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

October 4, 2011

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS REGARDING REVISION TO TECHNICAL SPECIFICATION TO ELIMINATE MAIN STEAM ISOLATION VALVE CLOSURE AND LOW CONDENSER VACUUM SCRAM FUNCTIONS DURING STARTUP MODE (TAC NOS. ME4844 AND ME4845)

Dear Mr. Pacilio:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 239 to Renewed Facility Operating License No. DPR-19 and Amendment No. 232 to Renewed Facility Operating License No. DPR-25 for Dresden Nuclear Power Station, Units 2 and 3. The amendments are in response to an application submitted by Exelon Generating Company, LLC (EGC, the licensee) dated October 4, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102800136), as supplemented by letter dated April 6, 2011 (ADAMS Accession No. ML110960658).

The amendment revises the applicability of Technical Specification 3.3.1.1, "Reactor Protection System Instrumentation" Function 5 ("Main Steam Isolation valve – Closure") and Function 10 ("Turbine Condenser Vacuum – Low") for Dresden Nuclear Power Station Units 2 and 3. The change removes the requirement to have these functions available in Mode 2 with reactor pressure greater than or equal to 600 pounds per square inch (psig).

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,

Joel S. Wiebe

Joel S. Wiebe, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-237 and 50-249

Enclosures:
1. Amendment No. 239 to DPR-19
2. Amendment No. 232 to DPR-25
3. Safety Evaluation

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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. 239
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 October 4, 2010 as supplemented by letter dated April 6, 2011, 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. 239, 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 during the next refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Jacob I. Zimmerman, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Renewed Facility Operating License

Date of Issuance: October 4, 2011

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.232
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 October 4, 2010 as supplemented by letter dated April 6, 2011, 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. 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 during the next refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Jacob I. Zimmerman

Jacob I. Zimmerman, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Renewed Facility Operating License

Date of Issuance: October 4, 2011

ATTACHMENT TO LICENSE AMENDMENT NOS. 239 AND 232

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 License and Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Remove	Insert
License DRP-19 Page 3	License DPR-19 Page 3
License DPR-25 Page 4	License DPR-25 Page 4
TSs 3.3.1.1-2 3.3.1.1-10 3.3.1.1-11	TSs 3.3.1.1-2 3.3.1.1-10 3.3.1.1-11

(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) <u>Maximum Power Level</u>

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) <u>Technical Specifications</u>

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.

(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. 232, 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.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 38.5% RTP	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	8 hours

(continued)

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux-High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 122% RTP
d. Inop	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.18	NA
3. Reactor Vessel Steam Dome Pressure—High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 1045 psig
4. Reactor Vessel Water Level—Low	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≥ 2.65 inches
5. Main Steam Isolation Valve—Closure	1	8	F	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 9.5% closed
6. Drywell Pressure—High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 1.94 psig

(continued)

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level—High					
a. Thermal Switch (Unit 2) Level Indicating Switch (Unit 3)	1,2	2	G	SR 3.3.1.1.1[c] SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12[c] SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 38.7 gallons (Unit 3)
	5[a]	2	H	SR 3.3.1.1.1[c] SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12[c] SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 38.7 gallons (Unit 3)
b. Differential Pressure Switch	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 38.7 gallons (Unit 3)
	5[a]	2	H	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 38.7 gallons (Unit 3)
8. Turbine Stop Valve-Closure	≥ 38.5% RTP	4	E	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 9.5% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low	≥ 38.5% RTP	2	E	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≥ 466 psig
10. Turbine Condenser Vacuum—Low	1	2	F	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	≥ 20.5 inches Hg vacuum
11. Reactor Mode Switch—Shutdown Position	1,2	1	G	SR 3.3.1.1.16 SR 3.3.1.1.18	NA
	5[a]	1	H	SR 3.3.1.1.16 SR 3.3.1.1.18	NA
12. Manual Scram	1,2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.18	NA
	5[a]	1	H	SR 3.3.1.1.8 SR 3.3.1.1.18	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.
(c) Specified SR performance only required for Unit 3.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED

TO AMENDMENT NO. 239 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 232 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-25

EXELON GENERATION COMPANY, LLC

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated October 4, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102800136) as supplemented by letter dated April 6, 2011 (ADAMS Accession No. ML110960658), Exelon Generation Company, LLC (EGC, the licensee) submitted a license amendment request (LAR) to revise the applicability of Technical Specification (TS) 3.3.1.1, "Reactor Protection System Instrumentation" Function 5 ("Main Steam Isolation Valve (MSIV) - Closure") and Function 10 ("Turbine Condenser Vacuum - Low") for the Dresden Nuclear Power Station (DNPS), Units 2 and 3. The proposed change removes the requirement to have these functions available in Mode 2 with reactor pressure is greater than or equal to 600 pounds per square inch (psig).

The April 6, 2011, supplement, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

2.0 BACKGROUND

Currently DNPS TSs requires a reactor scram if the reactor vessel dome pressure is greater than or equal to 600 psig in Mode 2 and the MSIVs closed, or with a low turbine condenser vacuum condition. This requirement dates back to the startup of an earlier General Electric Hitachi Nuclear Energy (GEH) dual-cycle Boiling Water Reactor (BWR) in 1966 when operators found it difficult to control power above 600 psig without pressure control. Experience on plant startups with later BWR designs indicated that this early experience was not inherent to the BWR design and therefore, in order to demonstrate this, a test was conducted at the Browns Ferry Nuclear Plant (BF) plant in 1974 to assess the susceptibility of this later BWR design to this same pressure and power oscillation. The operating condition under which this is a concern is referred to as "bottled-up" startup operations, (startup with MSIVs closed or pressure control otherwise unavailable).

The conclusions from the BF test were (1) that the later BWR design could be controlled in a bottled-up condition at reactor vessel dome pressures well in excess of 600 psig and (2) that there was no reason for this to be an unacceptable operating region and require a reactor scram when vessel dome pressure exceeds 600 psig with the MSIVs closed. It was further concluded that the results of the BF test could be extended to cover all BWR/4 plants, but that the test results would not be generically applied to BWR/1, 2, or 3 plants without a test or engineering analysis.

Given the results of the BF test and to provide additional operating flexibility, GEH recommended that the bottled-up startup operation should be considered as an option to (but not as a replacement for) the more conventional pressure and steam flow heat-up with MSIVs open and under pressure control mode. The licensee requested that GEH review the requirements for DNPS Units 2 and 3 (both BWR/3 plants) to maintain such a scram function. GEH's review concluded that the requirement to establish pressure control prior to exceeding 600 psig reactor vessel dome pressure could be eliminated for the DNPS units.

By letter dated December 20, 2002 (ADAMS Accession Nos. ML023610228 and ML023610236), the licensee submitted a LAR for a TS change for DNPS to increase the setpoint vessel dome pressure in the STARTUP mode for the scram on MSIV closure. In response to the licensee's TS change request, the NRC staff required additional quantifiable information to demonstrate that DNPS would not be subject to the phenomena of concern. The licensee provided the NRC staff requested information in the form of comparing plant parameters relevant to the phenomena between the DNPS units and BF and between the DNPS units and the early dual-cycle BWR that experienced the problem behavior. The NRC staff, however, indicated that the licensee's response was inadequate, and that a detailed technical analysis was required to support applicability of the BF test to DNPS. By letter dated October 1, 2003 (ADAMS Accession No. ML032810614), the licensee withdrew their original LAR to eliminate MSIV closure and low condenser vacuum scram functions in Mode 2.

Subsequently, in their letter dated October 4, 2010, the licensee submitted "Dresden Units 2 and 3 TRACG Analysis to Support Elimination of Mode 2 Scram Requirement," a detailed technical analysis report using General Electric TRACG computer code to demonstrate the applicability of the BF test to the DNPS. In addition, the analysis examined a range of operating conditions outside the range of the test conditions, but still within the range of concern to demonstrate that the DNPS plant design is not susceptible to the phenomena experienced by the dual-cycle early BWR design plant.

3.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications," requires that the facility's TS include a section addressing limiting conditions for operation (LCO). In accordance with 10 CFR 50.36(c)(2)(ii), the LCO of a nuclear reactor must be established for each item meeting one or more the specified criteria. One of these criteria is Criterion 3 which requires an LCO for a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The reactor protection system is designed to initiate a reactor scram when one or more monitored parameters exceed their specified limits to preserve the integrity of the fuel cladding and the reactor coolant pressure boundary (fission product barriers) and minimize the energy that must be absorbed following an accident. The proposed change eliminates the requirement for a scram with the reactor switch in STARTUP when the MSIVs are closed or the condenser vacuum is low with vessel pressure greater than or equal to 600 psig.

The specific proposed changes are:

- TS Table 3.3.1.1-1, Function 5 and Function 10, Column 2, "Applicable Modes or Other Specified Conditions," will be revised to remove the term "2(c)."
- TS Table 3.3.1.1-1 will be revised to remove footnote (c) "With reactor pressure >600 psig," on pages 3.3.1.1-10 and 3.3.1-11.
- TS Table 3.3.1.1-1, Function 7.a, "Scram Discharge Volume Water Level – High Thermal Switch (Unit 2) Level Indicating Switch (Unit 3)," Column 5, "Surveillance Requirements," will be revised to change the footnote designation from (d) to (c).
- TS Table 3.3.1.1-1 will be revised to rename footnote (d) "Specified SR performance only required for Unit 3," to (c).
- TS 3.3.1.1, Required Action F will be revised to delete Required Action F.2.

The proposed amendment of the DNPS TS was based on safety analysis performed using NRC-approved TRACG computer code. The results of TRACG analysis demonstrated that the proposed TS amendment to eliminate MSIV closure and low condenser vacuum scram functions requirements during startup mode is not part of the primary success path to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier, and therefore, the amendment is consistent with 10 CFR 50.36 requirements. The NRC staff finds the proposed request to amend the TS is in conformance with the regulatory requirement.

4.0 TECHNICAL EVALUATION

4.1 Methodology

The NRC approved computer code TRACG04A version was used to perform the analysis. The application of this code to perform the analysis in support of elimination of MSIV closure and low condenser vacuum scram functions during Mode 2 (also known as "bottled-up" startup) is within the accepted code application capabilities (Reference 2).

4.2 Inputs and Assumptions

The TRACG base deck for Dresden-3 Cycle 18 turbine trip analysis (Reference 1) was used as the starting point for developing the TRACG model for this analysis. The TRACG DNPS model of Reference 3 was used to perform benchmarks to two turbine trip events that occurred at DNPS in January 2004, and the analyses demonstrated that the code provides realistic simulation of the plant behavior. Hence, there is reasonable assurance that the TRACG model used in this analysis predicted a realistic simulation of DNPS.

The channel grouping established in Reference 1 was used in this analysis. Therefore, the channel grouping from the original analysis is considered adequate to capture the neutronics feedback important to this analysis. This calculation is independent of steam dryer modification due to the very low steaming rate and pressure drop associated with a "bottled up" startup. The calculation is also independent of the installation of digital Electro Hydraulic Controls because pressure control is not credited in these calculations.

Assumption 1:

The BF test was conducted during a plant startup in which the reactor coolant is being heated up as power is increased; thus, the reactor power is primarily being used for the sensible heat addition to the coolant (and vessel/piping metal) as opposed to steam production. Steam production is assumed insignificant because the turbines are isolated from the vessel and gland seal steam and the steam jet air ejectors are being supplied from the auxiliary boiler. The reactor coolant is heating up and is expanding. Level control is accomplished by inventory letdown from the reactor water cleanup system. Any coolant addition is a result of leakage from the control rod drive system. The amount of leakage is small such that even if the additional water is introduced at a very low temperature, the amount of core inlet subcooling is small as well.

To achieve the desired low power initial condition, the DNPS TRACG model from a previous transient test benchmark at nearly full power was maneuvered to the low power Mode 2 condition. As a result, the coolant was already heated up and the reactor power was being used for latent heat addition to the coolant, thereby, generating steam. Since the reactor power was low (approximately 1 to 2 percent), the amount of steam production was small, but nonetheless, coolant makeup was required in order to maintain level. To provide this makeup water, the feedwater fill junction was used in the model with the fill enthalpy controlled to maintain nearly saturated conditions at the core inlet. This fill path effectively simulated the injection of makeup water from the reactor water cleanup system that would be introduced into the feedwater line in the real plant line-up test conditions. The larger magnitude of steam production compared to the BF test also required that a path exist for the steam to be discharged from the model. The discharge path used in the model was to maintain a small fixed demand on the turbine bypass valves. This required that the MSIVs remain open in this analysis, which was consistent with the test configuration for the pressure perturbation test, but not for the reactivity perturbation test. These differences in initial conditions between the BF test and the DNPS TRACG model are believed to have insignificant impact on the comparison effort.

Assumption 2:

Control rods were grouped within the TRACG model using a DNPS 3 Cycle 18 control rod map that reflects the actual core design conditions for initial criticality as a starting guideline. The choice of ten control rod groups in the model provided the needed flexibility to achieve the desired initial conditions, namely, just critical in Mode 2 at very low power and nearly saturated core inlet conditions. Control rod group 10 corresponds to a single control rod at position 34-27 for the control rod used for the reactivity perturbation. This control rod is in a similar central location in the core as that used in the BF test for the reactivity perturbation. The magnitude of reactivity worth of the test control rod was considered in the analysis by performing a sensitivity study.

4.3 Limitations and Conditions

The NRC staff approved the application of TRACG04 computer code for operating BWR plant designs (BWR/2-6) over the current range of plant operating conditions, including extended power uprate and maximum extended load line limit analysis plus (MELLA+) operating domains (ADAMS Accession No. ML091520088). The NRC staff found that the methodology is acceptable when exercised within a set of limitations and conditions provided in the staff safety evaluation (ADAMS Accession No. ML091751102). The specific condition that is relevant to the current application of TRACG04 code for DNPS is the Condition # 4.33 (Submittal Requirements Condition) which states that a generic licensing topical report describing a code such as TRACG cannot provide full justification for each specific individual plant application. When a licensee proposes to reference the TRACG methodology for use in a license amendment, the individual licensee or applicant must provide justification for the specific application of the code in its request which is expected to include:

- Nodalization: Specific guidelines used to develop the plant-specific nodalization. Deviations from the reference plant must be described and defended.

- Chosen Parameters and Conservative Nature of Input Parameters: A table that contains the plant-specific parameters and the range of the values considered for the selected parameter during the topical approval process. When plant-specific parameters are outside the range used in demonstrating acceptable code performance, the licensee or applicant will submit sensitivity studies to show the effects of that deviation.

- Calculated Resuts: The licensee or applicant using the approved methodology must submit the results of the plant-specific analyses reactor vessel peak pressure.

In the April 6, 2011, submittal, the licensee stated that the specific guidelines used to develop the plant-specific nodalization for the current TRACG application to support elimination of the Mode 2 scram requirement were the same as those applied when performing plant-specific anticipated operational occurrence (AOO) transient applications. The NRC staff determined that this nodalization is acceptable because the TRACG code has already been approved by NRC staff to perform AOO transient analyses for BWRs. All TRACG applications utilize guidelines established by applying standard BWR nodalization strategies for each TRACG application as an integral part of the qualification of the code.

The modifications to the base deck were limited to those inputs needed to define the thermal-hydraulic initial conditions. The licensee's submittal also provided information confirming that none of the inputs for the plant-specific parameters are outside the range used in demonstrating acceptable code performance for AOO transient applications, and that the key parameters for both application types, i.e., AOO and Start-up mode, cover similar ranges of the parameters. The NRC staff determined that the input parameters are acceptable because they are within the range used in demonstrating acceptable code performance.

The extension of the qualification basis to include the specific application of TRACG to support elimination of the Mode 2 scram requirement was demonstrated by comparison of the calculated results to relevant Browns Ferry transient test data. The NRC staff, therefore, considers the calculated results to acceptable because the results were submitted and compared to relevant BF transient test data.

Based on the above, the NRC staff determined that the limitations and conditions for the use of the TRACGO4 code for this application is acceptable.

4.4 Results

Steady-State Case:

The initial steady-state run was made to establish the near full power conditions from which the model was "maneuvered" to reduce power in order to arrive at the desired conditions for the analysis. The second steady-state condition of interest was at the Mode 2 startup condition. This condition was achieved by performing a downpower maneuver using a combination of rod insertion and recirculation pump speed reduction to achieve nearly shutdown conditions. Once the power was reduced in this manner, the feedwater enthalpy was controlled to minimize the core inlet subcooling and the turbine valves were closed so that the steam flow could be controlled via the turbine bypass valve demand position in the TRACG model. The TRACG model parameters at the end of the downpower transient indicated the critical condition of the reactor.

Transient Base Case – Comparison to BF Test:

The first transient simulates the BF pressure perturbation test. There were two different pressure perturbation tests conducted at BF. The first test opened one bypass valve rapidly (over about 0.1 sec), held it open for about 15 seconds, and then closed it rapidly (over about 0.1 sec). The second pressure perturbation test slowly opened and closed two bypass valves. The purpose of performing the second test was to approximate the opening of one relief valve since one fully open bypass valve has a capacity of approximately 400,000 pound-mass per hour (lbm/hr) and the capacity of one relief valve is approximately 800,000 lbm/hr. The test report stated that the rapid opening of the single bypass valve pressure perturbation test was the one of primary interest, and therefore, it was the one simulated in this analysis.

The results from the TRACG analysis for DNPS were compared against the BF test data. The results demonstrated that the TRACG model provided a reasonable, realistic response and compared well with the actual plant response. The overall trend in the parameters was similar, and differences were similar in magnitude of the pressure and level response as well as the timing of peaks and valleys. The comparisons were reasonable given the differences in plant designs between the computer model and the test plant that might contribute to differences in the plant response.

Low Pressure Analysis:

Three cases at different operating conditions were subjected to the pressure perturbation test.

The first case is at an initial vessel dome pressure of approximately 750 pounds-force per square inch absolute (psia) with a core flow and recirculation pump flow of roughly 26 percent and 17 percent of rated flows, respectively. The second case is at an initial vessel dome pressure of approximately 600 psia with a core flow and recirculation pump flow of roughly 26 percent and 17 percent of rated flows, respectively. The third case is at an initial vessel dome pressure of approximately 650 psia with a core flow and recirculation pump flow of roughly 37 percent and 28 percent of rated flows, respectively.

The system response was driven by a large number of interdependent factors. The results of the analysis for DNPS were similar to that of BF test data, and showed that the overall response of the model continued to demonstrate that the system is stable and not susceptible to the pressure-power oscillations experienced by the older design dual-cycle BWR plant.

Pressure-Induced Reactivity Analysis:

This section of the analysis was intended to evaluate the system response to a rapid pressure increase caused by the fast closure of the bypass valves. It was referred to as the pressure induced reactivity analysis because the increase in pressure will insert positive reactivity into the core due to the collapse of voids in the core region. The closure of the bypass valves results in a rapid increase in pressure and a subsequent collapse of voids in the core. The positive reactivity added by the reduced voiding causes the power to increase rapidly and thereby mitigates the drop in core voids. No power oscillations developed from the model's response to significant pressure perturbation.

Rod-Induced Reactivity Analysis:

Up to this point, the perturbations imposed on the model have been initiated by pressure disturbances using the turbine bypass valves. This section of the analysis examined the system response to perturbations initiated by reactivity changes due to rod movements. The first rod induced reactivity cases analyzed are similar to the reactivity test performed at BF. The second rod-induced reactivity cases analyzed were intended to challenge the system more than the first cases. As expected, these reactivity changes are seen to have significant effects on the power, vessel dome pressure, level, and so on. However, the results showed that the system response was well-behaved in both cases and no signs of power oscillations in the system were apparent. In both cases, the system response to the slow withdrawal of the control rods was well behaved with no signs of any pressure and power oscillations.

Steam Volume Analysis:

These cases were intended to evaluate the impact of operating in a bottled-up condition by closing the MSIVs so that the steam volume is reduced and the system response to pressure changes will be amplified. The sudden closure of the MSIVs results in a rapid decrease in void fraction and subsequent void reactivity addition. The simulated system responded as expected.

Overall, the simulated system response was very well-behaved and again exhibited no tendency to develop power oscillations as shown by the results for the longer-term responses.

The NRC staff reviewed the licensee's submittal, supplemental information provided in response to the staff's request for additional information, related documentation (e.g., TS, UFSAR, service information letter (SIL) 107, past correspondence between the licensee and the NRC staff on the subject), and the TRACG analysis. The licensee performed detail technical analysis using NRC-approved TRACG computer code to justify elimination of MSIV closure and low condenser vacuum scram functions during startup mode (Mode 2) of DNPS. The NRC staff concluded that the results of the analysis demonstrated that significant pressure-power oscillation did not exist for a "bottled-up" startup operation at DNPS.

The NRC staff finds the proposed license amendment request is acceptable based on the following: (1) the licensee's technical analysis was based on NRC staff approved TRACG methodology; (2) the analysis predicted that significant pressure-power oscillation did not exist; and even if any significant oscillation is to occur, it would result in a high pressure or high flux scram to terminate the event. Because of low power level during startup Mode 2, the resulting event is bounded by the limiting design-basis event analyzed for run Mode 1 at rated power level; and (3) removal of the MSIV closure and low condenser vacuum scram requirement would eliminate unnecessary scrams during Mode 2 as the systems are being brought online at higher pressures.

4.5 Conclusion

Based on the above, the NRC staff determined that without a requirement to have a reactor scram when the reactor vessel dome pressure exceeds 600 psig in Mode 2 and the MSIVs closed, or low condenser vacuum, the applicable regulatory requirements of 10 CFR 50.36 will continue to be met because the analysis demonstrated that significant pressure-power oscillation did not exist for a "bottled-up" startup operation at DNPS. In addition, the NRC staff concluded that adequate defense-in-depth will be maintained and sufficient safety margins will be maintained because even if any significant oscillation is to occur, it would result in a high pressure or high flux scram to terminate the event. As a result, the NRC staff determined that the scram functions that are proposed to be removed are not part of the primary success path to mitigate a design basis accident or transient. The NRC staff, therefore, concludes that the proposed change is acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

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 (76 FR 5619, dated February 1, 2011). 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.

7.0 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 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.

8.0 REFERENCES

1. GE-NE-0000-0030-9614-R1-P Revision 1, "Dresden Unit 3 TRACG Modeling for Transient HPCI Steam Line Flooding Evaluations", August 2005

2. NEDE-32176P Revision 4, "Licensing Topical Report TRACG Model Description", January 2008.

Principal Contributor: M. Razzaque

Date of issuance: October 4, 2011

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/

Joel S. Wiebe, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-237 and 50-249

Enclosures:
1. Amendment No. 239 to DPR-19
2. Amendment No. 232 to DPR-25
3. Safety Evaluation

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