testing of the PIVs is adequate to identify any abnormal condition that might affect closure capability. Performance of the separate 24 month PIV leak rate testing does not contribute any additional assurance of functional capability - it only determines the seat tightness of the closed valves.

PROPOSED ALTERNATIVE AND BASIS: Hatch proposes to perform PIV testing at intervals ranging from every refuel to every third refuel. The specific interval for each valve would be a function of its performance and would be established in a manner consistent with the Containment Isolation valve (CIV) process under 10CFR50 Appendix J Option B. 12 of the 36 valves listed are also classified as CIVs and are currently leak rate tested with air according to 10CFR50 Appendix J methodology every 2 years to satisfy their PIV leakage test requirement (with acceptance criteria correlated to water at function maximum pressure differential). Whether the valve is a CIV/PIV or PIV only, the valve must have two consecutive leakage tests which meet its acceptance criteria to be considered a good performer. That is, the test interval may be extended to every third refuel outage upon completion of two consecutive periodic PIV tests with results within prescribed acceptance criteria. The test interval will be extended to a specific value in a range of frequencies from 30 months up to a maximum of 75 months (as described in NEI 94-01 Revision 3-A). The test interval shall not exceed 75 months with a 3 month grace period (i.e., a total of 78 months). Any test failure will require a return to the initial (every RFO) interval until good performance can again be established.

The primary basis for this relief request is the historically good performance of the PIVs and desire to reduce personnel dose (ALARA). With the testing being performed every refueling outage has resulted in approximately 180 tests with 2 failures which yields a failure rate of approximately 1 percent.

Additional basis for this relief request is provided below:

- Separate functional testing of power-operated PIVs and Condition Monitoring of Check Valve PIVs per ASME OM Code.
- Low likelihood of valve mispositioning during power operations (procedures, interlocks).
- Air test vs. water test - degrading seat conditions tend to be identified sooner with air testing.
- Relief valves in the low pressure (LP) piping - these relief valves may not provide Inner-System Loss of Coolant Accident (ISLOCA) mitigation for inadvertent PIV mispositioning but their relief capacity can accommodate conservative PIV seat leakage rates.

RR-V-10 (Cont.)

- Alarms that identify high pressure (HP) to LP leakage - Operators are highly trained to recognize symptoms of a present ISLOCA and to take appropriate actions.

DURATION: 5th Interval beginning January 1, 2016 and ending December 31, 2025.

PRECEDENTS:

1. This Relief Request was approved as RR VR-013 for Fermi 2 as documented in SER dated 10/28/2010 (TAC ME2558, ME2557 and ME2556)
2. Approved Quad Cities Relief Request RV03- NRC SER dated February 14, 2013 (ML13042A348)

REFERENCES: None