

RS-07-070

May 7, 2007

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

Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," Annual Report

Reference: Letter from Patrick R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "10 CFR 50.46, 'Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,' Annual Report," dated May 5, 2006

This letter provides the annual report required by 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The attachments describe the changes in accumulated Peak Cladding Temperature (PCT) since the previous annual submittal (Reference).

Should you have any questions concerning this letter, please contact Mr. David Gullott at (630) 657-2819.

Respectfully,



Patrick R. Simpson
Manager – Licensing

Attachments:

- Attachment A: Quad Cities Nuclear Power Station Unit 1, 10 CFR 50.46 Report
- Attachment B: Quad Cities Nuclear Power Station Unit 2, 10 CFR 50.46 Report (GE Fuel)
- Attachment C: Quad Cities Nuclear Power Station Unit 2, 10 CFR 50.46 Report (Westinghouse Fuel)
- Attachment D: Quad Cities Nuclear Power Station Units 1 and 2, 10 CFR 50.46 Report Assessment Notes

Attachment A
Quad Cities Nuclear Power Station Unit 1
10 CFR 50.46 Report

PLANT NAME: Quad Cities Unit 1
 ECCS EVALUATION MODEL: SAFER/GESTR-LOCA
 REPORT REVISION DATE: 05/07/07
 CURRENT OPERATING CYCLE: 19

ANALYSIS OF RECORD

Evaluation Model:

The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology, NEDE-23785-1-PA, General Electric Company, Revision 1, October 1984.

Calculations:

"SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis for Dresden Nuclear Station 2 and 3 and Quad Cities Nuclear Station Units 1 and 2," NEDC-32990P, Revision 2, GE Nuclear Energy, September 2003.

Fuel Analyzed in Calculation: GE9/10, ATRIUM-9B and GE14

Limiting Fuel Type: GE14

Limiting Single Failure: Diesel Generator

Limiting Break Size and Location: 1.0 Double-Ended Guillotine in a Recirculation Suction Pipe

Reference Peak Cladding Temperature (PCT)

PCT = 2110°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

10 CFR 50.46 Report dated December 6, 2002 (See Note 2)	$\Delta PCT = 0^\circ F$
10 CFR 50.46 Report dated May 8, 2003 (See Note 4)	$\Delta PCT = 0^\circ F$
10 CFR 50.46 Report dated May 5, 2004 (See Note 5)	$\Delta PCT = 0^\circ F$
10 CFR 50.46 Report dated May 5, 2005 (See Note 6)	$\Delta PCT = 0^\circ F$
10 CFR 50.46 Report dated May 5, 2006 (See Note 7)	$\Delta PCT = 0^\circ F$
Net PCT	2110 °F

B. CURRENT LOCA MODEL ASSESSMENTS

Acoustic Side Branch (See Note 8)	$\Delta PCT = 0^\circ F$
Axial Power Shape Impact on Small Break LOCA (See Note 9)	$\Delta PCT = 0^\circ F$
Total PCT change from current assessments	$\Sigma \Delta PCT = 0^\circ F$
Cumulative PCT change from current assessments	$\Sigma \Delta PCT = 0^\circ F$
Net PCT	2110 °F

Attachment B
Quad Cities Nuclear Power Station Unit 2
10 CFR 50.46 Report (GE Fuel)

PLANT NAME: Quad Cities Unit 2
 ECCS EVALUATION MODEL: SAFER/GESTR-LOCA
 REPORT REVISION DATE: 05/07/07
 CURRENT OPERATING CYCLE: 19

ANALYSIS OF RECORD

Evaluation Model:

The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology, NEDE-23785-1-PA, General Electric Company, Revision 1, October 1984.

Calculations:

"SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis for Dresden Nuclear Station 2 and 3 and Quad Cities Nuclear Station Units 1 and 2," NEDC-32990P, Revision 2, GE Nuclear Energy, September 2003.

Fuel Analyzed in Calculation: GE9/10, ATRIUM-9B and GE14

Limiting Fuel Type: GE14

Limiting Single Failure: Diesel Generator

Limiting Break Size and Location: 1.0 Double-Ended Guillotine in a Recirculation Suction Pipe

Reference Peak Cladding Temperature (PCT) PCT = 2110°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

10 CFR 50.46 Report dated March 28, 2002 (See Note 1)	$\Delta PCT = 0^\circ F$
10 CFR 50.46 Report dated May 9, 2002 (See Note 3)	$\Delta PCT = 0^\circ F$
10 CFR 50.46 Report dated May 8, 2003 (See Note 4)	$\Delta PCT = 0^\circ F$
10 CFR 50.46 Report dated May 5, 2004 (See Note 5)	$\Delta PCT = 0^\circ F$
10 CFR 50.46 Report dated May 5, 2005 (See Note 6)	$\Delta PCT = 0^\circ F$
10 CFR 50.46 Report dated May 5, 2006 (See Note 7)	$\Delta PCT = 0^\circ F$
Net PCT	2110°F

B. CURRENT LOCA MODEL ASSESSMENTS

Axial Power Shape Impact on Small Break LOCA (See Note 9)	$\Delta PCT = 0^\circ F$
Total PCT change from current assessments	$\sum \Delta PCT = 0^\circ F$
Cumulative PCT change from current assessments	$\sum \Delta PCT = 0^\circ F$
Net PCT	2110°F

Attachment C
Quad Cities Nuclear Power Station Unit 2
10 CFR 50.46 Report (Westinghouse Fuel)

PLANT NAME: Quad Cities Unit 2
 ECCS EVALUATION MODEL: USA5
 REPORT REVISION DATE: 05/07/07
 CURRENT OPERATING CYCLE: 19

ANALYSIS OF RECORD

Evaluation Model:

"Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," WCAP-16078-P-A, November 2004.

Calculations:

"Task Report for TSD DQW04-21 Final LOCA Analysis for Quad Cities 1 & 2 and Dresden 2 & 3," NF-BEX-06-44-P, Westinghouse Electric Company, LLC. April 2006."

Fuel Analyzed in Calculation: SVEA-96 Optima2
 Limiting Fuel Type: SVEA-96 Optima2
 Limiting Single Failure: LPCI injection valve
 Limiting Break Size and Location: 1.0 double-ended guillotine break in the recirculation pump suction line

Reference Peak Cladding Temperature (PCT) PCT = 2150°F

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

10 CFR 50.46 Report dated May 5, 2006 (See Note 7)	$\Delta PCT = 0^{\circ}F$
Net PCT	2150°F

B. CURRENT LOCA MODEL ASSESSMENTS

Hgap Correlation Input Error (See Note 10)	$\Delta PCT = 0^{\circ}F$
Total PCT change from current assessments	$\Sigma \Delta PCT = 0^{\circ}F$
Cumulative PCT change from current assessments	$\Sigma \Delta PCT = 0^{\circ}F$
Net PCT	2150°F

Attachment D
Quad Cities Nuclear Power Station Units 1 and 2
10 CFR 50.46 Report Assessment Notes

1. Prior LOCA Model Assessment

The 50.46 letter dated March 28, 2002 reported a new LOCA analysis to support extended power uprate (EPU) and transition to GE14 fuel for Quad Cities Unit 2.

[Reference: Letter from Timothy J. Tulon (Exelon) to U.S. NRC, "10 CFR 50.46, 30-Day Report for Quad Cities Unit 2," dated March 28, 2002]

2. Prior LOCA Assessment

A new LOCA analysis was performed to support EPU and transition to GE14 fuel for Quad Cities Unit 1. In the referenced letter, the impact of CS and LPCI leakage, GE LOCA error in the WEVOL code and change in DG start time requirement were reported. There is no assessment penalty.

[Reference: Letter from Timothy J. Tulon (Exelon) to U.S. NRC, "10 CFR 50.46, 30-Day Report for Quad Cities Nuclear Power Station, Unit 1," dated December 6, 2002]

3. Prior LOCA Assessment

In the referenced letter, no LOCA model assessment was reported for Unit 2 PCT.

[Reference: Letter from Timothy J. Tulon (Exelon) to U.S. NRC, "Transmittal of 10 CFR 50.46, 'Acceptance criteria for emergency core cooling systems for light water nuclear power reactors,' Annual Report for Quad Cities Units 1 and 2," dated May 9, 2002]

4. Prior LOCA Assessment

The referenced letter provided the annual 50.46 report for Units 1 and 2. This letter reported no LOCA model assessment for Unit 1 whereas it reported the impact of GE LOCA error in the WEVOL code and change in DG start time requirement for Unit 2. The PCT impact for these errors was determined to be 0°F.

[Reference: Letter from Timothy J. Tulon (Exelon) to U.S. NRC, "Transmittal of 10 CFR 50.46, 'Acceptance criteria for emergency core cooling systems for light water nuclear power reactors,' Annual Report for Quad Cities Nuclear Power Station, Units 1 and 2," dated May 8, 2003]

5. Prior LOCA Assessment

The referenced letter provided the annual 50.46 report for Units 1 and 2. This letter reported GE LOCA errors related to SAFER level/volume table and Steam Separator pressure drop and mid-cycle reload of GE14 fuel for Unit 1 (Cycle

Attachment D
Quad Cities Nuclear Power Station Units 1 and 2
10 CFR 50.46 Report Assessment Notes

18A). For Unit 2, this letter reported the same GE LOCA errors and second reload of GE14 fuel in Cycle 18 core. The PCT impact for these errors and reloads of GE14 fuel was determined to be 0°F.

[Reference: Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Transmittal of 10 CFR 50.46, 'Acceptance criteria for emergency core cooling systems for light water nuclear power reactors,' Annual Report for Quad Cities Nuclear Power Station, Units 1 and 2," dated May 5, 2004]

6. Prior LOCA Assessment

The referenced letter provided the annual 50.46 report for Units 1 and 2. This letter reported GE LOCA error due to new heat source of Units 1 & 2 and Quad Cities Unit 1 Cycle 19 with a new reload of GE14 fuel.

[Reference: Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Transmittal of 10 CFR 50.46, 'Acceptance criteria for emergency core cooling systems for light water nuclear power reactors,' Annual Report for Quad Cities Nuclear Power Station, Units 1 and 2," dated May 5, 2005]

7. Prior LOCA Assessment

The referenced letter provided the annual 50.46 report for Units 1 and 2. This letter reported LOCA evaluations for installation of new steam dryers during mid-cycle outages for Q1C19 and Q2C18, respectively. Also, the letter reported Q2C19 startup in April 2006 with the first reload of Westinghouse Optima2 fuel and implementation of the Westinghouse LOCA analysis. Additionally, LOCA evaluations by both GE and Westinghouse were reported for Unit 2 modification to the inlet configuration of the 6" inlet standpipe of eight main steam safety valves and four Electromatic relief valves, which replaced the previously installed inlet pipe and flange with a 6" Tee, flange and an Acoustic Side Branch (ASB). The PCT impact due to the plant modifications was determined to be 0°F.

[Reference: Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Transmittal of 10 CFR 50.46, 'Acceptance criteria for emergency core cooling systems for light water nuclear power reactors,' Annual Report for Quad Cities Nuclear Power Station, Units 1 and 2," dated May 5, 2006]

8. Current LOCA Assessment

During a mid-cycle outage, in May 2006, Quad Cities Unit 1 installed a modification to the inlet configuration of the 6" inlet standpipe of eight main steam safety valves and four Electromatic relief valves. The modification replaces the previously installed inlet pipe and flange with a 6" Tee, flange and an Acoustic Side Branch (ASB). The ASB consists of 6" pipe filled with metal screen material. GE performed an evaluation on the effects of this modification; the results showed that the licensing basis PCT was unaffected, thus the PCT impact was zero.

Attachment D
Quad Cities Nuclear Power Station Units 1 and 2
10 CFR 50.46 Report Assessment Notes

9. Current LOCA Assessment

Past GE small break analysis assumed a mid-peaked power shape consistent with the GE methodology. Recently, GE has determined that for small break cases, a top-peaked power shape can result in higher calculated PCT than a mid-peaked shape. GE has implemented a change in performing analysis of small break LOCA cases, which requires considering both mid-peaked as well as top-peaked power shapes. GE has evaluated effect of this change for Quad Cities LOCA analysis and determined the impact on the licensing basis PCT is zero. This is because the limiting PCT is based on the DBA large break and the revised small break PCT remains below the limiting large break. Therefore, there is no impact on the limiting large break PCT and as a result licensing basis PCT remains unaffected.

10. Current LOCA Assessment

In the referenced letter, Westinghouse has identified Hgap correlation input error. Westinghouse has assessed the error and determined that the BWR LOCA evaluation model uses a bounding Hgap value. Thus, the effect of this error on peak cladding temperature (PCT) is determined to be zero.