November 7, 2003

Mr. H. L. Sumner, Jr.
Vice President - Nuclear
Hatch Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 RE: RELIEF REQUEST NUMBER RR-V-18 FOR HIGH PRESSURE COOLANT INJECTION SYSTEM CHECK VALVES RELATED TO THE THIRD 10-YEAR INTERVAL INSERVICE TESTING PROGRAM (TAC NOS. MC0109 AND MC0110)

Dear Mr. Sumner:

By letter dated July 11, 2003, as supplemented by letter dated September 12, 2003, Southern Nuclear Operating Company, (the licensee) submitted a proposed alternative to the requirements of American Society of Mechanical Engineers (ASME) Code for the Operation and Maintenance of Nuclear Power Plants (OM Code), 1990 Edition, paragraph ISTC 4.5.4(c). The alternative is proposed for use at Edwin I. Hatch Nuclear Plant, Units 1 and 2, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(i) for the third 10-year interval inservice testing (IST) program.

ASME OM Code, 1990 Edition, paragraph ISTC 4.5.4(c) allows disassembly every refueling outage to verify operability of check valves as an alternative to the exercising requirements of paragraphs ISTC 4.5.4(a) and (b). This relief request applies to the high pressure coolant injection system check valves 1/2E41-F045. The proposed alternative would permit the IST activity to be performed during normal plant operation or during refueling outages in lieu of the Code requirement that limits the IST activity to refueling outages. The relief request was previously submitted as Relief Request RR-V-17 on July 11, 2001. At that time, the relief request for these check valves was denied because it lacked information necessary for the Nuclear Regulatory Commission (NRC) staff to reach a safety finding. The licensee re-submitted a new relief request, RR-V-18, regarding these check valves with additional information. In response to the NRC staff's request for additional information, the licensee revised its relief request, RR-V-18, in the September 12, 2003, letter.

Based on the information provided by the licensee, the NRC staff concludes that the proposed alternative provides an acceptable level of quality and safety. Therefore, the use of the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year Interval IST Program. The NRC staff's Safety Evaluation is enclosed.

Sincerely,

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosure: As stated

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST RR-V-18 FOR HIGH PRESSURE COOLANT INJECTION SYSTEM

CHECK VALVES FOR THIRD 10-YEAR INTERVAL INSERVICE TESTING PROGRAM

SOUTHERN NUCLEAR OPERATING COMPANY, INC

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated July 11, 2003, as supplemented by letter dated September 12, 2003, Southern Nuclear Operating Company (SNC, the licensee) for Edwin I. Hatch Nuclear Plant, Units 1 and 2, submitted Relief Request RR-V-18 for its high pressure coolant injection (HPCI) system check valves 1E41-F045 and 2E41-F045. These check valves were previously the subject of Relief Request RR-V-17 on July 11, 2001. At that time, the request for relief for these check valves was denied because the submittal lack certain information necessary for the Nuclear Regulatory Commission (NRC) staff to reach a safety finding. The licensee re-submitted its relief request for these check valves in a new relief request, RR-V-18, with additional information. In response to the NRC staff's request for additional information (RAI), the licensee submitted revised relief request RR-V-18 on September 12, 2003. This relief request is associated with the third 10-year interval inservice testing (IST) program for Hatch Nuclear Plant, Units 1 and 2.

The Hatch Nuclear Plant IST program plan for the third 10-year interval is based on the requirements in the American Society of Mechanical Engineers (ASME) *Code for the Operation and Maintenance* of Nuclear Power Plants (OM Code), 1990 Edition. The alternative is proposed for use at the Hatch Nuclear Plant, Units 1 and 2, pursuant to Title 10 of the *Code of Federal Regulation* (10 CFR) Section 50.55a(a)(3)(i).

Specifically, in Relief Request RR-V-18, the licensee proposes an alternative to the ASME OM Code, 1990 Edition, paragraph ISTC 4.5.4(c) for performing IST of HPCI system check valves, 1/2E41-045. The check valves will be tested using a disassembly-and-inspection method during normal plant operation or during refueling outages in lieu of the Code requirement that limits the IST activities to refueling outages only.

2.0 REGULATORY EVALUATION

The *Code of Federal Regulations*, 10 CFR 50.55a, requires that IST of certain ASME Code Class 1, 2, and 3 pumps and valves be performed in accordance with the ASME OM Code and applicable addenda, except where alternatives have been authorized or relief has been

ENCLOSURE

requested by the licensee and granted by the Commission pursuant to Sections (a)(3)(i), (a)(3)(ii), or (f)(6)(i) of 10 CFR 50.55a. In proposing alternatives or requesting relief, the licensee must demonstrate that: (1) the proposed alternatives provide an acceptable level of quality and safety; (2) compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety; or (3) conformance is impractical for its facility. Section 50.55a authorizes the Commission to approve alternatives and to grant relief from ASME Code requirements upon making the necessary findings. Guidance related to the development and implementation of IST programs is given in Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," issued April 3, 1989, and its Supplement 1 issued April 4, 1995, as well as in NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," and NUREG/CR-6396, "Examples, Clarifications, and Guidance on Preparing Requests for Relief from Pump and Valve Inservice Testing Requirements."

The 1990 Edition of the ASME OM Code is the applicable Code of record for the third 10-year interval IST program at the Hatch Nuclear Plant, Units 1 and 2. The ASME OM Code specifies the requirements for IST of valves and pumps.

The NRC's findings with respect to authorizing alternatives and granting or denying the IST program relief requests are discussed below.

3.0 TECHNICAL EVALUATION

The licensee's regulatory and technical analyses in support of its requests for relief from ASME OM Code IST requirements are described in the licensee's submittals dated July 11, 2003, and September 12, 2003. A description of the relief request and the staff evaluation follows.

3.1 Relief Request RR-V-18

The licensee has requested relief for the HPCI system check valves 1E41-045 and 2E41-045 from the ASME OM Code IST requirements as specified in paragraph ISTC 4.5.4(c) of 1990 Edition of OM Code.

ASME OM Code, paragraph 4.5.4 addresses methods that may be used to perform IST activities for valves. Paragraph 4.5.4(c) states, "As an alternative to the testing in (a) or (b) above, disassembly every refueling outage to verify operability of check valves may be used."

Position 2 of the GL 89-04 and its supplement 1 provide an alternative to full-stroking a check valve or for verifying closure capability through the use of sample disassembly and inspection requirements.

3.1.1 Licensee's Basis for Relief:

> NRC Generic Letter (GL) 89-04, Position 2, provides guidance for the grouping of check valves and sample disassembly as an alternative to the OM Code, subsection ISTC requirements. GL 89-04, Position 2, paragraph 2.b states: ".....Since this frequency differs from the Code required frequency, this deviation must be specifically noted in the IST program." The above listed check valves are specifically identified in the existing Hatch IST program for application of the

guidelines of GL 89-04, Position 2. Each check valve is scheduled for disassembly, visually examination, and manual full-stroke exercising each refueling outage. Therefore, the regulatory guidance and the OM Code requirements, associated with check valve disassembly, are incorporated into the exiting Hatch IST program.

These check valves are located in the respective unit's HPCI pump suction from the suppression pool. The HPCI pump suction is normally aligned to the Condensate Storage Tank (CST) during normal operation and the system is provided with automatic controls which swap the suction to the suppression pool should CST level fall below a specific set-point or on suppression pool high level. The suction line from the suppression pool is provided with two power operated valves (POVs, one Air-Operated Valve (AOV) and one Motor-Operated Valve (MOV)) between the suppression pool and check valve 1/2E41-F045, and one MOV between check valve and CST suction line. These POVs provide for normal isolation and the system automatic swap features. Neither POV (1/2E41-F042 or F051) from the suppression pool is required to be leakrate tested in accordance with 10 CFR 50 Appendix J because the plant licensing basis assumes the suppression pool to remain water filled post accident. The MOV, 1/2E41-F041, downstream from the check valve is not required to be leakrate tested to satisfy any code or regulatory requirements. Reference drawings H-16332 and H-26020 for Units 1 and 2, respectively.

In order to isolate check valve 1/2E41-045 for disassembly, SNC will close and disable MOV 1/2E41-F042 and AOV 1/2E41-F051 on the suppression pool side of the check valve and MOV 1/2E41-F041 on the CST side of the check valve. Closing and disabling these POVs provide a high level of confidence that the check valve is adequately isolated from the suppression pool and the CST to prevent any significant leakage and ensures that inadvertent operation while the check valve is disassembled does not occur. Additionally, SNC will perform a leakrate type test of valve 1/2E41-F041 at least once each cycle. This leakrate type test will be performed at containment accident pressure and the acceptance criteria of the ASME OM Code, 1990 Edition, paragraph ISTC 4.3.3(e)(1) (i.e., 0.5D gal/min or 5 gal/min, whichever is less) will be utilized for evaluation of leakrate test data. The disassembly procedure also includes requirements for maintenance personnel to ensure the check valve is adequately isolated before complete removal of the valve cover plate (bonnet). No disassembly will be attempted unless the above leakage rate test criteria are satisfied.

Additionally, the Code of Federal Regulations, Title 10, Part 50, paragraph 65(a)(4) (i.e., 10 CFR 50.65(a)(4)) requires Licensees to assess and manage the increase of risk that may result from proposed maintenance activities. SNC complies with the 10 CFR 50.65(a)(4) requirements at Plant Hatch via the application of a procedure governing maintenance scheduling. This procedure dictates the requirements for risk evaluations as well as the necessary levels of action required for risk management in each case. The procedure also controls operation of the on-line risk monitoring system which is based on the Hatch Probabilistic Risk Assessment (PRA). In addition, this procedure provides methods for risk assessing maintenance activities for components not directly in the Hatch Probabilistic Risk Assessment (PRA) model. With the use of risk evaluation for virtually all aspects of nuclear plant operation, SNC has initiated efforts to accomplish additional maintenance, surveillance, and testing activities during normal operation. Planned activities are evaluated utilizing risk insights to determine the impact on safe operation of the plant and the ability to maintain associated safety margins. Individual system components, a system train, or a complete system may be planned to be out-of-service to allow maintenance, or other activities, during normal operation.

All activities associated with disassembly of the check valves are performed in accordance with plant procedures which meet 10 CFR 50.65(a)(4) requirements. These procedures provide detailed instructions for the pre-assembly leakrate test of the isolation MOVs, and disassembly, visual examination, and full-stroke exercising of respective check valves. Closing and disabling the isolation MOVs will be controlled in accordance with the site administrative control procedures. Additionally, considerations for corrective actions are factored into the planning process. Therefore, the use of risk assessment, MOV closure, and leakrate testing to ensure check valve isolation prior to disassembly during normal operation, provides an acceptable level of quality and safety and is thus authorized by 10 CFR 50.55a(3)(i) [10 CFR 50.55a(a)(3)(i)].

In a response to NRC staff's RAIs, the licensee submitted additional information in letter dated September 12, 2003. The some of the significant information items are as follows:

1. To support disassembly and inspection of the check valve 1/2E41-F045, the air-operated valve 1/2E41-F051, which is normally-open and fail-open will be disabled in the closed position by utilizing a maintenance procedure which inserts a gagging device. The valve is closed using the control switch procedure and then maintenance personnel insert the mechanical gag which prevents the valve from moving from the closed position.

2. For conservatism, this activity would be treated the same as a planned HPCI outage. Such outages are performed periodically while the Units are on-line. Technical Specification (TS) preclude planned maintenance that would cause reactor core isolation cooling (RCIC) or the Automatic Depressurization System (ADS) to be inoperable during the HPCI "at power" outage. With HPCI out-of-service, the unavailability of RCIC or ADS would require the Unit to enter a shutdown action statement. It is not plant practice to perform planned maintenance activities through entry into the shutdown action statement of the TS. The plant is therefore, administratively assured of having a steam driven injection system as well as automated access to low pressure emergency core cooling system injection, thus minimizing the qualitative risk of removal of HPCI from service.

3. Limiting condition for operation (LCO) 3.5.1 is applicable for the HPCI System. Action 3.5.1.C states that if the HPCI System is inoperable that RCIC must be verified operable within 1-hour and that HPCI must be restored to the operable condition within 14 days. The licensee states that for a planned HPCI System Outage during normal plant operation, operations personnel would confirm RCIC operability prior to declaring HPCI inoperable to perform the scheduled maintenance activities. Typical on-line system

outages are scheduled utilizing only 50 percent of the allowable LCO time limit. Therefore, HPCI System maintenance activities would be scheduled to be completed by 7 of the 14 days allowable window, with the remaining 7 days allocated to system restoration, testing, documentation review, closeout and return to service. Based on previous history, the disassembly/inspection and reassembly of the 1/2E41-F045 check valves typically requires 1-shift (i.e., 12 hours).

4. Even though no significant valve degradation has been identified during previous inspections, repair parts are maintained in site warehouse inventory. Therefore, should degradation be identified that required replacement of internals, replacement parts are readily available and are staged prior to beginning the disassembly. Adequate measures are taken, and contingencies are in place to provide assurance that disassembly, inspection, and repair, if necessary, can be accomplished in the TS allowable time period.

5. The HPCI pump system has a duel suction path from either the (1) condensate storage tank, or (2) the suppression pool. This redundancy makes the risk of losing one pathway very small. The licensee states that the average change in risk for such event is as follows:

ΔCDF (change in average core damage frequency) = 4.18E-08, and

ΔLERF (change in large early release frequency) = Insignificant.

These numbers were derived by totally failing the suppression chamber suction pathway and leaving the other pathway's failure probabilities based on average failure intact. The core damage probabilities was derived in a similar manner. The time frame of consideration used the probability calculation was conservatively set at 2 years. These results are as follows:

ICCDP (Incremental Conditional Core Damage Probability) = 8.36E-08, and
ICLERP (Incremental Conditional Large Early Release Probability) = Insignificant.

These numbers are well below guidance provided in USNRC Regulatory Guide 1.174 and 1.177 for small risk.

3.1.2 Licensee Proposed Alternative Testing (as stated by the licensee):

Check valve disassembly, visual examination, and manual exercising will continue to be performed utilizing the guidance contained in NRC GL 89-04, Position 2. However, such disassembly, visual examination, and manual exercising will be performed during normal operation, in conjunction with appropriate system outages, or during refueling outages. In any case, disassembly, inspection, and manual exercising will be performed at least once each operating cycle (i.e., 24-months). Check valve disassembly during normal plant operation will be managed in accordance with the requirements of 10 CFR 50.65(a)(4) in conjunction with the isolation and leakrate testing described above.

3.1.3 NRC Staff's Evaluation

In relief request RR-V-18, SNC states that the Hatch Nuclear Plant IST check valve program for HPCI system valves 1-E41-E045 and 2-E41-F045 meets the requirements of the ASME OM Code, 1990 Edition paragraph ISTC 4.5.4(c). SNC also states that HPCI system check valves follow the guidance provided in Position 2 of NRC GL 89-04 with regard to sample disassembly and inspection of check valves. ASME OM Code, paragraph ISTC 4.5.4(c) as well as GL 89-04, Position 2 require that check valves IST activities (including disassembly) be performed during refueling outages.

The licensee proposes, as an alternative, to perform the IST disassembly and inspection activities during normal plant operation, in conjunction with appropriate system outages, or during refueling outages. In any case, disassembly, inspection, and manual exercising will be performed at least once each operating cycle (i.e., 24-months). Check valve disassembly during normal plant operation will be managed in accordance with the requirements of 10 CFR 50.65(a)(4) in conjunction with isolation and leakrate testing described above.

The HPCI system check valves 1-E41-F045 and 2-E41-F045 are 16-inch diameter check valves. These relatively large check valves are located in the respective unit's HPCI pump suction line from the suppression pool. The HPCI pump suction is normally aligned to the condensate storage tank (CST) during normal operation and the system is provided with automatic controls which swap the suction to suppression pool should CST level fall below a specific set-point or on suppression pool high level.

The NRC staff finds that disassembly and inspection of HPCI system check valves 1-E41-F045 and 2-E41-F045 are the appropriate methods to verify operability and can be accomplished during system outages when the plant is on-line or during refueling outages. The NRC staff's finding is based on the following considerations:

1. IST performed on a refueling outage frequency meets the intent of the ASME OM Code and Position 2 in GL 89-04. By specifying testing activities on a frequency commensurate with each refueling outage, the ASME OM Code recognizes and establishes an acceptable time period between testing. The refueling outages have provided a practical and definitive time period in which testing activities can be safely and effectively performed. An acceptable testing frequency can be maintained separately without being tied directly to a refueling outage. IST performed on a frequency (24 months) that maintains the acceptable time period between testing activities during the operating cycle (i.e., 24 months) is consistent with the intent of the ASME OM Code and GL-89-04.

2. Over time, approximately the same number of tests will be performed using the proposed operating cycle test frequency as would be performed using the current refueling outage frequency. Thus, inservice testing activities performed during the proposed operating cycle (i.e., 24 months) test frequency provide an equivalent level of quality and safety as IST performed at a refueling outage frequency.

3. During check valve 1/2E41-045 disassembly and inspection, SNC will close and disable MOV 1/2E41-F042 and AOV 1/2E41-F051 on the suppression pool side of the check valve and MOV 1/2E41-F041 on the CST side of the check valve. The associated

action for these valves will provide adequate isolation utilizing appropriate Code leakage criteria. As a result, all isolation valves will have been leak-tested and/or have double-isolation capability. The licensee states that its disassembly procedure also includes requirements for maintenance personnel to ensure the check valve is adequately isolated before complete removal of the check valve cover plate (bonnet). No disassembly will be attempted unless the Code specified leakage rate criteria are satisfied. The licensee's procedure provides adequate measures to ensure that the check valve will be properly isolated during disassembly and inspection activities.

4. There are no technical barriers to performing these IST activities during either the refueling outage or the operating cycle.

5. The licensee states that typical on-line HPCI system outages are scheduled utilizing only 50 percent of the allowable LCO time limit. Therefore, HPCI system maintenance activities would be scheduled to be completed within 7 of the 14 days allowable window, with the remaining 7 days allocated to system restoration, testing, documentation review, closeout and return to service. This provides adequate margin to complete disassembly and inspection activities in an orderly manner.

6. The licensee states that even though no significant valve degradation has been identified during previous inspections, repair parts are maintained in site warehouse inventory. Therefore, should degradation be identified that required replacement of internals, replacement parts are readily available and are staged prior to beginning the disassembly.

7. The potential risk impact in performing disassembly and inspection of these check valves while the plant is on-line is insignificant.

On the basis of these considerations, the NRC staff finds that the proposed alternative provides an acceptable level of quality and safety.

4.0 CONCLUSIONS

Based on the review of the information provided in the relief request RR-V-18, the NRC staff concludes that licensee's proposed alternative will provide an acceptable level of quality and safety. Therefore, the proposed alternative to disassemble and inspect the check valves 1/2E41-F045 once every operating cycle in lieu of once during each refueling outage is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year interval IST Program. The licensee may also continue to perform disassembly and inspection of these check valves during refueling outages as currently required by the Code without the need for further authorization.

Principal Contributor: G.Bedi

Date: November 7, 2003

Edwin I. Hatch Nuclear Plant

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