ENERGY NORTHWEST

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June 9, 2022

GO2-22-058 10 CFR 50.73

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 LICENSEE EVENT REPORT NO. 2021-002-01**

Dear Sir or Madam:

Transmitted herewith is Revision 1 to Licensee Event Report No. 2021-002-00 for Columbia Generating Station. This report is submitted pursuant to 10 CFR 50.73(a)(2)(ii)(A) and 10 CFR 50.73(a)(2)(v)(C).

There are no commitments being made to the Nuclear Regulatory Commission by this letter. If you have any questions or require additional information, please contact Ms. D.M. Wolfgramm, Regulatory Affairs Manager, at (509) 377-4792.

Executed on this 9 day of June, 2022.

Respectfully,

DocuSigned by:
Jeremy Hauger Acting for
B3D71514C4434A3...

J. Kent Dittmer
Vice President, Engineering

Attachment: Licensee Event Report 2021-002-01

cc: NRC Region IV Regional Admin
NRC Region IV Project Manager
NRC Senior Resident Inspector/988C
CD Sonoda – BPA/1399

NRC FORM 366 (08-2020)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See Page 3 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
Columbia Generating Station	05000 397	1 OF 3

4. Title Integrated Leak Rate Test

5. Event Date			6. LER Number			7. Report Date		
Month	Day	Year	Year	Sequential Number	Revision No.	Month	Day	Year
09	28	2021	2021	- 002 -	01	06	09	2022

8. Other Facilities Involved

Facility Name	Docket Number
	05000
	05000

9. Operating Mode	10. Power Level
5	0

11. This Report is Submitted Pursuant to the Requirements of 10 CFR §: *(Check all that apply)*

10 CFR Part 20	☐ 20.2203(a)(2)(vi)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(A)	**10 CFR Part 73**
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.69(g)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(4)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(i)(A)	☑ 50.73(a)(2)(v)(C)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(i)	**10 CFR Part 21**	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(1)(i)
☐ 20.2203(a)(2)(ii)	☐ 21.2(c)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(i)
☐ 20.2203(a)(2)(iii)	**10 CFR Part 50**	☑ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)	☐ 73.77(a)(2)(ii)
☐ 20.2203(a)(2)(iv)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)	
☐ 20.2203(a)(2)(v)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)	

☐ **OTHER** (Specify here, in abstract, or NRC 366A).

12. Licensee Contact for this LER

Licensee Contact	Phone Number (Include area code)
Valerie Lagen, Principal Licensing Engineer	(509) 372-5507

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

Cause	System	Component	Manufacturer	Reportable to IRIS	Cause	System	Component	Manufacturer	Reportable to IRIS
X	BO	ISV	V085	Y					

14. Supplemental Report Expected		15. Expected Submission Date	Month	Day	Year
☑ No	☐ Yes (If yes, complete 15. Expected Submission Date)				

16. Abstract (Limit to 1560 spaces, i.e., approximately 15 single-spaced typewritten lines)

At 12:31 PDT on November 3, 2021, it was determined that the primary containment leakage rate limits specified in Technical Specifications were exceeded principally due to the as-found leakage values obtained during local leak testing associated with suppression pool spray isolation valve RHR-V-27B. The tests were conducted during the R24 and R25 refueling outages at which time primary containment was not required to be operable. On May 23, 2019, during refueling outage R24, required test pressure could not be established for RHR-V-27B, resulting in an as-found leakage assumed to be greater than the maximum allowable containment leakage rate La. On June 5, 2021, during refueling outage R25, testing of RHR-V-27B resulted in a total as-found type B and C summation leakage rate of greater than 0.6 La (maximum allowable containment leakage rate).

The most likely cause of failure was determined to be due to the seats drying out and increased susceptibility to seat leakage. In both cases RHR-V-27B was flushed and retested satisfactorily prior to entering the mode of applicability.

This event was reported to the NRC as an eight-hour, non-emergency Event Notification Number 55560 on November 3, 2021, per 10 CFR 50.72(b)(3)(ii)(A), “event or condition that results in the nuclear power plant, including its principal safety barriers, being seriously degraded”.

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.
Columbia Generating Station	05000- 397	2021	- 002	- 01

NARRATIVE

Plant Conditions

At 12:31 PDT on November 3, 2021 it was determined that the local leak rate tests (LLRT) conducted on RHR-V-27B [ISV] suppression pool spray isolation valve during refueling outages R24 and R25 demonstrated leakage rates that exceeded primary containment leakage limits. This valve is used to provide spray cooling to the wet-well portion of the primary containment [NH] to control containment pressure during an accident. The first occurrence was during the R24 refueling outage on May 23, 2019 when the plant was in Mode 5 at 0% power. The second occurrence was during the R25 refueling outage on June 5, 2021 when the plant was in Mode 5 at 0% power. There were no safety related structures, systems, or components that were inoperable at the time of these occurrences that could have contributed to the event.

Event Description

On September 28, 2021, discrepancies were identified in the refueling outage R25 LLRT documentation for RHR-V-27B as-left (AL) and as-found (AF) values. A past operability assessment was conducted and on November 3, 2021 concluded that LLRT testing exceeded AF Technical Specification (TS) acceptance criteria for refueling outages R24 and R25.

On May 23, 2019, during the R24 refueling outage, RHR-V-27B failed its local leak rate test. The required test pressure could not be established, so the AF leakage was assumed to be greater than 1.0 La, the maximum allowable containment leakage rate. The assumed leakage exceeded both the leakage rate acceptance criteria for summation of Type B and Type C AF leakage of 0.6 La as well as primary containment leakage rate acceptance criterion of TS 5.5.12, which is less than or equal to 1.0 La. The valve was cycled to flush with water and the LLRT was repeated which resulted in an AL leakage rate within the acceptance criterion.

During the R25 refueling outage, an LLRT of RHR-V-27B was completed on June 05, 2021. The AF leakage rate was approximately 70,000 standard cubic centimeters per minute (sccm). The total summation for Type B and Type C AF leakage exceed 0.6 La, with RHR-V-27B measured AF leakage representing the primary leakage contributor. The valve was cycled to flush with water and the LLRT was repeated which resulted in an AL leakage rate within the acceptance criterion.

Operability is maintained through a combination of visual and leak testing described in the Columbia Generating Station (CGS) Primary Containment Leakage Rate Testing Program (LLRT-01). The LLRT-01 test program references ANSI/ ANS-56.8-2002 Section 6.4.4 surveillance acceptance criteria which states that the combined AF leakage rate of all Type B and Type C tests shall be less than or equal to 0.6 La when evaluated on a minimum path leak rate basis.

This event is reportable per 10 CFR 50.73(a)(2)(ii)(A) due to one of the plant's principal safety barriers being seriously degraded. This event is also reportable per 10 CFR 50.73(a)(2)(v)(C) as an event or condition that could have prevented fulfillment of a safety function of structures or systems that are needed to control the release of radioactive material.

This event was reported to the NRC as an eight-hour, non-emergency Event Notification Number 55560 on November 3, 2021, per 10 CFR 50.72(b)(3)(ii)(A), “event or condition that results in the nuclear power plant, including its principal safety barriers, being seriously degraded”.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER YEAR	SEQUENTIAL NUMBER	REV NO.
Columbia Generating Station	05000- 397	2021	- 002	- 01

NARRATIVE

Immediate Corrective Action

After the local leak rate test failures of RHR-V-27B, the valve seats were flushed and cycled prior to successfully retesting while the unit remained in Mode 5. Containment integrity is not required in Mode 5. The as-left leakage rates met the acceptance criterion prior to restarting the plant in both R24 and R25.

Cause of Event

As part of the R24 refueling outage, the required test pressure for LLRT testing of RHR-V-27B could not be obtained. During the R25 refueling outage, LLRT testing of RHR-V-27B resulted in a total as-found Type B and C summation leakage rate of greater than 0.6 La (maximum allowable containment leakage rate).

Previous experience with valves that are similar in style and seat to RHR-V-27B has shown drying out of the seats due to prolonged absence of water, increasing the susceptibility to seat leakage. The downstream (primary containment) side of RHR-V-27B is routinely exposed to prolonged periods of exposure to air. This drying out effect could cause adverse leakage results. In both R24 and R25 the valve was stroked and flushed with water, prior to the as-left LLRT testing which resulted in an acceptable leakage rates.

Assessment of Safety Consequences

The failure of RHR-V-27B occurred in an operating mode where it’s safety function of preventing the release of radioactive material from the containment structure was not required and the valve was repaired and satisfactorily retested prior to entering Mode 1, 2, or 3. Leakage past RHR-V-27B did not represent an actual open pathway in the physical integrity of reactor containment. The suppression pool spray line represents a closed system which takes suction from the suppression pool and discharges back to the suppression pool. There was no actual safety consequence associated with this event since there was no loss of safety function and no potential for radiological release.

Similar Events

A review of CGS LERs for the past three years was performed and did not identify any previously reported issues of this type.

Further Corrective Actions

During the next planned outage, the Primary Containment Isolation Valve will be subject to diagnostic testing, contingency repairs and required LLRT testing prior to being returned to operability.

Energy Industry Identification System codes from IEEE Standards 805-1984 and 803-1983 are represented in brackets throughout the body of the narrative.