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

August 11, 2009

Mr. Charles G. Pardee
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION, UNIT NO. 1; DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3; LASALLE COUNTY STATION, UNITS 1 AND 2; PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3; AND QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: APPLICATION FOR TECHNICAL SPECIFICATION CHANGE REGARDING REVISION OF CONTROL ROD NOTCH SURVEILLANCE TEST FREQUENCY, CLARIFICATION OF SOURCE RANGE MONITOR INSERT CONTROL ROD ACTION, AND CLARIFICATION OF A FREQUENCY EXAMPLE USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS (TAC NOS. MD8927, MD8928, MD8929, MD8930, MD8931, MD8933, MD8934, MD8935, AND MD8936)

Dear Mr. Pardee:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 188 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1, and Amendment No. 232 to Renewed Facility Operating License No. DPR-19 and Amendment No. 225 to Renewed Facility Operating License No. DPR-25 for the Dresden Nuclear Power Station, Units 2 and 3, respectively, and Amendment No. 193 to Facility Operating License No. NPF-11 and Amendment No. 180 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively, and Amendment No. 272 to Facility Operating License No. DPR-44 and Amendment No. 276 to Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station, Units 2 and 3, respectively, and Amendment No. 244 to Renewed Facility Operating License No. DPR-29 and Amendment No. 239 to Renewed Facility Operating License No. DPR-30 for the Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The amendments are in response to your application dated June 9, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081620236), as supplemented by letter dated March 30, 2009 (ADAMS Accession No. ML090890777). Changes to the Oyster Creek Nuclear Generating Station technical specifications (TSs) were also included with the June 9, 2008, application. Those changes are still under review by the staff and the results of that review will be included in a separate letter.

The amendments revise the TS surveillance requirement frequency in TS 3.1.3, "Control Rod OPERABILITY." The amendments also clarify the requirement to fully insert all insertable control rods for the limiting condition for operation in TS 3.3.1.2, Required Action E.2, "Source Range Monitoring Instrumentation" (Clinton Power Station only). Finally, the amendments revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension.

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

Sincerely,

Christopher Gratton, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-461, 50-237,
50-249, 50-373, 50-374, 50-277,
50-278, 50-254, and 50-265

Enclosures:
1. Amendment No. 188 to DPR-62
2. Amendment No. 232 to DPR-19
3. Amendment No. 225 to DPR-25
4. Amendment No. 193 to NPF-11
5. Amendment No. 180 to NPF-18
6. Amendment No. 272 to DPR-44
7. Amendment No. 276 to DPR-56
8. Amendment No. 244 to DPR-29
9. Amendment No. 239 to DPR-30
10. Safety Evaluation

cc w/encls: Distribution via Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 188
License No. NPF-62

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 June 9, 2008, as supplemented by letter dated March 30, 2009, 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 to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C. (2) of Facility Operating License No. NPF-62 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 188 are hereby incorporated into this license. Exelon Generation Company, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen J. Campbell

Stephen J. Campbell, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: August 11, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 188

FACILITY OPERATING LICENSE NO. NPF-62

DOCKET NO. 50-461

Replace the following pages of the Facility Operating License and Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License NPF-62	License NPF-62
Page 3	Page 3
TSs	TSs
1.0-27	1.0-27
1.0-28	1.0-28
3.1-7	3.1-7
3.1-9	3.1-9
3.1-10	3.1-10
3.3-11	3.3-11

(4) Exelon Generation Company, pursuant to the Act and to 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;

(5) Exelon Generation Company, 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

(6) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40, 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 Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the facility at reactor core power levels not in excess of 3473 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 188, are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-2 (continued)

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- Not required to be performed until 12 hours after ≥ 25% RTP. -----	
Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------NOTE------------------ Only required to be met in MODE 1. -- Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour (plus the extension allowed by SR 3.0.2) interval, but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LCO 3.1.3 Each control rod shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each control rod.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	------------NOTE------------ A stuck rod may be bypassed in the Rod Action Control System (RACS) in accordance with SR 3.3.2.1.9 if required to allow continued operation. ----------------------------	
	A.1 Disarm the associated control rod drive (CRD).	2 hours
	AND	
	A.2 Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the Rod Pattern Control System (RPCS)
	AND	
	A.3 Perform SR 3.1.1.1.	72 hours

(continued)

ACTIONS (Continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Required Action and associated Completion Time of Condition A, C, or D not met. OR Nine or more control rods inoperable.	E.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	DELETED	

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.3	------------------NOTE------------------ Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RPCS. -- Insert each withdrawn control rod at least one notch.	31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to notch position 13 is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4
SR 3.1.3.5	Verify each control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position <u>AND</u> Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more required SRMs inoperable in MODE 3 or 4.	D.1 Fully insert all insertable control rods.	1 hour
	AND	
	D.2 Place reactor mode switch in the shutdown position.	1 hour
E. One or more required SRMs inoperable in MODE 5.	E.1 Suspend CORE ALTERATIONS except for control rod insertion.	Immediately
	AND	
	E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No.232
Renewed License No. DPR-19

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

   A. The application for amendment by the Exelon Generation Company, LLC (the licensee) dated June 9, 2008, as supplemented by letter dated March 30, 2009, 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 to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C. (2) of Renewed Facility Operating License No. DPR-19 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 232 , are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen J. Campbell

Stephen J. Campbell, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: August 11, 2009

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 225
Renewed License No. DPR-25

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

   A. The application for amendment by the Exelon Generation Company, LLC (the licensee) dated June 9, 2008, as supplemented by letter dated March 30, 2009, 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 to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B. of Renewed Facility Operating License No. DPR-25 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 225 , are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen J. Campbell, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: August 11, 2009

ATTACHMENT TO LICENSE AMENDMENT NOS. 232 AND 225

RENEWED FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25

DOCKET NOS. 50-237 AND 50-249

Replace the following pages of the Facility Operating Licenses and Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License DPR-19 Page 3	License DPR-19 Page 3
License DPR-25 Page 4	License DPR-25 Page 4
TSs 1.4-4 3.1.3-2 3.1.3-4	TSs 1.4-4 3.1.3-2 3.1.3-4

(2) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear materials as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) Exelon Generation Company, LLC, 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 Company, LLC, 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 Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2957 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Techical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 232, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Operation in the coastdown mode is permitted to 40% power.

f. Surveillance Requirement 4.9.A.10 - Diesel Storage Tank Cleaning (Unit 3 and Unit 2/3 only)

Each of the above Surveillance Requirements shall be successfully demonstrated prior to entering into MODE 2 on the first plant startup following the fourteenth refueling outage (D3R14).

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

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state power levels not in excess of 2957 megawatts (thermal), except that the licensee shall not operate the facility at power levels in excess of five (5) megawatts (thermal), until satisfactory completion of modifications and final testing of the station output transformer, the auto-depressurization interlock, and the feedwater system, as described in the licensee's telegrams; dated February 26, 1971, have been verified in writing by the Commission.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 225, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. Restrictions

Operation in the coastdown mode is permitted to 40% power.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------NOTE------------------ Not required to be performed until 12 hours after ≥ 25% RTP. -------------------------------------- Perform channel adjustment.	 7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	AND		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	--------NOTE--------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. --------------------	
		Fully insert inoperable control rod.	3 hours
	AND		
	C.2	Disarm the associated CRD.	4 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	DELETED	
SR 3.1.3.3	------------------NOTE------------------ Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. --------------------------------------- Insert each withdrawn control rod at least one notch.	 31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to 90% insertion is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 193
License No. NPF-11

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

   A. The application for amendment filed by the Exelon Generation Company, LLC (the licensee) dated June 9, 2008, as supplemented by letter dated March 30, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

   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 to the Technical Specifications as indicated in the enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-11 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 193 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen J. Campbell

Stephen J. Campbell, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: August 11, 2009

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 180
License No. NPF-18

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

   A. The application for amendment filed by the Exelon Generation Company, LLC (the licensee), dated June 9, 2008, as supplemented by letter dated March 30, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

   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 to the Technical Specifications as indicated in the enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-18 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 180 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen J. Campbell

Stephen J. Campbell, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: August 11, 2009

ATTACHMENT TO LICENSE AMENDMENT NOS. 193 AND 180

FACILITY OPERATING LICENSE NOS. NPF-11 AND NPF-18

DOCKET NOS. 50-373 AND 50-374

Replace the following pages of the Facility Operating Licenses and Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License NPF-11	License NPF-11
Page 3	Page 3
License NPF-18	License NPF-18
Page 3	Page 3
TSs	TSs
1.4-4	1.4-4
3.1.3-2	3.1.3-2
3.1.3-4	3.1.3-4

(4) Exelon Generation Company, LLC, 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 Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of LaSalle County Station, Units 1 and 2.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3489 megawatts thermal).

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 193 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Conduct of Work Activities During Fuel Load and Initial Startup

The licensee shall review by committee all Unit 1 Preoperational Testing and System Demonstration activities performed concurrently with Unit 1 initial fuel loading or with the Unit 1 Startup Test Program to assure that the activity will not affect the safe performance of the Unit 1 fuel loading or the portion of the Unit 1 Startup Program being performed. The review shall address, as a minimum, system interaction, span of control, staffing, security and health physics, with respect to performance of the activity concurrently with the Unit 1 fuel loading or the portion of the Unit 1 Startup Program being performed. The committee for the review shall be composed of at least three members, knowledgeable in the above areas, and who meet the qualifications for professional-technical personnel specified by

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70 possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of LaSalle County Station Units 1 and 2.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3489 megawatts thermal). Items in Attachment 1 shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 180 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Conduct of Work Activities During Fuel Load and Initial Startup

The licensee shall review by committee all Unit 2 Preoperational Testing and System Demonstration activities performed concurrently with Unit 2 initial fuel loading or with the Unit 2 Startup Test Program to assure that the activity will not affect the safe performance of the Unit 2 fuel loading or the portion of the Unit 2 Startup Program being performed. The review shall address, as a minimum, system interaction, span of control, staffing, security and health physics, with respect to performance of the activity concurrently with the Unit 2 fuel loading or the portion of the Unit 2 Startup Program being performed. The committee for the review shall be composed of at least three members, knowledgeable in the above areas, and who meet the qualifications for professional-technical personnel specified by section 4.4 of ANSI N18.7-1971. At least one of these three shall be a senior member of the Assistant Superintendent of Operation's staff.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------NOTE------------------ Not required to be performed until 12 hours after ≥ 25% RTP. --	
Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod. AND A.4 Perform SR 3.1.1.1.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM 72 hours
B. Two or more withdrawn control rods stuck.	B.1 Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1 --------NOTE--------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. -------------------- Fully insert inoperable control rod. AND C.2 Disarm the associated CRD.	3 hours 4 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	DELETED	
SR 3.1.3.3	-------------------NOTE------------------- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. --- Insert each withdrawn control rod at least one notch.	 31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to notch position 05 is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

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

EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 272
Renewed License No. DPR-44

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), and PSEG Nuclear LLC (the licensees), dated June 9, 2008, as supplemented by letter dated March 30, 2009, 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 to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C (2) of Renewed Facility Operating License No. DPR-44 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 272 , are hereby incorporated in the license. Exelon Generation Company 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

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and Technical Specifications

Date of Issuance: August 11, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 272

RENEWED FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following pages of the Facility Operating License and Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License DPR-44 Page 3	License DPR-44 Page 3
TSs	TSs
1.4-4	1.4-4
1.4-5	1.4-5
3.1-8	3.1-8
3-1-10	3.1-10

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

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the 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 Company is authorized to operate the Peach Bottom Atomic Power Station, Unit 2, at steady state reactor core power levels not in excess of 3514 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 272 are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Peach Bottom Atomic Power Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program, Revision 3." The set contains Safeguards Information protected under 10 CFR 73.21.

(4) Fire Protection

The Exelon Generation Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility, and as approved in the NRC Safety Evaluation Report (SER) dated May 23, 1979, and Supplements dated August 14, September 15, October 10 and November 24, 1980, and in the NRC SERs dated September 16, 1993, and August 24, 1994, subject to the following provision:

The Exelon Generation Company may-make changes to the approved

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-2 (continued)

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------NOTE------------------ Not required to be performed until 12 hours after ≥ 25% RTP. -- Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----------------NOTE----------------- Only required to be met in MODE 1. -------------------------------------- Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	AND		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	--------NOTE--------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. ---------------------	
		Fully insert inoperable control rod.	3 hours
	AND		
	C.2	Disarm the associated CRD.	4 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	Deleted	
SR 3.1.3.3	-------------------NOTE------------------- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. -- Insert each withdrawn control rod at least one notch.	31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to notch position 06 is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

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

EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 276
Renewed License No. DPR-56

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), and PSEG Nuclear LLC (the licensees), dated June 9, 2008, as supplemented by letter dated March 30, 2009, 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 to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C (2) of Renewed Facility Operating License No. DPR-56 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 276 , are hereby incorporated in the license. Exelon Generation Company 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

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and Technical Specifications

Date of Issuance: August 11, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 276

RENEWED FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following pages of the Facility Operating License and Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License DPR-56	License DPR-56
Page 3	Page 3
TSs	TSs
1.4-4	1.4-4
1.4-5	1.4-5
3.1-8	3.1-8
3.1-10	3.1-10

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

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the 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 Company is authorized to operate the Peach Bottom Atomic Power Station, Unit No. 3, at steady state reactor core power levels not in excess of 3514 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No276 , are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.[1]

(3) Physical Protection

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[2], submitted by letter dated May 17, 2006, is entitled: Peach Bottom Atomic Power Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program, Revision 3." The set contains Safeguards Information protected under 10 CFR 73.21.

(4) Fire Protection

The Exelon Generation Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility, and as approved in

[1]Licensed power level was revised by Amendment No. 250, dated November 22, 2002, and will be implemented following the 14^{th} refueling outage currently scheduled for Fall 2003.

[2]The training and Qualification Plan and Safeguards Contingency Plan and Appendices to the Security Plan.

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-2 (continued)

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------NOTE------------------ Not required to be performed until 12 hours after ≥ 25% RTP. -- Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----------------NOTE----------------- Only required to be met in MODE 1. ---------------------------------------	
Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	AND		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	--------NOTE--------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. ---------------------	
		Fully insert inoperable control rod.	3 hours
	AND		
	C.2	Disarm the associated CRD.	4 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	Deleted	
SR 3.1.3.3	-------------------NOTE------------------- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. -- Insert each withdrawn control rod at least one notch.	31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to notch position 06 is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

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

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 244
Renewed License No. DPR-29

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

   A. The application for amendment by the Exelon Generation Company, LLC (the licensee) dated June 9, 2008, as supplemented by letter dated March 30, 2009, 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 to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Renewed Facility Operating License No. DPR-29 is hereby amended to read as follows:

B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 244 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen J. Campbell

Stephen J. Campbell, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: August 11, 2009

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

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO.50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 239
Renewed License No. DPR-30

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

   A. The application for amendment by the Exelon Generation Company, LLC, et al. (the licensees) dated June 9, 2008, as supplemented by letter dated March 30, 2009, 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 to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-30 is hereby amended to read as follows:

B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 239 , are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen J. Campbell

Stephen J. Campbell, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: August 11, 2009

ATTACHMENT TO LICENSE AMENDMENT NOS. 244 AND 239

RENEWED FACILITY OPERATING LICENSES NOS. DPR-29 AND DPR-30

DOCKET NOS. 50-254 AND 50-265

Replace the following pages of the Facility Operating Licenses and Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by number and contain marginal lines indicating the areas of change.

Remove	Insert
License DPR-29	License DPR-29
Page 4	Page 4
License DPR-30	License DPR-30
Page 4	Page 4
TSs	TSs
1.4-4	1.4-4
3.1.3-2	3.1.3-2
3.1.3-4	3.1.3-4

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 244, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. The licensee shall maintain the commitments made in response to the March 14, 1983, NUREG-0737 Order, subject to the following provision:

The licensee may make changes to commitments made in response to the March 14, 1983, NUREG-0737 Oder without prior approval of the Commission as long as the change would be permitted without NRC approval, pursuant to the requirements of 10 CFR 50.59. Consistent with this regulation, if the change results in an Unreviewed Safety Question, a license amendment shall be submitted to the NRC staff for review and approval prior to implementation of the change.

D. Equalizer Valve Restriction

Three of the four valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation with one bypass valve open to allow for thermal expansion of water.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined sets of plans[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Quad Cities Nuclear Power Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 2," submitted by letter dated May 17, 2006.

F. The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Reports dated July 27, 1979 with supplements dated November 5, 1980, and

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 239, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. The license shall maintain the commitments made in response to the March 14, 1983, NUREG-0737 Order, subject to the following provision:

> The licensee may make changes to commitments made in response to the March 14, 1983, NUREG-0737 Order without prior approval of the Commission as long as the change would be permitted without NRC approval, pursuant to the requirements of 10 CFR 50.59. Consistent with this regulation, if the change results in an Unreviewed Safety Question, a license amendment shall be submitted to the NRC staff for review and approval prior to implementation of the change.

D. Equalizer Valve Restriction

Three of the four valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation with one bypass valve open to allow for thermal expansion of water.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Quad Cities Nuclear Power Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 2," submitted by letter dated May 17, 2006.

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[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----------------NOTE----------------- Not required to be performed until 12 hours after ≥ 25% RTP. --------------------------------------- Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	<u>AND</u>	
	A.4 Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1 Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1 --------NOTE--------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. -------------------- Fully insert inoperable control rod.	3 hours
	<u>AND</u>	
	C.2 Disarm the associated CRD.	4 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	DELETED	
SR 3.1.3.3	------------------NOTE------------------ Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. -- Insert each withdrawn control rod at least one notch.	31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to 90% insertion is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 188 TO FACILITY OPERATING LICENSE NO. NPF-62,

AMENDMENT NO. 232 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-19,

AMENDMENT NO. 225 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-25,

AMENDMENT NO. 193 TO FACILITY OPERATING LICENSE NO. NPF-11,

AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. NPF-18,

AMENDMENT NO. 272 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-44,

AMENDMENT NO. 276 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-56,

AMENDMENT NO. 244 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-29,

AMENDMENT NO. 239 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-30,

EXELON GENERATION COMPANY, LLC

CLINTON POWER STATION, UNIT NO. 1

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

LASALLE COUNTY STATION, UNITS 1 AND 2

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-461, 50-237, 50-249, 50-373, 50-374,

50-277, 50-278, 50-254, AND 50-265

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (the Commission, NRC) dated June 9, 2008, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081620236), as supplemented by letter dated March 30, 2009 (ADAMS Accession No. ML090890777), Exelon Generation Company, LLC (EGC), and AmerGen Energy Company, LLC (AmerGen), requested changes to the technical specifications (TSs) and surveillance requirements (SRs) for the following units: Clinton Power Station, Unit No. 1 (CPS), Dresden Nuclear Power Station, Units 2 and 3 (DNPS), LaSalle County Station, Units 1 and 2 (LSCS), Oyster Creek Nuclear Generating Station (Oyster Creek), Peach Bottom Atomic Power Station, Units 2 and 3 (PBAPS), and Quad Cities Nuclear Power Station, Units 1 and 2 (QCNPS). EGC is the licensee for DNPS, LSCS, PBAPS, and QCNPS. At the time of the application, AmerGen was the licensee for CPS and Oyster Creek.

The supplement dated March 30, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 12, 2008 (73 FR 46928).

The changes are consistent with the NRC-approved Technical Specification Task Force (TSTF) change traveler TSTF-475, Revision 1, "Control Rod Notch Testing and [Source Range Monitor] SRM Insert Control Rod Action," with some deviations as discussed below. TSTF-475, Revision 1, was approved for use by the NRC on November 5, 2007 (ADAMS Accession No. ML073050017). This operating license improvement was made available to the industry by the NRC on November 13, 2007 (72 FR 63935) through the consolidated line item improvement process (CLIIP).

The licensee is proposing plant-specific deviations from the TS changes described in the TSTF-475, Revision 1, and the NRC staff's model safety evaluation dated November 13, 2007. The deviations are discussed in the Technical Evaluation Section (Section 2.0) of the attachments to this safety evaluation for the applicable plants. The deviations do not affect the applicability of either the safety evaluation or the no significant hazards consideration determination published in the *Federal Register* as part of the CLIIP.

Changes to the Oyster Creek TSs that were included with the June 9, 2008, application are still under review by the NRC staff. The results of that review will be included in a separate letter.

AmerGen was a wholly-owned subsidiary of EGC. On January 8, 2009, EGC eliminated AmerGen and transferred the operating licenses of the AmerGen reactor plants to EGC. By letter dated January 9, 2009, EGC adopted and endorsed docketed submittals that requested specific licensing actions that were made by AmerGen, and requested that the NRC staff continue to process those pending actions on the schedules previously agreed to by AmerGen.

2.0 REGULATORY EVALUATION

Attachments A thru E contain the regulatory evaluations for the nine reactor operating licenses covered by these amendments.

3.0 TECHNICAL EVALUATION

Attachments A thru E contain the technical evaluations for the nine reactor operating licenses covered by these amendments.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, Illinois and Pennsylvania State officials were notified of the proposed issuance of the amendments. The States' officials had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility's components located within the restricted area as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20 or a change to SRs. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (73 FR 46928). Accordingly, the amendments meet 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 amendments.

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) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Attachment A: Regulatory and Technical Evaluation for Clinton Power Station, Unit 1
Attachment B: Regulatory and Technical Evaluation for Dresden Nuclear Power Station, Units 2 and 3
Attachment C: Regulatory and Technical Evaluation for LaSalle County Station, Units 1 and 2
Attachment D: Regulatory and Technical Evaluation for Peach Bottom Atomic Power Station, Units 2 and 3
Attachment E: Regulatory and Technical Evaluation for Quad Cities Nuclear Power Station, Units 1 and 2

Principal Contributor: Ravi Grover, NRR

Date: August 11, 2009

REGULATORY AND TECHNICAL EVALUATIONS

CLINTON POWER STATION, UNIT NO. 1

DOCKET NO. 50-461

1.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix A, General Design Criterion (GDC) 26 - "Reactivity control system redundancy and capability," states that:

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences [AOOs], and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

GDC 29, "Protection against anticipated operational occurrences," states, "the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences."

The design relies on the Control Rod Drive (CRD) System to function in conjunction with the protection systems under AOOs, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRD System provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during AOOs. Compliance with GDCs 26 and 29 for the CRD System prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during AOOs. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier. Per Clinton Power Station's (CPS) Updated Safety Analysis Report, CPS Unit 1 fully satisfies and is in compliance with GDCs 26 and 29.

Section 50.36(c)(3) of 10 CFR states that Technical Specifications (TSs) shall contain Surveillance Requirements (SRs) "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 limiting conditions for operation will be met." As discussed in Section 2.0 of this attachment, revising the SR frequency for notch testing of each fully withdrawn control rod from weekly to monthly, clarifying in a TS example that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to time periods discussed in SR Notes, and also, clarifying a requirement to fully insert control rods in TS Limiting Condition for Operation (LCO) 3.3.1.2, still assures 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.

2.0 TECHNICAL EVALUATION

2.1 Revise the SR Frequency for notch testing of each fully withdrawn control rod from weekly to monthly (TS 3.1.3, "Control Rod OPERABILITY")

The CRD system consists of CRDs, which are hydraulically operated stepping mechanisms mounted in CRD housings, which extend below the reactor vessel bottom head.

The collet retainer tube (CRT) is a short tube welded to the upper end of the CRD, which houses the collet mechanism. The collet mechanism performs the locking and unlocking functions that allow the insertion and withdrawal of the control rod. The latch, or locking collet, is a ratchet device that allows the control rod to be freely inserted but requires a specific unlock signal for rod withdrawal.

Control rod insertion capability is demonstrated by notch testing (i.e., inserting a control rod by at least one notch). Notch testing is currently performed weekly for fully withdrawn control rods and monthly for partially withdrawn control rods. During power operation, most control rods in the core are fully withdrawn, and subjected to notch testing at weekly intervals. Notch testing can also detect a CRT that is totally severed, e.g., from a 360-degree from Intergranular Stress Corrosion Cracking (IGSCC)-initiated crack, and can identify most postulated causes of mechanical binding.

Notch testing is designed to verify the ability to move rods. The ability to scram may be inferred from notch test results; but this is confirmed through scram time testing. Scram time testing can also detect problems in CRD performance resulting from IGSCC-initiated cracks and mechanical binding. Unlike the notch tests, these single rod scram tests cover additional mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod. The Hydraulic Control Units, CRDs, and control rods are also tested during refueling outages, approximately every 18 to 24 months. Based on the data collected during the preceding cycle of operation, selected CRDs are inspected and their internal components are replaced, as required.

In 1975, cracking was observed in some CRTs (Ref. 1). Circumferential cracking could lead to failure of a CRT that would prevent movement of its CRD. Notch testing, which requires movement of CRDs, is used to demonstrate CRT integrity. Since there have been no CRT failures since cracking was first observed in 1975 (Ref. 1) and since the CRT crack growth rate is slow (Ref. 1), the applicant maintains that it would be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly.

IGSCC growth rates were evaluated using General Electric's PLEDGE model (Ref. 1), based on fundamental principles of stress corrosion cracking that can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. This report states that adding 24 days to the surveillance interval could result in an additional 10 mils of growth in total crack length before the next surveillance. The small addition in crack length would not amount to a significant difference in the results of two notch tests, performed 31 days apart.

The applicant also states that reducing the surveillance frequency for notch testing of fully withdrawn rods would (1) reduce the duty on the Reactor Manual Control System and CRD hardware, and (2) reduce the likelihood of reactivity control errors (e.g., incorrect insertion of control rods), since there would be fewer operator actions. The change in surveillance frequency would result in a substantial reduction in the number of rod movements. For example, TSTF-475 (Ref. 2) indicates that for BWR/4 plants, the number of notch tests per year would decrease from 6,590 to 1,613, and for BWR/6 plants, the number of notch tests per year would decrease from 9,284 to 2,272.

The NRC staff concludes that it would be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly, based on the following reasons:

(1) The accumulation of operating experience, as reviewed by the NRC staff, indicates there have been no immovable control rods identified via performance of rod notch surveillance for either partially or fully withdrawn control rods.

(2) The predicted crack growth rate is slow. The proposed surveillance interval (31 days) remains short enough to be effective in detecting failed CRTs.

The NRC staff finds that increasing the surveillance interval from 7 days to 31 days would not compromise the CRD system's capability to reliably control reactivity changes under normal operations, including AOOs, such that specified acceptable fuel design limits are not exceeded.

The NRC staff notes that General Electric recommends a limited sampling of several CRDs removed for maintenance, for evidence of discernable corrosion that is different from corrosion that was observed in the past when weekly notching was performed, and an evaluation of CRT maintenance data to assess the actual extent of CRT cracking (Ref.1). The NRC staff's conclusions are not based upon implementation of either of these recommendations. Operational experience, slow crack growth, and potential benefits of reduced control rod movements were sufficient. Therefore, implementation of either or both of these recommendations remains at the discretion of the user.

2.1.1 References

1. Letter BWROG–06036, "Response to NRC Request for Additional Information Regarding TSTF–475, Revision 0," dated November 16, 2006, with enclosure of GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," (report originally prepared in 2004), GE-NE-0000-0024-9858, Revision 3, November 2006 (ADAMS Accession No. ML063250258).

2. Letter TSTF-07-19, Response from the Technical Specifications Task Force to the NRC, "Request for Additional Information (RAI) Regarding TSTF-475 Revision 0, "Control Rod

Notch Testing Frequency and SRM Insert Control Rod Action," dated February 28, 2007, (TSTF-475, Revision 1 is an enclosure) (ADAMS Accession No. ML071420428).

2.2 Clarify in TS Example 1.4-3 that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to Time Periods discussed in SR Notes

The NRC staff has reviewed the proposal to amend Example 1.4-3 in Section 1.4 "Frequency," to clarify that the 1.25 provision in SR 3.0.2 is equally applicable to time periods specified in the Note in the "Surveillance" column. The NRC staff finds this change acceptable because the revision clarifies the example to make it consistent with the definition of "specified Frequency" provided in the second paragraph of Section 1.4, which states that "the 'specified Frequency' is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The 'specified Frequency' consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements."

2.3 Clarify the requirement to fully insert all insertable control rods for the LCO in TS 3.3.1.2, Required Action E.2, "Source Range Monitoring Instrumentation"

The NRC staff has reviewed the proposal to clarify the Required Action E.2 in TS LCO 3.3.1.2 by adding the word "fully" before "insert all insertable control rods." Currently, it states, "initiate action to insert all insertable control rods..." The NRC staff finds this change acceptable because the requirement to insert control rods is meant to require control rods to be fully inserted and adding "fully" does not change, but clarifies the intent of the action.

3.0 <u>SUMMARY</u>

The NRC staff has reviewed the licensee's proposal to amend Clinton Power Station TSs and has concluded that the TS revisions will have a minimal effect on the reliability of the CRD System while reducing the opportunity for potential reactivity events, and will clarify the applicability of the 1.25 provision in SR 3.0.2, and the application of Required Action E.2 in TS LCO 3.3.1.2. Therefore, the NRC staff concludes that the amendment request is acceptable.

REGULATORY AND TECHNICAL EVALUATIONS

DRESDEN NUCLEAR POWER STATION, UNITS 2, AND 3

DOCKET NO. 50-237 AND 50-249

1.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix A, General Design Criterion (GDC) 26 - "Reactivity control system redundancy and capability," states that:

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences [AOO], and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

GDC 29, "Protection against anticipated operational occurrences," states, "the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences."

Dresden Nuclear Power Station (DNPS), Units 2 and 3, were designed and construction was commenced prior to the codification of the current GDCs, thus the current GDCs are not part of the original design basis of the plant. However, the Updated Final Safety Analysis Report for DNPS Units 2 and 3 contains an evaluation of the design bases of the nuclear facility as measured against the General Design Criteria for Nuclear Power Plant Construction Permits that were proposed to be added to 10 CFR Part 50 as Appendix A in July 1967. The licensee concluded that DNPS Units 2 and 3 satisfied the intent of the proposed General Design Criteria for Nuclear Power Plants issued by the Atomic Energy Commission in July 1967.

The design relies on the Control Rod Drive (CRD) System to function in conjunction with the protection systems under AOOs, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRD System provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during AOOs. Compliance with GDCs 26 and 29 for the CRD System prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during AOOs. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier.

Section 50.36(c)(3) of 10 CFR states that Technical Specifications (TSs) shall contain Surveillance Requirements (SRs) "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 limiting conditions for operation will be met." As discussed in Section 2.0 of this attachment, revising the SR frequency for notch testing of each fully withdrawn control rod from weekly to monthly, as well as clarifying in a TS example that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to time periods discussed in SR Notes, still assures that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

2.0 TECHNICAL EVALUATION

2.1 Revise the SR Frequency for notch testing of each fully withdrawn control rod from weekly to monthly (TS 3.1.3, "Control Rod OPERABILITY")

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Notch testing is designed to verify the ability to move rods. The ability to scram may be inferred from notch test results; but this is confirmed through scram time testing. Scram time testing can also detect problems in CRD performance resulting from IGSCC-initiated cracks and mechanical binding. Unlike the notch tests, these single rod scram tests cover additional mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod. The Hydraulic Control Units, CRDs, and control rods are also tested during refueling outages, approximately every 18 to 24 months. Based on the data collected during the preceding cycle of operation, selected CRDs are inspected and their internal components are replaced, as required.

In 1975, cracking was observed in some CRTs (Ref. 1). Circumferential cracking could lead to failure of a CRT that would prevent movement of its CRD. Notch testing, which requires movement of CRDs, is used to demonstrate CRT integrity. Since there have been no CRT failures since cracking was first observed in 1975 (Ref. 1) and since the CRT crack growth rate is slow (Ref. 1), the applicant maintains that it would be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly.

IGSCC growth rates were evaluated using General Electric's PLEDGE model (Ref. 1), based on fundamental principles of stress corrosion cracking that can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. This report states that adding 24 days to the surveillance interval could result in an additional 10 mils of growth in total crack length before the next surveillance. The small addition in crack length would not amount to a significant difference in the results of two notch tests, performed 31 days apart.

The applicant also states that reducing the surveillance frequency for notch testing of fully withdrawn rods would (1) reduce the duty on the Reactor Manual Control System and CRD hardware, and (2) reduce the likelihood of reactivity control errors (e.g., incorrect insertion of control rods), since there would be fewer operator actions. The change in surveillance frequency would result in a substantial reduction in the number of rod movements. For example, TSTF-475 (Ref. 2) indicates that for BWR/4 plants, the number of notch tests per year would decrease from 6,590 to 1,613, and for BWR/6 plants, the number of notch tests per year would decrease from 9,284 to 2,272.

The NRC staff concludes that it would be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly, based on the following reasons:

(1) The accumulation of operating experience, as reviewed by the NRC staff, indicates there have been no immovable control rods identified via performance of rod notch surveillance for either partially or fully withdrawn control rods.

(2) The predicted crack growth rate is slow. The proposed surveillance interval (31 days) remains short enough to be effective in detecting failed CRTs.

The NRC staff finds that increasing the surveillance interval from 7 days to 31 days would not compromise the CRD system's capability to reliably control reactivity changes under normal operations, including anticipated operational occurrences, such that specified acceptable fuel design limits are not exceeded.

The NRC staff notes that General Electric recommends a limited sampling of several CRDs, removed for maintenance, for evidence of discernable corrosion that is different from corrosion that was observed in the past when weekly notching was performed, and an evaluation of CRT maintenance data to assess the actual extent of CRT cracking (Ref.1). The NRC staff's conclusions are not based upon implementation of either of these recommendations. Operational experience, slow crack growth, and potential benefits of reduced control rod movements were sufficient. Therefore, implementation of either or both of these recommendations remains at the discretion of the user.

2.1.1 References

1. Letter BWROG–06036, "Response to NRC Request for Additional Information Regarding TSTF–475, Revision 0," dated November 16, 2006, with enclosure of GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," (report originally prepared in 2004), GE-NE-0000-0024-9858, Revision 3, November 2006 (ADAMS Accession No. ML063250258).

2. Letter TSTF-07-19, Response from the Technical Specifications Task Force to the NRC, "Request for Additional Information (RAI) Regarding TSTF-475 Revision 0, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action," dated February 28, 2007, (TSTF-475, Revision 1 is an enclosure) (ADAMS Accession No. ML071420428).

2.2 Clarify in TS Example 1.4-3 that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to Time Periods discussed in SR Notes

The NRC staff has reviewed the proposal to amend Example 1.4-3 in Section 1.4 "Frequency," to clarify that the 1.25 provision in SR 3.0.2 is equally applicable to time periods specified in the Note in the "Surveillance" column. The NRC staff finds this change acceptable because the revision clarifies the example to make it consistent with the definition of "specified Frequency" provided in the second paragraph of Section 1.4, which states that "the 'specified Frequency' is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The 'specified Frequency' consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements."

3.0 SUMMARY

The NRC staff has reviewed the licensee's proposal to amend Dresden Nuclear Power Station, Units 2 and 3 TS to revise the TS surveillance requirement frequency in TS 3.1.3, "Control Rod Operability," and revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The NRC staff has concluded that the TS revisions will have a minimal effect on the reliability of the CRD System while reducing the opportunity for potential reactivity events, and will clarify the applicability of the 1.25 provision in SR 3.0.2. Therefore, the NRC staff concludes that the amendment request is acceptable.

REGULATORY AND TECHNICAL EVALUATIONS

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix A, General Design Criterion (GDC) 26 - "Reactivity control system redundancy and capability," states that:

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences [AOO], and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

GDC 29, "Protection against anticipated operational occurrences," states, "the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences."

In accordance with LaSalle County Station Units 1 and 2 Updated Final Safety Analysis Report, the design fully satisfies and is in compliance with GDCs 26 and 29.

The design relies on the Control Rod Drive (CRD) System to function in conjunction with the protection systems under AOOs, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRD System provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during AOOs. Compliance with GDCs 26 and 29 for the CRD System prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during AOOs. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier.

Section 50.36(c)(3) of 10 CFR states that Technical Specifications (TSs) shall contain Surveillance Requirements (SRs) "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 limiting conditions for operation will be met." As discussed in Section 2.0 of this attachment, revising the SR frequency for notch testing of each fully withdrawn control rod from weekly to monthly, as well as clarifying in a TS example that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to time periods discussed in SR Notes, still assures that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

2.0 TECHNICAL EVALUATION

2.1 Revise the SR Frequency for notch testing of each fully withdrawn control rod from weekly to monthly (TS 3.1.3, "Control Rod OPERABILITY")

The CRD system consists of CRDs, which are hydraulically operated stepping mechanisms mounted in CRD housings, which extend below the reactor vessel bottom head.

The collet retainer tube (CRT) is a short tube welded to the upper end of the CRD, which houses the collet mechanism. The collet mechanism performs the locking and unlocking functions that allow the insertion and withdrawal of the control rod. The latch, or locking collet, is a ratchet device that allows the control rod to be freely inserted but requires a specific unlock signal for rod withdrawal.

Control rod insertion capability is demonstrated by notch testing (i.e., inserting a control rod by at least one notch). Notch testing is currently performed weekly for fully withdrawn control rods and monthly for partially withdrawn control rods. During power operation, most control rods in the core are fully withdrawn, and subjected to notch testing at weekly intervals. Notch testing can also detect a CRT that is totally severed, e.g., from a 360-degree from Intergranular Stress Corrosion Cracking (IGSCC)-initiated crack, and can identify most postulated causes of mechanical binding.

Notch testing is designed to verify the ability to move rods. The ability to scram may be inferred from notch test results; but this is confirmed through scram time testing. Scram time testing can also detect problems in CRD performance resulting from IGSCC-initiated cracks and mechanical binding. Unlike the notch tests, these single rod scram tests cover additional mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod. The Hydraulic Control Units, CRDs, and control rods are also tested during refueling outages, approximately every 18 to 24 months. Based on the data collected during the preceding cycle of operation, selected CRDs are inspected and their internal components are replaced, as required.

In 1975, cracking was observed in some CRTs (Ref. 1). Circumferential cracking could lead to failure of a CRT that would prevent movement of its CRD. Notch testing, which requires movement of CRDs, is used to demonstrate CRT integrity. Since there have been no CRT failures since cracking was first observed in 1975 (Ref. 1) and since the CRT crack growth rate is slow (Ref. 1), the applicant maintains that it would be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly.

IGSCC growth rates were evaluated using General Electric's PLEDGE model (Ref. 1), based on fundamental principles of stress corrosion cracking that can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. This report states that adding 24 days to the surveillance interval could result in an additional 10 mils of growth in total crack length before the next surveillance. The small addition in crack length would not amount to a significant difference in the results of two notch tests, performed 31 days apart.

The applicant also states that reducing the surveillance frequency for notch testing of fully withdrawn rods would (1) reduce the duty on the Reactor Manual Control System and CRD hardware, and (2) reduce the likelihood of reactivity control errors (e.g., incorrect insertion of control rods), since there would be fewer operator actions. The change in surveillance frequency would result in a substantial reduction in the number of rod movements. For example, TSTF-475 (Ref. 2) indicates that for BWR/4 plants, the number of notch tests per year would decrease from 6,590 to 1,613, and for BWR/6 plants, the number of notch tests per year would decrease from 9,284 to 2,272.

The NRC staff concludes that it would be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly, based on the following reasons:

(1) The accumulation of operating experience, as reviewed by the NRC staff, indicates there have been no immovable control rods identified via performance of rod notch surveillance for either partially or fully withdrawn control rods.

(2) The predicted crack growth rate is slow. The proposed surveillance interval (31 days) remains short enough to be effective in detecting failed CRTs.

The NRC staff finds that increasing the surveillance interval from 7 days to 31 days would not compromise the CRD system's capability to reliably control reactivity changes under normal operations, including anticipated operational occurrences, such that specified acceptable fuel design limits are not exceeded.

The NRC staff notes that General Electric recommends a limited sampling of several CRDs, removed for maintenance, for evidence of discernable corrosion that is different from corrosion that was observed in the past when weekly notching was performed, and an evaluation of CRT maintenance data to assess the actual extent of CRT cracking (Ref.1). The NRC staff's conclusions are not based upon implementation of either of these recommendations. Operational experience, slow crack growth, and potential benefits of reduced control rod movements were sufficient. Therefore, implementation of either or both of these recommendations remains at the discretion of the user.

2.1.1 References

1. Letter BWROG–06036, "Response to NRC Request for Additional Information Regarding TSTF–475, Revision 0," dated November 16, 2006, with enclosure of GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," (report originally prepared in 2004), GE-NE-0000-0024-9858, Revision 3, November 2006 (ADAMS Accession No. ML063250258).

2. Letter TSTF-07-19, Response from the Technical Specifications Task Force to the NRC, "Request for Additional Information (RAI) Regarding TSTF-475 Revision 0, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action," dated February 28, 2007, (TSTF-475, Revision 1 is an enclosure) (ADAMS Accession No. ML071420428).

2.2 Clarify in TS Example 1.4-3 that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to Time Periods discussed in SR Notes

The NRC staff has reviewed the proposal to amend Example 1.4-3 in Section 1.4 "Frequency," to clarify that the 1.25 provision in SR 3.0.2 is equally applicable to time periods specified in the Note in the "Surveillance" column. The NRC staff finds this change acceptable because the revision clarifies the example to make it consistent with the definition of "specified Frequency" provided in the second paragraph of Section 1.4, which states that "the 'specified Frequency' is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The 'specified Frequency' consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements."

3.0 SUMMARY

The NRC staff has reviewed the licensee's proposal to amend LaSalle County Station, Units 1 and 2 TSs to revise the TS surveillance requirement frequency in TS 3.1.3, "Control Rod Operability," and revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The NRC staff has concluded that the TS revisions will have a minimal effect on the reliability of the CRD System while reducing the opportunity for potential reactivity events, and will clarify the applicability of the 1.25 provision in SR 3.0.2. Therefore, the NRC staff concludes that the amendment request is acceptable.

1.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix A, General Design Criterion (GDC) 26 - "Reactivity control system redundancy and capability," states that:

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences [AOOs], and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

GDC 29, "Protection against anticipated operational occurrences," states, "the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences."

Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, were designed and construction was commenced prior to the codification of the current GDCs, thus the current GDCs are not part of the original design basis of the plant. However, the Updated Final Safety Analysis Report for PBAPS Units 2 and 3 contains an evaluation of the design bases of the nuclear facility as measured against the General Design Criteria for Nuclear Power Plant Construction Permits that were proposed to be added to 10 CFR Part 50 as Appendix A in July 1967. The licensee concluded that PBAPS Units 2 and 3 conforms with the intent of the proposed General Design Criteria for Nuclear Power Plants, issued by the Atomic Energy Commission in July 1967.

The design relies on the Control Rod Drive (CRD) System to function in conjunction with the protection systems under AOOs, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRD System provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during AOOs. The CRD System prevents the occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during AOOs. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier.

Section 50.36(c)(3) of 10 CFR states that Technical Specifications (TSs) shall contain Surveillance Requirements (SRs) "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 limiting conditions for operation will be met." As discussed in Section 2.0 of this attachment, revising the SR frequency for notch testing of each fully withdrawn control rod from weekly to monthly, as well as clarifying in a TS example that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to time periods discussed in SR Notes, still assures that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

2.0 TECHNICAL EVALUATION

2.1 Revise the SR Frequency for notch testing of each fully withdrawn control rod from weekly to monthly (TS 3.1.3, "Control Rod OPERABILITY")

The CRD system consists of CRDs, which are hydraulically operated stepping mechanisms mounted in CRD housings, which extend below the reactor vessel bottom head.

The collet retainer tube (CRT) is a short tube welded to the upper end of the CRD, which houses the collet mechanism. The collet mechanism performs the locking and unlocking functions that allow the insertion and withdrawal of the control rod. The latch, or locking collet, is a ratchet device that allows the control rod to be freely inserted but requires a specific unlock signal for rod withdrawal.

Control rod insertion capability is demonstrated by notch testing (i.e., inserting a control rod by at least one notch). Notch testing is currently performed weekly for fully withdrawn control rods and monthly for partially withdrawn control rods. During power operation, most control rods in the core are fully withdrawn, and subjected to notch testing at weekly intervals. Notch testing can also detect a CRT that is totally severed, e.g., from a 360-degree from Intergranular Stress Corrosion Cracking (IGSCC)-initiated crack, and can identify most postulated causes of mechanical binding.

Notch testing is designed to verify the ability to move rods. The ability to scram may be inferred from notch test results; but this is confirmed through scram time testing. Scram time testing can also detect problems in CRD performance resulting from IGSCC-initiated cracks and mechanical binding. Unlike the notch tests, these single rod scram tests cover additional mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod. The Hydraulic Control Units, CRDs, and control rods are also tested during refueling outages, approximately every 18 to 24 months. Based on the data collected during the preceding cycle of operation, selected CRDs are inspected and their internal components are replaced, as required.

In 1975, cracking was observed in some CRTs (Ref. 1). Circumferential cracking could lead to failure of a CRT that would prevent movement of its CRD. Notch testing, which requires movement of CRDs, is used to demonstrate CRT integrity. Since there have been no CRT failures since cracking was first observed in 1975 (Ref. 1) and since the CRT crack growth rate is slow (Ref. 1), the applicant maintains that it would be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly.

IGSCC growth rates were evaluated using General Electric's PLEDGE model (Ref. 1), based on fundamental principles of stress corrosion cracking that can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. This report states that adding 24 days to the surveillance interval could result in an additional 10 mils of growth in total crack length before the next surveillance. The small addition in crack length would not amount to a significant difference in the results of two notch tests, performed 31 days apart.

The applicant also states that reducing the surveillance frequency for notch testing of fully withdrawn rods would (1) reduce the duty on the Reactor Manual Control System and CRD hardware, and (2) reduce the likelihood of reactivity control errors (e.g., incorrect insertion of control rods), since there would be fewer operator actions. The change in surveillance frequency would result in a substantial reduction in the number of rod movements. For example, TSTF-475 (Ref. 2) indicates that for BWR/4 plants, the number of notch tests per year would decrease from 6,590 to 1,613, and for BWR/6 plants, the number of notch tests per year would decrease from 9,284 to 2,272.

The NRC staff concludes that it would be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly, based on the following reasons:

(1) The accumulation of operating experience, as reviewed by the NRC staff, indicates there have been no immovable control rods identified via performance of rod notch surveillance for either partially or fully withdrawn control rods.

(2) The predicted crack growth rate is slow. The proposed surveillance interval (31 days) remains short enough to be effective in detecting failed CRTs.

The NRC staff finds that increasing the surveillance interval from 7 days to 31 days would not compromise the CRD system's capability to reliably control reactivity changes under normal operations, including anticipated operational occurrences, such that specified acceptable fuel design limits are not exceeded.

The NRC staff notes that General Electric recommends a limited sampling of several CRDs, removed for maintenance, for evidence of discernable corrosion that is different from corrosion that was observed in the past when weekly notching was performed, and an evaluation of CRT maintenance data to assess the actual extent of CRT cracking (Ref.1). The NRC staff's conclusions are not based upon implementation of either of these recommendations. Operational experience, slow crack growth, and potential benefits of reduced control rod movements were sufficient. Therefore, implementation of either or both of these recommendations remains at the discretion of the user.

2.1.1 References

1. Letter BWROG–06036, "Response to NRC Request for Additional Information Regarding TSTF–475, Revision 0," dated November 16, 2006, with enclosure of GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," (report originally prepared in 2004), GE-NE-0000-0024-9858, Revision 3, November 2006 (ADAMS Accession No. ML063250258).

2. Letter TSTF-07-19, Response from the Technical Specifications Task Force to the NRC, "Request for Additional Information (RAI) Regarding TSTF-475 Revision 0, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action," dated February 28, 2007, (TSTF-475, Revision 1 is an enclosure) (ADAMS Accession No. ML071420428).

2.2 Clarify in TS Example 1.4-3 that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to Time Periods discussed in SR Notes

The NRC staff has reviewed the proposal to amend Example 1.4-3 in Section 1.4, "Frequency," to clarify that the 1.25 provision in SR 3.0.2 is equally applicable to time periods specified in the Note in the "Surveillance" column. The NRC staff finds this change acceptable because the revision clarifies the example to make it consistent with the definition of "specified Frequency" provided in the second paragraph of Section 1.4, which states that "the 'specified Frequency' is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The 'specified Frequency' consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements."

3.0 SUMMARY

The NRC staff has reviewed the licensee's proposal to amend Peach Bottom Atomic Power Station Units 2 and 3 TSs to revise the TS surveillance requirement frequency in TS 3.1.3, "Control Rod Operability," and revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The NRC staff has concluded that the TS revisions will have a minimal effect on the reliability of the CRD System while reducing the opportunity for potential reactivity events, and will clarify the applicability of the 1.25 provision in SR 3.0.2. Therefore, the NRC staff concludes that the amendment request is acceptable.

REGULATORY AND TECHNICAL EVALUATIONS

QUAD CITIES NUCLEAR POWER STATION UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

1.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix A, General Design Criterion (GDC) 26 - "Reactivity control system redundancy and capability," states that:

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences [AOOs], and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

GDC 29, "Protection against anticipated operational occurrences," states, "the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences."

Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, were designed and construction was commenced prior to the codification of the current GDCs, thus the current GDCs are not part of the original design basis of the plant. However, the Updated Final Safety Analysis Report for QCNPS Units 1 and 2 contains an evaluation of the design bases of the nuclear facility as measured against the General Design Criteria for Nuclear Power Plant Construction Permits that were proposed to be added to 10 CFR Part 50 as Appendix A in July 1967. The licensee concluded that QCNPS Units 1 and 2 satisfied the intent of the proposed General Design Criteria for Nuclear Power Plants issued by the Atomic Energy Commission in July 1967.

The design relies on the Control Rod Drive (CRD) System to function in conjunction with the protection systems under AOOs, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRD System provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during AOOs. Compliance with GDCs 26 and 29 for the CRD System prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during AOOs. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier.

Section 50.36(c)(3) of 10 CFR states that Technical Specifications (TSs) shall contain Surveillance Requirements (SRs) "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 limiting conditions for operation will be met." As discussed in Section 2.0 of this attachment, revising the SR frequency for notch testing of each fully withdrawn control rod from weekly to monthly, as well as clarifying in a TS example that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to time periods discussed in SR Notes, still assures that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

2.0 TECHNICAL EVALUATION

2.1 Revise the SR Frequency for notch testing of each fully withdrawn control rod from weekly to monthly (TS 3.1.3, "Control Rod OPERABILITY")

The CRD system consists of CRDs, which are hydraulically operated stepping mechanisms mounted in CRD housings, which extend below the reactor vessel bottom head.

The collet retainer tube (CRT) is a short tube welded to the upper end of the CRD, which houses the collet mechanism. The collet mechanism performs the locking and unlocking functions that allow the insertion and withdrawal of the control rod. The latch, or locking collet, is a ratchet device that allows the control rod to be freely inserted but requires a specific unlock signal for rod withdrawal.

Control rod insertion capability is demonstrated by notch testing (i.e., inserting a control rod by at least one notch). Notch testing is currently performed weekly for fully withdrawn control rods and monthly for partially withdrawn control rods. During power operation, most control rods in the core are fully withdrawn, and subjected to notch testing at weekly intervals. Notch testing can also detect a CRT that is totally severed, e.g., from a 360-degree from Intergranular Stress Corrosion Cracking (IGSCC)-initiated crack, and can identify most postulated causes of mechanical binding.

Notch testing is designed to verify the ability to move rods. The ability to scram may be inferred from notch test results; but this is confirmed through scram time testing. Scram time testing can also detect problems in CRD performance resulting from IGSCC-initiated cracks and mechanical binding. Unlike the notch tests, these single rod scram tests cover additional mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod. The Hydraulic Control Units, CRDs, and control rods are also tested during refueling outages, approximately every 18 to 24 months. Based on the data collected during the preceding cycle of operation, selected CRDs are inspected and their internal components are replaced, as required.

In 1975, cracking was observed in some CRTs (Ref. 1). Circumferential cracking could lead to failure of a CRT that would prevent movement of its CRD. Notch testing, which requires movement of CRDs, is used to demonstrate CRT integrity. Since there have been no CRT failures since cracking was first observed in 1975 (Ref. 1) and since the CRT crack growth rate is slow (Ref. 1), the applicant maintains that it would be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly.

IGSCC growth rates were evaluated using General Electric's PLEDGE model (Ref. 1), based on fundamental principles of stress corrosion cracking that can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. This report states that adding 24 days to the surveillance interval could result in an additional 10 mils of growth in total crack length before the next surveillance. The small addition in crack length would not amount to a significant difference in the results of two notch tests, performed 31 days apart.

The applicant also states that reducing the surveillance frequency for notch testing of fully withdrawn rods would (1) reduce the duty on the Reactor Manual Control System and CRD hardware, and (2) reduce the likelihood of reactivity control errors (e.g., incorrect insertion of control rods), since there would be fewer operator actions. The change in surveillance frequency would result in a substantial reduction in the number of rod movements. For example, TSTF-475 (Ref. 2) indicates that for BWR/4 plants, the number of notch tests per year would decrease from 6,590 to 1,613, and for BWR/6 plants, the number of notch tests per year would decrease from 9,284 to 2,272.

The NRC staff concludes that it would be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly, based on the following reasons:

(1) The accumulation of operating experience, as reviewed by the NRC staff, indicates there have been no immovable control rods identified via performance of rod notch surveillance for either partially or fully withdrawn control rods.

(2) The predicted crack growth rate is slow. The proposed surveillance interval (31 days) remains short enough to be effective in detecting failed CRTs.

The NRC staff finds that increasing the surveillance interval from 7 days to 31 days would not compromise the CRD system's capability to reliably control reactivity changes under normal operations, including AOOs, such that specified acceptable fuel design limits are not exceeded.

The NRC staff notes that General Electric recommends a limited sampling of several CRDs, removed for maintenance, for evidence of discernable corrosion that is different from corrosion that was observed in the past when weekly notching was performed, and an evaluation of CRT maintenance data to assess the actual extent of CRT cracking (Ref.1). The NRC staff's conclusions are not based upon implementation of either of these recommendations. Operational experience, slow crack growth, and potential benefits of reduced control rod movements were sufficient. Therefore, implementation of either or both of these recommendations remains at the discretion of the user.

2.1.1 References

1. Letter BWROG–06036, "Response to NRC Request for Additional Information Regarding TSTF–475, Revision 0," dated November 16, 2006, with enclosure of GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," (report originally prepared in 2004), GE-NE-0000-0024-9858, Revision 3, November 2006 (ADAMS Accession No. ML063250258).

2. Letter TSTF-07-19, Response from the Technical Specifications Task Force to the NRC, "Request for Additional Information (RAI) Regarding TSTF-475 Revision 0, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action," dated February 28, 2007, (TSTF-475, Revision 1 is an enclosure) (ADAMS Accession No. ML071420428).

2.2 Clarify in TS Example 1.4-3 that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to Time Periods discussed in SR Notes

The NRC staff has reviewed the proposal to amend Example 1.4-3 in Section 1.4, "Frequency," to clarify that the 1.25 provision in SR 3.0.2 is equally applicable to time periods specified in the Note in the "Surveillance" column. The NRC staff finds this change acceptable because the revision clarifies the example to make it consistent with the definition of "specified Frequency" provided in the second paragraph of Section 1.4, which states that "the 'specified Frequency' is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The 'specified Frequency' consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements."

3.0 <u>SUMMARY</u>

The NRC staff has reviewed the licensee's proposal to amend Quad Cities Nuclear Power Station, Units 1 and 2 TSs to revise the TS surveillance requirement frequency in TS 3.1.3, "Control Rod Operability," and revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The NRC staff has concluded that the TS revisions will have a minimal effect on the reliability of the CRD System while reducing the opportunity for potential reactivity events, and will clarify the applicability of the 1.25 provision in SR 3.0.2. Therefore, the NRC staff concludes that the amendment request is acceptable.

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

Sincerely,

/RA/
Christopher Gratton, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-461, 50-237, 50-249, 50-373, 50-374, 50-277, 50-278, 50-254, and 50-265

Enclosures:
1. Amendment No. 188 to DPR-62
2. Amendment No. 232 to DPR-19
3. Amendment No. 225 to DPR-25
4. Amendment No.193 to NPF-11
5. Amendment No. 180 to NPF-18
6. Amendment No. 272 to DPR-44
7. Amendment No. 276 to DPR-56
8. Amendment No. 244 to DPR-29
9. Amendment No. 239 to DPR-30
10. Safety Evaluation

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ADAMS Accession No. ML091270818 NRR-058

OFFICE	LPL3-2/PM	LPL3-2/LA	ITSB/BC	SRXB/BC	OGC (NLO)	LPL1-2/BC	LPL3-2/BC
NAME	CGratton	THarris	RElliott	GCranston	DRoth	HChernoff	SCampbell
DATE	07/28/09	07/28/09	07/29/09	07/28/09	08/03/09	8/6/09	8/11/09

OFFICIAL AGENCY RECORD