Exelon Generation.

Oyster Creek
Route 9 South
P.O. Box 388
Forked River, NJ 08731

10 CFR 50.73

RA-16-094

November 16, 2016

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk or O-8B1
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

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

Subject: Licensee Event Report (LER) 2016-004-00, "Technical Specification Violation Due to Main Steam Safety Valve Setpoint Discovered Out of Tolerance".

Enclosed is LER 2016-004-00 reporting the Technical Specification violation due to a main steam safety valve setpoint discovered out of tolerance during as-found testing, which occurred on September 29, 2016.

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 regulatory commitments made in this LER submittal.

Should you have any questions concerning this report, please contact Mike McKenna, Regulatory Assurance Manager, at (609) 971-4389.

Respectfully,

Michael Gillin

Michael Gillin
Plant Manager
Oyster Creek Nuclear Generating Station

Enclosure: NRC Form 366, LER 2016-004-00

cc: Administrator, NRC Region I
NRC Senior Resident Inspector - Oyster Creek Nuclear Generating Station
NRC Project Manager - Oyster Creek Nuclear Generating Station

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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(06-2016)

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2018

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

(See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Oyster Creek	05000219	1 OF 4

4. TITLE
Technical Specification Violation Due to Main Steam Safety Valve Setpoint 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
09	29	2016	2016 -	004 -	00	11	15	2016	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: N

10. POWER LEVEL: 000

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)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
	☐ 50.73(a)(2)(i)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A	

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Michael McKenna, Regulatory Assurance Manager	609-971-4389

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 15. EXPECTED SUBMISSION DATE)* ☒ NO				

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

Oyster Creek Nuclear Generating Station was in Cold Shutdown on 09/29/16 for Refueling Outage 1R26 when a condition was discovered during routine laboratory as-found testing that a Safety Valve (SV) (EIIC:RV) removed on 09/27/16 as part of refueling outage maintenance activities did not meet required setpoint tolerances. There were no structures, systems or components out of service that contributed to this event.

In accordance with American Society of Mechanical Engineers (ASME) Operation and Maintenance (O&M) Code requirements, two (2) of the nine (9) SVs installed in the plant were scheduled for removal during the refueling outage (1R26) and sent for as-found laboratory testing. Based on information received from the laboratory performing SV as-found testing, Site Engineering personnel determined that SV setpoint deficiencies existed with one (1) of the two (2) SVs removed and sent for testing. Per ASME O&M Code Mandatory Appendix I Section I-1320, two additional SVs were removed from the plant and sent for as-found testing and both met the required setpoint acceptance criteria. One (1) of the four (4) SVs tested exceeded the setpoint tolerance of +/-3% (+/-36 psig) as specified in the Technical Specifications (TS), Paragraph 2.3.F. This resulted in a condition prohibited by TS and is considered reportable pursuant to 10 CFR 50.73(a)(2)(i)(B).

NARRATIVE

Unit Condition Prior to Discovery of the Event

Oyster Creek Nuclear Generating Station (OCNGS) was in Cold Shutdown on 09/29/16 for Refueling Outage 1R26 when a condition was discovered during routine laboratory as-found testing that a Safety Valve (SV) removed on 09/27/16 as part refueling outage maintenance activities did not meet required setpoint tolerances. There were no structures, systems or components out of service that contributed to this event.

Description of Event

In accordance with American Society of Mechanical Engineers (ASME) Operation and Maintenance (O&M) Code requirements, two (2) of the nine (9) SVs installed in the plant were scheduled for removal during the refueling outage (1R26) and sent for as-found laboratory testing. Based on information received from the laboratory performing SV as-found testing, Site Engineering personnel determined that SV setpoint deficiencies existed with one (1) of the two (2) SVs that were removed and sent for testing. Pursuant to ASME O&M Code Mandatory Appendix I Section I-1320, two (2) more SVs were removed from the plant and sent for as-found laboratory testing and both met the required setpoint acceptance criteria. One (1) of the four (4) SVs tested exceeded the setpoint tolerance of +/-3% (+/-36 psig) as specified in the Technical Specifications (TS), Paragraph 2.3F. One (1) valve, V-1-164 (Serial number BW05085) was determined to have an as-found setpoint value of -3.3% (-40 psig). The ASME Boiler and Pressure Vessel (B&PV) Code stipulates that relief valves (pressure relief function) have an as-found setpoint tolerance of +/- 3%.

This report is being submitted pursuant to 10 CFR 50.73(a)(2)(i)(B) for a condition prohibited by Technical Specifications (TS), since one (1) of the SVs removed and tested exceeded its allowable TS setpoint tolerance of +/- 36 psi (+/- 3%). The setpoint for the one (1) SV was found out-of- tolerance after removal from operation. The safety limit was not exceeded. The applicable transient analysis was bounded by previous analysis results and, therefore, the safety limit would not have been exceeded.

Equipment Description

There are a total of nine (9) SVs installed to prevent failure of the Reactor Pressure Vessel (RPV) on an over-pressurization event (Code-required – refer to ASME B&PV Code). Each valve is designed to limit RPV pressure to 1375 psig (110%) with a Main Steam Isolation Valve closure while operating at 1930 MWt, under the conditions where the:

- Drywell Recirculation Pumps fail to trip
- Turbine Bypass valves fail to open
- Isolation Condensers fail to initiate
- Electromatic Relief Valves fail to open

These conditions assume a reactor SCRAM on High Flux where all nine (9) SVs are required to turn the pressure transient. The ASME B&PV Code allows an as-found +/- 3% of setpoint pressure variation in the lift point of the valves. Four (4) out of nine (9) SVs have a setpoint of 1212 psig, and the remaining five (5) out of nine (9) valves have a setpoint of 1221 psig.

Analysis of Event

NARRATIVE

There was no actual safety consequences associated with this event.

The ASME B&PV Code requires that the RPV (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 (9) 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-actuated safety valves.

During Cycle 25 operations, there were no plant transients that required SV operation. The as-found setpoint for the one (1) SV that tested outside its TS allowable range was low. Even though the valve had setpoint below the - 36 psi (- 3%) TS limit, the valve would have functioned properly to provide pressure relief capability.

An evaluation of the condition with regard to the Overpressure Protection Analysis does not have to be performed since the nine (9) valves would have limited overpressure to below 110% (1375 psig) of design pressure (1250 psig). The Bases of TS 4.3E states that: "...with all safety valves set 36 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 SV being outside of its allowable as-found setpoint is attibuted 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%. [The TS specify that the testing is done per TS 4.3C.]

Corrective Actions

In accordance with ASME O&M Code Mandatory Appendix I Section I-1320, two (2) more SVs were removed from the plant and sent for as-found laboratory testing. Pursuant to ASME Code requirements all four (4) SVs that were removed during 1R26 refuel outage were replaced with refurbished SVs that met the Technical Specification 4.3E requirement of an as-left setpoint tolerance of +/- 1%.

Assessment of Safety Consequences

The SV being outside of the allowable as-found setpoint did not directly impact the valve's ability to maintain RPV pressure below the 1375 psig limit. The valve lifted at 1181 psig during testing, or 40 psig lower than the ASME Code setpoint, and is no longer installed in the plant. In accordance with the plant's TS, an analysis was performed which shows that with all nine (9) safety valves set 36 psig higher, the safety limit is not exceeded.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER YEAR	SEQUENTIAL NUMBER	REV NO.
Oyster Creek, Unit 1	05000-219	2016	- 004	- 00

NARRATIVE

Updated Final Safety Analysis Report (UFSAR) Section 5.2 states this analysis is performed each fuel cycle and the results reported in the supplemental reload licensing report. As the SVs present distinctly different concerns than those related to the relief valves, the plant's TS separately discuss the actions to be taken upon inoperability. The actuation of an SV will be immediately detectable by an observed increase in drywell pressure. Further confirmation can be gained by observing reactor pressure and water level. Operator action in response to these symptoms would be taken regardless of the acoustic monitoring system status, used to alert control room operators of a SV which is stuck open.

A review of Bases for TS Section 3.13 was performed to ensure that the issue was not indicative of a common mode failure. After receiving the results of the SV as-found testing, two (2) additional valves were removed for laboratory testing, one (1) with a setpoint 1221 psig setpoint, and one (1) with a setpoint of 1212 psig. Both valves were confirmed to meet the required as-found setpoint tolerances.

Additional Information:

A. Failed Components:

One Main Steam Line SV determined to have setpoint out-of-tolerance.

B. Previous Similar Events:

A similar event was identified in October 2005 and reported under LER 2005-005-00. In that specific event, three (3) SVs failed to meet as-found testing. At that time, the plant's TS required SVs setpoint to have a tolerance of +/- 1%. In 2006 OCNGS TS were revised (reference License Amendment No. 261) to align with ASME Code requirements and the lift setpoint tolerances were changed to reflect a +/- 3% variation (+/- 36 psi) for each valve.

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