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

July 28, 2021

Mr. David P. Rhoades
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. 145 RE: TECHNICAL SPECIFICATIONS CHANGE TO MAKE A ONE-TIME EXCEPTION TO THE STEAM GENERATOR TUBE INSPECTION REQUIREMENTS (EPID L-2020-LLA-0207)

Dear Mr. Rhoades:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 145 to Renewed Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant in response to your application dated September 21, 2020, as supplemented by letter dated March 19, 2021.

The amendment revises Technical Specification (TS) 5.5.8, "Steam Generator (SG) Program," to reflect a proposed change to the required SG tube inspection frequency for performing SG tube inspections and plugging. The licensee requested a one-time exception to the steam generator (SG) tube inspection requirements for the fall 2021 outage. Since the fall 2021 outage does not include a full core offload, a SG inspection would result in increased personnel dose and higher risk outage activities (e.g., a plant hold at mid-loop level for SG nozzle dam installation) that the licensee considered to not be commensurate with the derived safety benefits.

The NRC staff concludes that the licensee has demonstrated there is reasonable assurance that the structural and leakage integrity of the Ginna SG tubes will be maintained with the one-time exception to TS Section 5.5.8.

A copy of the safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's monthly *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. 145 to
   Renewed License No. 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. 145
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 September 21, 2020, as supplemented by letter dated March 19, 2021, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 145, 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

Digitally signed by James G. Danna
Date: 2021.07.28 11:18:58 -04'00'

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: July 28, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 145

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18

R. E. GINNA NUCLEAR POWER PLANT

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 page of the Appendix A, Technical Specifications, with the attached revised page. The revised page is identified by amendment numbers and marginal lines indicating the area of change.

Remove	Insert
5.5-6	5.5-6

(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. 145, 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. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial, and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected, with the exception that each SG is to be inspected during the fourth refueling outage in G1R44 following inspections completed in refueling outage G1R40.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 145 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-18

EXELON GENERATION COMPANY, LLC

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated September 21, 2020 (Reference 1), and as supplemented by letter dated March 19, 2021 (Reference 2), Exelon Generation (the licensee) requested a change to the Technical Specifications (TS) for R.E. Ginna Nuclear Power Plant (Ginna). The proposed change would make a one-time exception to the steam generator (SG) tube inspection requirements in TS Sections 5.5.8 for Ginna. The proposed change would mean the next SG tube inspection would take place in refueling outage 44, which is in the spring of 2023.

The licensee requested the change since the SG inspection scheduled during spring 2020 Ginna refueling outage 42 (G1R42)) was deferred due to COVID-19 precautions. The current Ginna TS state, in part, “No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.” However, because the licensee was inspecting more frequently and to a greater extent than the prescriptive requirements in its technical specifications, the deferral of the spring 2020 inspections did not require a license amendment. Since the fall 2021 outage (G1R43) does not include a full core offload, a SG inspection would result in increased personnel dose and higher risk outage activities (e.g., a plant hold at mid-loop level for SG nozzle dam installation) that the licensee considered to not be commensurate with the derived safety benefits.

This staff evaluation considered multiple factors including that the licensee’s historical SG inspections have been timed to avoid SG inspections during outages without full core offloads. This SG inspection timing has resulted in SG inspections being performed more frequently than required by the TSs. A key staff consideration was the increased time between inspections for pressurized water reactors (PWRs) with Alloy 690 tubing, like Ginna, that are allowed for units adopting the recently approved Technical Specification Task Force (TSTF)-577, due to favorable operating experience with Alloy 690 tubing (Reference 3). This safety evaluation (SE) considers overall SG tubing integrity and does not rely on avoiding a mid-loop evolution, since some PWRs perform SG inspections after installing nozzle dams during a mid-loop hold. Likewise, impacts to outage duration or the performance of heavy lifts that are performed routinely in every outage with SG inspections are also not considered in this SE. Therefore, the

U. S. Nuclear Regulatory Commission (NRC) staff does not consider this SE as setting a precedent for a SG inspection skip due to avoiding a mid-loop hold during an outage or for avoiding routine outage activities that are associated with SG tube inspections.

2.0 REGULATORY EVAULATION

2.1 Description of System

The tubes within a SG function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, isolate fission products in the primary coolant from the secondary coolant and the environment. For the purpose of this SE, SG tube integrity means the tubes are capable of performing this safety function in accordance with the plant design and licensing basis.

2.2 Regulatory Requirements and Guidance

Fundamental regulatory requirements with respect to the integrity of the SG tubing are established in 10 CFR Part 50. Specifically, the general design criteria (GDC) in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 provide regulatory requirements that state, in part, the RCPB shall have "…an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture" (GDC 14), "…shall be designed with sufficient margin…" (GDCs 15 and 31), shall be of "…the highest quality standards practical" (GDC 30), and shall be designed to permit "…periodic inspection and testing…to assess their structural and leaktight integrity…" (GDC 32).

For plants that were issued construction permits before the effective date of 10 CFR Part 50, Appendix A, the plant-specific principal design criteria in the plant design basis established similar fundamental regulatory requirements pertaining to the integrity of the SG tubing. Ginna received a construction permit prior to May 21, 1971, which is the date the GDC in Appendix A of 10 CFR Part 50 became effective. The Ginna Nuclear Plant Updated Final Safety Analysis Report (UFSAR) addresses the general design criteria of Appendix A to 10 CFR Part 50 in Section 3.1.2 of the UFSAR.

Section 50.55a to 10 CFR specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), except as provided in 10 CFR 50.55a(c)(2), (3), and (4). Section 50.55a further requires that throughout the service life of PWR facilities like Ginna, ASME Code Class 1 components must meet the Section XI requirements of the ASME Code to the extent practical, except for design and access provisions, and pre-service examination requirements. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. The Section XI requirements pertaining to in-service inspection of SG tubing are augmented by additional requirements in the TS.

Section 182(a) of the Atomic Energy Act requires nuclear power plant operating licenses to include TSs as part of any license. In 10 CFR 50.36, "Technical specifications," the NRC regulatory requirements related to the content of the TS are established. The TSs for all current PWR licenses require that an SG Program be established and implemented to ensure that SG tube integrity is maintained. Programs established by the licensee, including the SG Program, are listed in the administrative controls section of the TSs to operate the facility in a safe manner.

2.3 SG Tube Integrity Requirements in the Ginna Technical Specifications

For Ginna, the requirements for performing SG tube inspections and plugging are in TS Section 5.5.8, while the requirements for reporting the SG tube inspections and plugging are in TS Section 5.6.7.

SG tube integrity is maintained by meeting the performance criteria specified in Ginna TS Section 5.5.8.b for structural and leakage integrity, consistent with the plant design and licensing basis. TS Section 5.5.8.a requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. TS Section 5.5.8.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that may be present along the length of a tube and that may satisfy the applicable tube plugging criteria. The applicable tube plugging criteria, specified in TS Section 5.5.8.c, are that tubes found during in-service inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged.

Ginna TS Section 3.4.13 includes a limit on operational primary-to-secondary leakage, beyond which the plant must be promptly shutdown. Should a flaw exceeding the tube plugging limit not be detected during the periodic tube surveillance required by the plant TSs, the operational leakage limit provides added assurance of timely plant shutdown before tube structural and leakage integrity are impaired, consistent with the design and licensing bases.

As part of the plant's licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents, such as a SG tube rupture and a steam line break. These analyses consider primary-to-secondary leakage that may occur during these events and must show that the radiological consequences do not exceed the applicable limits of 10 CFR 50.67 or 10 CFR 100.11 for offsite doses; GDC 19 of 10 CFR Part 50, Appendix A for control room operator doses (or some fraction thereof as appropriate to the accident); or the NRC-approved licensing basis (e.g., a small fraction of these limits). No accident analyses for Ginna are being changed because of the proposed amendment and, thus, no radiological consequences of any accident analysis are being changed. The proposed change maintains the accident analyses and consequences that the NRC has reviewed and approved for the postulated design-basis accidents for SG tubes.

3.0 TECHNICAL EVALUATION

3.1 Background

3.1.1 Steam Generator Design

Ginna has two recirculating Babcock and Wilcox Canada (BWI) design replacement SGs that were installed in May 1996 (G1R26). Each BWI SG contains 4,765 thermally treated Alloy 690 tubes, which have a nominal outside diameter of 0.75 inches and a nominal wall thickness of 0.043 inches. The tubes have a full depth hydraulic expansion into the tubesheet, and the straight portion of the tubes are supported by eight Type 410 stainless steel, lattice grid supports comprised of high, medium, and low bars. The U-bend portions of the tubes are supported by ten Type 410 stainless steel fan bar/collector bar assemblies and the number of supports vary by row. Tube Rows 1 through 18 received an additional thermal stress relief following the tube bending process. The Row 1 and Row 2 U-bends use a crossover design to maximize the tube bend radius and minimize tube ovality.

3.1.2 Operating Experience

The two most recent tube inspections at Ginna were performed in 2014 and 2017 (References 4 and 5). The only degradation mechanism detected in the Ginna replacement SGs is mechanical tube wear from interaction with lattice grid support structures and from foreign objects. An operating experience review of tube degradation noted during inspections is provided after Table 1.

Tube Plugging

Table 1 contains the plugging summary for the Ginna SGs. Two tubes that were plugged before service during manufacturing are shown in the April 1996 row. Since the replacement SGs were placed into service in 1996, only four tubes have been removed from service by plugging (in 2005), due to foreign object wear.

Table 1: Ginna Plugging History by Degradation Mechanism

Date	Outage	Fan Bar Wear	Lattice Grid Wear	Foreign Object Wear	Other	Total
April 1996	G1R26	0	0	0	2*	2
March 1999	G1R28	0	0	0	0	0
March 2002	G1R30	0	0	0	0	0
March 2005	G1R32	0	0	4	0	4
April 2008	G1R34	0	0	0	0	0
April 2011	G1R36	0	0	0	0	0
April 2014	G1R38	0	0	0	0	0
April 2017	G1R40	0	0	0	0	0
April 2020	G1R42	Inspection Skip				
	Total	0	0	4	2	6

* Two tubes were plugged before service due to a manufacturing defect

Tube Wear

Table 1 provides historical tube plugging numbers while this section discusses more recent tube wear experience. The only tube wear mechanisms that have been detected through the G1R40 inspection are lattice grid wear and foreign object wear. Only four indications of lattice grid wear were detected during the G1R40 inspection and all four had been reported during the previous G1R38 inspection. Based on +Point™ probe inspection results, three of the four indications showed no change in depth and one indication grew by 1 percent through-wall (TW), from 6 percent TW to 7 percent TW. The maximum depth of any lattice grid wear indication measured during the G1R40 inspection was 10 percent TW.

The Ginna SGs have three tubes with foreign object wear indications. All three indications are located at lattice grid support intersections and were identified prior to the most recent (G1R40) inspection. No foreign objects remain at these locations and therefore no growth was observed in these indications between the G1R38 and the G1R40 inspections. Primary side inspections during G1R40 included array probe examination of $\geq$4 tubes deep from the periphery and no tube lane from the tube end to the first support on the hot and cold legs.

Corrosion Degradation

Ginna has not reported any indications of corrosion degradation, such as stress corrosion cracking (SSC), and to date, the NRC staff is unaware of any corrosion degradation in any operating SGs with Alloy 690TT tubing. In addition to the bobbin inspections during 1R40, Ginna performed +Point™ inspection of all dents and over-expansions at the top-of-the-tubesheet and array probe inspections of at least four tubes in from the periphery and no tube lane.

Secondary Side Inspections

Degradation of SG secondary side components that could affect the structural or leakage integrity of the SG tubes must be considered. The secondary side inspection activities during G1R40 included the top of tubesheet region and secondary moisture separator inspections. Sludge lancing and foreign object search and retrieval (FOSAR) performed during G1R40 removed 18 pieces of Flexitallic gasket, three pieces of weld slag, and one small machine curl. To date, no foreign object wear has been detected at the top-of-the tubesheet in the Ginna SGs.

Previous secondary side inspections have identified flow accelerated corrosion (FAC) of the secondary moisture separator baseplates. During G1R40, the baseplates of all 85 secondary separators in both steam generators were inspected both visually and with laser profilometry. Table 4-3 in Attachment 1 to Reference 1 provides the five most degraded secondary separator baseplate FAC TW thickness depths, that ranged from 32 percent to 51 percent TW.

3.2. Proposed TS Change

3.2.1 Current TS Requirements

The "Steam Generator (SG) Program," Section 5.5.8 of the Ginna TS, provides the SG tube inspection requirements for Ginna. Ginna TS Section 5.5.8.d.2 requires, in part, that, "No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected."

The Ginna TS Section 5.5.8.d.2 also states, in part, "Inspect 100% of the tubes at sequential periods of 144, 108, 72, and thereafter, 60 effective full power months." Ginna is currently in the 108 EFPM sequential period and last performed a 100 percent tube inspection during G1R38.

3.2.2 Description of Proposed TS Change

The license amendment request proposes to revise Ginna TS Section 5.5.8.d.2, "Steam Generator (SG) Program, to add an exception to the following requirement "No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected." The proposed revised TS Section 5.5.8.d.2, with the exception shown in bold, states:

"No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected, **with the exception that each SG is to be**

inspected during the fourth refueling outage, in G1R44, following inspections completed in refueling outage G1R40."

In essence, the amendment requests that the fall 2021 (G1R43) inspection be performed in the spring of 2023 (G1R44). There are no TS Bases changes associated with this amendment.

3.3 Staff Evaluation of Proposed TS Change

The NRC staff evaluation of the proposed TS change focused on the potential for the change to affect SG tube integrity, since maintaining SG tube integrity is a current TS requirement that plays a key role in protecting the public's health and safety. In particular, the evaluation assessed whether the technical justification in Reference 1 demonstrated that the structural integrity performance criterion (SIPC) and accident-induced leakage performance criterion (AILPC) will continue to be met with the revised inspection intervals proposed in Reference 1. These tube integrity criteria are defined in TS Section 5.5.8.b.

As noted previously in Section 3.1.2, the only degradation mechanisms detected in the Ginna SGs are tube wear from interaction at lattice grid supports, and from foreign objects in the SGs. Since being placed in service in 1996, only four tubes have been plugged (in 2005), due to foreign object wear. This operating experience shows minimal tube degradation but is consistent with industry experience at similar SGs that have Alloy 690TT tubing, where wear from tubing support structures is the main tubing degradation mechanism (Reference 6).

In the operational assessment (OA) that was performed to support one additional cycle for the Ginna amendment request, tube wear at lattice grid supports and from foreign objects were the existing tube degradation mechanisms considered. Tube wear from fan bar support and from tube-to-tube interaction were included as potential mechanisms. Since the G1R40 bobbin probe inspection included 75% of the tubes, the remaining 25% of the tubes were most recently examined in G1R38. Consequently, the OA considered degradation that was detected during the G1R40 inspections and returned to service, along with degradation that was considered to be present but undetected during G1R38. It is more conservative (i.e., limiting) to consider undetected degradation during G1R38 since that results in a longer time for indication growth relative to degradation that was undetected during G1R40.

3.3.1 Evaluation of Existing Tube Degradation Mechanisms

Wear at Lattice Grid Supports

No tubes have been plugged in the Ginna SGs due to wear at lattice grid supports. Only four indications of lattice grid support wear were detected during G1R40 in April 2017 and all four were reported during the previous G1R38 inspection. The deepest wear indication returned to service after G1R40 measured 10 percent TW. The greatest growth measured between the two most recent tube inspections occurred in a tube in SG B that grew from 6 percent TW to 7 percent TW.

The licensee evaluation for existing indications used the maximum measured depth, adjusted it conservatively to account for nondestructive examination sizing uncertainty to obtain an upper bound depth at the beginning of Cycle 40. This adjustment was performed by applying the eddy current sizing uncertainty parameters for the appropriate technique specification to generate the upper 95th percentile probability/50th percentile confidence depth estimate. A bounding growth rate of 2 percent per Effective Full Power Year (EFPY) was assumed since there is insufficient wear to establish a statistically valid upper 95/50 growth rate. Adjusting the beginning of Cycle 40 upper bound depth for growth until the next inspection in G1R44 results in a bounding wear depth of 36 percent TW. This depth is significantly less than the structural limit depth of 51 percent TW, assuming a bounding length, uniformly deep wear scar.

To account for undetected wear, the licensee assumed the depth of a missed wear indication during G1R38 was equal to the depth corresponding to a probability of detection of 0.95 for the applicable bobbin probe technique. Using the model assisted probability of detection (POD) technique and Ginna specific tubing eddy current noise, the licensee determined the 0.95 POD to be 7 percent TW. This value was conservatively adjusted by assuming a 20 percent TW wear depth was undetected in G1R38. The same 2 percent per EFPY bounding wear growth rate was applied to the 20 percent beginning wear depth. The resulting bounding wear depth at the next inspection in G1R44 was calculated to be 37 percent TW, which is also significantly below the structural limit, assuming a bounding length uniform depth wear scar.

Evaluation Summary for Wear at Lattice Grid Supports

The NRC staff finds the licensee's evaluation of tube wear at lattice grid supports to be acceptable. Wear at these locations in the SGs has been minimal and effectively managed for many cycles without challenging tube integrity. Only four tubes have wear indications at the lattice grid supports and no tubes have been plugged due to lattice grid support wear. Wear at support structures is readily detected with standard eddy current examination techniques and wear sizing errors are considered in the projection of existing flaws until G1R44. Wear projections were made based on the existing indications with a conservatively assumed wear depth that was missed during G1R38. The limiting wear analysis showed that the projections of G1R44 TW depths meet the SIPC with significant margin. The staff found the determination of beginning flaw depth and growth rates acceptable because they are based on industry guidelines (Reference 7) and conservative assumptions. For flaws of this type, satisfying the SIPC demonstrates that the AILPC will also be satisfied since the limiting accident induced pressure differentials are much less than three times the normal operating pressure differential (3xNOPD). Therefore, both SIPC and AILPC are satisfied.

Foreign Object Wear

Ginna has three tubes with wear from foreign objects, all located at lattice grid support intersections. These indications were identified during or before the G1R38 inspection, have no foreign object remaining, and showed no growth between the G1R38 and G1R40 inspections. These indications have no mechanism to cause future growth since the objects that caused the wear in the tubes are no longer present. The licensee's evaluation for the structural limit for volumetric wear and a bounding assumed length of 0.5 inches shows significant margin between the actual wear depths and the structural limit.

The G1R40 inspection scope for foreign objects included both eddy current and visual examinations. Eddy current examinations included array probe inspection of all periphery tubes between the tubesheet and first support, at least four tubes deep including the no tube lane locations. Visual examinations performed following sludge lancing viewed all rows and columns into the tube bundle at all peripheral and no tube lane locations. The licensee also removed some foreign objects that were detected during the G1R40 FOSAR. To date, no foreign object wear has ever been detected at the top-of-the-tubesheet.

The NRC staff finds the licensee's analysis of foreign object wear acceptable. The only three tubes with foreign object wear in the Ginna SGs no longer have the object present so tube degradation will not continue in these tube locations. The results of FOSAR, array probe examinations, bobbin coil probe examinations, and the Ginna history with foreign objects provide reasonable assurance that any new foreign object generated prior to G1R44 will not violate the structural integrity performance criterion. The staff also acknowledges that predicting future loose part generation is not possible since past fleet-wide operating experience has shown that new loose part generation, transport to the SG tube bundle, and interactions with the tubes cannot be reliably predicted. However, plants can reduce the probability of loose parts by maintaining robust foreign material exclusion programs and applying lessons learned from previous industry operating experience with loose parts. Plants in general, including Ginna, have demonstrated the ability to conservatively manage loose parts once they are detected by eddy current examinations or by secondary side FOSAR inspections. If unanticipated aggressive tube wear from new loose parts should occur in a Ginna SG, operating experience has shown that a primary-to-secondary leak will probably occur, rather than a loss of tube integrity. As an added compensatory measure, Ginna will administratively reduce the allowable SG leakage until the next tube inspection from the current 100 gallons per day (gpd) per SG to 30 gpd per SG. This will ensure Ginna will quickly respond to any increase in primary-to-secondary leakage if it were to occur. In the event of a primary-to-secondary leak, the NRC staff will interact with the licensee in accordance with established procedures in Inspection Manual Chapter (IMC) 0327, "Steam Generator Tube Primary-to-Secondary Leakage," dated January 1, 2019 (Reference 8), to confirm the licensee's conservative decision making.

3.3.2 Evaluation of Potential Tube Degradation Mechanisms

Since the baseplates of the Ginna secondary moisture separators are experiencing FAC, the staff questioned (Reference 9) if the FAC could progress during operation until G1R44 to the extent that a loose part could be generated that could impact tube integrity. The licensee responded (Reference 2) with a discussion of conservative assumptions that are contained in the baseplate FAC analysis and with operating experience from other units that have experienced greater amounts of secondary moisture separator baseplate FAC without any challenges to tube integrity.

Along with the active tube degradation mechanisms, Ginna evaluated two potential degradation mechanisms that have not occurred in the SGs: (1) fan bar support wear, and (2) tube-to-tube wear. The licensee's OA fan bar support wear analysis assumed that wear was occurring but was not detected during the G1R38 inspection. The licensee determined that the 0.95 POD for the appropriate inspection technique was 15 percent TW and then assumed that an undetected fan bar support wear indication was 20% TW for additional margin. The licensee assumed a 2 percent TW/EFPY growth rate, which resulted in a flaw depth of 37 percent TW at G1R44. This is significantly below the structural limit for a bounding flaw length.

Tube-to-tube wear was also postulated to be occurring even though there is no history of tube-to-tube wear at Ginna or in any other recirculating SGs manufactured by BWI. Even though all proximity indications were inspected with a +Point™ Probe during G1R38, the licensee assumed a 20 percent TW tube-to-tube wear indication was missed. The licensee applied a 2 percent TW/EFPY growth rate and assumed a 30 inch wear length since tube-to-tube wear can occur over greater lengths than tube-to-tube support wear. The resulting wear indication was calculated to be 37 percent TW, which is significantly less than the structural integrity depth limit.

Evaluation Summary for Potential Mechanisms

The staff finds the licensee's secondary moisture separator baseplate FAC analysis acceptable for providing reasonable assurance that secondary moisture baseplate FAC will not affect tube integrity. The licensee's analysis included conservative assumptions. For example, the licensee evaluation assumed an FAC growth rate double the upper 95th percentile growth rate of an international plant that experienced baseplate perforations. In response to a staff question, the licensee stated that a domestic plant that experienced baseplate degradation, to a depth greater than the worst projected Ginna baseplate degradation at the G1R44 outage, ran for an additional two operating cycles without experiencing loose parts that affected the tube bundle. In addition, plants that have experienced complete baseplate perforation have not generated loose parts that caused tube wear. Therefore, the staff concludes that there is reasonable assurance that secondary moisture separator baseplate FAC will not affect tube integrity.

The NRC staff reviewed the licensee's evaluation of potential mechanisms and compared them to the SG tube integrity criteria contained in the Ginna TS. The staff considers these evaluations to be acceptable for Ginna since they assumed that wear was present but undetected during G1R38, assumed initial bounding lengths and depths, and applied conservative growth rates. The calculated end of cycle wear sizes for each evaluated potential mechanism satisfied the structural integrity requirements until the next inspection with significant margin. The accident induced leakage requirement is also satisfied until G1R44 with zero projected leakage. Therefore, the staff concludes there is reasonable assurance that both the tube structural integrity and leakage integrity performance criteria will be met for all tubes until G1R44.

3.4 Technical Evaluation Conclusion

Based on the information submitted, the NRC staff finds that the licensee has demonstrated that there is reasonable assurance that the structural and leakage integrity of the Ginna SGs will be maintained, in accordance with the TS requirements in Section 5.5.8.b., until the next scheduled SG tube inspections in G1R44, as proposed in Reference 1. The staff also finds that the proposed change to the Ginna SG requirements is acceptable because continues to meet the requirements of 10 CFR 50.36 by providing administrative controls necessary to assure operation of the facility in a safe manner. Therefore, the NRC staff concludes that the licensee's proposed change to the Ginna TS Section 5.5.8 is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment on June 23, 2021. The State official had no comments.

5.0 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 December 1, 2020 (85 FR 77275). 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.

6.0 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.

7.0 REFERENCES

1. Exelon Generation letter to U.S. Nuclear Regulatory Commission (NRC), "Application to Revise R.E. Ginna Technical Specifications for Steam Generator Tube Inspection Frequency," dated September 21, 2020 (ADAMS Accession No. ML20265A198).

2. Exelon Generation letter to U. S. Nuclear Regulatory Commission (NRC), "Response to Request for Additional Information by the Office of Nuclear Reactor Regulation to Support Review of R.E. Ginna Nuclear Power Plant License Amendment Request to Modify the Steam Generator Tube Inspection Frequency," dated March 19, 2021 (ADAMS Accession No. ML21078A002).

3. U.S. Nuclear Regulatory Commission (NRC) Final Safety Evaluation by the Office of Nuclear Reactor Regulation Technical Specifications Task For Traveler, TSTF-577, Revision 1 "Revised Frequencies for Steam Generator Tube Inspections," (ADAMS Accession No. ML21098A188).

4. Exelon Generation letter to U. S. Nuclear Regulatory Commission (NRC) – R. E. Ginna Nuclear Power Plant, "2014 Steam Generator Tube Inspection Report," dated November 13, 2014 (ADAMS Accession No. ML14322A149).

5. Exelon Generation letter to U. S. Nuclear Regulatory Commission (NRC) – Re. E. Ginna Nuclear Power Plant, "2017 Steam Generator Tube Inspection Report," dated September 26, 2017 (ADAMS Accession No. ML17276A348).

6. NUREG-1841, "U.S. Operating Experience with Thermally Treated Alloy 690 Steam Generator Tubes," dated August 2007 (ADAMS Accession No. ML072330588).

7. Electric Power Research Institute (EPRI) letter to U. S. Nuclear Regulatory Commission (NRC), Request for Withholding of the following Proprietary Documents: " Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines, Revision 4, dated June 18, 2016 (ADAMS Accession No. ML16208A272).

8. U. S. Nuclear Regulatory Commission (NRC) Inspection Manual Chapter 0327 - Steam Generator Tube Primary-to-Secondary Leakage, dated January 1, 2019 (ADAMS Accession No. ML18093B067).

9. U. S. Nuclear Regulatory Commission (NRC), V. Sreenivas email to J. Hodge, Exelon Generation, "Ginna: RAI-Steam Generator LAR to revise Technical Specifications for Steam Generator Tube Inspection Frequency – EPID L-2020-LLA-0207," dated March 17, 2021, (ADAMS Accession No. ML21076A515).

Principal Contributors: P. Klein

Date: July 28, 2021

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT – ISSUANCE OF AMENDMENT NO. 145 RE: TECHNICAL SPECIFICATIONS CHANGE TO MAKE A ONE-TIME EXCEPTION TO THE STEAM GENERATOR TUBE INSPECTION REQUIREMENTS (EPID L 2020 LLA 0207) DATED JULY 28, 2021

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ADAMS Accession No. ML21175A001 ***by memorandum**

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