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RS-11-077

10 CFR 50.46

May 6, 2011

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 J. L. Hansen (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 7, 2010

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. Kenneth M. Nicely at (630) 657-2803.

Respectfully,

Jeffrey L. Hansen
Manager – Licensing

Attachments:
1. Quad Cities Nuclear Power Station, Unit 1, 10 CFR 50.46 Report (GE Fuel)
2. Quad Cities Nuclear Power Station, Unit 1, 10 CFR 50.46 Report (Westinghouse Fuel)
3. Quad Cities Nuclear Power Station, Unit 2, 10 CFR 50.46 Report (Westinghouse Fuel)
4. Quad Cities Nuclear Power Station, Units 1 and 2, 10 CFR 50.46 Report Assessment Notes

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector, Quad Cities Nuclear Power Station

ATTACHMENT 1
Quad Cities Nuclear Power Station, Unit 1,
10 CFR 50.46 Report (GE Fuel)

PLANT NAME: Quad Cities Nuclear Power Station, Unit 1
ECCS EVALUATION MODEL: SAFER/GESTR-LOCA
REPORT REVISION DATE: April 18, 2011
CURRENT OPERATING CYCLE: 21

ANALYSIS OF RECORD

Evaluation Model:	NEDE-23785-1-PA, Revision 1, "GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident (Volume III), SAFER/GESTR Application Methodology," October 1984
Calculation:	NEDC-32990P, Revision 2, "SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis for Dresden Nuclear Station 2 and 3 and Quad Cities Nuclear Station Units 1 and 2," 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 the 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)	ΔPCT = 0°F
10 CFR 50.46 Report dated May 8, 2003 (See Note 3)	ΔPCT = 0°F
10 CFR 50.46 Report dated May 5, 2004 (See Note 4)	ΔPCT = 0°F
10 CFR 50.46 Report dated May 5, 2005 (See Note 5)	ΔPCT = 0°F
10 CFR 50.46 Report dated May 5, 2006 (See Note 6)	ΔPCT = 0°F
10 CFR 50.46 Report dated May 7, 2007 (See Note 7)	ΔPCT = 0°F
10 CFR 50.46 Report dated May 7, 2008 (See Note 8)	ΔPCT = 0°F
10 CFR 50.46 Report dated May 7, 2009 (See Note 9)	ΔPCT = 0°F
10 CFR 50.46 Report dated May 7, 2010 (See Note 10)	ΔPCT = 0°F
Net PCT	**2110°F**

B. CURRENT LOCA MODEL ASSESSMENTS

None – See Note 11	ΔPCT = 0°F
Total PCT change from current assessments	ΣΔPCT = 0°F
Cumulative PCT change from current assessments	Σ |ΔPCT| = 0°F
Net PCT	**2110°F**

ATTACHMENT 2
Quad Cities Nuclear Power Station, Unit 1,
10 CFR 50.46 Report (Westinghouse Fuel)

PLANT NAME:	Quad Cities Nuclear Power Station, Unit 1
ECCS EVALUATION MODEL:	USA5
REPORT REVISION DATE:	April 18, 2011
CURRENT OPERATING CYCLE:	21

ANALYSIS OF RECORD

Evaluation Model:	WCAP-16078-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," November 2004
Calculation:	OPTIMA2-TR021QC-LOCA, Revision 5, "Quad Cities 1 & 2 LOCA Analysis for SVEA-96 Optima2 Fuel," September 2009
Fuel Analyzed in Calculation:	SVEA-96 Optima2
Limiting Fuel Type:	SVEA-96 Optima2
Limiting Single Failure:	Low Pressure Coolant Injection System Injection Valve
Limiting Break Size and: Location	1.0 Double-Ended Guillotine in the Recirculation Suction Pipe

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

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

10 CFR 50.46 Report dated May 7, 2010 (See Note 10)	ΔPCT = 11°F
Net PCT	**2161°F**

B. CURRENT LOCA MODEL ASSESSMENTS

None – See Note 11	ΔPCT = 0°F
Total PCT change from current assessments	ΣΔPCT = 0°F
Cumulative PCT change from current assessments	Σ |ΔPCT| = 0°F
Net PCT	**2161°F**

ATTACHMENT 3
Quad Cities Nuclear Power Station, Unit 2,
10 CFR 50.46 Report (Westinghouse Fuel)

PLANT NAME:	Quad Cities Nuclear Power Station, Unit 2
ECCS EVALUATION MODEL:	USA5
REPORT REVISION DATE:	April 18, 2011
CURRENT OPERATING CYCLE:	21

ANALYSIS OF RECORD

Evaluation Model:	WCAP-16078-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," November 2004
Calculations:	OPTIMA2-TR021QC-LOCA, Revision 5, "Quad Cities 1 & 2 LOCA Analysis for SVEA-96 Optima2 Fuel," September 2009.
Fuel Analyzed in Calculation:	SVEA-96 Optima2
Limiting Fuel Type:	SVEA-96 Optima2
Limiting Single Failure:	Low Pressure Coolant Injection System Injection Valve
Limiting Break Size and: Location	1.0 Double-Ended Guillotine in the Recirculation Suction Pipe

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

MARGIN ALLOCATION

A. PRIOR LOCA MODEL ASSESSMENTS

10 CFR 50.46 Report dated May 7, 2010 (See Note 10)	ΔPCT =11°F
Net PCT	**2161°F**

B. CURRENT LOCA MODEL ASSESSMENTS

None – See Note 11	ΔPCT = 0°F
Total PCT change from current assessments	ΣΔPCT = 0°F
Cumulative PCT change from current assessments	Σ |ΔPCT| = 0°F
Net PCT	**2161°F**

ATTACHMENT 4
Quad Cities Nuclear Power Station, Units 1 and 2,
10 CFR 50.46 Report Assessment Notes

1. Prior Loss-of-Coolant Accident (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 Nuclear Power Station (QCNPS) Unit 2.

   [Reference: Letter from T. J. Tulon (Exelon Generation Company, LLC) 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 the transition to GE14 fuel for Unit 1. In the referenced letter, the impact of Core Spray system and Low Pressure Coolant Injection system leakage, a General Electric (GE) LOCA error in the WEVOL code, and a change in the diesel generator (DG) start time requirement were reported. There was no assessment penalty.

   [Reference: Letter from T. J. Tulon (Exelon Generation Company, LLC) 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

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

   [Reference: Letter from T. J. Tulon (Exelon Generation Company, LLC) 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]

4. 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 the SAFER level/volume table and steam separator pressure drop, and the mid-cycle reload of GE14 fuel for Unit 1 Cycle 18A. For Unit 2, this letter reported the same GE LOCA errors and a second reload of GE14 fuel in the Cycle 18 core. The PCT impact for these errors and reloads of GE14 fuel was determined to be 0°F.

   [Reference: Letter from P. R. Simpson (Exelon Generation Company, LLC) 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]

5. Prior LOCA Assessment

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

   [Reference: Letter from P. R. Simpson (Exelon Generation Company, LLC) 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.]

6. Prior LOCA Assessment

   The referenced letter provided the annual 50.46 report for Units 1 and 2. This letter reported LOCA evaluations for the installation of new steam dryers during mid-cycle outages for Unit 1 Cycle 19 and Unit 2 Cycle 18. Also, the letter reported Unit 2 Cycle 19 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 a Unit 2 modification to the 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. The PCT impact due to the plant modifications was determined to be 0°F.

   [Reference: Letter from P. 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]

7. Prior LOCA Assessment

   The referenced letter provided the annual 50.46 report for Units 1 and 2. This letter reported a GE LOCA evaluation for installation of a modification to the configuration of the 6" inlet standpipe of eight main steam safety valves and four electromatic relief valves for Unit 1. This letter also reported an evaluation of a change in the GE small break analysis assumption for axial power shape and the Westinghouse LOCA analysis Hgap correlation input error. The PCT impact due to these changes was determined to be 0°F.

   [Reference: Letter from P. 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 7, 2007]

8. Prior LOCA Assessment

   For the GE LOCA analysis, the referenced letter reported no PCT assessment. For the Westinghouse analysis, it reported a revision to the QCNPS LOCA analysis report (i.e., a new plant-specific LOCA analysis). This new analysis applies to operation of the Westinghouse Optima2 fuel in the Unit 1 and Unit 2 reactors. This analysis applies specific inputs and assumptions in the LOCA calculation approved in the licensed Westinghouse methodology. Also, a second reload of Optima2 fuel was implemented with the Unit 2

Cycle 20 core. The limiting PCT for Optima2 fuel as analyzed under the Westinghouse LOCA method is 2150°F, and the limiting PCT for GE14 fuel as analyzed under the GE LOCA method is 2110°F.

[Reference: Letter from J. L. Hansen (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 7, 2008]

9. Prior LOCA Assessment

For the GE LOCA analysis, the referenced letter reported no PCT assessment. For the Westinghouse analysis, it reported a revision to the LOCA analysis report. The revision was required to clarify the low-pressure Core Spray flow and leakage model because of an error in the assumed Dresden Nuclear Power Station (DNPS) Unit 2 Core Spray flow in the DNPS LOCA analysis. The Westinghouse QCNPS LOCA analysis was not affected by the change due to this revision and the PCT impact due to these changes was determined to be 0°F.

[Reference: Letter from J. L. Hansen (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 7, 2009]

10. Prior LOCA Assessment

The referenced letter identified the PCT impact due to an evaluation of the adjustable speed drive (ASD) modification installed in both Units 1 and 2 and updated vessel leakage on both Westinghouse Optima2 and GE14 fuel types. The ASD implementation resulted in a 0°F PCT impact; whereas, the updated vessel leakage had a 0°F PCT impact for GE14 fuel and a 2°F PCT impact for Optima2 fuel. Also, the referenced letter identified the PCT impact due to the corrected bypass hole flow coefficient for the Westinghouse LOCA analysis of Optima2 fuel. The PCT impact due to this correction was determined to be 9°F.

[Reference: Letter from J. L. Hansen (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 7, 2010]

11. Current LOCA Assessment

Since the last annual report (see Note 10), no vendor notifications of Emergency Core Cooling System (ECCS) model errors/changes that are applicable to QCNPS have been issued. Also, no ECCS-related changes or modifications have occurred at QCNPS that affect the assumptions of the ECCS analyses.