April 3, 2020

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT – ISSUANCE OF AMENDMENT NO. 139 RE: ADD A ONE-TIME NOTE FOR USE OF ALTERNATIVE RESIDUAL HEAT REMOVAL METHOD (EPID L-2020-LLA-0031)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 139 to Renewed Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant (Ginna) in response to your application dated February 25, 2020, as supplemented by letters dated March 5, 2020; March 12, 2020; and March 20, 2020.

The amendment revises Technical Specifications 3.4.7, "RCS [Reactor Coolant System] Loops – MODE 5, Loops Filled"; 3.4.8, "RCS Loops – MODE 5, Loops Not Filled"; 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – Water Level ≥ 23 Ft"; and 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Water Level < 23 Ft," to allow the use of alternative means for residual heat removal. This one-time change is requested to support Ginna in the shutdown of the reactor during the upcoming refueling outage scheduled to start on April 5, 2020.

A copy of the safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

V. Sreenivas, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
1. Amendment No. 139 to Renewed DPR-18
2. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 139
Renewed License No. DPR-18

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (the licensee), dated February 25, 2020, as supplemented by letters dated March 5, 2020; March 12, 2020; and March 20, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-18 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 139, are hereby incorporated in the renewed license. Exelon Generation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License and Technical
Specifications

Date of Issuance: April 3, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 139

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Replace the following page of Renewed Facility Operating License No. DPR-18 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
3	3

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the areas of change.

Remove	Insert
3.4.7-1	3.4.7-1
3.4.8-1	3.4.8-1
3.9.4-1	3.9.4-1
3.9.5-1	3.9.5-1

(b) Exelon Generation pursuant to the Act and 10 CFR Part 70, to possess and use four (4) mixed oxide fuel assemblies in accordance with the RG&E's application dated December 14, 1979 (transmitted by letter dated December 20, 1979), as supplemented February 20, 1980, and March 5, 1980;

(3) Exelon Generation pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required.

(4) Exelon Generation pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Exelon Generation pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

(1) Maximum Power Level

Exelon Generation is authorized to operate the facility at steady-state power levels up to a maximum of 1775 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 139, are hereby incorporated in the renewed license. Exelon Generation shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

Exelon Generation shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee's amendment request dated March 28, 2013, supplemented by letters dated December 17, 2013; January 29, 2014; February 28, 2014; September 5, 2014; September 24, 2014; December 4, 2014; March 18, 2015; June 11, 2015; August 7, 2015; and as approved in the safety evaluation report dated November 23, 2015. Except where NRC approval for changes or deviations is required

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:*

a. One additional RHR loop shall be OPERABLE;* or

b. The secondary side water level of at least one steam generator (SG) shall be ≥ 16%.

- NOTE -

1. The RHR pump of the loop in operation may be de-energized for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. One required RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

3. No reactor coolant pump shall be started with one or more RCS cold leg temperatures less than or equal to the LTOP enable temperature specified in the PTLR unless:

   a. The secondary side water temperature of each SG is ≤ 50°F above each of the RCS cold leg temperatures; or

   b. The pressurizer water volume is < 324 cubic feet (38% level).

4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

*Beginning April 3, 2020, an alternative means of RHR as approved in Amendment No. 139 may be used until June 30, 2020. No increase in Mode changes will be permitted while utilizing the alternate approved means for RHR.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.*

- NOTE -

1. All RHR pumps may be de-energized for ≤ 15 minutes when switching from one loop to another provided:
   a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1; and
   b. Core outlet temperature is maintained at least 10°F below saturation temperature; and
   c. No draining operations to further reduce the RCS water volume are permitted.
2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately
B. Both RHR loops inoperable. <u>OR</u> No RHR loop in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1. <u>AND</u>	Immediately

*Beginning April 3, 2020, an alternative means of RHR as approved in Amendment No. 139 may be used until June 30, 2020. No increase in Mode changes will be permitted while utilizing the alternate approved means for RHR.

3.9 REFUELING OPERATIONS

3.9.4 Residual Heat Removal (RHR) and Coolant Circulation - Water Level ≥ 23 Ft

LCO 3.9.4 One RHR loop shall be OPERABLE and in operation.*

- NOTE -

The required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction of coolant into the Reactor Coolant System (RCS) with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	AND	

*Beginning April 3, 2020, an alternative means of RHR as approved in Amendment No. 139 may be used until June 30, 2020. No increase in Mode changes will be permitted while utilizing the alternate approved means for RHR.

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft

LCO 3.9.5 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.*

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than the required number of RHR loops OPERABLE.	A.1 Initiate action to restore RHR loop(s) to OPERABLE status.	Immediately
	<u>OR</u>	
	A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately
B. No RHR loop in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	<u>AND</u>	
	B.2 Initiate action to restore one RHR loop to operation.	Immediately
	<u>AND</u>	
	B.3 Close all containment penetrations providing direct access from containment to outside atmosphere.	4 hours

*Beginning April 3, 2020, an alternative means of RHR as approved in Amendment No. 139 may be used until June 30, 2020. No increase in Mode changes will be permitted while utilizing the alternate approved means for RHR.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

R. E. GINNA NUCLEAR POWER PLANT

LICENSE AMENDMENT REQUEST FOR A ONE-TIME USE OF

ALTERNATE RESIDUAL HEAT REMOVAL METHODS

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated February 25, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20056E958), as supplemented by letters dated March 5, 2020 (ADAMS Accession No. ML20066E040); March 12, 2020 (ADAMS Accession No. ML20073E229); and March 20, 2020 (ADAMS Accession No. ML20080N889), Exelon Generation Company, LLC (the licensee) submitted a license amendment request (LAR) to revise R. E. Ginna Nuclear Power Plant (Ginna) limiting conditions for operation (LCOs) in Technical Specifications (TSs) 3.4.7, “RCS [Reactor Coolant System] Loops – MODE 5, Loops Filled”; 3.4.8, “RCS Loops – MODE 5, Loops Not Filled”; 3.9.4, “Residual Heat Removal (RHR) and Coolant Circulation – Water Level ≥ 23 Ft”; and 3.9.5, “Residual Heat Removal (RHR) and Coolant Circulation – Water Level < 23 Ft,” to allow the use of alternative means for RHR.

The supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on March 2, 2020 (85 FR 12349).

The proposed amendment would allow the use of approved alternate means of RHR in the event that the licensee is not able to open the normal RHR suction valve Motor Operated Valve (MOV) 700, in which case neither RHR loop will be able to be made operable. This one-time change is being requested to provide an alternate form of plant cooldown, which may be needed during the shutdown of the reactor during the upcoming refueling outage scheduled for April 2020.

To support its safety evaluation, the NRC staff conducted a regulatory audit on March 17-18, 2020. A summary of the audit is available in ADAMS Accession No. ML20085L688.

2.0 REGULATORY EVALUATION

2.1 System Description

The RHR system, as described in Section 5.4.5 of the Ginna Updated Final Safety Analysis Report (UFSAR), is designed to remove residual and sensible heat from the core and reduce the temperature of the RCS during the second phase of a plant cooldown. The system consists of two heat exchangers, two pumps, piping, and associated valves and instrumentation.

As stated in the LAR, after the steam generators have been used to reduce the reactor coolant temperature to 350 degrees Fahrenheit (°F), decay heat cooling is initiated by aligning the RHR pumps to take suction from the RCS loop A hot leg through MOVs 700 and 701 and discharge through the RHR heat exchangers to the loop B cold leg through MOVs 720 and 721. With both pumps and heat exchangers in operation, RHR flow is adjusted to maintain a cooldown rate of less than 50 °F/hour (hr). If only one pump and heat exchanger are available, cooldown is accomplished at a lower rate.

The licensee stated that following the isolation of the RHR system during heatup from the RFO in 2018, MOV 700 was closed with plans to re-open the valve as part of the interlock and leakage testing. However, during the startup, the licensee indicated that the testing was delayed, and heating and distortion of the valve disc in MOV 700 caused the motor actuator to be unable to open the valve (i.e., the valve is stuck in the closed position). The licensee has evaluated the temperature conditions for MOV 701 (which is in series with MOV 700) and has determined that MOV 701 does not experience significant heating from the RCS following its closure. Therefore, MOV 701 is not susceptible to the same mechanism of uneven heating and distortion that affects MOV 700. The licensee is preparing strategies to assist in the opening of MOV 700. The licensee is also preparing a decay heat removal strategy in the event that MOV 700 only opens partially. Therefore, the licensee is requesting alternate RHR methods in the event that heating and distortion of the valve disc in MOV 700 might cause the motor actuator to be unable to open the valve (or use of other contingency methods to open the valve) to support the normal decay heat removal process. The licensee states that MOV 700 will be refurbished during the upcoming RFO in the spring of 2020.

2.2 Proposed Change

In the event that MOV 700 does not open (or only partially opens), the licensee has proposed to revise the following TSs to support use of a temporary alternative RHR method:

> LCO 3.4.7 (“RCS Loops-Mode 5, Loops Filled”) requires that one RHR loop shall be operable and in operation, and either: a) one additional RHR loop shall be operable, or b) the secondary side water level of at least one steam generator shall be greater than or equal to 16%. If MOV 700 is not able to be opened, neither RHR loop will be able to be made operable. The Required Action associated with inability to have two loops of RHR operable is to (B.2) “initiate Action to Restore one RHR loop to OPERABLE status and Operation” immediately. The ability to use an alternate RHR loop will allow this action to be met.
>
> LCO 3.4.8 (“RCS Loops – MODE 5, Loops Not Filled”) requires that two RHR loops shall be OPERABLE and one RHR loop be in operation during MODE 5 with RCS loops not filled. As discussed previously, if MOV 700 is not able to be

opened, neither RHR loop will be able to be made operable. The Required Action associated with inability to have two loops of RHR operable is to (B.2) "initiate Action to Restore one RHR loop to OPERABLE status and Operation" immediately. The ability to use an alternate RHR loop will allow this action to be met.

LCO 3.9.4 ("Residual Heat Removal (RHR) and Coolant Circulation - Water Level ≥ 23 Ft") applicable during refueling operations, requires among other things, to (A.3) initiate action to satisfy RHR loop requirements. If MOV 700 is not able to be opened, RHR loop requirements could not be met. The ability to use an alternate RHR loop will allow this action to be met.

LCO 3.9.5 ("Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft") applicable during refueling operations, requires that when there are less than the required number of RHR loops operable, (A.1) initiate action to restore RHR loop(s) to operable status, and if there is no RHR loop in operation, (B.2) initiate action to restore one loop to operation. If MOV 700 is not able to be opened, RHR loop requirements could not be met. The ability to use an alternate RHR loop will allow this action to be met.

The proposed change would modify the LCOs above by allowing the usage of alternate means for RHR to meet the LCO. To accomplish this, an asterisked note is added to each of the above-listed LCOs stating:

> * Beginning April 3, 2020, an alternate means of RHR as approved in Amendment No. 139 may be used until June 30, 2020. No increase in Mode changes will be permitted while utilizing the alternate approved means for RHR.

2.3 <u>Applicable Regulatory Requirements and Guidance</u>

Under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.90, whenever a holder of a license wishes to amend the license, including TSs in the license, an application for amendment must be filed, fully describing the changes desired. Under 10 CFR 50.92(a), determinations on whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. Both the common standards for licenses in 10 CFR 50.40(a), and those specifically for issuance of operating licenses in 10 CFR 50.57(a), provide that there must be 'reasonable assurance' that the activities at issue will not endanger the health and safety of the public, and will meet the regulations.

In accordance with 10 CFR 50.36(c)(2)(i), the TSs proposed by the applicant and issued by the NRC will include LCOs, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When LCOs are not met, the licensee must shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met. The staff reviewed the application to determine if the modified LCOs were correctly determined to be the lowest functional capability or performance level for the systems or components in question necessary for safe operation of the facility in the applicable TS modes of operation.

TSs are derived from a plant's safety analysis report, which contains the plant's design. As described in the UFSAR, Ginna's design is based on the design criteria in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 (GDC). The staff's review considered the following criteria:

- Criterion 15, "Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- GDC Criterion 34, "Residual heat removal." A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

  Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

- GDC 35, "Emergency core cooling." This criterion requires that an emergency core cooling system with the capability for accomplishing abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the core following any loss of reactor coolant such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

- GDC Criterion 37, "Testing of emergency core cooling system." The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

- GDC Criterion 54, "Piping systems penetrating containment." Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the LAR to determine if the proposed changes are consistent with the regulations, guidance, and licensing and design-basis information discussed in Section 2 of this safety evaluation. The NRC staff reviewed the proposed changes to verify that the proposed alternative RHR will be adequate to remove reactor coolant system and core decay heat. This includes the following aspects of the alternate RHR cooling method, including RHR pump suction flow and injection paths and flow requirements. The review included a regulatory audit where NRC staff examined supporting analysis, calculations, and documentation associated with use of alternate means of RHR due to the possibility of not being able to open the normal RHR suction valve MOV 700.

The license and regulations set forth requirements associated with plant modifications. Further, regulations in 10 CFR 50.59 require licensees to determine if any changes to their facilities or procedures described in the UFSAR will need prior NRC approval through a license amendment. The licensee did not identify that the changes to the facility and procedures associated with this LAR required prior NRC approval. If, during the finalization of the design and installation, the licensee subsequently determines that prior approval is required, then the licensee will need to make a separate amendment request.

3.1 Evaluation of LCOs

As stated in Section 5.4.5.3.4 of the UFSAR, the single RHR cooling suction line from the RCS renders the RHR susceptible to single failure of the in-line suction valves (MOVs 700 and 701). During the upcoming spring outage, the licensee plans to bring the plant down to Mode 4 (hot standby, 350 °F > Tavg > 200 °F) using the normal cooldown strategy that employs auxiliary feedwater and the condenser steam dump valves and atmospheric relief valves for cooldown. Once in Mode 4, the low temperature overpressure protection system is put into place, and the normal shutdown cooling RHR will be attempted to be put into place. In order to put the RHR in service, the RHR pumps are aligned to take suction from the RCS loop A hot leg through MOVs 700 and 701 and discharge through the RHR heat exchangers to the loop B cold leg through MOVs 720 and 721. If during this process MOV 700 cannot be opened or only partially opens, alternative RHR system modifications would be installed. Once installed, tested, and shown to be operating correctly, the alternative RHR cooling method will be used for entry into Mode 5 (cold shutdown, Tavg ≤ 200 °F). If for some reason the alternative RHR cooling method is not operating correctly, the licensee has other means of decay heat removal when MOV 700 is inoperable as described in Section 5.4.5.3.4 of the UFSAR.

The licensee has proposed two different alternate RHR flow paths, both consisting of temporary piping to bypass MOV 700. These flow paths are shown in Figures 2 and 3 of Attachment 1 of the LAR. If MOV 700 fails to open at all, the alternative flow path for RHR suction includes the addition of piping to connect from the loop B cold leg (between MOVs 720 and 721) to normal RHR suction inlet between MOVs 700 and 701. The discharge flow will then be routed through the RHR pumps, heat exchangers, and finally to the reactor vessel upper plenum injection nozzles via the existing low pressure ECCS flow path. Note that the suction flow path is backwards from the normal RHR flow path, which goes from hot leg A to cold leg B, while the alternative goes from cold leg B to the upper plenum. This is not a new situation as the RHR pumps inject directly to the reactor vessel in the upper plenum region during the cold leg injection phase of a loss-of-coolant accident (LOCA), as noted in Section 15.6.4.1.4.2 of the UFSAR. During the audit, the NRC staff examined calculations that show this flow path is an

effective core cooling method. Therefore, the NRC staff finds injection of the alternative RHR into the upper plenum acceptable.

If MOV 700, when demanded to open, gets stuck at less than 10 percent open, a second alternative flow path is proposed, which takes suction from cold leg B (between MOVs 720 and 721) like the alternative described above and routes flow through the containment sump B to the RHR pump suction. As with the first option, the normal RHR flow path is reversed.

As stated in the LAR, in order to implement either of the alternative flow paths, additional piping, fittings, connections, and hoses will need to be added. The licensee stated that these additional components will be designed, procured, installed, and tested to rigorous standards consistent with safety-related Class 1 or 2 components, as appropriate. In the supplement dated March 5, 2020, the licensee stated that the alternative flow paths would include the use of stainless steel hose between the tees. During the audit, the licensee stated that the design, as noted in the LAR, has changed, and is currently in the process of being finalized. The NRC staff's conclusion is based on the final design being able to meet the appropriate flow and heat transfer capabilities in order to cool down the RCS to Mode 6.

In the supplement dated March 20, 2020, the licensee stated that the minimum required flow in the alternate RHR configurations is 1,500 gallons per minute (gpm). This is based on the flow rate used in a RELAP analysis, which demonstrates that the RCS can be vented, depressurized, and drained to 84 inches (reactor vessel flange level) when at least 1,500 gpm of flow is provided through the deluge nozzles. The use of a single RHR pump at this flow rate was found to be adequate to cool down the RCS to refueling conditions. The NRC staff examined the RELAP calculation during the audit, which supported the staff's determination that 1,500 gpm is a sufficient flow rate through the RHR system to cool the RCS down to Mode 5.

The licensee stated that the updated design using multiple 3-inch diameter lines in parallel was analyzed to determine the minimum number of runs required to meet at least 1,500 gpm from one operating RHR pump at all plant conditions prior to defueling the reactor. The most limiting condition for flow was found to be when the RCS is vented and drained to reactor vessel flange level. Under these conditions, the licensee found that a single operating RHR pump can recirculate an RCS flow rate of greater than 1,500 gpm without loss of net positive suction head (NPSH) or flashing in the suction piping with six 3-inch diameter lines installed in parallel. Prior to venting the RCS, two RHR pumps can be operated together to achieve a faster cooldown rate in accordance with existing procedural guidance. The combined flow rate from two RHR pumps operating in parallel would exceed 2,800 gpm under these conditions.

As stated by the licensee in the supplement dated March 20, 2020, during the upcoming RFO, the plan is to drain the RCS to the reactor vessel flange. This level allows for head removal and entering Mode 6 (refueling) when the cavity can be flooded. While the alternate RHR system is in service, the RCS will not be drained beyond what is required to remove the reactor vessel head and will not enter into reduced inventory, which is defined as less than 64 inches, or into mid-loop conditions. If the alternate RHR system is in service, then the RHR flow will be limited to between 1,500 gpm and 1,600 gpm as the RCS is drained to 84 inches.

The NRC staff finds that it is conservative to keep the water level at the reactor vessel flange in order to support removal of the reactor head and not allow the level to drop to a reduced inventory condition (defined as 20 inches below the reactor vessel flange).

Given the licensee calculations demonstrate that a minimum flow can be obtained when using the alternative flow paths, the NRC staff finds that the alternative RHR cooling method is capable of removing reactor coolant pump heat and core decay heat and is adequate to cooldown the RCS to refueling conditions.

In the supplement dated March 20, 2020, the licensee stated that if MOV 700 is at least 10 percent open, there is adequate RHR pump flow to cool down the RCS using the normal shutdown cooling flow path. Under these conditions, two RHR pumps would be capable of operating at a combined flow of greater than 2,500 gpm, and a single RHR pump operating alone can exceed 2,000 gpm while the RCS is pressurized above 100 pounds per square inch gauge (psig). If the valve were to stick at a position less than 10 percent open, then NPSH could be lost and flashing could occur on the suction of the RHR pump. In this case, since MOV 700 could not be relied upon as an isolation boundary with the valve partially open, the alternate RHR cooling with suction from the connection between MOVs 700 and 701 could not be implemented. In the event that MOV 700 sticks partially open, but less than 10 percent open, the alternate RHR cooling flow path utilizing the containment sump would need to be implemented. Note that in either the alternative configuration where the RHR pump suction is connected between MOVs 700 and 701 or the suction is connected to the containment sump, the RHR system requirements are the same (i.e., minimum flow, heat transfer capability, and NPSH considerations.)

In the supplement dated March 20, 2020, the licensee states the temporary alternate RHR piping is being designed to meet the RHR system design pressure of 660 psig. This pressure rating exceeds the peak pressure for shutdown transients caused by either mass addition or an unintentional heat addition. The analysis for these events determined that the peak pressure in the RHR system would be limited to less than 650 psig on the discharge of the RHR pumps. The temporary piping will be located on the suction of the RHR pumps. As stated in Section 5.4.5.3.2.1 of the UFSAR, the RHR relief valve has a nominal setpoint of 600 psig and a capacity of 70,000 pounds (lb)/hr. In addition, the RHR system is provided with a 550 psig high-pressure alarm and an RCS interlock pressure alarm at 410 psig. Therefore, the NRC staff finds the additional piping acceptable if it is designed to 660 psig.

The licensee also stated that the portion of the modification that would be left permanently installed, the 10-inch tees between MOVs 700 and 701 and MOVs 720 and 721, are designed to meet the design pressure of that section of piping of 2,485 psig.

On the basis that these additional components will be designed, procured, installed, and tested to rigorous standards consistent with safety-related Class 1 or 2 components, as appropriate, the NRC staff finds this acceptable.

In the supplement dated March 5, 2020, the licensee provided details on the decay heat level versus time, as well as heat removal capability of the alternative RHR configurations. The resulting RHR heat transfer was found to correspond to a decay heat level less than 12 hours post-shutdown. Since the modification would take at least several days to install, the decay heat would be reduced to a much lower value by the time the modification is installed.

Therefore, the NRC staff finds that the alternative RHR cooling method, including revised flow paths, flow rates, and heat removal capability, is capable of cooldown to refueling conditions.

Given that the alternative RHR configurations would be installed and implemented during Mode 4 (hot standby, 350 °F > Tavg > 200 °F), the licensee evaluated the following accidents

relevant to a shutdown reactor: Mode 4 LOCA, startup of an inactive loop, seismic event, and tornado event.

The licensee states that a LOCA in Mode 4 would not be impacted by the installation and use of the alternative RHR flow paths. TS LCO 3.5.3 requires that one ECCS train be available in Mode 4. In a LOCA from Mode 4, the RHR pumps would take suction from the refueling water storage tank or from the containment sump after the refueling water storage tank has drained and would not be made inoperable from the alternative RHR flow paths. In the case of the second alternative RHR flow path, which connects to containment sump B, this would render one of the sump valves inoperable for ECCS; however, the other sump valve is still available. Therefore, the NRC staff determined that a LOCA from Mode 4 would not be impacted, as the safety injection and RHR pumps and flow paths will still be available with the alternative RHR flow path modifications in place.

As stated in Section 15.4.3.1 of the UFSAR, the re-start of the idle reactor coolant pump, without bringing the loop temperature closer to the average temperature, would result in the injection of cooler water into the core. Since the moderator temperature coefficient can be negative, the resulting feedback can cause a subsequent increase in reactor power. The licensee states that this event is bounded by the UFSAR analyzed scenario. The UFSAR scenario is analyzed at 8.5 percent of full power, since operation above this power with one inactive loop is prohibited by the plant TSs.

Given that in Mode 4 there is a smaller temperature difference between the cold leg and hot leg, the NRC staff finds that the startup of an inactive loop event would be expected to be bounded by the UFSAR analysis at 8.5 percent full power.

The licensee also discussed seismic and tornado events in the LAR. For these events, the concerns were due to the use of a water-solid steam generator cooldown. However, in the supplement dated March 12, 2020, the licensee stated that the water-solid steam generator cooldown method was no longer being pursued. The seismic and tornado analysis do not apply to the modes in which the alternate RHR systems will be operated; therefore, the NRC staff finds that the use of either alternative RHR cooldown flow path does not result in changes to the existing seismic or tornado analyses.

3.2 Evaluation of Surveillance Requirements and Action Statements

Surveillance requirements (SR) are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met. The licensee did not propose any changes to SRs. The post-installation testing performed as part of the modification will confirm functionality prior to taking advantage of the new TS note and declaring the alternate cooling loop operable. The NRC staff considered whether new SRs were needed. The staff concluded that the existing SRs still apply to the new equipment. The proposed alternate RHR methods are designed to meet or exceed the lowest functional capability or performance levels of equipment required for safe operation of the facility. All TS SRs remain applicable, and no additional SRs are needed.

When LCOs cannot be met, the licensee must shut down the reactor or take whatever actions are in the TSs. The applicant did not propose any new actions.

4.0 NRC TECHNICAL CONCLUSION

The licensee has proposed use of alternative RHR cooling flow paths in the event that RHR suction inlet valve MOV 700 fails to open. The NRC staff finds that the proposed alternative flow paths can provide for the required minimum RHR flow rate to remove both RCP and core decay heat and result in cooling the RCS from Mode 4 to Mode 6. The NRC staff finds that the alternative RHR cooling flow paths in the revised LCOs meet or exceed the lowest functional capability or performance levels of equipment required for safe operation of the facility.

5.0 NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission may issue a license amendment before the expiration of the 60-day period provided that its determination is that the amendment involves no significant hazards consideration. This amendment is being issued prior to the expiration of the 60-day period. Therefore, a finding of no significant hazards consideration follows.

The Commission has made a determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), in its letter dated February 25, 2020, the licensee provided its analysis of the issue of no significant hazards consideration, which is presented below.

Exelon has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment, as discussed below.

1. Does the proposed amendment involve a significant probability or consequences of an accident previously evaluated?

   Response: No.

   This one-time change is requested to support the station in the shutdown of the reactor during the upcoming refueling outage scheduled to start in April 2020. The proposed method of cooldown during Mode 5 is the water solid Steam Generator cooldown method. This method involves removing residual heat by filling the steam lines with water and using the Steam Generators as water-to-water heat exchangers. The proposed method to achieve Mode 5, loops not filled, utilizes portions of the normal RHR loop, additional piping, fittings, hoses, and connections meeting to safety-related Class 1 or 2 criteria, and portions of the low pressure ECCS system. These proposed alternative methods will not act as a precursor or an initiator for any transient or design basis accident; therefore, the proposed change does not significantly increase the probability of any accident previously evaluated.

   The proposed change provides an alternate means to remove decay heat and is intended to mitigate the consequences of an initiating event within the

assumed acceptance limits. This alternative method has been analyzed to ensure that it does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Implementation of this method does not affect the integrity of the fission product barriers utilized for mitigation of radiological dose consequences as a result of an accident. Plant response as modeled in the safety analyses is unaffected. Hence, the releases used as input to the dose calculations are unchanged from those previously assumed.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed alternative methods do not affect accident initiation sequences or response scenarios as modeled in the safety analyses. This method will not create a new failure scenario. In addition, no new failure modes are being created for any plant equipment. The proposed alternative methods have been designed to applicable regulatory and industry standards. Fault conditions, failure detection, reliability and equipment qualification have been considered. The new methods do not result in any new or different accident scenarios. The types of accidents defined in the UFSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   No safety analyses were changed or modified as a result of the proposed TS changes. The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. Margins associated with the current safety analyses acceptance criteria are unaffected. The current safety analyses remain bounding since their conclusions are not affected by the new method.

   Therefore, the proposed change does not result in a significant reduction in a margin of safety.

Based on its review of the licensee's no significant hazards consideration analysis quoted above, the NRC staff has determined that the proposed amendment involves no significant hazards consideration.

Accordingly, the Commission has determined that this amendment involves no significant hazards information.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment on March 20, 2020. The State official had comments/questions by e-mail dated March 25, 2020, and the NRC responded to those questions. The questions/responses are listed below:

1. Is the plant operating in an unanalyzed condition?

   No. From an event response perspective, Ginna is required to show the ability to achieve and maintain hot standby. The shutdown cooling system is not required to support that mode of operation.

2. If an accident occurred right now and Exelon needed to cool down the reactor, how would they perform residual heat removal? And did NRC evaluate that as part of the audit?

   For a LOCA type event, ECCS injection and subsequent containment recirculation are operable. For non-LOCA events, natural circulation is available, and if needed, a bleed-and-feed approach using the emergency operating procedures (EOPs) to support core cooling. The audit only evaluated the acceptability of the license amendment request associated with the alternate shutdown cooling proposal. We briefly considered the issue and determined they were not significantly impacted by the valve.

3. Why was the plant not required to address the lack of RHR capability as soon as it was discovered?

   When discovered, the RHR loop was not required to be operable. See response to question 1.

4. Will the delay in addressing this issue be addressed as an inspection finding? The entire sequence from discovery to resolution is open for inspection, and if a performance deficiency is identified, its significance will be assessed using the reactor oversight process.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on March 2, 2020 (85 FR 12349). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: April 3, 2020

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT – ISSUANCE OF AMENDMENT NO. 139 RE: ADD A ONE-TIME NOTE FOR USE OF ALTERNATIVE RESIDUAL HEAT REMOVAL METHOD (EPID L-2020-LLA-0031) DATED APRIL 3, 2020

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ADAMS Accession No.: ML20057E091 *by e-mail **by memorandum

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