Exelon Generation

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December 2, 2021

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

RE. Ginna Nuclear Power Plant
Renewed Facility Operating License No. DPR-18
NRC Docket No. 50-244

Subject: LER 2021-002, Valid Auxiliary Feedwater System Actuation on Lowered Steam Generator Level due to Failure to Control Main Feed Water Flow and Delay in Closing Main Steam Isolation Valves

The attached Licensee Event Report (LER) 2021-002 is submitted under the provisions of NUREG-1022, Event Reporting Guidelines. There are no new commitments contained in this submittal. This submittal is for revision 0 of the LER.

Should you have any questions regarding this submittal, please contact Chris Bradshaw at (315) 791-3246.

Sincerely,

Paul Swift

Attachment: LER 2021-002

cc: NRC Regional Administrator, Region 1
NRC Project Manager, Ginna
NRC Resident Inspector, Ginna (e-mail)

Attachment

LER 2021-002, Revision 0

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/)

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

1. Facility Name	2. Docket Number	3. Page
R.E. Ginna Nuclear Power Plant, Unit 1	05000 244	1 OF 4

4. Title
Valid Auxiliary Feedwater System Actuation on Lowered Steam Generator Level due to Failure to Control Main Feed Water Flow and Delay in Closing Main Steam Isolation Valves

5. Event Date			6. LER Number			7. Report Date			8. Other Facilities Involved	
Month	Day	Year	Year	Sequential Number	Revision No.	Month	Day	Year	Facility Name	Docket Number
10	04	2021	2021	- 002 -	00	12	02	2021		05000
									Facility Name	Docket Number
										05000

9. Operating Mode	10. Power Level
3	000

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)
Christopher Bradshaw, Regulatory Assurance Manager	3157913246

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
N/A	N/A	N/A	N/A	N/A						

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)

On 10/4/21 during a shutdown for the 2021 Refueling Outage, following the planned reactor trip, Reactor Coolant Average Temperature (Tavg) lowered due to overfeeding the 'B' Steam Generator (S/G). To maintain Tavg, feed to the S/Gs was minimized. During the cooldown, 'A' S/G lowered and a valid actuation of the Auxiliary Feedwater (AFW) System occurred. The valid actuation of the AFW System is reportable per 10CFR50.72(b)(3)(iv)(A) and 10CFR50.73.(a)(2)(iv)(A).

No equipment position changes were noted as a result of the actuation, which is the expected response for the given plant conditions. The causes of this event were Operators failed to control Main Feedwater Flow following Reactor Trip resulting in a cooldown and did not promptly close the Main Steam Isolation Valves (MSIVs).

Corrective actions include procedures revisions to include specific guidance on FRV Bypass Valve and MSIV closure. Training will be conducted to address gaps in Tavg control following Reactor Trip.

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.
R.E. Ginna Nuclear Power Plant, Unit 1	05000- 244	2021	- 002	- 00

NARRATIVE

I. PRE-EVENT PLANT CONDITIONS

At the time of the event, the plant was in MODE 3 following a planned manual reactor trip during a plant shutdown.

II. DESCRIPTION OF EVENT

A. EVENT

On 10/4/21 during a shutdown for the 2021 Refueling Outage, following the planned reactor trip, Reactor Coolant Average Temperature (Tavg) lowered due to overfeeding the 'B' Steam Generator (S/G). To maintain Tavg, feed to the S/Gs was minimized. During the cooldown, 'A' S/G lowered and resulted in an unplanned entry into LCO 3.4.5, Reactor Coolant System (RCS) Loops - MODES 1, </=8.5% rated thermal power (RTP), 2, and 3.

As S/G Level lowered, a valid Auxiliary Feedwater (AFW) actuation signal was generated. The 'A' Train AFW System was already in service and aligned when the actuation signal was generated. No equipment position changes were noted as a result of the actuation, which is the expected response for the given plant conditions. Prior to the reactor trip, the 'B' Feedwater Regulating Valve (FRV) Bypass Valve was placed in manual. Following the reactor trip, the 'B' FRV Bypass Valve did not close. When in manual, FRVs and FRV Bypass Valves will not automatically close on a reactor trip. 'B' S/G level rose to a maximum of 63% over 8 minutes prior to the FRV Bypass Valve being manually closed by Operators.

Feedwater was secured to the S/Gs; however, Tavg lowered and stabilized at approximately 535 degrees. During this time the 'B' Main Feedwater Pump was secured so AFW flow could be established and was subsequently minimized to reduce the cooldown effects on Tavg. 'A' S/G level lowered to 6% and the Main Steam Isolation Valves (MSIVs) were closed to limit secondary side steam flows and raise Tavg. After closing the MSIVs, Tavg stabilized at approximately 547 degrees and 'A' S/G level was restored to approximately 52%.

B. INOPERABLE STRUCTURES, COMPONENTS, OR SYSTEMS THAT CONTRIBUTED TO THE EVENT:

None

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NARRATIVE

C. DATES AND APPROXIMATE TIMES OF MAJOR OCCURENCES:

0010 10/04/2021 Manual Reactor Trip
0031 10/04/2021 AFW System actuation signal due to low 'A' S/G water level
0032 10/04/2021 'A' RCS Loop declared inoperable due to low 'A' S/G water level
0041 10/04/2021 Main Steam Isolation Valves Closed
0046 10/04/2021 'A' RCS Loop declared Operable

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

'A' RCS Loop declared inoperable due to low water level in the 'A' S/G in MODE 3

E. METHOD OF DISCOVERY:

Self-revealing: At 0031 on 10/04/2021 in MODE 3, S/G 'A' Narrow Range Water Level went low causing an AFW System actuation signal. AFW was in service at the time of the event providing decay heat removal.

F. SAFETY SYSTEM RESPONSES:

The 'A' Train AFW System was already in service and aligned when the actuation signal was generated. No equipment position changes were noted as a result of the actuation, which is the expected response for the given plant conditions.

III. CAUSE OF EVENT:

Operators did not close the 'B' Main Feedwater Regulating Valve Bypass Valve following Reactor Trip resulting in a cooldown and did not promptly close the Main Steam Isolation Valves

IV. ASSESSMENT OF THE SAFETY CONSEQUENCES OF THE EVENT:

In Mode 1 </= 8.5% Power and Mode 2, the Reactor Coolant Pumps (RCP) are used to provide forced circulation of the reactor coolant to ensure mixing of the coolant for proper boration and chemistry control and to remove the limited amount of reactor heat. In MODE 3, the RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The Mode 1 </= 8.5% Power, Mode 2, and Mode 3 reactor and decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be Operable to ensure redundant capability for decay heat removal.

Operator action restored the 'A' RCS Loop to Operable within 14 minutes. RCS decay heat removal was not challenged during the event. As such this event was not significant with respect to the health and safety of the public.

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NARRATIVE

V. CORRECTIVE ACTIONS

Operations guidance for closure of both the Main Feedwater Regulating Valves and Main Steam Isolation Valves will be added to the shutdown procedures. Training solutions will be used to address the gaps in temperature control following Reactor Trip.

VI. ADDITIONAL INFORMATION:

None

A. FAILED COMPONENTS:

None

B. PREVIOUS LERs ON SIMILAR EVENTS:

A LER historical search was conducted and no similar LER events were identified.

C. THE ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) COMPONENT FUNCTION IDENTIFIER AND SYSTEM NAME OF EACH COMPONENT OR SYSTEM REFERRED TO IN THIS LER:

COMPONENT - Flow Control Valve

IEEE 803 FUNCTION NUMBER - FCV

IEEE 805 SYSTEM IDENTIFICATION - SJ