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NL-03-1744

Docket Nos.: 50-321
50-366

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

Edwin I. Hatch Nuclear Plant
Third 10-Year Interval Inservice Testing Program
Submittal of Revised Relief Request RR-V-18 and
Response to Request for Additional Information (RAI)

Ladies and Gentlemen:

On July 11, 2003 Southern Nuclear Operating Company (SNC) submitted Inservice Testing (IST) Program Relief Request RR-V-18 (ref. NL-03-1380). This relief request concerns High Pressure Coolant Injection (HPCI) System check valves 1/2E41-F045.

During the NRC review of the submittal, clarification and additional information was requested. The revised relief request (Enclosure 1) clarifies the correct type valve for 1/2E41-F051. Additionally, Enclosure 2 provides the SNC response to the Request for Additional Information.

With this additional information, SNC requests approval of Relief Request RR-V-18 in accordance with 10 CFR 50.55a(a)(3)(i).

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

HLS/whc

A047

Enclosures: Enclosure 1 - Revised Relief Request RR-V-18
Enclosure 2 - SNC Response to Request for Additional Information

cc: Southern Nuclear Operating Company
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Enclosure 1

Edwin I. Hatch Nuclear Plant, Units 1 and 2
Inservice Inspection Program

Revised Relief Request RR-V-18

Edwin I. Hatch Nuclear Plant, Units 1 and 2
Inservice Inspection Program
Relief Request RR-V-18

SYSTEM(S): High Pressure Coolant Injection System (HPCI - E41)

COMPONENTS:

Unit 1	<u>MPL</u>	<u>ASME CLASS</u>	<u>SIZE</u>
	1E41-F045	2	16"
Unit 2	<u>MPL</u>	<u>ASME CLASS</u>	<u>SIZE</u>
	2E41-F045	2	16"

CATEGORY: C

ASME CLASS: 2

TEST REQUIREMENT:

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).

REQUIREMENT FOR WHICH RELIEF IS REQUESTED:

Relief is requested from the ASME OM Code requirement that check valve disassembly be performed only during a refueling outage.

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 existing 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 the check valve and the 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.

BASIS FOR RELIEF (continued):

In order to isolate check valve 1/2E41-F045 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 provides 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 Safety Assessment (PSA) 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 listed 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-disassembly leakrate test of the isolation MOVs, and disassembly, visual examination, and full-stroke exercising of the respective check valve. Closing and disabling the isolation MOVs will be controlled in accordance with 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).

ALTERNATE TESTING:

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.

Enclosure 2

Edwin I. Hatch Nuclear Plant, Units 1 and 2
Inservice Inspection Program

Request for Additional Information (RAI) Response
Relief Request RR-V-18

Request for Additional Information (RAI)
Edwin I. Hatch Nuclear Plant, Units 1 and 2
Relief Request No. RR-V-18
High Pressure Coolant Injection (HPCI) System Check Valves 1/2E41-F045
Docket Nos. 50-321 and 50-366 (MC0109 and MC0110)

Reference: Southern Nuclear Operating Company, Inc., Edwin I. Hatch Nuclear Plant, Units 1 and 2, "Re-submittal of Relief Request RR-V-18 for Third 10-year Interval Inservice Testing Program," Docket No. 50-321 and 50-366, July 11, 2003.

Background: In Valve Relief Request RR-V-17, the NRC staff authorized the sample disassembly and inspection of several check valves during power operation (on-line). These check valve were all relatively small check valves (i.e., 4-inch nominal pipe size and smaller). The staff in RR-V-17 denied on-line IST of a 16" HPCI check valve on the basis that the safety and risk significance of on-line testing the relatively large check valve was not addressed. The licensee resubmitted a relief request for this 16" HPCI check valve (RR-V-18) in the above referenced letter and addressed certain safety and risk issues associated with on-line testing of the valve. However, because of the relatively large size of this valve and the potential flow path from the high-energy feedwater system, the staff needs the following additional information to complete its review.

<u>RAI 1</u>: The licensee is proposing to disassemble and inspect the 16" HPCI check valve (1/2E41-F045) "during normal operation in conjunction with appropriate system outages or during refueling outages." Please clarify what frequency this check valve will be tested (disassembled and inspected). If there are no appropriate system outages, when will this check valve be tested? If it is practical to perform IST of this check valve on-line, explain why it is not practical to perform IST (i.e., disassemble and inspect) these valves on a quarterly basis as required by the ASME OM Code.

<u>SNC Response to RAI 1:</u>

NRC GL 89-04 Position 2 and the ASME OM Code-1990 Edition require check valve disassembly and inspection each refueling outage. Plant Hatch operates on a 24-month fuel cycle; therefore, each check valve will be disassembled, visually examined, and manually exercised at least once every 2 years. SNC is not requesting a change in the frequency; only the option to perform testing during system outages that are scheduled during normal plant operation or during a refueling outage if no system outage was scheduled during the previous operating cycle. In any case, the proposed testing frequency is every 2 years.

The ASME OM Code does not require disassembly and inspection quarterly. The ASME OM Code-1990 Edition is applicable for Plant Hatch (see Relief Request RR-G-1 granted in NRC SE dated April 12, 1996 TAC NOS. M93072 and M93073). Paragraph ISTC 4.5.4 (c) states:

> "As an alternative to the testing in paragraph ISTC 4.5.4(a) or ISTC 4.5.4(b), disassembly every refueling outage to verify operability of check valves may be used."

<u>SNC Response to RAI 1 (continued):</u>

The burdens associated with check valve disassembly and inspection were previously identified by the NRC in GL 89-04, Position 2 which allows grouping of check valves and disassembly of one in each group every refueling outage. SNC previously provided the NRC with justification why these check valves could not be exercised with flow as required by the Code (ISTC 4.5.1 and ISTC 4.5.2) in Relief Request RR-V-7 (per SE dated April 12, 1996 TAC NOS. M93072 and M93073, relief was not required). SNC is not aware of any Code or NRC initiatives which would require licensees to perform check valve disassembly and inspection on a quarterly frequency for valves that cannot be exercised with flow.

<u>RAI 2</u>: Under the section entitled <u>Basis for Relief</u>, the licensee describes valve 1/2E41-F051 as a motor-operated valve (MOV) that the licensee would close and disable in order to isolate the check valve 1/2E41-F045. However, the P&ID depicts this valve as an air-operated valve (normally open and fail open). Please clarify the correct valve type for 1/2E41-F051. If the valve is an AOV that fails in the open position, please explain how you intend to disable the valve in the closed position.

<u>SNC Response to RAI 2:</u>

The P&ID (H-16332, H-26020) is correct in showing valve 1/2E41-F051 as an AOV that is normally open and fails open. To support disassembly/inspection of 1/2E41-F045, the valve 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 and then maintenance personnel insert the mechanical gag which prevents the valve from moving from the closed position. In addition to the maintenance procedure controls, the activity is also administratively controlled via the site clearance and tagging procedure which ensures that the valve will be maintained closed during the disassembly/inspection evolution of the associated F045 check valve.

<u>RAI 3</u>: Under the Section entitled <u>Basis for Relief</u>, the licensee states that it intends to isolate the HPCI check valve 1/2E-E41-F045 by closing the [POVs] on the suppression pool side of the check valve in order to perform the proposed IST (disassembly and inspection). Please address (either in a qualitative or quantitative manner) the potential risk significance of disassembly and inspection of this check valve on-line compared to while the plant is shutdown.

SNC Response to RAI 3:

The HPCI pump system has a dual 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 average change in risk for such an 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. Core damage probability was derived in a similar manner. The time frame of consideration used for the probability calculations 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 Guides 1.174 and 1.177 for small risk.

These calculations take into account the potential for failure of the HPCI system during power operations. Despite this the risk numbers are low. Taking a unit to cold shutdown to perform the required inspection of the HPCI suction check valve brings into play a different set of risk considerations. HPCI is not available or required in Mode 4, and in general, the redundancy of high pressure feed availability is lost in Mode 4. In Mode 4, the unit enters an area where coping with the LOSP case becomes more severe because of the unavailability of non-electrically driven water pumps. Qualitatively, risk may be increased by specifically going to cold shutdown to perform the inspection task in question.

RAI 4: Based on the risk significance discussed in RAI 3 above, discuss what preventive or compensatory measures are necessary to maintain safety and minimize risk while performing on-line IST.

SNC Response to RAI 4:

As a matter of conservatism, this particular maintenance activity would be treated the same as a planned HPCI outage. Such outages are performed periodically while the units are on-line. Technical Specifications (TS) preclude planned maintenance that would cause RCIC or the Automatic Depressurization System (ADS) to be inoperable during

SNC Response to RAI 4 (continued):

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 statements of the TS. The plant is therefore administratively assured of having a steam driven injection system as well as automated access to low pressure ECCS injection, thus minimizing the qualitative risk of removal of HPCI from service.

RAI 5: Under the section entitled Basis for Relief, the licensee states that the maintenance rule 10 CFR 50.65(a)(4) requires licensees to assess and manage the increase of risk that may result from proposed maintenance activities. However, in order for the staff to evaluate whether the proposed IST alternative is acceptable, the licensee must demonstrate that the alternative provides an acceptable level of quality and safety pursuant to 10 CFR 50.55a(a)(3)(i). Performing a risk assessment of the proposed on-line testing at the time of IST does not address why on-line testing provides an acceptable level of quality and safety before it is performed. Meeting the maintenance rule is a separate regulatory requirement. Nonetheless, discuss how risk insights, as well as other factors, will be used to establish when IST should be performed either on-line or during refueling outages.

SNC Response to RAI 5:

The response to RAI 3 explains that, qualitatively, risk tends to increase when redundancy of equipment is reduced, such as during the cold shutdown case. In addition, it becomes one of several maintenance tasks to manage. On the other hand, the combination of risk insights being quantitatively and qualitatively low, the excellent isolation capabilities surrounding the 1E41/2E41-F045 check valves, management of the task as a HPCI outage, focused attention on this particular job, and the fact that the HPCI function can still be made available through its normal suction path, makes on-line maintenance in this case acceptable.

RAI 6: Explain how Technical Specification requirements for the HPCI system will be satisfied while performing on-line IST of this check valve. Specifically, address the limiting condition for operation (LCO) and describe the actions the licensee will take to ensure that on-line IST will be accomplished within the allowed outage time. Discuss the typical amount of time needed to complete the IST of this check valve based on previous testing experience. Similarly, describe any contingency plans that will be in effect to provide reasonable confidence that the AOT will not be exceeded if the check valve is found to be in a significantly degraded or unacceptable condition.

SNC Response to RAI 6:

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. 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% of the allowable LCO time limit. Therefore, HPCI System maintenance activities would be scheduled to be completed by day 7 of the 14 day allowable window, with the remaining 7 days allocated to system restoration, testing, documentation review, closeout and return to service.

Review of previous maintenance history indicates that disassembly/inspection and reassembly of the 1/2E41-F045 check valves typically requires 1-shift (i.e., 12-hours). 1/2E41-F045 are swing check valves. NRC GL 89-04, Position 2, and the ASME OM Code only require disassembly to the extent required to inspect the valve internals and manually exercise the disc full-swing. To accomplish this, these valves only require that the valve bonnet be removed and complete disassembly of the disc assembly (i.e., hinge pin, disc, swing arm, etc.) is not required unless repairs are necessary. Both of these valves have been disassembled and inspected seven (7) times each with no significant degradation having been identified. These valves are located in the suppression pool suction line and HPCI would not be operated utilizing this source except for emergency conditions. Since the suppression pool water is relatively clean, and the valves are not routinely operated, no degradation is expected.

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. The Maintenance department at Plant Hatch is adequately staffed and has equipment to support repair of any degradation that would require specialized machining. Therefore, adequate measures are taken, and contingencies are in place to provide assurance that disassembly, inspection, and repair, if necessary, can be accomplished in the Technical Specification allowable time period.