NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
OYSTER CREEK, UNIT 1	0 5 0 0 0 2 1 9	1 OF 0 3

TITLE (4)

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 NAMES	DOCKET NUMBER(S)
0 8	2 1	8 5	8 5	0 1 9	0 0	0 9	2 0	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10):

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

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)	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
N. Ertle, Plant Engineering	AREA CODE 6 0 9 9 7 1 - 4 4 6 5

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)

YES (If yes, complete EXPECTED SUBMISSION DATE) — NO

EXPECTED SUBMISSION DATE (15) — MONTH / DAY / YEAR

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

The calculated differential pressure equivalent to the Technical Specification limit for 120% steam flow, in the main steam lines, was found to be incorrect. The correct value was calculated, reviewed and implemented. Previous surveillances were reviewed against the new value and are discussed in this report.

IE22

NRC Form 366
(9-83)

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

DATE OF DISCOVERY

The date of discovery was August 21, 1985.

IDENTIFICATION OF DISCOVERY

A discrepancy was identified in the established setpoints of the main steam line high flow detectors (RE-22) in a non-conservative direction.

CONDITIONS PRIOR TO DISCOVERY

The plant was shutdown for maintenance.

DESCRIPTON OF DISCOVERY

While troubleshooting a low indication problem with RE-22F, baseline calculations were made for steam flow vs. differential pressure provided by the main steam flow venturis which conflicted with the existing calculation 97.5 psid = 120% steam flow (Technical Specification limit) as recorded in Standing Order #1, "Instrument Setpoints".

Technical Functions Division performed an independent review of the calculations, which concurred that a discrepancy existed and documented the correct values. The correct psid equivalent of 120% steam flow was calculated to be 96.5 psid. The instrument setpoint of 87.5 $\pm$ 2.5 psid was calculated to be 114.18% of steam flow, instead of 108% as recorded in Standing Order #1.

Subsequently, past surveillances from November 6, 1981 through August 21, 1985 were reviewed for violation of 96.5 psid setpoint limit. One surveillance was determined to be reportable as a result of the new psid calculation.

On January 24, 1985, RE-22 sensors C and E setpoints exceeded the previous Technical Specification equivalent limit of 97.5 psid but were not reportable under the reporting criteria of 10CFR50.73. By the equivalent new Technical Specification limit of 96.5 psid, RE-22 sensor B, G, and H setpoints additionally exceeded the Technical Specification limit for the same date and caused this surveillance to become reportable. At no time did the actual steam flow exceed the Technical Specification limit.

APPARENT CAUSE OF OCCURRENCE

The previously calculated value of psid, equivalent to the Technical Specification limit of 120% steam flow, was incorrect in a non-conservative direction.

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

ANALYSIS OF OCCURRENCE AND SAFETY SIGNIFICANCE

A reactor isolation will occur upon high flow sensed in either main steam line. The Technical Specification limit for high flow is established at a value of 120% of rated steam flow. The basis for this setoint is to ensure a reactor isolation occurs, to limit the loss of reactor water inventory, and minimize the dose to the thyroid of an individual at the site boundary in the event of a main steam line break.

Based upon a review of surveillance and calibration records, the worst case deviation from the 120% Technical Specification limit was 122.6% of rated steam flow. The deviation was a result of setpoint drift. The time required to reach 122.6% as compared to 120% of rated flow following a large break would be insignificant when considering instrument response times.

The safety significance of this occurrence is minimal, because a reactor isolation would have occurred, if required. The time delay in actuation would have been insignificant with no appreciable affect on LOCA analysis.

CORRECTIVE ACTIONS

1. Appropriate procedures, log sheets, and documents were properly revised to implement the correct Technical Specification equivalent value in psid for 120% of flow.

2. Appropriate procedures, log sheets, and documents were properly revised to implement the correct value of flow, 114.18% at 87.5 psid for Main Steam Isolation.

3. To preclude recurrence, all Technical Specification related instrument setpoints that measure a process via a derivative variable, (i.e. flow measured by differential pressure) will be checked for accuracy and documented.

4. A method will be implemented in which the verification of Standing Order #1 setpoint can be traced to a GPUN documentation file.

5. The development of the 97.5 psid setpoint referenced in this LER will be researched to determine whether this calculation is isolated to Oyster Creek or generic in nature, (i.e. established by NSSS vendor.) The outcome of this investigation will determine if any additional corrective actions will be taken.

GPU Nuclear

GPU Nuclear Corporation
Post Office Box 388
Route 9 South
Forked River, New Jersey 08731-0388
609 971-4000
Writer's Direct Dial Number:

September 20, 1985

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Licensee Event Report

This letter forwards one (1) copy of Licensee Event Report (LER) No. 85-019.

Very truly yours,

Peter B. Fiedler
Vice President and Director
Oyster Creek

PBF:JR:dam(0085A)
Enclosures

cc: Dr. Thomas E. Murley, Administrator
Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

Mr. Jack N. Donohew, Jr.
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue, Phillips Bldg.
Bethesda, MD 20014
Mail Stop No. 314

NRC Resident Inspector
Oyster Creek Nuclear Generating Station
Forked River, NJ 08731

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