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10 CFR 50.73

February 2, 2007
2130-07-20450

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555 - 0001

Oyster Creek Generating Station
Facility Operating License No. DPR-16
NRC Docket No. 50-219

Subject: Licensee Event Report 2006-004-00, Operation Exceeding Maximum Power Level

Enclosed is Licensee Event Report 2006-004-00, Operation Exceeding Maximum Power Level. This event did not affect the health and safety of the public or plant personnel. This event did not result in a safety system functional failure. There are no new regulatory commitments made in this LER submittal.

If any further information or assistance is needed, please contact Richard Milos, Regulatory Assurance at 609-971-4973 or Sylvain Schwartz, Engineering, at 609-971-4558.

Sincerely,

Timothy S. Rausch, Vice President
Oyster Creek Generating Station

Enclosure: NRC Form 366, LER 2006-004-00

cc: Administrator, USNRC Region I
USNRC Project Manager, Oyster Creek
USNRC Senior Resident Inspector, Oyster Creek
File No. 07050

IE22

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(6-2004)

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@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.

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4. TITLE
Operation Exceeding Maximum Power Level

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
12	08	2006	2006	- 004 -	00	02	02	2007	FACILITY NAME	DOCKET NUMBER 05000
										05000

9. OPERATING MODE: N

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

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Sylvain Schwartz, Engineering	(609) 971-4558

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	SB	PS	B070	N					

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

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

An unplanned reactor power increase to 102.46 % of rated thermal power occurred due to opening and re-closing of one of the five Electromatic Relief Valves (EMRVs). After being open for approximately 57 seconds, the EMRV re-closed without operator action. Subsequent to the EMRV closure, the reactor power increased to 102.46 % and was immediately reduced to less than 100% as a result of operator action to decrease the recirculation flow rate. The most probable cause of this event was the malfunction of the pressure switch connected to the subject EMRV. The pressure switch was subsequently replaced.

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

Unit Conditions Prior to the Event.

The unit was in the Power Operation Operational Condition at 100% power. There were no structures, systems or components out of service that contributed to this event.

Description of the Event

At 13:51 on 12/8/06 the 'D' EMRV opened for about 57 seconds and re-closed without operator action. EMRV actuation was confirmed by operator monitoring of the valve's acoustic monitors, EMRV solenoid actuated control room indication, relief valve tailpiece temperature, decrease in reactor power, decrease in generator output, decrease in reactor pressure and various alarms. Prior to the event the reactor was at 100% power, 1020 psig and normal water level. When the EMRV re-closed, the resultant pressure increase caused reactor power to increase to 102.46%, as indicated on the Average Power Range Monitors (APRMs), for approximately 9 seconds. Reactor power was reduced below 100% when the control room operator decreased the reactor recirculation flow rate.

EMRVs actuate on one of three signals: manually from the Control Room; reactor high pressure, as sensed on its associated IA83 pressure switch; and Automatic Depressurization System (ADS) initiation, which requires three simultaneous signals of Reactor Triple Lo Level, High Drywell Pressure, and Core Spray Booster Pump pressure differential. None of these conditions for ADS initiation existed at the time of actuation.

Prior to the event Instrument Technicians were in the process of performing the Reactor Triple Lo Water Level Calibration, 619.3.006, and had just completed the calibration of level sensor RE18B, which was valved out of service when the EMRV actuated. The Instrument Technicians' actions were confirmed to be consistent with the surveillance procedure and could not explain EMRV's actuation. RE18B was returned to service after the EMRV closed. The IA83D pressure switch, which actuates the 'D' EMRV on high pressure, was inspected for signs of unintentional impact from concurrent work in the area. No sign of impact was found, and the personnel who had been working in the area were interviewed but no inadvertent contact with the pressure switch was reported.

Procedure ABN-40, Stuck Open EMRV, was entered, but the EMRV closed without operator action prior to reaching the steps that would have required repositioning the EMRV's control switch to the OFF position.

Analysis of the Event

There were no actual safety consequences associated with this event. The potential safety consequences of this event were also minimal. Operators took prompt action to decrease reactor power, which terminated the overpower condition in approximately 40 seconds.

The peak reactor power of 102.46% was of very short duration and would have a negligible impact on the decay heat for a Loss of Coolant Accident (LOCA). The LOCA analysis assumes that the initial conditions of the event are at 102% power at steady state, due to uncertainties in measurement and detection of core power. These analysis assumptions bound this event.

Opening of an EMRV at power is an evaluated event that is significantly bounded by the limiting events evaluated each reload and the associated operating limits documented in the Core Operating Limits Report. As such, no reactor vessel (pressure boundary) or fuel safety limits or other Specified Acceptable Fuel Design Limits (SAFDLs) were challenged

A Prompt Investigation has been performed in accordance with current plant procedure "Event Response Guidelines". The investigation team determined that there was no evidence that this event was influenced by human performance. The investigation determined that the most likely cause of this event was a malfunction of the IA83D Pressure Sensor. A review of the as left data for EMRV Pressure Sensor test and calibration surveillance indicates that during the last surveillance test prior to this event, the set point was within specification. However, set point drift of the pressure switch since last surveillance occurred, lowering the set points by approximately 20-25 psig. The as found set point for IA83D after the event was still above the operating pressure at the time of the event. Component history review indicates that in July and August 2005, the IA83D pressure switch had excessive drift identified during the EMRV Pressure Sensor Test and Calibration. This component had a history of drifting in the past year. Furthermore, Operating Experience from Oyster Creek in 1993 indicates an EMRV pressure switch failed due to dirty contacts from natural aging. Based on the above, the degraded pressure switch was considered to be the most probable cause of the spurious opening of the D EMRV.

Cause of the Event

The most probable cause of the event was malfunction of the pressure switch connected to the D EMRV.

LICENSEE EVENT REPORT (LER)

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

Corrective Action Completed.

The IA83D pressure switch and its associated relay were replaced.

Corrective Action Planned.

The new pressure switch performance will be monitored for a year to determine if periodic replacement of the pressure switches is warranted.

Previous Similar Occurrences

There were no previous occurrences of a spurious EMRV opening that caused reactor power to exceed the licensed maximum power level.

Component Data.

Component:	IA83D, Reactor High Pressure
Cause:	Pressure switch degradation
System:	SB (Main/Reheat Steam System)
Component:	PS (Pressure Switch)
Manufacturer:	Barksdale
Model number:	B2S-M12SS