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10 CFR 50.73

December 12, 2005

2130-05-20240

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 2005-005-00, Technical Specification Violation Due to Main Steam Safety Valves Setpoints Discovered Out of Tolerance

Enclosed is Licensee Event Report 2005-005-00, Technical Specification Violation Due to Main Steam Safety Valves Setpoints Discovered Out of Tolerance. 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 Dave Fawcett at 609-971-4284.

Sincerely,

C. N. Swenson
Vice President, Oyster Creek Generating Station

CNS/DIF
Enclosure: NRC Form 366, LER 2005-005-00

cc: S. J. Collins, Administrator, USNRC Region I
P. S. Tam, USNRC Senior Project Manager, Oyster Creek
M. S. Ferdas, USNRC Senior Resident Inspector, Oyster Creek
File No. 05054

IE22

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

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Oyster Creek, Unit 1	05000 219	1 OF 4

4. TITLE
Technical Specification Violation due to Main Steam Safety Valves Setpoints Discovered Out of Tolerance

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
										05000
10	13	2005	2005	- 005	- 00	12	12	2005	FACILITY NAME	DOCKET NUMBER 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)
Jim Correa, Engineering	(609) 971-2208

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	RV	DRESSER	Y					

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)*

Oyster Creek Generating Station was in RUN at 100% power on 10/13/05 when a condition was discovered during routine laboratory as-found testing for Safety Valves (SVs) (EIIC: RV) removed during the 1R20 refueling Outage in November 2004. There were no structures, systems or components out of service that contributed to this event.

In accordance with ASME Boiler & Pressure Vessel Code Criteria the nine SVs removed in refueling outage 1R20 were sent for as-found setpoint testing within the one-year time frame. Based on information received from the laboratory performing SV as-found testing, Site Engineering personnel determined that SV setpoint deficiencies existed with three SVs that were installed during the 1R19 refueling outage. Three (3) of the nine (9) valves exceeded the setpoint tolerance of +/-1% (+/-12 psig) as specified in the Technical Specifications paragraph 2.3F.

All three SVs were within the ASME Code allowable +/-3% tolerance for as found values and there were no actual safety consequences associated with this event.

LICENSEE EVENT REPORT (LER)

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Oyster Creek, Unit 1	05000219	2005	- 005 -	00	2 OF 4

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

Unit Condition Prior to Discovery of the Event

Oyster Creek Generating Station was in RUN at 100% power on 10/13/05 when a condition was discovered during routine laboratory as-found testing for Safety Valves (SVs) (EIIC: RV) removed during the 1R20 refueling Outage in November 2004. There were no structures, systems or components out of service that contributed to this event.

Description of Event

Based on information received from the laboratory performing SV as-found testing, Site Engineering personnel determined that SV setpoint deficiencies existed with three SVs that were installed during the 1R19 refueling outage. In accordance with ASME Code Criteria the nine SVs removed in refueling outage 1R20 were sent for as-found setpoint testing within the one-year time frame. Three (3) of the nine (9) valves exceeded the setpoint tolerance of +/-1% (+/-12 psig) as specified in the Technical Specifications paragraph 2.3F. The three valves, V-1-160 (Serial number BW05087), V-1-165 (Serial number BY08713) and V-1-166 (Serial number BY08714) had as-found values of -2.5% (-30 psig), -1.4% (-17 psig) and -1.9% (-23 psig) respectively. All three SVs were within the ASME Code allowable +/-3% tolerance.

This report is being submitted pursuant to:

10 CFR 50.73 (a)(2)(i)(B): The relief valves (pressure relief function) are required by Technical Specifications to have a +/-1% setpoint tolerance. The setpoints were found out of tolerance after removal from operation. The safety limit was not exceeded. The applicable transient analysis was bounded by previous analysis results therefore the safety limit would not have been exceeded.

Analysis of Event

There were no actual safety consequences associated with this event.

The ASME Boiler and Pressure Vessel Code requires that the Reactor Pressure Vessel (EIIS: RCT) be protected from overpressure during upset conditions by self-actuated relief valves. As part of the nuclear pressure relief system, the size and number of SVs are selected such that the peak pressure in the nuclear system will not exceed the ASME Code limits for Reactor Coolant Pressure Boundary. The nine installed SVs discharge steam directly to the Drywell. The SVs are located on the two main steam lines (EIIS: SB) within the Drywell. The SVs are spring safety valves.

During Cycle 19 operations, there were no plant transients that required SV operation. The as-found setpoints for the three SVs that tested outside of their Technical Specification allowable range were all low. Even though the three valves have setpoints below the -1% Technical Specification limit the valves would have functioned properly (opened and passed full flow at 3% accumulation) to provide pressure relief capability. The average setpoint tolerance for all nine valves is within the +/-1% limit and therefore the valves would have limited overpressure to below 110% of design pressure as required.

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

Analysis of Event (cont'd)

An evaluation of the condition with regard to the Overpressure Protection Analysis does not have to be performed since the valves would have limited overpressure to below 110% (1375 psig) of design pressure (1250 psig). The average setpoint tolerance of all nine valves was -0.69% (-8.3 psig) which is within the +/-1% limit. The Basis of Technical Specification 4.3E states that "with all safety valves set 12 psig higher the safety limit of 1375 psig is not exceeded".

This event is not considered risk significant. The applicable transient was bounded by previous analysis results therefore the safety limit would not have been exceeded.

This event is reportable under 10 CFR 50.73(a)(2)(i)(B)

Cause of Event

The cause of the three SVs being outside of their allowable as-found setpoint is due to setpoint drift. The ASME Code acknowledges setpoint drift by requiring the as-left setpoint to be +/-1% and allowing the as-found setpoint to be +/-3%.

Corrective Actions:

The nine SVs removed during the 1R20 refueling Outage in November 2004 were replaced with refurbished SVs that met the Technical Specification 4.3E requirement of an as-left setpoint tolerance of +/-1%.

Additional Information:

A. Failed Components:

Three Main Steam Line Safety Valves (SVs) were determined to have setpoints out of tolerance.

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

B. Previous Similar Events:

Based on further review of this event, it has been determined that during previous operating cycles, Oyster Creek has found similar out of tolerance results from previous laboratory testing. At the time of those discoveries it was determined that those events were not reportable in accordance with the guidance found in NUREG 1022 regarding criteria for not reporting routine surveillance results having no safety consequences. In previous events and this current event, it was determined that the capability of performing the safety functions of the SVs with the out of tolerance setpoints was maintained.

C. Identification of Components referred to in this Report:

Components	IEEE 805 System ID	IEEE 803A Function
Safety Valves	EIIS-SB	EIIC-RV
Reactor Pressure Vessel	EIIS-RCT	EIIC-RPV