Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381

January 7, 2021
WBL-20-068

10 CFR 50.73

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

Watts Bar Nuclear Plant, Unit 2
Facility Operating License No. NPF-96
NRC Docket No. 50-391

Subject: **Licensee Event Report 391/2020-004-00, Steam Generators Degraded Due to Axial Outside Diameter Stress Corrosion Cracking**

This submittal provides Licensee Event Report (LER) 391/2020-004-00. This LER provides details concerning a degraded condition associated with the steam generator tubes at Watts Bar Nuclear Plant (WBN) Unit 2. This condition is being reported as a condition of one of the plant's principal safety barriers being seriously degraded in accordance with 10 CFR 50.73(a)(2)(ii)(A).

There are no regulatory commitments contained in this letter. Please direct any questions concerning this matter to Tony Brown, WBN Licensing Manager, at (423) 365-7720.

Respectfully,

Anthony L. Williams IV
Site Vice President
Watts Bar Nuclear Plant

Enclosure:

LER 391/2020-004-00 “Steam Generators Degraded Due to Axial Outside Diameter Stress Corrosion Cracking”

cc (Enclosure):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Watts Bar Nuclear Plant
NRC Project Manager - Watts Bar Nuclear Plant

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ENCLOSURE
Tennessee Valley Authority
Watts Bar Nuclear Plant
Unit 2

LER 391/2020-004-00 "Steam Generators Degraded Due to Axial Outside Diameter Stress Corrosion Cracking"

NRC FORM 366 (08-2020) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2023

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 https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

1. Facility Name	2. Docket Number	3. Page
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4. Title
Steam Generators Degraded Due to Axial Outside Diameter Stress Corrosion Cracking

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
11	11	2020	2020	- 004 -	00	1	7	2021	N/A	05000
									NA	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 in NRC 366A).

12. Licensee Contact for this LER

Licensee Contact	Phone Number (Include Area Code)
Dean Baker, Licensing Engineer	(423) 452-4589

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
B	AB	HX	W120	Y					

14. Supplemental Report Expected		15. Expected Submission Date	Month	Day	Year
☒ No	☐ **Yes** (If yes, complete 15. Expected Submission Date)		N/A	N/A	N/A

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

At 1311 EST on November 11, 2020, it was determined, after evaluation of the Watts Bar Nuclear Plant (WBN) Unit 2 Steam Generator (SG) tube eddy current test data collected during the ongoing refueling outage, that the WBN Unit 2 Reactor Coolant System pressure boundary did not meet the performance criteria for SG tube structural integrity. Specifically, SG number 3 failed the condition monitoring assessment for conditional burst probability.

The cause of the degradation in the SGs, and particularly SG number 3, is axial outside diameter stress corrosion cracking (ODSCC) of the Alloy 600 mill annealed (MA) SG tubing coincident with the carbon steel tube support plate intersections. Corrective actions taken include plugging, and stabilizing if required, all SG tubes as required by the SG program. Corrective actions to prevent recurrence include a planned mid-cycle SG inspection and steam generator replacement.

This event is being reported to the Nuclear Regulatory Commission (NRC) under 10 CFR 50.73(a)(2)(ii)(A) as a condition that resulted in the plant's principal safety barriers being seriously degraded.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER YEAR	SEQUENTIAL NUMBER	REV NO.
Watts Bar Nuclear Plant, Unit 2	05000-391	2020	- 004	- 00

NARRATIVE

I. Plant Operating Conditions Before the Event

Watts Bar Nuclear Plant (WBN) Unit 2 was in Mode 5 at 0 percent rated thermal power (RTP).

II. Description of Event

A. Event Summary

At 1311 EST on November 11, 2020, it was determined, after evaluation of the Watts Bar Nuclear Plant (WBN) Unit 2 Steam Generator (SG)[EIIS:HX] tube eddy current test data collected during the U2R3 refueling outage, that the WBN Unit 2 Reactor Coolant System (RCS)[EIIS:AB] pressure boundary did not meet the performance criteria for SG tube [EIIS:TBG] structural integrity. Specifically, SG number 3 failed the condition monitoring assessment for conditional burst probability in accordance with Generic Letter (GL) 95-05, “Voltage Based Repair Criteria.” The conditional probability of burst limit in GL 95-05 is 1E-02; however, this limit was exceeded for SG 3. WBN has completed tube plugging and additional corrective actions are in progress.

This event is being reported to the Nuclear Regulatory Commission (NRC) under 10 CFR 50.73(a)(2)(ii)(A) as a condition that resulted in the plant’s principal safety barriers being seriously degraded.

B. Status of structures, components, or systems that were inoperable at the start of the event and that contributed to the event

No inoperable structures, systems, or components contributed to this condition.

C. Dates and approximate times of occurrences

Date	Time (EST)	Event
11/11/20	1311	Received preliminary condition monitoring report that indicated SG#3 failed accident induced conditional burst probability..
11/11/20	1611	Event Notification 54994 made to NRC.

D. Manufacturer and model number of each component that failed during the event

The degraded steam generators are Model D3 SGs manufactured by Westinghouse.

E. Other systems or secondary functions affected

No other systems or secondary functions were affected.

F. Method of discovery of each component or system failure or procedural error

The degraded condition of the steam generators was discovered during periodic eddy current testing (ECT) of the steam generator tubes.

G. Failure mode, mechanism, and effect of each failed component

The failure mechanism of the SG tubes that resulted in failing to meet condition monitoring is axial outside diameter stress corrosion cracking (ODSCC) of the SG tubing at the carbon steel tube support plate (TSP) intersections.

H. Operator actions

No operator actions were required.

I. Automatically and manually initiated safety system responses

None required.

III. Cause of the Event

A. Cause of each component or system failure or personnel error

The degradation mechanism leading to the equipment failure event is axial ODSCC of the Alloy 600 mill annealed (MA) SG tubing coincident with carbon steel TSP intersections. The potential for this degradation to occur in Alloy 600MA tubing has been widely documented through industry operating experience (OE). Although the degradation mechanism was an expected occurrence at Watts Bar Unit 2, the growth rate identifed was greater than projected. Operating temperatures of the tube material are elevated at WBN Unit 2 and coupled with localized crevice chemistry at the TSP intersections creates an undesirable condition that leads to initiation and growth of ODSCC.

B. Cause(s) and circumstances for each human performance related root cause

No human performance root causes are associated with this event.

IV. Analysis of the Event

WBN Unit 2 is a four loop Pressurized Water Reactor (PWR) provided by Westinghouse. The SGs installed at WBN Unit 2 are Westinghouse Model D3 SGs with Alloy 600 MA tubing. This particular SG Model and tubing has a history of tube degradation that requires them to be replaced relatively early in plant life. Replacement SGs have already been procured for WBN Unit 2 and they were scheduled for replacement during the fifth refueling outage for WBN Unit 2.

During the WBN Unit 2 third refueling outage, a 100 percent eddy current inspection of all four SGs was performed. This inspection identified higher than projected degradation from axial ODSCC of the SG tubing coincident with carbon steel TSP intersections. The results show that the End-of-Cycle 3 condition monitoring SG primary-to-secondary leak rates and conditional probabilities of tube burst for SG1, SG2 and SG4 are well within their respective allowable accident analysis limits of 3 gallons per minute (gpm)/SG and 1E-02. The SG3 condition monitoring primary-to-secondary leak rate is also within the allowable limit, however, the conditional probability of tube burst for SG3 was calculated to be 3.0050E-02 which exceeds the limit. The conditional probability of burst refers to the probability that the SG tube burst pressure associated with one or more flaw indications in the faulted SG will be less than the maximum pressure differential associated with a postulated Main Steam Line Break assumed to have occurred over the prior operating interval.

V. Assessment of Safety Consequences

An assessment of safety significance using the guidance of GL 95-05 was performed for the as found condition of the SGs. This assessment considered the probability of a MSLB, the probability of a SG TSP being displaced, and the probability of a tube rupture from the TSP displacement. This resulted in a change in the large early release frequency (LERF) of much less than 1E-7 per year, which is very small.

A. Availability of systems or components that could have performed the same function as the components and systems that failed during the event

Not applicable.

B. For events that occurred when the reactor was shut down, availability of systems or components needed to shutdown the reactor and maintain safe shutdown conditions, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident

Not applicable.

C. For failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from the discovery of the failure until the train was returned to service

Not applicable.

VI. Corrective Actions

This event was entered into the Tennessee Valley Authority's (TVA) Corrective Action Program and is being tracked under Condition Report (CR) 1651444.

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A. Immediate Corrective Actions

Upon identifying the condition, a condition report was generated and a report was made to the NRC. SG tubes were plugged, and stabilized if required, prior to returning the plant to MODE 4 in accordance with program requirements.

B. Corrective Actions to Prevent Recurrence or to reduce probability of similar events occurring in the future

As a result of the identified condition, a mid-cycle inspection of SG tubes will be performed at WBN Unit 2. Actions are in progress to move up the planned replacement of the WBN Unit 2 steam generators.

VII. Previous Similar Events at the Same Site

No previous similar events have been reported at WBN.

VIII. Additional Information

There is no additional information.

IX. Commitments

There are no new commitments.