NEXTera ENERGY SEABROOK

April 5, 2018

10 CFR 50.73

Docket No. 50-443
SBK-L-18065

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

Seabrook Station
Licensee Event Report (LER) 2018-001-00
Pressurizer Safety Valve Outside of Technical Specification Limits Discovered During As-Found Set Point Testing

Enclosed is Licensee Event Report (LER) 2018-001-00. This LER reports an event that occurred at Seabrook Station on February 06, 2018. This event is being reported pursuant to the requirements of 10 CFR 50.73(a)(2)(i)(B).

Should you require further information regarding this matter, please contact me at (603) 773-7932.

Sincerely,

NextEra Energy Seabrook, LLC

Kenneth Browne
Licensing Manager

cc: D. Lew, Acting NRC Region I Administrator
J. Poole, NRC Project Manager
P. Cataldo, NRC Senior Resident Inspector

IE22
NRR

Enclosure to SBK-L-18065

NRC FORM 366
(02-2018)

U.S. NUCLEAR REGULATORY COMMISSION

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/)

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 03/31/2020

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Information Services Branch (T-2 F43), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@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 an 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.

1. Facility Name	2. Docket Number	3. Page
Seabrook Station	05000 443	1 OF 3

4. Title

Pressurizer Safety Valve Outside of Technical Specification Limits Discovered During As-Found Set Point Testing

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
02	06	2018	2018	- 001	- 00	04	05	2018	Facility Name	Docket Number 05000

9. Operating Mode	11. This Report is Submitted Pursuant to the Requirements of 10 CFR §: *(Check all that apply)*			
1	☐ 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)
10. Power Level	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
100	☐ 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)(C)	☐ 73.77(a)(2)(ii)
	☐ 20.2203(a)(2)(vi)	☑ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(iii)
		☐ 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)
Joshua Greene, Licensing Engineer	603-773-7851

13. Complete One Line for each Component Failure Described in this Report

Cause	System	Component	Manufacturer	Reportable to ICES	Cause	System	Component	Manufacturer	Reportable to ICES
X	AB	RV	C711	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 14 single-spaced typewritten lines)

On February 6, 2018, with Seabrook Station at 100% power, it was determined that one of the Pressurizer Safety Valves (PSVs) had a high as-found set point pressure which was discovered during off site testing after the valve was removed from service during the previous refueling outage (04/08/17). Technical Specifications (TS) require three pressurizer safety valves to be operable, with a lift setting of +/- 3% in Modes 1, 2 & 3. One safety valve had as-found set point of +6.1%, the other two were within the +/- 3% tolerance. Although the TS limit was surpassed, the ASME code limitation of 110% design pressure was not exceeded. The contributing cause of the excessive set point pressure is attributed to set point drift. A direct cause of the excessive set point drift was not able to be determined. The Corrective Action was to replace the spring. Following that, additional testing was performed to ensure the valve meets design specifications and would be suitable for future service. The additional testing was satisfactory and within design specifications.

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

(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. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Seabrook Station	05000- 443	2018	- 001	- 00

NARRATIVE

Event Description:

During testing of the Pressurizer Safety Valves (PSV) (EEIS: RV) performed by an off-site vendor, the as-found set point for one of the three PSVs removed during the stations most recent refueling outage did not meet the Technical Specifications (TS) 3.4.2.2 requirement of being with +/- 3% of design lift pressure. The subject PSV had an as-found set point pressure that was +6.1%, which was validated on February 06, 2018. The PSV was last installed in the plant from October 17, 2015 until April 08, 2017 when it was removed from service.

TS 3.4.2.2 has an action statement completion time of 15 minutes for one Inoperable Pressurizer Code Safety Valve, after which the plant must be in Hot Standby within 6 hours, and in at least Hot Shutdown within the following 6 hours. Since it is assumed the condition existed prior to the as-found testing, and for a duration longer than the TS action completion time, it is assumed that the condition existed greater than the allowed out of service time and completion time. This event is being reported in accordance with 50.73(a)(2)(i)(B), "Any operation or condition which was prohibited by the plant's Technical Specifications."

Safety Consequences:

There were no actual safety consequences for this event. No safety system responses occurred.

The Pressurizer Code Safety Valves have a lift setting of 2485 psig, +/- 3%.

The Pressurizer Code Safety Valves operate to prevent the Reactor Coolant System (RCS) (EEIS: AB) from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown.

During operation, all pressurizer Code safety valves must be operable to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Trip System Trip Set point is reached (i.e., no credit is taken for a direct Reactor trip on the loss of load) and also assuming no operation of the power-operated relief valves or steam dump valves.

The Reactor Coolant System (RCS) is protected from overpressure regardless of reactor trip, assuming all safety valves function properly. Should the reactor trip at the first protection grade trip (high pressurizer pressure), then only 40 percent of the total valve capacity is required. This 40 percent capacity readily falls within that provided by two of the three safety valves.

Although the TS limit was surpassed, the ASME code limitation of 110% design pressure was not exceeded.

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NARRATIVE

In conclusion, the RCS is adequately protected from overpressure with only two (of the three) safety valves if the reactor trips at the first protection grade trip set point.

Cause of Event:

The contributing cause of the excessive set point pressure is attributed to set point drift. A direct cause of the excessive set point drift was not able to be determined as material conditions were within OEM specifications. However, some spring characteristics had changed sufficiently that a decision was made to install a new one.

Corrective Actions:

The Corrective Action was to replace the spring. After the spring was installed, testing of the valve reoccurred and the as-left conditions of the valve were within required specifications. Additionally, procedures are being revised to add conservative thresholds and evaluation criteria to determine if subsequent springs are suitable for future service.

The currently installed PSVs all have springs that measured within the acceptable limits during their most recent inspection. Their test results were also reviewed and it was noted that the valves did not contain similar behaviors. Therefore, there are no concerns with the currently installed PSVs.

Previous Similar Events:

Similar conditions were documented at Indian Point 3, LER Number 2012-003-01, and Palo Verde Nuclear Generating Station 1, LER 50-528/2003-001-01, where as-found PSV set point testing occurred.

The PSV is manufactured by Crosby Valve and Gage Co., Size 6M6, Style HB-BP 86, Type E.