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10 CFR 50.46

April 8, 2022

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

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

Subject: 2022 10 CFR 50.46 Annual Report

Reference: 1. Letter from D. Gudger (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "2021 10 CFR 50.46 Annual Report," dated April 8, 2021

The purpose of this letter is to submit the 10 CFR 50.46 annual reporting information for R.E. Ginna Nuclear Power Plant. The referenced letter is the most recent annual 10 CFR 50.46 Report submitted to the U.S. Nuclear Regulatory Commission.

Two attachments are included with this letter that provide the current Ginna 10 CFR 50.46 status. Attachment 1 provides the Peak Cladding Temperature (PCT) "rack-up" sheets. Attachment 2, "Assessment Notes," contains a detailed description of each change/error reported.

There are no commitments contained in this letter. If you have any questions, please contact Ron Reynolds at 610-765-5247.

Respectfully,

David T. Gudger

David T. Gudger
Senior Manager - Licensing
Exelon Generation Company, LLC

Attachments: 1) Peak Cladding Temperature Rack-Up Sheets for R.E. Ginna Nuclear Power Plant
2) Assessment Notes

cc: U.S. NRC Administrator, Region I
U.S. NRC Project Manager, Ginna
U.S. NRC Senior Resident Inspector, Ginna

ATTACHMENT 1

10 CFR 50.46

"Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors"

Annual Report of the Emergency Core Cooling System Evaluation Model Changes and Errors

Assessments as of April 8, 2022

Peak Cladding Temperature Rack-Up Sheets for

R.E. Ginna Nuclear Power Plant

PLANT NAME: Ginna

ECCS EVALUATION MODEL: Small Break Loss of Coolant Accident (SBLOCA)

REPORT REVISION DATE: 4/8/2022

CURRENT OPERATING CYCLE: 43

ANALYSIS OF RECORD

Evaluation Model: NOTRUMP

Calculation: Westinghouse CN-LIS-04-206, April 2005

Fuel: 422 Vantage+

Limiting Fuel Type: 422 Vantage+

Limiting Single Failure: Diesel Generator Failure to Start

Limiting Break Size and Location: 2-inch Equivalent High T_{avg} Cold Leg Break

Reference Peak Cladding Temperature (PCT) PCT = 1167.0°F

MARGIN ALLOCATION

A. PRIOR LOSS OF COOLANT ACCIDENT (LOCA) MODEL ASSESSMENTS

10 CFR 50.46 report dated April 30, 2007 (Note 1)	ΔPCT = 0°F
10 CFR 50.46 report dated February 10, 2009 (Note 3)	ΔPCT = 0°F
10 CFR 50.46 report dated March 4, 2011 (Note 5)	ΔPCT = 0°F
10 CFR 50.46 report dated March 27, 2012 (Note 6)	ΔPCT = 0°F
10 CFR 50.46 report dated April 1, 2013 (Note 8)	ΔPCT = 0°F
10 CFR 50.46 report dated April 9, 2014 (Note 9)	ΔPCT = 0°F
10 CFR 50.46 report dated April 9, 2015 (Note 10)	ΔPCT = 0°F
10 CFR 50.46 report dated April 7, 2016 (Note 11)	ΔPCT = 0°F
10 CFR 50.46 report dated April 6, 2018 (Note 13)	ΔPCT = 0°F
10 CFR 50.46 report dated April 8, 2019 (Note 14)	ΔPCT = 0 °F
10 CFR 50.46 report dated April 8, 2020 (Note 15)	ΔPCT = 0 °F
10 CFR 50.46 report dated April 8, 2021 (Note 16)	ΔPCT = 0 °F
NET PCT	**PCT =1167.0°F**

B. CURRENT LOCA MODEL ASSESSMENTS

General Code Maintenance (Note 17)	ΔPCT = 0°F
Reduction in Flow Area to the Bottom of the Barrel/Baffle Region (Note 17)	ΔPCT = 0 °F
Reconstituted Fuel Rod Evaluation (Note 17)	ΔPCT = +3°F
Total PCT change from current assessments	$\sum \Delta$PCT = +3°F
Cumulative PCT change from current assessments	$\sum \lvert \Delta$PCT $\rvert$ = 3°F
NET PCT	**PCT =1170.0°F**

PLANT NAME: Ginna

ECCS EVALUATION MODEL: Large Break Loss of Coolant Accident (LBLOCA)

REPORT REVISION DATE: 4/8/2022

CURRENT OPERATING CYCLE: 43

ANALYSIS OF RECORD

Evaluation Model: ASTRUM (2004)

Calculation: Westinghouse CN-LIS-05-11, April 2005

Fuel: 422 Vantage+

Limiting Fuel Type: 422 Vantage+

Limiting Single Failure: Loss of one train of ECCS flow

Limiting Break Size and Location: Cold Leg Split Break

Reference PCT PCT = 1870.0°F

MARGIN ALLOCATION

A. PRIOR LOSS OF COOLANT ACCIDENT (LOCA) MODEL ASSESSMENTS

10 CFR 50.46 report dated April 30, 2007 (Note 1)	ΔPCT = 0°F
10 CFR 50.46 report dated May 4, 2008 (Note 2)	ΔPCT =+37°F
10 CFR 50.46 report dated February 10, 2009 (Note 3)	ΔPCT = 0°F
10 CFR 50.46 report dated March 26, 2010 (Note 4)	ΔPCT = 0°F
10 CFR 50.46 report dated March 4, 2011 (Note 5)	ΔPCT = 0°F
10 CFR 50.46 report dated March 27, 2012 (Note 6)	ΔPCT = 0°F
30-Day 10 CFR 50.46 report dated August 16, 2012 (Note 7)	ΔPCT = +134°F
10 CFR 50.46 report dated April 1, 2013 (Note 8)	ΔPCT = +75°F
10 CFR 50.46 report dated April 9, 2014 (Note 9)	ΔPCT = +2°F
10 CFR 50.46 report dated April 9, 2015 (Note 10)	ΔPCT = 0°F
10 CFR 50.46 report dated April 7, 2016 (Note 11)	ΔPCT = +1°F
10 CFR 50.46 report dated April 7, 2017 (Note 12)	ΔPCT = 0°F
10 CFR 50.46 report dated April 6, 2018 (Note 13)	ΔPCT = 0°F
10 CFR 50.46 report dated April 8, 2019 (Note 14)	ΔPCT = 0 °F
10 CFR 50.46 report dated April 8, 2020 (Note 15)	ΔPCT = 0 °F
10 CFR 50.46 report dated April 8, 2021 (Note 16)	ΔPCT = 0 °F
NET PCT	**PCT =2119.0°F**

B. CURRENT LOCA MODEL ASSESSMENTS

General Code Maintenance (Note 17)	ΔPCT = 0 °F
Reconstituted Fuel Rod Evaluation (Note 17)	ΔPCT = +6 °F
Reconstituted Fuel Rod Evaluation Superseded (Note 17)	ΔPCT = -1 °F
Total PCT change from current assessments	$\sum \Delta$PCT = +5 °F
Cumulative PCT change from current assessments	$\sum \lvert \Delta$PCT $\rvert$ = 5 °F
NET PCT	**PCT =2124.0 °F**

ATTACHMENT 2

10 CFR 50.46

"Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors"

Annual Report of the Emergency Core Cooling System Evaluation Model Changes and Errors

Assessments as of April 8, 2022

Assessment Notes

R.E. Ginna Nuclear Power Plant

1. Prior LOCA Model Assessment

   The 10 CFR 50.46 report dated April 30, 2007, reported new licensing basis peak cladding temperature (PCT) for small break loss of coolant accident (SBLOCA) and large break loss of coolant accident (LBLOCA) analyses to support fuel assembly transition from OFA to 422 Vantage+ and extended power uprate. The new licensing basis PCT reported for SBLOCA and LBLOCA are 1167°F and 1870°F, respectively.

2. Prior LOCA Model Assessment

   The 10 CFR 50.46 report dated May 4, 2008, reported an evaluation for LBLOCA related to HOTSPOT fuel relocation error which resulted in a 37°F PCT assessment.

3. Prior LOCA Model Assessment

   The 10 CFR 50.46 report dated February 10, 2009, reported evaluations for SBLOCA and LBLOCA model changes which resulted in 0°F PCT change.

4. Prior LOCA Model Assessment

   The 10 CFR 50.46 report dated March 26, 2010, reported evaluations for LBLOCA model changes which resulted in 0°F PCT change.

5. Prior LOCA Model Assessment

   The 10 CFR 50.46 report dated March 4, 2011, reported evaluations for SBLOCA and LBLOCA model changes which resulted in 0°F PCT change.

6. Prior LOCA Model Assessment

   The 10 CFR 50.46 report dated March 27, 2012, reported evaluations for SBLOCA and LBLOCA model changes which resulted in 0°F PCT change.

7. Prior LOCA Model Assessment

   The 30-day 10 CFR 50.46 report dated August 16, 2012, reported evaluations for fuel pellet thermal conductivity degradation (TCD) and peaking factor burndown, and design input change assessments which resulted in a 134°F PCT impact for LBLOCA.

8. Prior LOCA Model Assessment

   The 10 CFR 50.46 report dated April 1, 2013, reported evaluations for SBLOCA model changes which resulted in 0°F PCT impact. A LBLOCA assessment for the evaluation of an elevated initial containment and accumulator temperature was submitted in a License Amendment Request for NRC review and approval. The assessment resulted in a 75°F PCT impact. This increase in temperature was approved in an NRC Safety Evaluation Report (SER) (ML14232A331) dated August 21, 2014. The SER (ML14232A331) evaluated the 10 CFR 50.46 reporting criteria explicitly.

9. Prior LOCA Model Assessment

   The 10 CFR 50.46 report dated April 9, 2014, reported evaluations for SBLOCA model changes which resulted in 0°F PCT impact. A LBLOCA assessment was reported related to revised heat transfer multiplier distribution which resulted in a 2°F PCT assessment.

10. Prior LOCA Model Assessment

    The 10 CFR 50.46 report dated April 9, 2015, reported general code maintenance for both LBLOCA and SBLOCA. An error in Decay Group Uncertainty Factors against the LBLOCA model was reported. Additionally, it reported errors in Fuel Rod Gap Conductance, Radiation Heat Transfer Model, and SBLOCTA Pre-DNB Cladding Surface Heat Transfer Coefficient Calculation for the SBLOCA model. All changes resulted in 0°F PCT impact.

11. Prior LOCA Model Assessment

    The 10 CFR 50.46 report dated April 7, 2016, reported General Code Maintenance for the SBLOCA which led to a PCT impact of 0°F. Additionally, Ginna began inserting reconstituted fuel with 5 stainless steel filler rods starting in Cycle 39. The effects to SBLOCA are 0°F and the effects to LBLOCA are 1°F for as long as reconstituted fuel with 5 stainless steel filler rods remain in the core.

12. Prior LOCA Model Assessment

    The 10 CFR 50.46 report dated April 7, 2017, reported General Code Maintenance for the LBLOCA to enhance the usability of codes and to streamline future analyses which led to a PCT impact of 0°F. There were two errors assessed to the LBLOCA analysis related to the calculation of high temperature oxidation within a realistic LBLOCA calculation and to the use of the American Society of Mechanical Engineers (ASME) steam tables to calculate the steady-state upper head liquid temperature as a function of the pressure and specific enthalpy in the ASTRUM software program. Both errors each resulted in an estimated PCT impact of 0°F for 10 CFR 50.46. There were no impacts or assessments to SBLOCA.

13. Prior LOCA Model Assessment

    The 10 CFR 50.46 report dated April 6, 2018, reported General Code Maintenance for the SBLOCA and LBLOCA which each led to a PCT impact of 0°F. The SBLOCA also reported one error pertained to the upper plenum fluid volume calculation with an estimated PCT impact of 0°F. The LBLOCA reported three assessments with the first involving an evaluation of inconsistent application of numerical ramp applied to the entrained liquid / vapor interfacial drag coefficient, the second involving an evaluation of inappropriate resetting of transverse liquid mass flow, and the third involving an evaluation of steady-state fuel temperature calibration method. All three errors each resulted in an estimated PCT impact of 0°F for 10 CFR 50.46.

14. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated April 8, 2019, reported one error pertaining to fuel rod heat-up calculations for the SBLOCA which led to a PCT impact of 0°F. The LBLOCA also reported two errors, one pertaining to the CCTF model used in the WCOBRA/TRAC calculation and the second involved the modeling of vapor temperature in the WCOBRA/TRAC codes. upper plenum fluid volume calculation with an estimated PCT impact of 0°F. Both errors each resulted in an estimated PCT impact of 0°F for 10 CFR 50.46.

15. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated April 8, 2020, reported one change pertaining to main steam safety valve (MSSV) setpoint pressure tolerance for the SBLOCA which led to a PCT impact of 0°F. The LBLOCA also reported three errors or changes, one pertaining to the general code maintenance, one concerning the vessel interfacial heat transfer implementation and validation basis for modeling upper plenum injection (UPI) phenomena. Each error or change resulted in an estimated PCT impact of 0°F for 10 CFR 50.46.

16. Prior LOCA Model Assessment

The 10 CFR 50.46 report dated April 8, 2021, reported no error assessed to the SBLOCA analysis and two errors assessed for the LBLOCA. One error was identified in the support column metal thickness modeling. The second error was a discrepancy where unheated conductors used node sizes inconsistent with the WCOBRA/TRAC two-loop vessel model input guidelines. Both errors each resulted in an estimated LBLOCA PCT impact of 0°F for 10 CFR 50.46.

17. Current LOCA Model Assessment

For the current LBLOCA and SBLOCA analyses, two changes and one error were reported.

Various changes were made to the code used to evaluate both the Small Break and Large Break LOCA analyses. Regular code maintenance resulted in multiple changes to enhance the usability of the code and streamline future analyses. Changes include improving the input diagnostic check, enhancing the code output, optimizing active coding, and eliminating inactive coding. These changes represent Discretionary Changes that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451. The nature of these changes leads to an estimated peak cladding temperature impact of 0°F.

An error was discovered related to the modeling of the flow area to the bottom of the barrel/baffle region for the Small Break LOCA model. For plants without holes in the edge of the lower core plate, the flow area from the bottom of the core to the barrel/baffle region has historically been modeled as the gap between the baffle plate and the lower core plate, and this flow area did not consider the reduced flow area due to the presence of the bottom nozzle flow skirt. The impact of reducing the flow area between the core and barrel baffle region due to including the bottom nozzle flow skirt has been qualitatively evaluated. This

item represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451. The evaluation determined that considering a reduced flow area from the bottom of the core to the barrel/baffle region when considering the bottom nozzle flow skirt has a negligible effect on the SBLOCA analysis results. This leads to an estimated peak cladding temperature impact of 0°F.

A change to both the Large Break LOCA and Small Break LOCA evaluations was made to add an evaluation of up to 45 stainless steel filler rods being added. As a precaution against and to address the possibility of leaking fuel rods beginning in Cycle 43, Exelon requested Westinghouse to perform an evaluation considering the impact of inserting up to 45 stainless steel filler rods on the large and small break loss-of-coolant accident (LOCA) analyses of record. A leaking fuel rod was discovered during the most recent outage and stainless steel filler rods were added in the now current Cycle 43. This item represents a change in plant configuration or associated setpoints, distinguished from an evaluation model change in Section 4 of WCAP-13451. This addition is further addressed in Reference 3. The estimated effect of up to 45 reconstituted rods was evaluated as a 3°F peak cladding temperature impact on the small break LOCA NOTRUMP analysis of record and a 6°F peak cladding temperature impact on the large break LOCA ASTRUM analysis of record.

This above-mentioned 6°F peak cladding temperature impact on the large break LOCA analysis from the evaluation of up to 45 stainless steel filler rods replaces a previous evaluation. The previous evaluation was for an addition of up to 5 stainless steel filler rods and is described in Note 11. This previous evaluation resulted in a 1°F LBLOCA impact. This 1°F impact is now replaced with the current 6°F impact, as the new 45 stainless steel filler rod addition encompasses any impact resulting from the previous 5 filler rod addition. This has been confirmed by Westinghouse.