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SOUTHERN COMPANY
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November 22, 2002

Docket No. 50-321 HL-6326

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

Edwin I. Hatch Nuclear Plant - Unit 1
Licensee Event Report
Calculation Error Results in Incorrect Steam Line High Flow Setpoints

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(i)(B), Southern Nuclear Operating Company is submitting the enclosed revision to a Licensee Event Report (LER) concerning a calculation error which resulted in incorrect steam line high flow setpoints. The revision is submitted to provide clarification on the safety analysis. Additionally, a typographical error is corrected.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

OCV/eb

Enclosure: LER 50-321/2002-003, R1

cc: Southern Nuclear Operating Company
Mr. P. H. Wells, Nuclear Plant General Manager
SNC Document Management (R-Type A02.001)

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NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Main Steam Line High Flow Setpoint Not Within Technical Specification Limits

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)
									Plant Hatch, Unit 2	05000-366
08	08	2002	2002	003	1	11	22	2002	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)		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)		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) 367-7851

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

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 8/8/2002, at 1400 EDT, with Unit 1 and 2 in the Run mode at a power level of 2763 CMWt, the plant Architect Engineer notified Southern Nuclear that the setpoints for the main steam line high flow Group I isolation had previously been non-conservative, due to the use of an incorrect discharge coefficient for the main steam line flow nozzles. The error was discovered by General Electric during an independent review of Technical Specifications allowable values and analytical limits for a thermal power optimization program. Previous investigations into the potential problem had resulted in a conservative resetting of the actual in-plant main steam line high flow setpoints on 8/6/2002. At the time of the notification, therefore, the actual plant setpoints were set correctly.

The cause of this event could not be conclusively determined. It is believed to be personnel error that involved a transposition of numbers.

Corrective actions included a pre-emptive reset of the setpoints and continuing a review of Instrument and Controls calculation.

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 8/8/2002, at 1400 EDT, Unit 1 and Unit 2 were in the Run mode at a power level of 2763 CMWT (100% of rated thermal power). At that time, Southern Company Services (SCS), the Plant architect/engineer, notified Southern Nuclear (SNC) that the high flow isolation setpoints for the main steam line (MSL, EIIS Code SB) flow instruments 1B21-N686A-D, 1/2B21-N687A-D, 1/2B21-N688A-D, and 1/2B21-N689A-D had previously not been in compliance with the requirements of the Technical Specifications. Unit 1 and 2 Technical Specifications Table 3.3.6.1-1 requires the Group I primary containment isolation system (EIIS Code JM) main steam line high flow isolation setpoints to be $\leq$ 138% of rated steam line flow. However, A/E personnel determined that one of the design input values for the setpoint calculation, specifically the discharge coefficient for the main steam flow nozzles, had been incorrect. The value used in the setpoint calculation was .955; the correct number is .987. The discharge coefficient is a parameter that accounts for pipe flow losses due to friction. It is essentially a ratio of the actual to ideal volumetric flow rate. Substituting the correct discharge coefficient into the calculation resulted in a setpoint that corresponded to 144.65% of Nuclear Boiler Rated (NBR) steam flow. The Technical Specifications allowable value is, as previously mentioned, $\leq$ 138% of rated flow, and the analytical limit is 140%.

Investigations into this potential problem began on 7/24/2002 when General Electric (GE) was in the process of performing instrumentation reviews for a Thermal Power optimization program. In this program, GE , SCS and SNC personnel are exploring the possibility of an increase in the Plant Hatch Units 1 and 2 rated thermal power to 115% of the original licensed power level.

GE personnel noticed that the Appendix K value for this function did not seem to correlate with the Hatch specific calculations. An investigation ensued and, on 7/30/2002, GE notified SCS that a discharge coefficient of .987 had been used for the Appendix K values and the subsequent setpoints, while the specific Hatch calculations had used a different value. SCS then requested GE to assist in the investigation of the origin of the discharge coefficient (.955) being used in the Hatch Specific setpoint calculations. At this point, SCS notified Southern Nuclear of the discrepancy and the ensuing investigation into the adequacy of the .955 discharge coefficient.

On 8/1/2002, with the investigation continuing, a design change request was initiated to change the Hatch Units 1 and 2 setpoints in the more conservative direction. This action was taken as a pre-emptive conservative measure while the investigation ensued. The setpoints provided in the DCR would be based on the same discharge coefficient used in GE's Appendix K calculations, specifically .987. Main steam line high flow calculations based on this coefficient did not present an unacceptable risk of spurious isolations, and so the setpoints were physically changed on 8/6/2002.

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As part of the investigation, the original manufacturer of the Unit 1 main steam line flow restrictors was contacted. They were unable to substantiate a discharge coefficient of .955 and in fact stated that it appeared to be too low. As a result of this and upon review of all the information, it was determined, on 8/8/2002, that the .955 discharge coefficient was incorrect.

This calculational error had been introduced in 1994 as a result of the recalculation of MSL high flow setpoints, prompted by the issuance of GE SIL No. 438, R1. This SIL noted potentially incorrect MSL high flow setpoints resulting from the use of design steam flow conditions for the turbine rather than the rated steam flow conditions for the nuclear boiler. This condition was applicable to Hatch and was reported in LER 50-321/1994-009, R1.

CAUSE OF EVENT

The cause of this event could not be conclusively determined. It is believed that a human error was made when the discharge coefficient was input into the calculation, possibly a transposition of numbers. GE SIL 438 listed a suggested discharge coefficient of .995 for a venturi. It is possible that .995 is the number that the engineers performing the calculation intended to use, and it was simply miscopied as .955.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73(a)(2)(i)(B) because a condition existed which was prohibited by the Technical Specifications. Specifically, the high flow isolation setpoint for the main steam line flow instruments was not in compliance with the requirements of the Unit 1 and 2 Technical Specifications Table 3.3.6.1-1, item 1c.

The flow signal in each of the four main steam lines is derived from four independent differential pressure instruments connected to each of the steam line flow limiters (EIIS Code SB). The flow instruments are designed to detect and isolate breaks in the main steam lines by sending a high flow isolation signal to the Group I PCIS isolation valves, which includes the main steam isolation valves (MSIVs, EIIS Code JM). The signal will close the MSIVs thereby isolating the break and minimizing reactor vessel inventory loss and radioactive material.

The flow limiters, besides providing a differential pressure signal for flow measurement, act to limit the maximum flow in each steam line to about 200% of rated steam flow for that line. Thus, the flow limiters serve to limit the reactor vessel inventory loss and radioactive material release during the time the MSIVs are closing.

Since the flow limiters limit flow to 200% of rated, the setpoint for the high flow isolation must be less than this maximum, or choke, flow. The setpoint also must be sufficiently above 100% of rated flow to prevent spurious trips and to allow on-line testing of the MSIVs. With these considerations in mind, 140% of rated steam flow was chosen as the analytical limit and 138% of rated flow was chosen as the allowable value. Maintaining the setpoint below the allowable value insures that the analytical limit is not exceeded.

In this event, an incorrect discharge coefficient was used in the determination of the high flow isolation setpoint. Specifically, a value of .955 was used, versus the correct value of .987. With the incorrect setpoint, the differential pressure transmitters would have initiated the signal to the PCIS valves at 144.65% of rated steam flow, a value higher than that allowed by the Unit 1 and 2 Technical Specifications.

There are additional leakage detection systems, for example temperature monitors, in the turbine and reactor buildings that would also initiate MSIV closure on a main steam line break. Since their isolation setpoints were unaffected by this error, they would have been capable of closing the MSIVs.

Additionally, the high flow instruments were themselves capable of initiating an isolation, albeit at a later point than assumed in the accident analysis. An evaluation performed by GE for the 1994 event, in which the setpoint was seven percent over the analytical limit, determined that there was no safety significance. GE determined that, without reliance on additional leakage detection instrumentation, the seven percent setpoint difference resulted in less than a ten percent increase in the release of radioactive material. This is significantly less than the 10 CFR 100 limits. The conclusions of that analysis are not adversely affected by the recent Hatch power uprates for the reasons provided below.

As previously described, the MSL high flow instrumentation setpoint is designed to send an isolation signal to the MSIVs for the design basis MSLBA described in the FSAR Section 15.3. For this event, flow through the broken steam lines rises to the flow limiter point (roughly 200 % of rated) in a matter of milliseconds. Therefore, if the postulated break were to occur, the slightly non-conservative adjustment of the MSL high flow setpoints would delay the isolation signal an insignificant amount relative to the 5.5 second closure time assumed in the accident analysis. Because the closure time is not significantly affected, the mass flow out of the break and the radiological consequences of the break are not affected.

In support of the first 5% power uprate, the MSLBA was analyzed in NEDC-32405P, "Power Uprate Safety Analysis Report for Edwin I. Hatch Plant Units 1 and 2", December 1994. The initial conditions for the analysis of the MSLBA are based on hot standby conditions because this condition maximizes the release of coolant through the break and, therefore, the radiological consequences of the event. (Significant, non-mechanistic fuel failures are not assumed for this event). Analyses were performed at two different power levels. Case 1 was performed at a hot standby condition following power operation at 102 % of the original rated power level. Case 2 was also performed at a hot standby condition following power operation at 112.2% of the original rated power level. The calculated total mass release differed by only a fraction of a percent. Radiological consequences did not differ for the two cases and remained well below 10 CFR 100 limits. Based on these results, it was concluded that the MSLBA consequences for the limiting hot standby case were not impacted by an increase in the rated power level. Since increasing from 102 % to 112 % of the

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original rated power had no effect, it was concluded that an increase in rated power to the extended power uprate conditions (approximately 115.7 % of original rated power) would also not have a significant effect. The results of this evaluation were reported to the NRC in NEDC-32749P, "Extended Power Uprate Safety Analysis Report for E.I. Hatch Plant Units 1 and 2 , July 1997", and documented in Unit 2 FSAR section 15.3.4.4.2.2.

In summary, the inventory releases and dose consequences of the MSLBA are not significantly affected by rated power level or the MSL high flow setpoint (as a percent of rated steam flow) as long as the setpoint remains well below the flow allowed by the restrictor. Therefore, this slightly non-conservative setpoint does not affect the dose analysis reported in Table 15.3-8 of the Unit 2 FSAR.

CORRECTIVE ACTIONS

The main steam line high flow isolation setpoints were recalculated based on a discharge coefficient of .987. The new setpoints were provided to the site and were physically adjusted on 8/6/2002.

As previously noted, this condition was initially discovered by General Electric as they were performing a review of Hatch analytical limits and allowable values for a possible future power uprate to 115% of the original licensed power level. As a result, GE has already reviewed many Technical Specifications values and continues to do so. Their review will be completed by December 31, 2002.

ADDITIONAL INFORMATION

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

No failed components caused or resulted from this event.

There was one previously identified event in the last two years in which a setpoint was not in compliance with the Technical Specifications; it was reported in LER 50-321/2001-003. In this event, General Electric notified SNC that setpoints for the Oscillating Power Range Monitor setpoints were potentially non-conservative and thus could potentially fail to protect the Safety Limit Minimum Critical Power Ratio. The setpoints were potentially non-conservative as a result of an incorrect generic analysis.

Corrective actions for the previous event could not have prevented the current event because the causes are different. The previous event was caused by an erroneous generic analysis supplied to the industry by General Electric and the BWR Owner's Group. This particular event was more than likely the result of an individual personnel error, not a faulty generic industry analysis.