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 4

TITLE (4)

VIOLATION OF APLHGR LIMIT

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 1	0 2	8 5	8 5	0 0 4	0 0	0 3	0 4	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 9 1

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)	[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 (AREA CODE)
Joseph R. Molnar, Core Manager	6 0 9 9 7 1 - 4 6 9 9

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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 fifteen single-space typewritten lines) (16)

The Power Shape Monitoring System (PSMS) is a new core monitoring system which is being used for the first time at Oyster Creek Nuclear Generating Station for Technical Specification thermal limits compliance. During the period of January 2 to January 30, 1985, the Oyster Creek core was highly bottom peaked during high power/flow operation for the first time in the current cycle. During this period no measured Local Power Range Monitor (LPRM) or Traversing Incore Probe (TIP) data feedback adjustments were made to the model. As a result, PSMS power distribution and thermal limits calculations were inadequately monitoring core conditions due to the flux peaking, which was outside the range of calibration of the model. The bottom peaks violated APLHGR limits. APLHGR violations went undetected until such time when PSMS model performance was adjusted to be within the established acceptance criteria. Immediate action was then taken to reduce APLHGR below the Technical Specification limit.

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

Date of Occurrence

The condition described herein was verified on January 30, 1985.

Identification of Occurrence

Technical Specification limits for APLHGR, as given in Section 3.10.A, were exceeded.

This event is reportable as required by 10CFR50.73(a)(2)(i)B.

Description of Occurrence

On January 10, 1985, it was noted that the PSMS calculated TIP traces were under-calculating when compared to measured TIP traces and PSMS model performance was beyond established acceptance criteria. An investigation commenced immediately to confirm the observation and determine the cause of the different flux values. The reactor rod pattern was adjusted on January 15, 1985 in an attempt to reduce the high flux peaks. On January 24, 1985, a complete set of TIP traces were taken to determine if the adjustment reduced the peaks and improved model performance. Upon review of this TIP set, it was noted that flux peaks remained high and PSMS model performance was still outside the established acceptance criteria. At this time, it was suspected that the APLHGR Technical Specification limit was violated. Analysis revealed that PSMS was calculating flux peaks which were approximately 20% lower than the measured peaks. In an attempt to improve PSMS model accuracy, the axial averaged LPRM measurements were fed back into the model via LPRM Feedback Function. Initially, the LPRM feedback option did not function. Software changes corrected this problem and subsequent analysis of the thermal limits resulted in APLHGR values which exceeded Technical Specifications by approximately 10% for about thirty four (34) fuel bundles. Similar analysis of plant data prior to the time of discovering the violation indicated the APLHGR violation first occurred on January 2, 1985.

DETAILS

Core power distributions are calculated by a computer program within PSMS which does not normally require LPRM or TIP data to calculate local flux data. This program is a 3-dimensional nuclear code which is based on the Neutron Diffusion Theory. PSMS can analyze the absolute Root Mean Square (RMS) error between measured and calculated TIP traces. The established acceptance criteria for an absolute error is 7.5%. This acceptance criteria is based on the GETAB uncertainty evaluation for process computers. If the

PSMS absolute error is greater than the established acceptance criteria, the code has the capability to utilize average axial LPRM measurements and/or measured TIP data to reduce the error to an acceptable value.

The bottom flux peaks which existed at Oyster Creek during the month of January were beyond the limits of the PSMS Cycle 10 model and resulted in the under-calculation of the peaks. The statistical analysis function available in PSMS indicated that the code was performing outside established acceptance criteria. When the absolute error was discovered to be greater than 7.5%, the model was adjusted by turning on LPRM and TIP data feedback options. Upon investigation of the LPRM feedback option, it was ascertained the PSMS software did not permit the LPRM feedback option to function, even though the option was turned on. After the software was corrected, both LPRM feedback and the TIP data feedback adjustments confirmed the high flux peaks and violation of APLHGR Technical Specification limits. At that time, January 30, 1985, a second control rod pattern adjustment was implemented to reduce the flux peaks and APLHGRs. A final complete set of TIP traces were taken on January 31, 1985.

Analysis of Occurrence

Technical Specifications (TS) limits on MAPHLGR assure that, in postulated loss of coolant events, reactor response conservatively calculated from the approved Appendix K model would remain within the criteria specified in 10CFR50.46. Since the TS MAPLHGR's were exceeded in about 2% of the core, the calculated loss of coolant response for this part of the core would have been above the 10CFR50.46 criteria.

It should be noted, however, that the approved Appendix K model contains several conservatisms. Estimates of a few of the known conservatisms is in excess of 20% when compared with realistically calculated reactor performance for postulated loss of coolant events. Therefore, from the viewpoint of realistically expected performance, the reactor response would have been below the 10CFR50.46 criteria with ample margin for the entire core.

Corrective Action

Once it was determined that the Technical Specification limits on APLHGR had been exceeded, core thermal power was reduced and the control rod pattern was reconfigured to reduce power peaking. The flattening of the power distribution was sufficient to eliminate the Technical Specification violation.

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

The long term solution of preventing recurrence of this problem is threefold:

1. Improved procedural control will be implemented to more frequently evaluate the PSMS nodal model accuracy and performance. This action will provide us with timely information to quickly determine the code's capability to accurately calculate power distributions and hence thermal limits.

2. If PSMS performance is determined to be outside the established acceptance criteria, immediate corrective action will be taken to ensure ample margin to Technical Specification thermal limits and to adjust the PSMS model. Margins to Technical Specification thermal limits may be increased by reducing recirculation flow or adjustment of the control rod configuration. Model adjustments result in upgrading the accuracy and performance of the nodal model on the basis of comparison with measured plant data. Various established methods for model calibration include feeding measured LPRM and/or TIP data back into the model.

3. Subsequent core operation will be conducted to adhere to the following operational guidelines:

   a. reduce measured TIP peaks,

   b. reduce average relative axial power shape, and

   . perform individual TIP traces during power maneuvering more frequently. Upon completion of each full set of TIP traces the PSMS Statistical Analysis Function will be used to determine nodal model performance.

GPU Nuclear

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Writer's Direct Dial Number:

March 4. 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. 85-004.

Very truly yours,

Peter B. Fiedler
Vice President and Director
Oyster Creek

PBF:PFC:dam (#0714A)
Enclosures

cc: Dr. Thomas E. Murley, Administrator
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NRC Resident Inspector
Oyster Creek Nuclear Generating Station
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