H. L. Sumner, Jr.
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SOUTHERN COMPANY
Energy to Serve Your World℠

January 15, 2004

Docket No.: 50-321

NL-04-0013

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

Edwin I. Hatch Nuclear Plant
Licensee Event Report
Residual Heat Removal Service Water Pump 1C
Inoperable for Failure to Meet its Seismic Requirements

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(i) and 10 CFR 50.73(a)(2)(v), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning an inoperable Residual Heat Removal Service Water pump.

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

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

HLS/IL/daj

Enclosure: LER 1-2003-003

cc: Southern Nuclear Operating Company
Mr. J. B. Beasley, Jr., Executive Vice President
Mr. G. R. Frederick, General Manager – Plant Hatch
Document Services RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. S. D. Bloom, NRR Project Manager – Hatch
Mr. D. S. Simpkins, Senior Resident Inspector – Hatch

JE22

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7/31/2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Edwin I. Hatch Nuclear Plant - Unit 1	05000-321	1 OF 7

4. TITLE
Residual Heat Removal Service Water Pump 1C Inoperable For Its Failure To Meet Seismic Requirements

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S) 05000
11	17	2003	2003	003	0	01	16	2004	FACILITY NAME	DOCKET NUMBER(S) 05000

9. OPERATING MODE (9)	1
10. POWER LEVEL	100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20-2203(a)(2)(ii)		50.36(c)(2)	X	50.73(a)(2)(v)(B)		OTHER
	20-2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)	X	50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Steven B. Tipps, Nuclear Safety and Compliance Manager, Hatch	(912) 537-5880

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
E	BS	P	J105	Y						

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)		NO X				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On 11/17/2003 at 1400 EST, Unit 1 was in the Run mode at a power level of approximately 2516 CMWT (91.1 percent rated thermal power). At that time, personnel replacing corroded anchor bolt nuts securing the 1C Residual Heat Removal Service Water (RHRSW) pump identified corrosion on two of the nuts to the extent that the adequacy of the seismic supporting capability of the nuts for that pump was questionable for the design loads. This postulated seismic failure of the 1C RHRSW pump could have prevented this pump from being able to perform its design function during a seismic event. Because the corrosion of the nuts occurred over time, and based upon the extent of the corrosion found on the nuts securing the pump, it was concluded that this condition may have existed for a period of time longer than the 30 days allowed by the plant's Technical Specifications 3.7.1 Condition A. The Residual Heat Removal Service Water pumps are used in the long-term containment heat removal mode.

This event was caused by untimely corrective action. Additional procedure guidance has been provided to address the Operability of components that are identified to exhibit progressive degradation. The corroded nuts on the 1B, 1C, and 1D RHRSW pumps were replaced. The bolts on 1A RHRSW pump were inspected and determined to not have any gross corrosion. The Unit 2 RHRSW pumps were inspected and found to have some degree of corrosion present. This was evaluated by a civil engineer and determined to not present an operability concern.

FACILITY NAME (1)	DOCKET	LER NUMBER (6) YEAR	SEQUENTIAL YEAR	REVISION NUMBER	PAGE (3)
Edwin I. Hatch Nuclear Plant - Unit 1	05000-321	2003	-- 003 --	0	2 OF 7

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes appear in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 11/17/2003 at 1400 EST, Unit 1 was in the Run mode at a power level of approximately 2516 CMWT (91.1 percent rated thermal power). At that time, personnel replacing corroded anchor bolt nuts securing the 1C Residual Heat Removal Service Water (RHRSW) pump identified corrosion on two of the nuts to the extent that the adequacy of the seismic supporting capability of the nuts for that pump was questionable for the design loads. This postulated seismic failure of the 1C RHRSW pump could have prevented this pump from being able to perform its design function during a seismic event. Because the corrosion of the nuts occurred over time, and based upon the extent of the corrosion found on the nuts securing the pump, it was concluded that this condition may have existed for a period of time longer than the 30 days allowed by the plant's Technical Specifications (TS) 3.7.1 Condition A.

On 3/28/2001, Condition Report 2001002557 was initiated and documented corrosion on some of the anchor bolt nuts for the 1B and 1C RHRSW pumps. An evaluation of this condition report was performed by Hatch Support Engineering. This evaluation concluded that the nuts with bolts were still operable and should be replaced in the next outage (Spring 2002) or earlier. The nuts were not replaced during the outage nor was an evaluation performed during the outage to reconsider the operability issues related to this condition.

On 11/17/2003, an anchor bolt nut was removed from the northeast corner of pump 1E11-C001C. The nut could not be removed by unscrewing it, but had to be cut off with a chisel. When the nut was removed, it was apparent by physical inspection that the corrosion was more severe than had been expected based on the system engineer's previous observations and conversations with seismic design engineers. In those conversations, it had been anticipated that the lower portion of the nut would be mostly intact with corrosion increasing toward the upper portion of the nut. No allowance was made for any corrosion penetrating the entire thickness of the nut. In fact, the nut was found with asymmetric corrosion penetrating all the way through its middle range. Although the lower portion of the nut had substantial amounts of steel still present, the corrosion was markedly worse on one side than the other. The judgment of the system engineer was that the nut may not have been able to carry its design loading. The asymmetrical wasting of the material may have contributed to the mistaken impression that there was more substance to the nut than was, in fact, remaining.

Since one other nut on this same pump (the northwest anchor) also showed evidence of significant wasting, the system engineer's judgment was that this nut also may not carry its design load. Both of these nuts were replaced per design with hex heavy nuts made from ASTM SA-194, Grade 3 steel. Work orders initiated to correct these conditions had been initially delayed while site engineering requested a contingency plan from corporate. This contingency plan was to have included a design change package to address the possibility that underlying anchor bolts may be damaged during nut removal. In lieu of a design change, the anchors were evaluated and it was determined that 3 of 4 anchors were adequate to establish operability. The system engineer then recommended that the work proceed. The unexpectedly severe extent of the corrosion was subsequently discovered.

The RHRSW pumps are mounted on base plates embedded in and anchored to the floor of the 111-ft. elevation of the intake structure. The Unit 1 pumps have four 1-inch diameter anchor bolts per pump. Since the base plates are embedded in the concrete, there is no shear loading on the anchor bolts either from pump discharge head thrust or from a postulated seismic event. Since the base plates rest on the floor, the floor receives all the downward loading of weight and pump thrust. The anchor bolts therefore provide only a straight tensile loading during a seismic event, preventing the pumps from overturning.

The anchor bolts and nuts prevent the RHRSW pumps from rocking or overturning during a seismic event. With one anchor bolt nut removed or unable to carry its design load, an analysis determined that the remaining three anchor bolts could carry the load. The load could be imparted by combinations of vertical and horizontal mass accelerations imparting an overturning moment to the cantilevered pump column at the floor. An analysis was performed by Hatch Support Engineering based on 3% damped ½ Seismic Margin Earthquake (SME) in-structure response spectra considering worst single horizontal with the vertical load combinations for an appropriate seismic operability assessment. The allowable stresses on the three remaining anchor bolts would not be exceeded if the fourth bolt was removed. However, in this case two anchor bolt nuts may not have been able to carry their design load.

Operations removed the 1C RHRSW pump for maintenance on 11/17/2003 at 0325 ET. A Required Action Sheet (RAS) 1-03-231 was initiated as required by plant procedures and the Unit 1 TS 3.7.1.

The RHRSW System is designed to provide cooling water for the Residual Heat Removal (RHR) System heat exchangers, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The RHRSW System is operated whenever the RHR heat exchangers are required to operate in the shutdown cooling mode or in the suppression pool cooling or spray mode of the RHR System. The RHRSW System consists of two independent and redundant subsystems. Each subsystem is made up of a header, two 4000 gpm pumps, a suction source, valves, piping, heat exchanger, and associated instrumentation. Either of the two subsystems is capable of providing the required cooling capacity with two pumps operating to maintain safe shutdown conditions. A review of the Required Action Sheet (RAS) entries and clearances for Unit 1 during this current operating cycle (since completion of the Unit 1 Spring 2002 refueling outage) was performed. This review determined that there were always at least two RHRSW pumps that were operable (not counting the 1C RHRSW pump) since the last refueling outage with the exception of approximately an 11 hour period that began at 1744 EST on 11/3/2003 and ended at 0426 EST on 11/4/2003. During this 11 hour

FACILITY NAME (1)	DOCKET	LER NUMBER (6) YEAR	SEQUENTIAL YEAR	REVISION NUMBER	PAGE (3)
Edwin I. Hatch Nuclear Plant - Unit 1	05000-321	2003	-- 003 --	0	4 OF 7

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

period of time, both the 1B and 1D RHRSW pumps were tagged out for an RHR system outage. A review of the clearance removing these pumps from service determined that it would have taken approximately 20 to 30 minutes for these pumps to have been restored. During the rest of the operating cycle, at least 2 pumps were always operable. If one pump in the "B" RHRSW subsystem was not operable, it was confirmed that the 1A RHRSW pump was operable. The 1A RHRSW pump could be used in conjunction with one of the two "B" RHRSW subsystem pumps by opening the RHRSW system crosstie valves 1E11-F119A and 1E11-F119B to connect the two RHRSW divisions.

CAUSE OF EVENT

The corrosion on the 1C RHRSW pump's anchor bolt and nuts was previously identified, but actions to correct the problem were untimely.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

Since the actual date on which the pump became inoperable is not known, this report is being made pursuant to 10 CFR 50.73(a)(2)(v) because a condition existed that alone could have prevented the fulfillment of the safety function of a system needed to remove residual heat and to mitigate the consequences of an accident. With the inadequate seismic support of the 1C RHRSW pump in conjunction with the 1B and 1D RHRSW pumps being removed from service to support a system outage, the ability of the RHRSW system to fulfill its safety function of removing containment heat following a Design Basis Loss-of-Coolant Accident could have been impaired. This report is also being submitted pursuant to 10 CFR 50.73(a)(2)(i) because a condition existed that was prohibited by the plant's Technical Specifications. The time the seismic supporting capability of the 1C RHRSW pump may have been inadequate probably exceeded the allowable out-of-service times given in the TS 3.7.1 Condition A for an inoperable RHRSW pump.

The RHRSW system is designed to provide cooling water to the RHR system heat exchangers which are required for a safe reactor shutdown following a design basis accident or transient. The RHRSW system consists of two independent and redundant subsystems. Each subsystem consists of a header, two 4000 gallon-per-minute pumps, a suction source, valves, piping, heat exchanger, and associated instrumentation. Either of the two subsystems is capable of providing the required cooling capacity with two pumps operating. The two subsystems are separated from each other by two normally closed, motor operated crosstie valves in series.

The RHRSW system is initiated manually and removes heat from the suppression pool to limit the suppression pool temperature and primary containment pressure following a loss-of-coolant accident. This ensures the primary containment can perform its function of limiting the release of radioactive materials to the environment and the emergency core cooling system pumps, whose suction source is the suppression pool, have adequate net positive suction head. As discussed in the Unit 1 Final Safety Analysis Report, manual initiation of an RHRSW subsystem and the associated RHR subsystem are

FACILITY NAME (1)	DOCKET	LER NUMBER (6) YEAR	SEQUENTIAL YEAR	REVISION NUMBER	PAGE (3)
Edwin I. Hatch Nuclear Plant - Unit 1	05000-321	2003	-- 003 --	0	5 OF 7

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

assumed to occur ten minutes after the onset of the accident. The RHRSW flow rate assumed in the analyses is 4000 gallons per minute per pump with two pumps providing cooling water to one RHR heat exchanger. In this case, the maximum suppression pool water temperature is approximately 210°F which is well below the design temperature of 281°F.

In this event, there were always at least two RHRSW pumps operable (not counting the 1C RHRSW pump) since the last refueling outage with the exception of approximately an 11 hour period that began at 1744 EST on 11/3/2003 and ended at 0426 EST on 11/4/2003. During this 11 hour period of time, both the 1B and 1D RHRSW pumps were tagged out for an RHR system outage. A review of the clearance removing these pumps from service determined that it would have taken approximately 20 to 30 minutes for these pumps to have been returned to an operable status. The plant's TS 3.7.1 Condition D does govern this condition and requires that if both RHRSW subsystems are inoperable, then one subsystem must be restored to an operable status within 8 hours. The 8 hour completion time for restoring one subsystem is based on the completion times provided for the RHR suppression pool cooling and spray functions which state that it is acceptable due to the low probability of a DBA and because alternative methods to remove heat from primary containment are available. If at least one subsystem is not restored within 8 hours, then TS 3.7.1 Condition E is entered. This condition requires that the plant must be placed in Mode 3 within 12 hours. This condition would have been exited 3 hours after entry because at that time the 1B and 1D pumps were restored. During the rest of the operating cycle, at least 2 pumps were always operable. If one pump in the "B" RHRSW subsystem was not operable, it was confirmed that the 1A RHRSW pump was operable. The 1A RHRSW pump could be used in conjunction with one of the two "B" RHRSW subsystem pumps by opening the RHRSW system crosstie valves 1E11-F119A and 1E11-F119B to connect the two RHRSW divisions. With two RHRSW pumps operable the RHRSW system is capable of fulfilling its safety function of long-term containment heat removal during the Design Basis Loss-of-Coolant Accident. Therefore, it may be concluded that this event did not adversely impact nuclear safety or the health and safety of the public. This analysis applies to all operating conditions and power levels where a Design Basis Loss-of-Coolant Accident may occur.

CORRECTIVE ACTIONS

The corroded nuts on the 1B, 1C, and 1D RHRSW pumps have been replaced. The anchor bolts and nuts on RHRSW pump 1A were also inspected by the system engineer and found to have no gross corrosion present. Additionally, the nuts on the Unit 2 RHRSW pumps have been inspected and found to have some degree of corrosion present. This was evaluated by a civil engineer and determined to not present an operability concern. The time frame for repair of these nuts will be reviewed as part of the corrective action involving the maintenance work order (MWO) backlog review described later in this section.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL YEAR	REVISION NUMBER	
Edwin I. Hatch Nuclear Plant - Unit 1	05000-321	2003	-- 003 --	0	6 OF 7

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

A review of open Maintenance Work Orders (originated prior to January 1, 2003) for safety-related systems, structures or Components (SSCs) will be performed by April 30, 2004. This review will confirm that the SSC remains operable and meets all applicable licensing requirements until such time that the repair can be completed. This review will also confirm that the scheduled date to perform the work is appropriate to support continued operability. Routine "walkdowns" of the affected systems will continue to be performed by the system engineers to identify further degradation of existing conditions or new conditions requiring corrective actions. Corrective actions will be prioritized commensurate with their impact on reactor safety and in accordance with the existing 12-week schedule.

A proposed modification has been submitted to the plant change control board (CCB) to consider the installation of drains in the RHRSW pump moats to remove water to an area outside the intake structure and preclude corrosion of these nuts and bolts.

The plant's Operability Determination procedure will be revised to specifically evaluate and establish corrective actions for safety related structures, systems or components that are identified with conditions that will continue to degrade, e.g. corrosion, oil / water leaks, etc. This revision will serve to ensure that the impact of the condition on the affected components are properly evaluated and corrective actions taken in a timely manner. It will also ensure that criteria is established at which the affected component(s) would be considered inoperable. This procedure revision will be implemented by February 6, 2004.

Carbon steel components in the interior of the intake structure will be repaired as necessary and repainted with a protective coating chosen as an interim action to provide reasonable assurance of retarding further corrosion. This will be completed by October 30, 2004.

ADDITIONAL INFORMATION

No systems other than those previously described in this report were affected by this event.

This LER does not contain any permanent licensing commitments.

There have been no previous similar events reported in which the plant entered a condition prohibited by Technical Specifications as the result of untimely corrective actions.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL YEAR	REVISION NUMBER	
Edwin I. Hatch Nuclear Plant - Unit 1	05000-321	2003	-- 003	-- 0	7 OF 7

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Failed Component Information:

Master Parts List: 1E11-C001C
Manufacturer: Johnston
Manufacturer Code: J105
Model Number: 18DC
Type: RHR Service Water Pump
EIIS System Code: BS
EIIS Component Code: P
Root Cause Code: E
Reportable to EPIX: Yes