NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
EDWIN I. HATCH NUCLEAR PLANT - UNIT 1	05000321	1 OF 5

TITLE (4)
MAIN STEAM LINE HIGH FLOW ISOLATION SETPOINT NOT WITHIN TECHNICAL SPECIFICATIONS LIMITS

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S)
07	28	94	94	009	01	09	22	94		05000
										05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR ?: (Check one or more of the following) (11)

OPERATING MODE (9): 1							
POWER LEVEL (10): 100							
	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (include area code)
STEVEN B. TIPPS, NUCLEAR SAFETY AND COMPLIANCE MANAGER	AREA CODE 912 367-7851

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

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

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

On 7/28/94 at 1130 EDT, Unit 1 was in the Run mode at a power level of 2436 CMWT (100% rated thermal power). At that time, it was determined that the high flow isolation setpoint for the main steam line flow instruments was not in compliance with the requirements of the Unit 1 Technical Specifications. Technical Specifications Table 3.2-1 requires the setpoint for the high flow isolation to be ≤ 138% rated steam line flow (101 psid). However, the actual isolation setpoint was based on 138% design steam line flow (110 psid). Consequently, the actual setpoint for the flow instruments was in excess of the limit. The flow instruments were declared inoperable and Limiting Condition for Operation (LCO) 1-94-254 was entered at 1210 EDT. The setpoint was adjusted to within limits and LCO 1-94-254 was closed at 1905 EDT on 7/28/94.

The cause of this event was personnel error. Nuclear Steam Supply System (NSSS) personnel developing the original main steam line high flow isolation setpoint mistakenly used design steam line flow instead of the required rated flow in the calculations.

The setpoint was recalculated based on rated flow and the setpoint for the flow instruments was adjusted to this new value.

PLANT AND SYSTEM IDENTIFICATION

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

DESCRIPTION OF EVENT

On 7/28/94 at 1130 EDT, Unit 1 was in the Run mode at a power level of 2436 CMWT (100% rated thermal power). At that time, Plant Hatch's Architect/Engineer (A/E) personnel were evaluating the high flow isolation setpoint as part of the power uprate project. While verifying setpoint calculations per General Electric Service Information Letter 438, Revision 1, "Main steam line high flow trip setting," A/E personnel determined that the high flow isolation setpoint for the 16 main steam line (EIIS Code SB) flow instruments (EIIS Code IJ), 1B21-N686A-D, 1B21-N687A-D, 1B21-N688A-D, and 1B21-N689A-D, was not in compliance with the requirements of the Unit 1 Technical Specifications. Specifically, Unit 1 Technical Specifications Table 3.2-1, item 6, requires the setpoint for the Group 1 Primary Containment Isolation System (PCIS, EIIS Code JM) isolation on main steam line high flow to be ≤ 138% rated steam line flow (equivalent to 101 psid as measured across each steam line flow limiter). However, A/E personnel determined that the actual trip setpoint had been calculated based on 138% design steam line flow (design steam line flow is 105% rated flow). Consequently, the actual high flow isolation setpoint for the main steam line flow instruments was approximately 144% rated flow (110 psid), which was more than the Unit 1 Technical Specification limit of 138% rated flow (101 psid).

At 1210 EDT, upon notification of this setpoint discrepancy, Operations personnel declared the 16 main steam line flow instruments inoperable and entered Limiting Condition for Operation (LCO) 1-94-254 as required by the Unit 1 Technical Specifications and plant procedures. Plant Instrument and Controls technicians adjusted the main steam line flow instruments' isolation setpoints to within Unit 1 Technical Specification limits based upon new setpoint calculations performed by A/E personnel and provided to the site on 7/28/94. The flow instruments were then declared operable and LCO 1-94-254 was closed at 1905 EDT.

CAUSE OF EVENT

The cause of this event was personnel error. Nuclear Steam Supply System (NSSS) personnel developing the main steam line high flow isolation setpoint mistakenly used design steam line flow instead of the required rated flow in the calculations. Since design steam line flow is 105% rated flow, this error resulted in a main steam line flow instrument isolation setpoint equivalent to

approximately 144% rated flow, which is above the Unit 1 Technical Specifications limit of 138% rated flow.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73(a)(2)(i) because a condition existed which was prohibited by the plant's Technical Specifications. Specifically, the high flow isolation setpoint for the main steam line flow instruments was not in compliance with the requirements of Unit 1 Technical Specifications Table 3.2-1, item 6. The setpoint is required by this specification to be ≤ 138% rated steam line flow, but, due to an error in the original calculation, the actual setpoint was approximately 144% rated flow.

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), that is, venturis. 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 1 PCIS isolation valves. This signal will close the main steam line isolation valves (EIIS Code JM) thereby isolating the break and minimizing reactor vessel inventory loss and radioactive material release.

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

Since the venturis limit flow to 200% rated flow, the setpoint for the high flow isolation must be less than this maximum (choke) flow. The setpoint also must be sufficiently above 100% rated flow to prevent spurious trips and to allow on-line testing of the main steam line isolation valves (closing one valve would result in each of the remaining three steam lines carrying approximately 133% of its normal flow at 100% power). With these considerations in mind, Plant Hatch's NSSS vendor chose 140% rated steam flow as the analytical limit for the high flow isolation signal; 138% rated flow is the Technical Specification limit, which ensures the analytical limit is not exceeded.

In this event, the NSSS vendor used the incorrect steam flow in determining the value of the differential pressure corresponding to the Technical Specification limit of 138% flow. They calculated the differential pressure at 138% design steam flow; however, design steam flow is 105% rated flow. Consequently, the calculated differential pressure used in setting the flow instruments' isolation setpoint was higher than allowed by the Unit 1 Technical Specifications. The actual instrument isolation setpoint was equivalent to approximately 144% rated steam flow.

APPROVED OMB NO. 3150-0104
EXPIRES: 5/31/95
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

As a result of this error, the size of the minimum steam line break capable of being detected by the flow (differential pressure) instruments increased. However, there are additional leakage detection monitors, for example, turbine building temperature sensors (EIIS Code IJ), which would initiate main steam line valve closure in the event of a main steam line break. Since their isolation setpoints were unaffected by the error in the high flow isolation setpoint, they were capable of closing the main steam line isolation valves at the point assumed in the main steam line break accident analysis.

It also is important to note that the steam line flow instruments were still capable of initiating a main steam line isolation signal, albeit later than assumed in the accident analysis. A conservative, generic evaluation of the higher isolation setpoint, performed by General Electric as part of Service Information Letter 438, Revision 1, indicated that the higher setpoint resulted in less than a ten percent increase in released radioactive material. This evaluation took no credit for any additional leakage detection monitors.

Assuming a ten percent increase in the main steam line break accident doses given in Unit 1 Final Safety Analysis Report section 14.4.5.3 due to the ten percent increase in released radioactive material, the doses at the site boundary from a main steam line break accident would be 1.32×10^{-3} rem "cloud gamma" (whole body) dose and 5.61 rem "thyroid inhalation" (thyroid) dose. Therefore, the resulting doses to the public from a main steam line break accident at the higher isolation setpoint remained significantly less than the 10 CFR 100 limits of 25 rem whole body dose and 300 rem thyroid dose.

On the basis of the preceding discussion, it is concluded that this event had no adverse effect on the health and safety of the public. This analysis is applicable to all power levels.

CORRECTIVE ACTIONS

Plant Hatch's A/E recalculated the main steam line high flow isolation setpoint based on rated steam flow. The new setpoint was provided to site personnel on 7/28/94. Plant Instrument and Controls personnel adjusted the setpoint on the 16 main steam line flow instruments to match the new value calculated by the A/E. The flow instruments were then declared operable and LCO 1-94-254 was closed at 1905 EDT on 7/28/94.

The A/E reviewed the Unit 2 main steam line high flow isolation setpoint calculations. They found the existing setpoint to be in compliance with the requirements of the Unit 2 Technical Specifications; therefore, no changes to the setpoint were necessary or made.

Since the error occurred in the original high flow isolation setpoint calculations some years ago, personnel counseling was not considered to be an effective corrective action.

ADDITIONAL INFORMATION

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

No failed components caused or resulted from this event.

Previous similar events reported in the last two years in which an as-found setpoint was not in compliance with Technical Specification requirements were reported in Licensee Event Reports 50-321/1993-002, dated 4/21/93, and 50-366/1992-019, dated 10/26/92. In these two events, routine testing of the safety relief valves revealed some of their lift setpoints were outside the plus or minus one percent nominal setpoint tolerance allowed by the plant's Technical Specifications. The cause of each of these events was corrosion-induced bonding of the safety relief valve pilot valve disc and seat.

Corrective actions for these previous events would not have prevented this event because their causes were different. This event was caused by personnel error resulting in setpoints being calculated and set incorrectly. However, the previous events were the result of a physical phenomenon causing the safety relief valve lift setpoints to drift from their correct values over the course of an operating cycle. Since the previous corrective actions addressed corrosion-induced bonding, they could not have prevented an event caused by personnel error.