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

October 20, 2009

Mr. Charles G. Pardee
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 (TAC NOS. MD7202 AND MD7203)

Dear Mr. Pardee:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 233 to Renewed Facility Operating License No. DPR-19 and Amendment No. 226 to Renewed Facility Operating License No. DPR-25 for Dresden Nuclear Power Station, Units 2 and 3. The amendments are in response to your application dated October 9, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072830088), as supplemented by letter dated January 30, 2009 (ADAMS Accession No. ML090350151).

The amendments would modify the technical specifications to risk-informed requirements regarding selected required action end states as provided in Technical Specification Task Force (TSTF) change traveler TSTF-423, Revision 0, "Technical Specifications End States, NEDC-32988-A, Revision 2."

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. 50-237 and 50-249

Enclosures:
1. Amendment No. 233 to DPR-19
2. Amendment No. 226 to DPR-25
3. Safety Evaluation

cc w/encls: Distribution via Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 233
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 9, 2007, as supplemented by letter dated January 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. 233 , 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 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

FOR

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 Renewed Facility Operating License

Date of Issuance: October 20, 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. 226
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 9, 2007, as supplemented by letter dated January 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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 226 , 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 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Marshall [illegible] FOR

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 Renewed Facility Operating License

Date of Issuance: October 20, 2009

ATTACHMENT TO LICENSE AMENDMENT NOS. 233 AND 226

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	License DPR-19
Page 3	Page 3
License DPR-25	License DPR-25
Page 4	Page 4
TSs	TSs
3.4.3-1	3.4.3-1
3.5.1-1	3.5.1-1
3.5.1-2	3.5.1-2
3.5.1-3	3.5.1-3
3.5.3-1	3.5.3-1
3.6.1.6-1	3.6.1.6-1
3.6.1.7-2	3.6.1.7-2
3.6.1.7-3	3.6.1.7-3
3.6.1.8-1	3.6.1.8-1
3.6.2.3-1	3.6.2.3-1
3.6.2.4-1	3.6.2.4-1
3.6.4.1-1	3.6.4.1-1
3.6.4.3-1	3.6.4.3-1
3.6.4.3-2	3.6.4.3-2
3.7.1-1	3.7.1-1
3.7.4-1	3.7.4-1
3.7.4-2	3.7.4-2
3.7.5-1	3.7.5-1
3.7.6-1	3.7.6-1
3.8.1-5	3.8.1-5
3.8.4-4	3.8.4-4
3.8.7-2	3.8.7-2

(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. 233, 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. <u>Maximum Power Level</u>

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. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 226, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. <u>Reports</u>

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

D. <u>Records</u>

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

E. <u>Restrictions</u>

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety and Relief Valves

LCO 3.4.3 The safety function of 9 safety valves shall be OPERABLE.

AND

The relief function of 5 relief valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One relief valve inoperable.	A.1 Restore the relief valve to OPERABLE status.	14 days
B. Required Action and associated Completion Time of Condition A not met.	------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 3. ---------------------------- B.1 Be in MODE 3.	12 hours
C. Two or more relief valves inoperable. OR One or more safety valves inoperable.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND ISOLATION CONDENSER (IC) SYSTEM

3.5.1 ECCS—Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of five relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

-- NOTE --

LCO 3.0.4.b is not applicable to HPCI.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Low Pressure Coolant Injection (LPCI) pump inoperable.	A.1 Restore LPCI pump to OPERABLE status.	30 days
B. One LPCI subsystem inoperable for reasons other than Condition A. OR One Core Spray subsystem inoperable.	B.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
C. One LPCI pump in each subsystem inoperable.	C.1 Restore one LPCI pump to OPERABLE status.	7 days

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. --------------------------- D.1 Be in MODE 3.	12 hours
E. Two LPCI subsystems inoperable for reasons other than Condition C.	E.1 Restore one LPCI subsystem to OPERABLE status.	72 hours
F. Required Action and associated Completion Time of Condition E not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4.	12 hours 36 hours
G. HPCI System inoperable.	G.1 Verify by administrative means IC System is OPERABLE. <u>AND</u> G.2 Restore HPCI System to OPERABLE status.	Immediately 14 days
H. One ADS valve inoperable.	H.1 Restore ADS valve to OPERABLE status.	14 days
I. Required Action and associated Completion Time of Condition G or H not met.	------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. --------------------------- I.1 Be in Mode 3.	12 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
J. Two or more ADS valves inoperable.	J.1 Be in Mode 3. AND J.2 Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours
K. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition C or E. OR HPCI System and one or more ADS valves inoperable. OR One or more low pressure ECCS injection/spray subsystems inoperable and one or more ADS valves inoperable. OR HPCI System inoperable and either one low pressure ECS injection/spray subsystem is inoperable or Condition C entered.	K.1 Enter LCO 3.0.3	Immediately

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND ISOLATION CONDENSER (IC) SYSTEM

3.5.3 IC System

LCO 3.5.3 The IC System shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

------------------------------------NOTE------------------------------------

LCO 3.0.4.b is not applicable to IC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. IC System inoperable.	A.1 Verify by administrative means High Pressure Coolant Injection System is OPERABLE.	Immediately
	AND	
	A.2 Restore IC System to OPERABLE status.	14 days
B. Required Action and associated Completion Time not met.	------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. ----------------------------	
	B.1 Be in MODE 3.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.1.6 Low Set Relief Valves

LCO 3.6.1.6 The low set relief function of two relief valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low set relief valve inoperable.	A.1 Restore low set relief valve to OPERABLE status.	14 days
B. Required Action and associated Completion Time of Condition A not met.	------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 3. ---------------------------- B.1 Be in MODE 3.	 12 hours
C. Two low set relief valves inoperable.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and Associated Completion Time of Condition C not met.	------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. -----------------------------	
	D.1 Be in MODE 3.	12 hours
E. Two lines with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening.	E.1 Restore all vacuum breakers in one line to OPERABLE status.	1 hour
F. Required Action and Associated Completion Time of Conditions A, B, or E not met.	F.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	F.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.7.1	-----------------NOTES----------------- 1. Not required to be met for vacuum breakers that are open during Surveillances. 2. Not required to be met for vacuum breakers open when performing their intended function. --------------------------------------- Verify each vacuum breaker is closed.	14 days
SR 3.6.1.7.2	Perform a functional test of each vacuum breaker.	92 days
SR 3.6.1.7.3	Verify the opening setpoint of each vacuum breaker is ≤ 0.5 psid.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.1.8 Suppression Chamber-to-Drywell Vacuum Breakers

LCO 3.6.1.8 Nine suppression chamber-to-drywell vacuum breakers shall be OPERABLE for opening.

<u>AND</u>

Twelve suppression chamber-to-drywell vacuum breakers shall be closed.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required suppression chamber-to-drywell vacuum breaker inoperable for opening.	A.1	Restore one vacuum breaker to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. -----------------------------		
	B.1	Be in MODE 3.	12 hours
C. One suppression chamber-to-drywell vacuum breaker not closed.	C.1	Close the open vacuum breaker.	4 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	D.2	Be in MODE 4.	36 hours

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Suppression Pool Cooling

LCO 3.6.2.3 Two suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One suppression pool cooling subsystem inoperable.	A.1 Restore suppression pool cooling subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 3. ---------------------------- B.1 Be in MODE 3.	 12 hours
C. Two suppression pool cooling subsystems inoperable.	C.1 Restore one suppression pool cooling subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Suppression Pool Spray

LCO 3.6.2.4 Two suppression pool spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One suppression pool spray subsystem inoperable.	A.1 Restore suppression pool spray subsystem to OPERABLE status.	7 days
B. Two suppression pool spray subsystems inoperable.	B.1 Restore one suppression pool spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 3. ----------------------------- C.1 Be in MODE 3.	 12 hours

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. -----------------------------	
	B.1 Be in MODE 3.	12 hours
C. Secondary containment inoperable during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	C.1 --------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	C.2 Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. ----------------------------- B.1 Be in MODE 3.	 12 hours
C. Required Action and associated Completion Time of Condition A not met during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- C.1 Place OPERABLE SGT subsystem in operation. OR	 Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1 Suspend movement of recently irradiated fuel assemblies in secondary containment. AND C.2.2 Initiate action to suspend OPDRVs.	Immediately Immediately
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Restore one SGT subsystem to OPERABLE status.	1 hour
E. Required Action and associated Completion Time of Condition D not met.	------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. ----------------------------- E.1 Be in MODE 3.	12 hours
F. Two SGT subsystems inoperable during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	F.1 --------NOTE-------- LCO 3.0.3 is not applicable. -------------------- Suspend movement of recently irradiated fuel assemblies in secondary containment. AND F.2 Initiate action to suspend OPDRVs.	Immediately Immediately

3.7 PLANT SYSTEMS

3.7.1 Containment Cooling Service Water (CCSW) System

LCO 3.7.1 Two CCSW subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CCSW pump inoperable.	A.1 Restore CCSW pump to OPERABLE status.	30 days
B. One CCSW pump in each subsystem inoperable.	B.1 Restore one CCSW pump to OPERABLE status.	7 days
C. One CCSW subsystem inoperable for reasons other than Condition A.	C.1 Restore CCSW subsystem to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Conditions A, B, or C not met.	------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 3. ---------------------------- D.1 Be in MODE 3.	 12 hours
E. Both CCSW subsystems inoperable for reasons other than Condition B.	E.1 Restore one CCSW subsystem to OPERABLE status.	8 hours
F. Required Action and associated Completion Time of Condition E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 4.	12 hours 36 hours

3.7 PLANT SYSTEMS

3.7.4 Control Room Emergency Ventilation (CREV) System

LCO 3.7.4 The CREV System shall be OPERABLE.

-----------------NOTE-----------------
The main control room envelope (CRE) boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CREV System inoperable in MODE 1, 2, or 3 for reasons other than Condition C.	A.1 Restore CREV System to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. ----------------------------- B.1 Be in MODE 3.	 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. CREV system inoperable due to inoperable CRE boundary in MODE 1, 2, or 3.	C.1	Initiate action to implement mitigating actions.	Immediately
	AND		
	C.2	Verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.	24 hours
	AND		
	C.3	Restore CRE boundary to OPERABLE status.	90 days
D. Required Action and associated Completion Time of Condition C not met in MODE 1, 2, or 3.	D.1	Be in MODE 3.	12 hours
	AND		
	D.2	Be in MODE 4.	36 hours
E. CREV System inoperable during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs. OR CREV System inoperable due to an inoperable CRE boundary during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------		
	E.1	Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND		
	E.2	Initiate action to suspend OPDRVs.	Immediately

3.7 PLANT SYSTEMS

3.7.5 Control Room Emergency Ventilation Air Conditioning (AC) System

LCO 3.7.5 The Control Room Emergency Ventilation AC System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Control Room Emergency Ventilation AC System inoperable in MODE 1, 2, or 3.	A.1 Restore Control Room Emergency Ventilation AC System to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 3. ---------------------------- B.1 Be in MODE 3.	 12 hours
C. Control Room Emergency Ventilation AC System inoperable during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- C.1 Suspend movement of recently irradiated fuel assemblies in the secondary containment. <u>AND</u> C.2 Initiate action to suspend OPDRVs.	 Immediately Immediately

3.7 PLANT SYSTEMS

3.7.6 Main Condenser Offgas

LCO 3.7.6 The gross gamma activity rate of the noble gases measured prior to the offgas holdup line shall be ≤ 252,700 μCi/second after decay of 30 minutes.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated and steam jet air ejector (SJAE) in operation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Gross gamma activity rate of the noble gases not within limit.	A.1 Restore gross gamma activity rate of the noble gases to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Isolate all main steam lines.	12 hours
	<u>OR</u>	
	B.2 Isolate SJAE.	12 hours
	<u>OR</u>	
	------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. -----------------------------	
	B.3 Be in MODE 3.	12 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met.	------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 3. ---------------------------- F.1 Be in MODE 3.	 12 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
H. Division 1 or 2 125 VDC electrical power subsystem inoperable for reasons other than Conditions E, F, or G.	H.1	Restore Division 1 or 2 125 VDC electrical power subsystem to OPERABLE status.	2 hours
	<u>OR</u>		
	H.2	--------NOTE--------- Only applicable if the opposite unit is not in MODE 1, 2, or 3. --------------------- Place associated OPERABLE alternate 125 VDC electrical power subsystem in service.	2 hours
I. Opposite unit Division 2 125 VDC electrical power subsystem inoperable.	I.1	Restore opposite unit Division 2 125 VDC electrical power subsystem to OPERABLE status.	7 days
J. Required Action and associated Completion Time not met.	------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. -----------------------------		
	J.1	Be in MODE 3.	12 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more DC electrical power distribution subsystems inoperable.	B.1 Restore DC electrical power distribution subsystems to OPERABLE status.	2 hours <u>AND</u> 16 hours from discovery of failure to meet LCO 3.8.7.a
C. One or more required opposite unit Division 2 AC or DC electrical power distribution subsystems inoperable.	-------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.1 when Condition C results in the inoperability of a required offsite circuit. ----------------------------- C.1 Restore required opposite unit Division 2 AC and DC electrical power distribution subsystems to OPERABLE status.	 7 days
D. Required Action and associated Completion Time of Condition A, B, or C not met.	------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. ----------------------------- D.1 Be in MODE 3.	 12 hours
E. Two or more electrical power distribution subsystems inoperable that, in combination, result in a loss of function.	E.1 Enter LCO 3.0.3.	Immediately

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED

TO AMENDMENT NO. 233 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 226 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 dated October 9, 2007, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072830088), as supplemented by letter dated January 30, 2009, (ADAMS Accession No. ML090350151), Exelon Generation Company (EGC, the licensee) submitted a license amendment request (LAR) which proposed changes to the technical specifications (TSs) for Dresden Nuclear Power Station (DNPS) Units 2 and 3. The LAR would modify the TSs to risk-informed requirements regarding required action end states. In the request, DNPS planned to adopt Technical Specification Task Force (TSTF) change traveler 423, "Technical Specifications End States, NEDC-32988-A," (Reference (Ref.) 8), to NUREG-1433, "Standard Technical Specifications [STS] General Electric Plants, BWR/4," Revision 3, and NUREG-1434, "Standard Technical Specifications [STS] General Electric Plants, BWR/6," Revision 3, which was proposed by the TSTF owners groups on August 12, 2003, on behalf of the industry. TSTF-423 incorporates the Boiling-Water Reactor (BWR) Owners Group (BWROG) approved topical report (TR) NEDC-32988-A, (Ref. 1), into the BWR STS (NOTE: The changes in TSTF-423 are made with respect to Revision 3 of the BWR STS NUREGs).

On March 30, 2001 (ADAMS Accession No ML011130121), the Nuclear Regulatory Commission (NRC, the Commission) staff approved the licensee's request to covert the DNPS TSs to the improved TSs design based on NUREG-1433, Revision 1, "Standard Technical Specifications, General Electric Plants BWR/4," dated April 1995, and on guidance provided in the Commission's "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," published on July 22, 1993 (58 FR 39132). The licensee's October 9, 2007, application states that DNPS TSs are based on NUREG-1433 though it is not identical to the Commission's Policy Statement guidance. Therefore, an adaptation of the referenced document was required.

The January 30, 2009, supplement contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

TSTF-423 is one of the industry's initiatives developed under the Risk Management Technical Specifications Program. These initiatives are intended to maintain or improve safety through the incorporation of risk assessment and risk management techniques in TS, while reducing unnecessary burden and making TS requirements consistent with the Commission's other risk-informed regulatory requirements, in particular, Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.65 (Ref. 3), the "Maintenance Rule." Section 50.36(c)(2)(i) of 10 CFR, states, in part: "When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow the remedial action permitted by the technical specification until the condition can be met." Plant TSs provide, as part of the remedial action, a completion time (CT) for the plant to either comply with remedial actions or restore compliance with the limiting conditions for operation (LCO). If the LCO or the remedial action cannot be met, then the reactor is required to be shutdown. When the STS and individual plant TSs were written, the shutdown condition, or end state specified, was usually cold shutdown. The BWROG TR provides the technical bases to change certain required end states when the TS Actions for remaining in power operation cannot be met within the CTs. The proposed TS changes permit an end state of hot shutdown (Mode 3), if risk is assessed and managed, rather than an end state of cold shutdown (Mode 4), contained in the current TSs. The proposed changes were limited to those end states where: (1) entry into the shutdown mode is for a short interval, (2) entry is initiated by inoperability of a single train of equipment or a restriction on a plant operational parameter, unless otherwise stated in the applicable TS, and (3) the primary purpose is to correct the initiating condition and return to power operation as soon as is practical.

The TSs for DNPS define five operational modes:

- Mode 1 - Power Operation. The reactor mode switch is in the RUN position.
- Mode 2 - Reactor Startup. The reactor mode switch is in REFUEL position (with all reactor vessel head closure bolts fully tensioned) or in STARTUP/HOT STANDBY position.
- Mode 3 - Hot Shutdown. The reactor coolant system (RCS) temperature is above 212 °F (TS specific) and the reactor mode switch is in SHUTDOWN position (with all reactor vessel head closure bolts fully tensioned).
- Mode 4 - Cold Shutdown. The RCS temperature is equal to or less than 212 °F and the reactor mode switch is in SHUTDOWN position (with all reactor vessel head closure bolts fully tensioned).
- Mode 5 – Refueling. The reactor mode switch is in the SHUTDOWN or REFUEL position, and one or more reactor vessel head closure bolts are less than fully tensioned.

Modifying the DNPS TSs consistent with TSTF-423 allows a Mode 3 end state rather than a Mode 4 end state for selected initiating conditions in order to perform short-duration repairs. Short duration repairs are on the order of 2-to-3 days, but not more than a week.

The licensee stated that the BWROG TR and TSTF-423, as well as the NRC safety evaluation (SE) of the TR (Ref. 6), were applicable to DNPS Units, and provided justifications for incorporation of the proposed changes into the DNPS Units 2 and 3 TSs.

2.0 REGULATORY EVALUATION

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36(c), TSs are required to include items in the following eight specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) surveillance requirements; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. The NRC staff did not review the LAR with respect to decommissioning, initial notification and written reports, as the licensee did not propose any changes to these specific requirements. The rule does not specify the particular requirements to be included in a plant's TSs. As stated in 10 CFR 50.36(c)(2)(i), LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications. In describing the basis for changing end states, the BWROG TR states:

> "Cold shutdown is normally required when an inoperable system or train cannot be restored to an operable status within the allowed time. Going to cold shutdown results in the loss of high pressure core cooling systems, challenges the shutdown heat removal systems, and requires restarting the plant. A more preferred operational mode is one that maintains adequate risk levels while repairs are completed without causing unnecessary challenges to plant equipment during shutdown and startup transitions."

In the end state changes under consideration, a problem with a component or train has, or will, result in a failure to meet TSs, and a controlled shutdown is directed because a TS Action statement cannot be met within the TS CT.

Most of today's TSs and the design basis analyses were developed under the perception that putting a plant in cold shutdown would result in the safest condition, and the design basis analyses would bound credible shutdown accidents. In the late 1980s and early 1990s, the NRC and licensees recognized that this perception was incorrect and took corrective actions to improve shutdown operation. At the same time, the STS were developed and many licensees took action to improve their TSs. Since enactment of a shutdown rule was expected, almost all TS changes involving power operation, including a revised end state requirement, were postponed (e.g., see Ref. 2). However, in the mid-1990s, the Commission decided a shutdown rule was not necessary in light of industry improvements. Controlling shutdown risk encompasses control of conditions that can cause potential initiating events and responses to those initiating events that do occur. Initiating events are a function of equipment malfunctions and human error. Responses to events are a function of plant sensitivity, ongoing activities, human error, defense-in-depth, and additional equipment malfunctions.

In practice, the risk during shutdown operations is often addressed via voluntary actions and application of the Maintenance Rule. Section 50.65(a)(4) of the Maintenance Rule states, in part:

> "Before performing maintenance activities ..., the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety."

Regulatory Guide (RG) 1.182 (Ref. 4) provides guidance on implementing the provisions of 10 CFR 50.65(a)(4) by endorsing the revised Section 11 (published separately) to NUMARC 93-01, Revision 2 (Ref. 7). The remainder of NUMARC 93-01, Revision 2, was previously endorsed by the NRC staff in RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," March 1997.

3.0 TECHNICAL EVALUATION

The changes proposed in the amendment are consistent with the changes proposed and justified in the BWROG TR, and that have been accepted by the NRC staff as documented in the NRC staff's TR SE. The evaluation included in the NRC staff's TR SE, as appropriate and applicable to the changes of TSTF-423, is reiterated here, and differences from the SE are justified. In its application, the licensee commits to the implementation guidance for TSTF-423 contained in TSTF-IG-05-02 (Ref. 9), which addresses a variety of issues, such as considerations and compensatory actions for risk-significant plant configurations. An overview of the generic evaluation and associated risk assessment is provided below, along with a summary of the associated TS changes justified by the BWROG TR.

3.1 Risk Assessment

The objective of the BWROG TR risk assessment was to show that any risk increases associated with the proposed changes in TS end states are either negligible or negative (i.e., a net decrease in risk). The BWROG TR documents a risk-informed analysis of the proposed TS change. Probabilistic Risk Assessment (PRA) results and insights are used, in combination with results of deterministic assessments, to identify and propose changes in "end states" for all BWR plants. This is in accordance with guidance provided in RG 1.174 (Ref. 10) and RG 1.177 (Ref. 5). The three-tiered approach documented in RG 1.177 was followed. The first tier includes the assessment of the risk impact of the proposed change for comparison to acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174. The first tier aims at ensuring that there are no unacceptable temporary risk increases as a result of the TS change, such as when equipment is taken out of service. The second tier addresses the need to preclude potentially high-risk configurations which could result if equipment is taken out of service concurrently with the equipment out of service as allowed by this TS change. The third tier addresses the application of 10 CFR 50.65 (a)(4) for identifying risk-significant configurations resulting from maintenance related activities and taking appropriate compensatory measures to avoid such configurations.

The proposed TS change invokes a risk assessment because 10 CFR 50.65(a)(4) is applicable to maintenance-related activities and does not cover other operational activities beyond the effect they may have on existing maintenance related risk.

As discussed in the NRC staff's TR SE, the NRC staff found that the BWROG's risk assessment approach used in the BWROG TR was comprehensive and acceptable. In addition, the analyses show that the three-tiered approach criteria for allowing TS changes are met as follows:

- Risk Impact of the Proposed Change (Tier 1): The risk changes associated with the TS changes in TSTF-423, in terms of mean yearly increases in core damage frequency (CDF) and large early release frequency (LERF), are risk neutral or risk beneficial. In addition, there are no significant temporary risk increases, as defined by RG 1.177 criteria, associated with the implementation of the TS end state changes.

- Avoidance of Risk-Significant Configurations (Tier 2): The risk analyses that were performed, which are based on single LCOs, indicate that there are no high-risk configurations associated with the TS end state changes. The reliability of redundant trains is normally covered by a single LCO. When multiple LCOs occur, which affect trains in several systems, the plant's risk-informed configuration risk management program, or the risk assessment and management program implemented in response to 10 CFR 50.65(a)(4), shall ensure that high-risk configurations are avoided. As part of the implementation of TSTF-423, the licensee has committed to follow Section 11 of NUMARC 93-01, Revision 3, and include guidance in appropriate plant procedures and/or administrative controls to preclude high-risk plant configurations when the plant is at the proposed end state. While the NRC staff has not endorsed Revision 3 to NUMARC 93-01, the NRC staff has endorsed a revised version of NUMARC 93-01, Revision 2 (Ref. 7), Section 11 in RG 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants" (Ref. 4). The NRC staff compared Section 11 of NUMARC 93-01, Revision 3, to the endorsed version in RG 1.182 and found that that the language in the two versions is consistent. The NRC staff finds that such guidance is adequate for preventing risk-significant plant configurations.

- Configuration Risk Management (Tier 3): The licensee has a commitment in place (as described below) to comply with 10 CFR 50.65 (a)(4) to assess and manage the risk from maintenance activities. This program can support the licensee's decision in selecting the appropriate actions to control risk for most cases in which a risk informed TS is entered.

The generic risk impact of the end state mode change was evaluated subject to the following assumptions which are incorporated into the TS, TS Bases, and TSTF-IG-05-02:

a. The entry into the end state is initiated by the inoperability of a single train of equipment or a restriction on a plant operational parameter, unless otherwise stated in the applicable TS.

b. The primary purpose for entering the end state is to correct the initiating condition and return to power as soon as is practical.

c. When Mode 3 is entered as the repair end state, the time the reactor coolant pressure is above 500 psig will be minimized. If reactor coolant pressure is above 500 psig for more than 12 hours, the associated plant risk will be assessed and managed.

These assumptions are consistent with typical entries into Mode 3 for short duration repairs, which is the intended use of the TS end state changes. The NRC staff concludes that, going to Mode 3 (hot shutdown) instead of going to Mode 4 (cold shutdown) to carry out equipment repairs that are of short duration, does not have any adverse effect on plant risk.

In its application, the licensee committed to follow the guidance established in Section 11 of NUMARC 93-01. NUMARC 93-01 provides guidance on implementing the provisions of 10 CFR 50.65(a)(4). The licensee also committed in the January 30, 2009, supplement to follow the guidance established in TSTF-IG-05-02. The commitments are restated in the following table:

COMMITMENT	COMMITTED DATE	COMMITMENT TYPE	
		ONE TIME ACTION (YES/NO)	PROGRAMMATIC (YES/NO)
EGC will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision 3, July, 2000.	Ongoing	No	Yes
EGC will follow the guidance established in TSTF-IG-05-02, "Implementation Guidance for TSTF-423, Revision 0, 'Technical Specifications End States,NEDC-32988-A,'" Revision 1, March 2007. The following statement on Page 2 no longer applies: "If Primary Containment is not operable, Secondary Containment and Standby Gas Treatment must be verified operable in order to remain in Mode 3."	Implement with amendment	No	Yes

The NRC staff notes that it has not endorsed NUMARC 93-01, Revision 3, referenced in the first commitment, but has evaluated the language in Section 11 of NUMARC 93-01, Revision 3, and finds that it is consistent with the version of Section 11 that has been endorsed by the NRC staff in RG 1.182. The NRC staff acceptance of this commitment relates only to Section 11 of NUMARC 93-01, Revision 3.

By following the implementation guidance, the licensee will ensure that defense-in-depth is maintained for key safety functions by ensuring availability of Tier 2 systems/equipment necessary for safe shutdown. Therefore, the NRC staff finds the licensee's commitments to be acceptable.

3.2 Request for Additional Information

During its review of the application, the NRC staff identified two concerns. First, revising the TSs to allow the licensee to remain in Mode 3 indefinitely with inoperable systems would also permit starting up using the allowance of LCO 3.0.4(a) with inoperable systems or equipment. This is inconsistent with the purpose of TSTF-423, which is to allow licensees to remain in Mode 3 (instead of proceeding to Mode 4) while conducting repairs, and then return to Mode 1.

The second concern is that primary containment should not have been treated the same as the other systems included in the TSTF-423. Primary containment was not included in the TSTFs for the pressurized-water reactor designs (i.e., TSTF-422 for Combustion Engineering Plants; TSTF-431 for Babcock and Wilcox plants). The reason for this is that, unlike the other systems included in TSTF-423, an inoperable primary containment constitutes a loss of one of the three fission product barriers. Staying at hot conditions in such an unanalyzed condition is not consistent with maintaining defense-in-depth, which is one of the five key principles of risk-informed regulations in RG 1.174. From the RG perspective, the core damage risks are found to be acceptable; however, the compensatory measures identified (i.e., availability of secondary containment, ventilation treatment systems, etc.) do not provide an acceptable defense-in-depth approach, and, therefore, an equivalent level of protection as provided by the primary containment, could not be attained by the compensatory measures.

To address these two concerns, the NRC staff issued a request for additional information (RAI) (ADAMS Accession No. ML090080309) to the licensee on January 8, 2009. While the RAI was related to the TSTF-423 review for Clinton Nuclear Power Station, the NRC staff requested that the licensee's response include information for the three facilities with TSTF-423 applications under NRC staff review. The following summarizes the NRC staff's questions, and the licensee's responses for DNPS:

<u>RAI 1:</u> The licensee was requested to demonstrate how they would prevent LCO 3.0.4(a) from being inappropriately invoked during startup to facilitate going up in mode with inoperable systems or equipment.

<u>RAI 2:</u> The licensee was requested to demonstrate how they would maintain an equivalent level of protection while operating in Mode 3 with an inoperable primary containment.

On January 30, 2009, the licensee provided their response to the NRC staff's RAI as follows:

<u>Response to RAI 1</u>: To prevent LCO 3.0.4(a) from being inappropriately invoked during startup to facilitate going up in mode with inoperable systems or equipment, EGC proposed the insertion of the following Note into those Required Actions affected by TSTF-423:

NOTE
LCO 3.0.4.a is not applicable when entering MODE 3.

In addition, the licensee indicated that some of the previously submitted TS pages have been amended since its request to adopt TSTF-423 at DNPS. Accordingly, the licensee provided revised versions of the TS pages that included the original TSTF-423 adoption markups and the above Note.

<u>Response to RAI 2</u>: The licensee indicated that it had evaluated its requests to amend station TS for primary containment and decided to withdraw its request to amend this TS. Because Mode 3 is no longer the requested end state for primary containment, the licensee determined that it is necessary to revise its original commitment to follow guidance established in TSTF-IG-05-02, to indicate that the following statement on Page 2 no longer applies:

"If Primary Containment is not operable, Secondary Containment and Standby Gas Treatment must be verified operable in order to remain in Mode 3."

Conclusion:

The NRC staff reviewed the licensee's response to the staff's RAIs, and found them to be acceptable since the amended station TSs prevent a) operation in Mode 3 without primary containment, and b) starting up with inoperable systems or equipment.

3.3 Assessment of TS Changes

NOTE: The BWROG Report and the NRC staff's TR SE identified several important design and operational differences between the various BWR plant types and the representative BWR-4 and -6 plants used in the quantitative risk assessment in TSTF-423. One such difference is that BWR-2 and early BWR-3 plants (including the DNPS units) are equipped with an Isolation Condenser (IC) instead of the Reactor Core Isolation Cooling (RCIC) system used in BWR-4 plants. The NRC staff compared the design of the RCIC system to the design of the IC system and provided the results below:

a. The RCIC system consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the reactor pressure vessel (RPV) through the feedwater sparger. The system is not part of the Emergency Core Cooling Systems (ECCS); however, it is included with the ECCS because of their similar functions. The system is designed to operate either automatically or manually following RPV isolation accompanied by a loss-of-coolant flow from the feedwater system to provide adequate core cooling and control of the RPV water level. Under these conditions, the High-Pressure Coolant Injection (HPCI) and RCIC systems perform similar functions.

   Per the Safety Analyses in the Bases for the STSs (NUREG-1433), the safety function of RCIC system is to respond to transient events by providing makeup coolant to the reactor. The system is not an Engineered Safety Feature System and no credit is taken in the safety analyses for RCIC system operation. The system satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

b. The IC system is a passive high pressure system comprised of one natural circulation heat exchanger, motor-operated and high point vent isolation valves to main steamline "A." The IC system is not part of the ECCS; however, it is included with the ECCS section because of their similar functions. The system is designed to operate either automatically or manually following RPV isolation to provide adequate core cooling. Under these conditions, the HPCI and IC systems perform similar functions.

   Per the Safety Analyses in the DNPS TS Bases, the safety function of the IC system is to respond to main steamline isolation events by providing core cooling to the reactor. The IC system is an Engineered Safety Feature system, and credit is taken in the loss of feedwater transient analysis for the IC system operation. Based on the contribution to overall plant risk, the system satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

As discussed in the BWROG TR, "[S]ome BWR-2, 3 plants are equipped with Isolation Condensers instead of RCIC. Following an isolation transient, Isolation Condensers condense the main steam and return it to the reactor. Isolation Condensers do not require any motive power for pumping. Just like RCIC, Isolation Condensers are also capable of mitigating station blackout events in Mode 3. Both Isolation Condenser and RCIC have similar system reliability, therefore, their contributions to Mode 3 results are very similar. Neither system is available during Mode 4, so the Mode 4 results are not impacted by the two systems. One difference between RCIC and Isolation Condenser is that in Mode 3, RCIC can mitigate small LOCA while Isolation Condenser cannot. However, small LOCA's are insignificant contributors to Mode 3 and 4 core damage frequency and this difference between RCIC and Isolation Condenser is considered negligible."

The NRC staff evaluated General Electric's conclusions regarding the variations in the RCIC and IC systems in the NRC staff's TR SE. Section 4.2.2 on "Quantitative Risk Assessment" in the staff's safety evaluation states the following:

> "Design and operational differences among the various BWR plants were identified and appropriate sensitivity studies were performed which show that the conclusions of the quantitative risk assessment apply to all BWR plants."

Based on the above, since, a) both (RCIC and IC) systems respond to transient events by providing core cooling to the reactor, b) both satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii), and c) specific differences between the both systems were identified in the BWROG TR, and evaluated in the NRC staff's TR SE, the NRC staff finds that from a risk perspective, the differences between the two systems are negligible, and, therefore, the quantitative risk assessment for RCIC in TSTF-423 also applies to the IC system at DNPS.

The following sections discuss the specific changes, and include a synopsis of the STS LCOs. The NRC staff discusses the acceptability of the proposed changes in Section 3.4.

3.3.1 LCO 3.4.3: Safety and Relief Valves (SRVs)

The function of the SRVs is to protect the plant against severe overpressurization events. The TSs provide the operability requirements for the SRVs as described below. The relief valves are also designed to rapidly depressurize the reactor vessel so that the Core Spray and Low Pressure Coolant Injection (LPCI) systems can function to mitigate small line break events.

<u>LCO</u>: The safety function of nine safety valves and the relief function of five relief valves shall be OPERABLE.

<u>Condition Requiring Entry into End State:</u> If the LCO cannot be met with one relief valve inoperable, the inoperable valve must be returned to operability within 14 days. If the relief valve cannot be returned to operable status within that time, the plant must be placed in Mode 3 within 12 hours and in Mode 4 within 36 hours.

<u>Modification for End State Required Actions</u>: If the LCO cannot be met with one relief valve inoperable, the inoperable valves must be returned to operability within 14 days. If the valve cannot be returned to operable status within 14 days, the plant must be placed in Mode 3 within

12 hours (Required Condition B.1). A Note is added to the TS Required Action for B.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3. Required Action B.2 is deleted allowing the plant to stay in Mode 3 while completing repairs. Condition B concerning inoperability of two or more relief valves or one or more safety valves is renumbered to a new Condition C with no changes to its original Required Actions and Completion Times.

Assessment:

In the BWROG TR, the BWROG documented a comparative PRA evaluation of the core damage risks of operation in the current end state and in the Mode 3 end state. The NRC staff reviewed the PRA evaluation and concluded in its TR SE, that the core damage risks are approximately the same or lower in Mode 3 than in Mode 4. For DNPS, going to Mode 4 would cause loss of HPCI, the IC, and the power conversion systems due to the plant cooldown, and would require activating the Residual Heat Removal (RHR) system. With one relief valve inoperable, the remaining valves are adequate to perform the required function. By remaining in Mode 3, the HPCI, the IC, and the power conversion systems remain available to ensure adequate core cooling, improving defense-in-depth compared to transitioning to Mode 4. The plant Emergency Operating Procedures (EOPs) direct the operator to take control of the depressurization function if low pressure injection/spray systems are needed for RPV water makeup and cooling. The NRC staff concluded in its TR SE that the change allows the repair of the inoperable relief valve to be performed in a plant operating mode with lower risks.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in the TS Required Action for B.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with this system inoperable.

3.3.2 LCO 3.5.1: Emergency Core Cooling Systems (ECCS)–Operating

The ECCS systems provide cooling water to the core in the event of a loss-of-coolant accident (LOCA). This set of ECCS TSs provides the operability requirements for the various ECCS subsystems as described below. This TS change would delete the secondary actions. The plant can remain in Mode 3 until the required repair actions are completed. The reactor is not depressurized.

LCO: Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of five relief valves shall be OPERABLE.

Condition Requiring Entry into End State: If the LCO cannot be met, the following actions must be taken for the listed conditions:

a. If one LPCI pump is inoperable, the LPCI pump must be restored to operable status in 30 days (Condition A).

b. If one LPCI subsystem is inoperable for reasons other than Condition A or one Core Spray subsystem is inoperable, the low pressure ECCS injection/spray subsystem must be restored to operable status within 7 days (Condition B).

c. If one LPCI pump in each subsystem is inoperable, one LPCI pump must be restored to operable status within 7 days (Condition C).

d. If two LPCI subsystems are inoperable for reasons other than Condition C, one LPCI subsystem must be restored to operable status within 72 hours (Condition D).

e. If the Required Action and associated Completion Time of Condition A, B, C, or D is not met, then place the plant in Mode 3 within 12 hours and in Mode 4 with in 36 hours (Condition E).

f. If the HPCI system is inoperable, verify immediately by administrative means that the IC system is operable and restore HPCI system to operable status within 14 days (Condition F).

g. If one ADS valve is inoperable, it must be restored to operable status within 14 days (Condition G).

h. If the Required Action and associated Completion Time of Condition F or G is not met or two or more ADS valves become inoperable, the plant must be placed in Mode 3 within 12 hours and the reactor steam dome pressure reduced to less than 150 psig within 36 hours (Condition H).

i. If two or more low pressure ECCS injection/spray subsystems are inoperable for reasons other than Condition C or D, or the HPCI system and one or more ADS valve are inoperable, or one or more low pressure ECCS injection/spray subsystems inoperable, and one or more ADS valves are inoperable or the HPCI system is inoperable, and either one low pressure ECCS injection/spray subsystems inoperable, or Condition C is entered, LCO 3.0.3 must be entered immediately (Condition I).

Modification for End State Required Actions:

a. No change in Required Action for Condition A.

b. No change in Required Action for Condition B.

c. No change in Required Action for Condition C.

d. If the Required Action and associated Completion Time of Condition A, B, or C is not met, then place the plant in Mode 3 within 12 hours (new Required Action D.1). The plant is not taken into Mode 4 (cold shutdown). A Note is added to the TS Required Action D.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3. Old Conditions D, E, F and G are renumbered to E, F, G and H with no changes to the Required Actions for these Conditions.

e. If the Required action and associated Completion Time of renumbered Conditions G or H is not met, then the plant must be placed in Mode 3 within 12 hours (new Condition I). The reactor is not depressurized and not taken to Mode 4. A Note is added to the TS Required Action I.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3. Old Conditions H and I are renumbered to J and K. The renumbered Condition J states that if two or more ADS valves inoperable, the plant must be placed in Mode 3 within 12 hours and the reactor steam dome pressure reduced to less than 150 psig within 36 hours. In renumbered Condition K, change Condition D to E in accordance with the renumbering format of Conditions with no changes to the Required Action and Completion Time for the Condition.

Assessment: The BWROG TR discusses a comparative PRA evaluation of the core damage risks of operation in the current end state and the Mode 3 end state. The NRC staff's conclusion on the BWROG TR's PRA evaluation described in the NRC staff's TR SE, indicates that the core damage risks are lower in Mode 3 than in the current end state of Mode 4. For DNPS, going to Mode 4 would cause loss of HPCI, the IC, and the power conversion systems due to the plant cooldown, and would require activating the RHR system. In addition, plant EOPs direct the operator to take control of the depressurization function if low-pressure injection/spray systems are needed for RPV water makeup and cooling.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Actions D.1 and I.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.3 LCO 3.5.3: The IC System

NOTE: DNPS's TS LCO 3.5.3 specifies the requirements for the operability of IC System. LCO 3.5.3 in TSTF-423 discusses the requirements for a typical BWR4-type plant's RCIC system. As the NRC staff has concluded in Section 3.3 above, since, a) both RCIC and IC systems respond to transient events by providing core cooling to the reactor, b) both satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii), and c) specific differences between both systems were identified in the BWROG TR, and evaluated in the NRC staff's TR SE, the NRC staff finds that from a risk perspective, the differences between the two systems are negligible, and, therefore, the quantitative risk assessment for the RCIC in TSTF-423 also applies to the IC system at DNPS.

The performance objective of the IC is to provide reactor core cooling in the event that the reactor becomes isolated from the turbine and the main condenser by closure of the main steam isolation valves. This TS provides the operability requirements for the IC system as described below. In the event of an inoperable IC system, the TS change allows the plant to remain in Mode 3 until the repairs are completed.

LCO: The IC system must be operable during Mode 1, and in Modes 2 and 3 when the reactor steam dome pressure is greater than 150 psig.

Condition Requiring Entry into End State: If the LCO cannot be met, the following actions must be taken: (a) verify immediately by administrative means that the HPCI system is operable (Required Action A.1), and (b) restore the IC system to operable status within 14 days (Required Action A.2). If either or both actions cannot be completed within the allotted time, the plant must be placed in Mode 3 within 12 hours and the reactor steam dome pressure reduced to less than 150 psig within 36 hours (Required Actions B.1 and B.2).

Modification for End State Required Actions: This TS change keeps the plant in Mode 3 until the required repairs are completed. A Note is added to the TS Required Action B.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3. The reactor steam dome pressure is not reduced to less than 150 psig (delete Required Action B.2).

Assessment: This change would allow the inoperable IC system to be repaired in a plant operating mode with lower risk and without challenging the normal shutdown systems. In the BWROG TR, the BWROG performed a comparative PRA evaluation of the core damage risks of operation in the current end state and in the Mode 3 end state. In the NRC staff's TR SE, the NRC staff concluded that the core damage risks are lower in Mode 3 than in Mode 4. For DNPS, going to Mode 3 with reactor steam dome pressure less than 150 psig would cause a loss of HPCI and a loss of the power conversion systems, and would require activating the RHR system. By remaining in Mode 3 above 150 psig steam dome pressure, HPCI and the power conversion systems remain available for coolant inventory control and decay heat removal. In addition, the plant EOPs direct the operator to take control of the depressurization function if low pressure injection/spray systems are needed for RPV water makeup and cooling.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Action B.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.4 LCO 3.6.1.6: Low Set Relief Valves

The function of low set relief valves is to prevent excessive short-duration SRV cycling during an overpressure event. This TS provides operability requirements for the two low set relief valves as described below. The TS change allows the plant to remain in Mode 3 until the repairs are completed.

LCO: The low set relief function of two relief valves shall be OPERABLE.

Condition Requiring Entry into End State: If one low set relief valve is inoperable, it must be returned to operability within 14 days. If the low set relief valve cannot be returned to operable status within the allotted time, the plant must be placed in Mode 3 within 12 hours and in Mode 4 within 36 hours.

Modification for End State Required Actions: The TS change would keep the plant in Mode 3 until the required repair actions are completed. The plant would not be taken into Mode 4 (cold shutdown) (delete Required Action B.2). A Note is added to the TS Required Action B.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3. (Note: The Required Action for two low set relief valves inoperable was changed from Condition B to new Condition C without changing the Required Action end state.)

Assessment: In the BWROG TR, the BWROG performed a comparative PRA evaluation of the core damage risks of operation in the current end state and the Mode 3 end state. In the NRC staff's TR SE, the NRC staff concluded that the core damage risks are lower in Mode 3 than in Mode 4. In Mode 3, with one low set relief valve inoperable, the remaining valves are adequate to perform the required function. In addition, the plant EOPs direct the operator to take control of the depressurization function if low pressure injection/spray systems are needed for RPV water makeup and cooling. With one low set relief valve inoperable, going to Mode 4 would cause loss of HPCI, the IC, and the power conversion systems due to the plant cooldown, and would require activating the RHR system. The NRC staff concluded in its TR SE that remaining in Mode 3 allows repairs of the inoperable low set relief valve to be performed in a plant operating mode with lower risks.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Action B.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.5 LCO 3.6.1.7: Reactor Building-to-Suppression Chamber Vacuum Breakers

The reactor building-to-suppression chamber vacuum breakers relieve vacuum when the primary containment depressurizes below the pressure of the reactor building, thereby preserving the integrity of the primary containment.

LCO: Each reactor building-to-suppression chamber vacuum breaker shall be OPERABLE.

Condition Requiring Entry into End State: If one line has one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening, the breaker(s) must be returned to operability within 7 days (Required Action C.1). If the vacuum breaker(s) cannot be returned to operability within the allotted time, the plant must be placed in Mode 3 within 12 hours (Required Action E.1) and in Mode 4 within 36 hours (Required Action E.2).

Modification for End State Required Actions: A new Condition D modifies the Required Actions so that, if the vacuum breaker(s) cannot be returned to operable status within the required Completion Times, the plant is placed in hot shutdown. A Note is added to the TS Required Action D.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3. Existing Conditions D and E are renumbered to E and F without changing the Required Action end state. Condition F would require shutting down the plant to Mode 3 (Required Action F.1) and then Mode 4 (Required Action F.2), to address an inability to comply with the required actions related to Conditions A, B, and E.

Assessment: In the BWROG TR, it was determined that the specific failure condition of interest is not risk significant in BWR PRAs. The reduced end state would only be applicable to the situation where the vacuum breaker(s) in one line are inoperable for opening, with the remaining operable vacuum breakers capable of providing the necessary vacuum relief function. The existing end state remains unchanged, as established by new Condition F, for conditions involving more than one inoperable line or vacuum breaker, since they are needed in Modes 1, 2, and 3. By remaining in Mode 3, HPCI, the IC, and the power conversion systems remain available to ensure adequate core cooling. Therefore, defense-in-depth is improved with respect to water makeup and decay heat removal by remaining in Mode 3. In addition, the plant EOPs direct the operators to take control of the depressurization function if low pressure injection/spray is needed for reactor coolant makeup and cooling.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Action D.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.6 LCO 3.6.1.8: Suppression Chamber-to-Drywell Vacuum Breakers

The function of the suppression chamber-to-drywell vacuum breakers is to relieve vacuum in the drywell, thereby preventing an excessive negative differential pressure across the wetwell/drywell boundary.

LCO: Nine suppression chamber-to-drywell vacuum breakers shall be OPERABLE for opening, AND twelve suppression chamber-to-drywell vacuum breakers shall be closed.

Condition Requiring Entry into End State: If one suppression chamber-to-drywell vacuum breaker is inoperable for opening, the vacuum breaker must be returned to operability within 72 hours (Required Action A.1). If the vacuum breaker cannot be returned to operability within the allotted time, the plant must be placed in Mode 3 within 12 hours (Required Action C.1) and in Mode 4 within 36 hours (Required Action C.2).

Modification for End State Required Actions: A new Condition B modifies the Required Actions so that if vacuum breaker(s) cannot be returned to operable status within the required Completion Time, the plant is placed in hot shutdown. A Note is added to the TS Required Action B.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3. Existing Conditions B and C are renumbered to C and D without changing the Required Action end state. Condition D would require shutting down the plant to Mode 3 (Required Action D.1) and Mode 4 (Required Action D.2), to address an inability to comply with the required actions related to Condition C.

Assessment: In the BWROG TR, it was determined that the specific failure condition of interest is not risk significant in BWR PRAs. The reduced end state would only be applicable to the situation where the vacuum breaker(s) in one line are inoperable for opening, with the remaining operable vacuum breakers capable of providing the necessary vacuum relief function. In the NRC staff's TR SE, the staff concluded that by remaining in Mode 3, HPCI, the IC, and the power conversion systems remain available to ensure adequate core cooling, maintaining defense-in-depth. In addition, the plant EOPs direct the operators to take control of the depressurization function if low pressure injection/spray is needed for reactor coolant makeup and cooling.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Action B.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.7 LCO 3.6.2.3: Suppression Pool Cooling

Some means must be provided to remove heat from the suppression pool so that the temperature inside the primary containment remains within design limits. This function is provided by two suppression pool cooling subsystems.

LCO: Two suppression pool cooling subsystems shall be OPERABLE.

Condition Requiring Entry into End State: If one suppression pool cooling subsystem is inoperable (Condition A), it must be restored to operable status within 7 days (Required Action A.1). If two suppression pool cooling subsystem are inoperable (Condition B), one suppression pool cooling system must be restored to operable status within 8 hours (Required Action B.1). If the suppression pool cooling subsystem cannot be restored to operable status within the allotted time (Condition C), the plant must be placed in Mode 3 within 12 hours (Required Action C.1), and in Mode 4 within 36 hours (Required Action C.2).

Modification for End State Required Actions: A new Condition B modifies the Required Actions so that, if the Required Action and associated Completion Time of Condition A are not met, the plant is placed in hot shutdown. A Note is added to the TS Required Action B.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3. Existing Conditions B and C are renumbered to C and D without changing the Required Action End State. Condition D would require shutting down the plant to Mode 3 (Required Action D.1) after 12 hours, and to Mode 4 (Required Action D.2) after 36 hours, to address an inability to comply with the required actions related to Condition C.

Assessment: In the BWROG TR, a comparative PRA evaluation of the core damage risks of operation in the Mode 3 end state was completed. The results described in the BWROG TR, and evaluated by the NRC staff in the TR SE, indicated that the core damage risks while operating in Mode 3 (assuming the individual failure conditions) are lower or comparable to the current end state. For DNPS, one loop of the suppression pool cooling system is sufficient to accomplish the required safety function. By remaining in Mode 3, HPCI, the IC, and the power conversion systems remain available to ensure adequate core cooling. Additionally, the EOPs direct the operators to take control of the depressurization function if low pressure injection/spray is needed for RCS makeup and cooling.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Action B.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.8 LCO 3.6.2.4: Suppression Pool Spray

Following a design-basis accident (DBA), the suppression pool spray system removes heat from the suppression chamber airspace. A minimum of one suppression pool spray subsystem is required to mitigate potential bypass leakage paths from the drywell and maintain the primary containment peak pressure below the design limits.

LCO: Two suppression pool spray subsystems shall be OPERABLE.

Condition Requiring Entry into End State: If one suppression pool spray subsystem is inoperable (Condition A), it must be restored to operable status within 7 days (Required Action A.1). If both suppression pool spray subsystems are inoperable (Condition B), one subsystem must be restored to operable status within 8 hours (Required Action B.1). If the suppression pool spray subsystem cannot be restored to operable status within the allotted time, the plant must be placed in Mode 3 within 12 hours (Required Action C.1), and in Mode 4 within 36 hours (Required Action C.2).

Modification for End State Required Actions: Required Action C.2 is deleted allowing the plant to stay in Mode 3 while completing repairs. A Note is added to the TS Required Action C.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3.

Assessment: The main function of the suppression pool spray system at DNPS is to remove heat from the suppression chamber so that the pressure and temperature inside primary containment remain within analyzed design limits. The suppression pool spray system was designed to mitigate potential effects of a postulated DBA, that is, a large LOCA which is assumed to occur concurrently with the most limiting single-failure and using conservative inputs, such as for initial suppression pool water volume and temperature. Under the conditions assumed in the DBA, steam blown down from the break could bypass the suppression pool and end up in the suppression chamber air space. The suppression pool spray system could be needed to condense such steam so that the pressure and temperature inside primary containment remain within analyzed design limits. However, the frequency of a DBA is very small and the containment has considerable margin to failure above the design limits. For these reasons, the unavailability of one or both suppression pool spray subsystems has no significant impact on CDF or LERF, even for accidents initiated during operation at power. Therefore, it is very unlikely that the suppression pool spray system will be challenged to mitigate an accident occurring during power operation. This probability becomes extremely unlikely for accidents that would occur during a small fraction of the year (less than 3 days) during which the plant would be in Mode 3 (associated with lower initial energy level and reduced decay heat load as compared to power operation) to repair the failed suppression pool spray system.

Section 6 of the NRC staff's TR SE summarizes the staff's risk argument for approval of the BWROG TR Section 4.5.1.11 and LCO 3.6.2.4, "Residual Heat Removal (RHR) Suppression Pool Spray," for a representative BWR-4 type plant. The NRC staff reviewed the updated final safety analysis report (UFSAR) and TS Bases for DNPS's Suppression Pool Spray System and determined that the system's design function is similar to the BWR-4's RHR Suppression Pool Spray System, and that the end state changes to both plant types' LCOs are the same. Therefore, the NRC staff determined that the risk arguments documented in Section 6 of NRC staff's TR SE can be applied to the DNPS Suppression Pool Spray System. Per the NRC staff's TR SE, the argument for staying in Mode 3 instead of going to Mode 4 to repair the RHR Suppression Pool Spray system is also supported by defense-in-depth considerations. Section 5.2 of the TR SE makes a comparison between the Mode 3 and the Mode 4 end states, with respect to the means available to perform critical functions (i.e., functions contributing to the defense-in-depth philosophy) whose success is needed to prevent core damage and containment failure and mitigate radiation releases, and precludes the need for RHR suppression spray subsystems.

In addition, the probability of a DBA (large-break) is much smaller during shutdown as compared to power operation. A DBA in Mode 3 would be considerably less severe than a DBA occurring during power operation, since Mode 3 is associated with a lower initial energy level and a reduced decay heat load. Under these extremely unlikely conditions, an alternate method that can be used to remove heat from the primary containment (in order to keep the pressure and temperature within the analyzed design basis limits) is containment venting. For more realistic accidents that could occur in Mode 3, several alternate means are available to remove heat from the primary containment, such as the RHR system in the suppression pool cooling mode and the containment spray mode. The risk and defense-in-depth arguments used according to the "integrated decision-making" process of RGs 1.174 and 1.177, support the conclusion that Mode 3 is as safe as Mode 4 for repairing an inoperable RHR suppression spray system.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Action B.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.9 LCO 3.6.4.1: Secondary Containment

Following a DBA, the function of the secondary containment is to contain, dilute, and stop radioactivity (mostly fission products) that may leak from primary containment. Its leak tightness is required to ensure that the release of radioactivity from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated by the Standby Gas Treatment (SGT) System prior to discharge to the environment.

LCO: The secondary containment shall be OPERABLE.

Condition Requiring Entry into End State: If the secondary containment is inoperable, it must be restored to operable status within 4 hours (Required Action A.1). If it cannot be restored to operable status within the allotted time, the plant must be placed in Mode 3 within 12 hours (Required Action B.1), and in Mode 4 within 36 hours (Required Action B.2).

Modification for End State Required Actions: Required Action B.2 is deleted allowing the plant to stay in Mode 3 while completing repairs. A Note is added to the TS Required Action B.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3.

Assessment: This LCO entry condition does not include gross leakage through an unisolable release path. In the BWROG TR, it was concluded that previous generic PRA work related to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," requirements has shown that containment leakage is not risk significant. The primary containment, and all other primary and secondary containment-related functions would still be operable, including the SGT system, thereby minimizing the likelihood of an unacceptable release. In addition, by remaining in Mode 3, HPCI, the IC, and the power conversion systems remain available to ensure adequate core cooling, improving defense-in-depth compared with transitioning to Mode 4.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Action B.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.10 LCO 3.6.4.3: SGT System

The function of the SGT system is to ensure that radioactive materials that leak from the primary containment into the secondary containment following a DBA are filtered and adsorbed prior to exhausting to the environment.

LCO: Two SGT subsystems shall be OPERABLE.

Condition Requiring Entry into End State: If one SGT subsystem is inoperable, it must be restored to operable status within 7 days (Required Action A.1). If the SGT subsystem cannot be restored to operable status within the allotted time, the plant must be placed in Mode 3 within 12 hours (Required Action B.1) and in Mode 4 within 36 hours (Required Action B.2). In addition, if two SGT subsystems are inoperable in Mode 1, 2, or 3, (Condition D), and one SGT system can not be restored to operable within 1 hour, then the plant must be placed in Mode 3 within 12 hours (Required Action E.1) and in Mode 4 within 36 hours (Required Action E.2).

Modification for End State Required Actions: Required Action B.2 and E.2 are deleted allowing the plant to stay in Mode 3 while completing repairs. A Note is added to TS Required Actions B.1 and E.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3.

<u>Assessment</u>: The unavailability of one or both SGT subsystems has no impact on CDF or LERF, irrespective of the mode of operation at the time of the accident. Furthermore, the challenge frequency of the SGT system (i.e., the frequency with which the system is expected to be challenged to mitigate offsite radiation releases resulting from materials that leak from the primary to the secondary containment above TS limits) is less than 1.0E-6/yr. Consequently, the conditional probability that this system will be challenged during the repair time interval while the plant is at either the current or the proposed end state (i.e., Mode 4 or Mode 3, respectively) is less than 1.0E-8. This probability is considerably smaller than probabilities considered "negligible" in RG 1.177 for much higher consequence risks, such as large early release.

Section 5.2 of the NRC staff's TR SE makes a comparison between the Mode 3 and the Mode 4 end states, with respect to the means available to perform critical functions (i.e., functions contributing to the defense-in-depth philosophy) whose success is needed to prevent core damage and containment failure and mitigate radiation releases. Section 6 of of the NRC staff's TR SE summarizes the NRC staff's risk argument for approval of TR Sections 4.5.1.13 and 4.5.2.11, and LCO 3.6.4.3. According to this evaluation, which applies to BWR plant types (including DNPS's BWR-3 design), staying in Mode 3 instead of going to Mode 4 to repair the SGT system (one or both trains) is also supported by defense-in-depth considerations. The risk and defense-in-depth arguments, used according to the "integrated decision-making" process of RGs 1.174 and 1.177, support the conclusion that Mode 3 is as safe as Mode 4 for repairing an inoperable SGT system.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Actions B.1 and E.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.11 LCO 3.7.1: Containment Cooling Service Water (CCSW) System

The design bases of the CCSW system are as follows:

a. To provide the containment cooling function to meet containment capability requirements;

b. To provide redundancy in critical components to meet reliability requirements;

c. To operate without reliance upon external sources of power; and

d. To provide for periodic testing and inspection of each component in the system to demonstrate system availability.

<u>LCO:</u> Two CCSW subsystems shall be OPERABLE.

<u>Condition Requiring Entry into End State:</u> If the LCO cannot be met, the following actions must be taken for the listed conditions:

a. If one CCSW pump is inoperable (Condition A), it must be restored to operable status within 30 days (Required Action A.1).

b. If one CCSW pump in each subsystem is inoperable (Condition B), one CCSW pump must be restored to operable status within 7 days (Required Action B.1).

c. If one CCSW subsystem is inoperable for reasons other than Condition A (Condition C), the CCSW subsystem must be restored to operable status within 7 days (Required Action C.1).

d. If both CCSW subsystems are inoperable for reasons other than Condition B (Condition D), then one CCSW subsystem must be restored to operable within 8 hours.

e. If the required action and associated completion time cannot be met within the allotted time (Condition E), the plant must be placed in Mode 3 within 12 hours (Required Action E.1) and in Mode 4 within 36 hours (Required Action E.2).

Modification for End State Required Actions: A new Condition D is added which establishes requirements for existing Conditions A, B, and C, that are similar to existing Condition E, but without Required Action E.2. A Note is added to the Required Actions D.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3. The existing Required Conditions D and E are renumbered to Conditions E and F, respectively. The renumbered Required Condition F now applies to renumbered Condition E, which maintains the existing requirements with respect to both CCSW subsystems being inoperable for reasons other than Condition B.

Assessment: In the BWROG TR a comparative PRA evaluation of the core damage risks of operation in the current end state versus operation in the Mode 3 end state was performed using the model BWR-4 design, which includes the RHR Service Water (RHRSW) System. The NRC staff reviewed the UFSAR and TS Bases for DNPS's CCSW system and determined that, because the system's design function is similar to the BWR-4's RHRSW system and the end state changes to both plant types' LCOs are the same, the conclusions reached by the NRC staff in its TR SE for the RHRSW system can be used to evaluate the CCSW system. The NRC staff conclusion described in its TR SE on the BWROG's PRA evaluation, indicates that the core damage risks are lower in Mode 3 than in Mode 4, the current end state. By remaining in Mode 3, HPCI, the IC, and the power conversion systems remain available to ensure adequate core cooling. Additionally, the EOPs direct the operators to take control of the depressurization function if low pressure injection/spray are needed for RCS makeup and cooling. Therefore, defense-in-depth is improved with respect to water makeup and decay heat removal by remaining in Mode 3, and the required safety function can still be performed with the CCSW subsystem components that are still operable.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Actions D.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.12 LCO 3.7.4: Control Room Emergency Ventilation (CREV) System

The CREV system provides a protected environment from which the plant can be safely operated following an uncontrolled release of radiation, hazardous chemicals, or smoke.

LCO: The CREV system shall be OPERABLE.

Condition Requiring Entry into End State: If the CREV system is inoperable in Mode 1, 2, or 3 for reasons other than an inoperable control room envelope boundary in Mode 1, 2, or 3, (Condition B), it must be restored to operable status within 7 days (Required Action A.1). If the

CREV system is inoperable in Mode 1, 2, or 3 due to inoperable control room envelope boundary, initiate mitigating actions immediately (Required Action B.1), and follow Required Actions B.2 and B.3 within 24 hours and 90 days, respectively. If the CREV system cannot be restored to operable status within the allotted time for Condition A or B, the plant must be placed in Mode 3 within 12 hours (Required Action C.1) and in Mode 4 within 36 hours (Required Action C.2).

Modification for End State Required Actions: The change adds a new Condition B with Required Action B.1 to be in Mode 3 within 12 hours when Required Action and associated Completion Time of Condition A are not met in Mode 1, 2, or 3. The change renumbers old Conditions B, C and D to Conditions C, D and E. The renumbered Condition D would require that if the CREV system cannot be restored to operable status within the allotted time for renumbered Condition C, the plant must be placed in Mode 3 within 12 hours (Required Action D.1) and in Mode 4 within 36 hours (Required Action D.2). A Note is added to the new Required Actions B.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3.

Assessment: The unavailability of the CREV system has no significant impact on CDF or LERF, irrespective of the mode of operation at the time of the accident. Furthermore, the challenge frequency of the CREV system (i.e., the frequency with which the system is expected to be challenged to provide a radiologically-controlled environment in the main control room following a DBA which leads to core damage and leaks of radiation from the containment that can reach the control room) is less than 1.0E-6/yr. Consequently, the conditional probability that this system will be challenged during the repair time interval while the plant is at either the current or the proposed end state (i.e., Mode 4 or Mode 3, respectively) is less than 1.0E-8. This probability is considerably smaller than probabilities considered "negligible" in RG 1.177 for much higher consequence risks, such as large early release.

Section 5 of the NRC staff's TR SE makes a comparison between the Mode 3 and the Mode 4 end states, with respect to the means available to perform critical functions (i.e., functions contributing to the defense-in-depth philosophy) whose success is needed to prevent core damage and containment failure and mitigate radiation releases. Section 6 of the TR SE summarizes the NRC staff's risk argument for approval of TR Section 4.5.1.16, and LCO 3.7.4, "Main Control Room Environmental Control (MCREC) System." (Note: The Control Room Emergency Ventilation system at DNPS serves a similar design purpose as the MCREC described in NUREG-1433, LCO 3.7.4). The argument for staying in Mode 3 instead of going to Mode 4 end state to repair the system is also supported by defense-in-depth considerations. The risk and defense-in-depth arguments, used according to the "integrated decision-making" process of RGs 1.174 and 1.177, support the conclusion that Mode 3 is as safe as Mode 4 for repairing an inoperable CREV system.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Actions B.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.13 LCO 3.7.5: Control Room Emergency Ventilation Air Conditioning System

The Control Room Emergency Ventilation Air Conditioning System provides temperature control for the control room following control room isolation during accident conditions.

LCO: The Control Room Emergency Ventilation Air Conditioning System shall be OPERABLE.

Condition Requiring Entry into End State: If the Control Room Emergency Ventilation Air Conditioning System is inoperable, the system must be restored to operable status within 30 days (Required Action A.1). If the required actions and associated completion times for Condition A cannot be met, the plant must be placed in Mode 3 within 12 hours (Required Action B.1) and in Mode 4 within 36 hours (Required Action B.2).

Modification for End State Required Actions: Required Action B.2 is deleted allowing the plant to stay in Mode 3 while completing repairs. A Note is added to the Required Action B.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3.

Assessment: The unavailability of the Control Room Emergency Ventilation Air Conditioning System has no significant impact on CDF or LERF, irrespective of the mode of operation at the time of the accident. Furthermore, the challenge frequency of the air conditioning system (i.e., the frequency with which the system is expected to be challenged to provide temperature control for the control room following control room isolation following a DBA) is less than 1.0E-6/yr. Consequently, the conditional probability that this system will be challenged during the repair time interval while the plant is at either the current or the proposed end state (i.e., Mode 4 or Mode 3, respectively) is less than 1.0E-8. This probability is considerably smaller than probabilities considered "negligible" in RG 1.177 for much higher consequence risks, such as large early release.

Section 5 of the NRC staff TR SE makes a comparison between the Mode 3 and Mode 4 end states, with respect to the means available to perform critical functions (i.e., functions contributing to the defense-in-depth philosophy) whose success is needed to prevent core damage and containment failure and mitigate radiation releases. Section 6 of the TR SE summarizes the NRC staff's risk argument for approval of TR Section 4.5.1.17, and LCO 3.7.5, "Control Room Air Conditioning (AC) System," (which functions similar to DNPS's Control Room Emergency Ventilation Air Conditioning System). Staying in Mode 3 instead of transitioning to the Mode 4 end state to repair the Control Room Emergency Ventilation Air Conditioning System is also supported by defense-in-depth considerations. The risk and defense-in-depth arguments, used according to the "integrated decision-making" process of RGs 1.174 and 1.177, support the conclusion that Mode 3 is as safe as Mode 4 for repairing an inoperable Control Room Emergency Ventilation Air Conditioning System.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Actions B.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.14 LCO 3.7.6: Main Condenser Offgas

The offgas from the main condenser normally includes radioactive gases. The gross gamma activity rate is controlled to ensure that accident analysis assumptions are satisfied and that offsite dose limits will not be exceeded during postulated accidents. The Main Condenser Offgas (MCOG) gross gamma activity rate is an initial condition of a DBA which assumes a gross failure of the MCOG system pressure boundary.

<u>LCO:</u> The gross gamma activity rate of the noble gases measured prior to the offgas holdup line shall be ≤252,700 μCi/second after decay of 30 minutes.

<u>Condition Requiring Entry into End State</u>: If the gross gamma activity rate of the noble gases in the MCOG system is not within limits, the gross gamma activity rate of the noble gases in the MCOG system must be restored to within limits within 72 hours (Required Action A.1). If the required action and associated completion time cannot be met, one of the following must occur:

a. All main steam lines must be isolated within 12 hours (Required Action B.1).

b. The steam jet air ejector must be isolated within 12 hours (Required Action B.2).

c. The plant must be placed in Mode 3 within 12 hours (Required Action B.3.1) and in Mode 4 within 36 hours (Required Action B.3.2).

<u>Modification for End State Required Actions:</u> Required Action B.3.2 is deleted allowing the plant to stay in Mode 3 while completing repairs. Required Action B.3.1 is renumbered to Required Action B.3 and a Note is added to the TS Required Action B.3 stating that LCO 3.0.4.a is not applicable when entering Mode 3.

<u>Assessment:</u> The failure to maintain the gross gamma activity rate of the noble gases in the MCOG within limits has no significant impact on CDF or LERF, irrespective of the mode of operation at the time of the accident. Furthermore, the challenge frequency of the MCOG system (i.e., the frequency with which the system is expected to be challenged to mitigate offsite radiation releases following a DBA) is less than 1.0E-6/yr. Consequently, the conditional probability that this system will be challenged during the repair time interval while the plant is at either the current or the proposed end state (i.e., Mode 4 or Mode 3, respectively) is less than 1.0E-8. This probability is considerably smaller than probabilities considered "negligible" in RG 1.177 for much higher consequence risks, such as large early release.

Section 5 of the NRC staff's TR SE makes a comparison between the Mode 3 and the Mode 4 end states, with respect to the means available to perform critical functions (i.e., functions contributing to the defense-in-depth philosophy) whose success is needed to prevent core damage and containment failure and mitigate radiation releases. Section 6 of the TR SE summarizes the NRC staff's risk argument for approval of TR Section 4.5.1.18 and LCO 3.7.6. The argument for staying in Mode 3 instead of going to Mode 4 end state to repair the MCOG system (one or both trains) is also supported by defense-in-depth considerations. The risk and defense-in-depth arguments, used according to the "integrated decision-making" process of RGs 1.174 and 1.177, support the conclusion that Mode 3 is as safe as Mode 4 for repairing an inoperable MCOG system.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Actions B.3.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.15 LCO 3.8.1: Alternating Current (AC) Sources - Operating

The purpose of the AC electrical system is to provide the power required to put and maintain the plant in a safe condition and prevent the release of radioactivity to the environment under normal and accident conditions. DNPS is a two unit facility, with an onsite AC power system that consists of two main generators, two main step-up transformers, two unit auxiliary transformers (UATs), two reserve auxiliary transformers (RATs) one 345/138-kV auto transformer, distribution buses, and three standby diesel generators (DGs). The distribution system has nominal ratings of 4160-V, 480-V, and 120/208-V.

DNPS's offsite AC power system supplies power to the onsite auxiliary power system through two RATs (one per unit). The UAT is also available to provide offsite power when back fed from the grid with the main generator offline. In the event of a loss of offsite power, the emergency electrical loads are automatically connected to the operating DGs in sufficient time to provide for a safe reactor shutdown and to mitigate the consequence of a DBA, such as a LOCA.

LCO: The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class1E AC Electric Power Distribution System.

b. Two DGs

c. One qualified circuit between the offsite transmission network and the opposite unit's onsite Class 1E AC Electrical Power Distribution System capable of supporting the equipment required to be OPERABLE by LCO 3.6.4.3., LCO 3.7.4 (Unit 3 only), and LCO 3.7.5. (Unit 3 only); and;

d. The opposite unit's DG capable of supporting the equipment required to be OPERABLE by LCO 3.6.4.3., LCO 3.7.4 (Unit 3 only), and LCO 3.7.5 (Unit 3 only).

Condition Requiring Entry into End State: Plant operators must bring the plant to Mode 3 within 12 hours and Mode 4 within 36 hours following the sustained inoperability of either or both required offsite circuits; one or two required DGs; or one required offsite circuit and one required DG.

Modification for End State Required Actions: Required Action F.2 is deleted allowing the plant to stay in Mode 3 while completing repairs. The plant will remain in Mode 3 (Required Action F.1). A Note is added to the TS Required Action F.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3.

Assessment: Entry into any of the conditions for the AC power sources implies that the AC power sources have been degraded and the single-failure protection for the safe shutdown equipment may be ineffective. Consequently, as specified in TS 3.8.1 at present, the plant operators must bring the plant to Mode 4 when the required action is not completed by the specified time for the associated action.

In the BWROG TR, a comparative PRA evaluation of the core damage risks of operation in the current end state and in the Mode 3 end state was performed. Events initiated by the loss of offsite power are dominant contributors to CDF in most BWR PRAs and the IC and HPCI play major roles in mitigating these events. In the NRC staff's TR SE, the NRC staff concluded that, based on the low probability of loss of the AC power and the number of systems available in Mode 3, the core damage risks are lower in Mode 3 than in Mode 4 for one inoperable AC power source. For DNPS, going to Mode 4 for one inoperable AC power source would cause loss of IC or HPCI, and loss of the power conversion system, and requires activating the RHR system.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Action for F.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.16 LCO 3.8.4: Direct Current (DC) Sources - Operating

The purpose of the DC power system is to provide a reliable source of DC power for both normal and abnormal conditions. It must supply power in an emergency for an adequate length of time until normal supplies can be restored.

LCO: For Modes 1, 2, and 3, The following DC electrical power subsystems shall be OPERABLE:

a. Two 250 VDC electrical power subsystems;

b. Division 1 and Division 2 125 VDC electrical power subsystems; and

c. The opposite unit's 125 VDC electrical power subsystem capable of supporting equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," LCO 3.7.4. "Control Room Emergency Ventilation (CREV) System" (Unit 3 only), LCO 3.7.5 "Control Room Emergency Ventilation Air Conditioning (AC) System" (Unit 3 only), and LCO 3.8.1 "AC Sources- Operating."

Condition Requiring Entry into End State: The plant operators must bring the plant to Mode 3 within 12 hours (Required Action J.1) and Mode 4 within 36 hours (Required Action J.2) following the sustained inoperability of (a) one 250 VDC electrical subsystem, or (b) Division 1 or 2, DC electrical power subsystem, or (c) opposite unit's 125 VDC electrical power subsystem.

Modification for End State Required Actions: Required Action J.2 is deleted, allowing the plant to stay in Mode 3 while Conditions A thru I are not met. A Note is added to the TS Required Action J.1 stating that LCO 3.0.4(a) is not applicable when entering Mode 3.

Assessment: If one of the DC electrical power subsystems is inoperable, the remaining DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate an accident condition. In the BWROG TR, a comparative PRA evaluation of the core damage risks of operation in the Mode 3 and Mode 4 end states was performed, assuming one DC system inoperable. Events initiated by the loss of offsite power are dominant contributors to CDF in most BWR PRAs, and the IC and HPCI systems play major roles in mitigating these events. In the NRC staff's TR SE, the staff concluded that the core damage risks are lower in Mode 3 than in Mode 4. For DNPS, going to Mode 4 for one inoperable DC power source would cause loss of the IC and HPCI systems, and the power conversion systems (condenser/feedwater), and would require activating the RHR system.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Action J.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.3.17 LCO 3.8.7: Distribution Systems-Operating

The onsite Class 1E AC electrical power distribution system is divided into redundant and independent AC electrical power distribution systems for each unit. The primary AC electrical power distribution subsystem for each unit consists of two 4.16-kV Essential Service System (ESS) buses having an offsite source of power as well as a dedicated onsite DG source. The secondary plant distribution subsystems include 480-VAC ESS buses and associated load centers, motor control centers, distribution panels and transformers.

The 120-VAC vital buses are arranged in three different subsystems: 120V reactor protection system, 120V instrumentation bus system, and 120V ESS. The 120V ESS bus is supplied by a static uninterruptible power supply. There are two independent 250 VDC station service electrical power distribution subsystems, one for each unit, and two independent 125 VDC electrical power distribution subsystems, one for each unit.

LCO: For Modes 1, 2, and 3, the following electrical power distribution subsystems shall be OPERABLE:

a. Division I and Division 2 AC and DC electrical power distribution subsystems; and

b. The portions of the opposite unit's Division 2 AC and DC electrical power distribution subsystem necessary to support equipment required to be OPERABLE by LCO 3.6.4.3., LCO 3.7.4 (Unit 3 only), LCO 3.7.5 (Unit 3 only), and LCO 3.8.1.

Condition Requiring Entry into End State: The plant operators must bring the plant to Mode 3 within 12 hours and Mode 4 within 36 hours following the sustained inoperability of one or more AC, DC, or one or more required opposite unit AC or DC electrical power distribution subsystems inoperable for a period of 8 hours, 2 hours and 7 days, respectively (with a maximum 16-hour Completion Time limit from initial discovery of failure to meet the LCO, to preclude being in the LCO indefinitely).

Modification for End State Required Actions: The TS change is to remove the requirement to place the plant in Mode 4 (Required Action D.2 is deleted). A Note is added to the TS Required Action D.1 stating that LCO 3.0.4.a is not applicable when entering Mode 3.

Assessment: If one of the AC/DC/AC ESS is inoperable, the remaining AC/DC/AC ESS subsystems have the capacity to support a safe shutdown and to mitigate an accident condition. In the BWROG TR, the BWROG performed a comparative PRA evaluation of the core damage risks of operation in the current end state and in the Mode 3 end state, with one of the AC/DC/AC ESS subsystems inoperable. Events initiated by the loss of offsite power are dominant contributors to CDF in most BWR PRAs, and the high pressure core cooling systems, HPCI or IC, play a major role in mitigating these events. In the NRC staff's TR SE, the staff concluded that the core damage risks are lower in Mode 3 than in Mode 4. For DNPS, going to Mode 4 would cause loss of HPCI, the IC, and the power conversion systems due to the plant cooldown, and would require activating the RHR system.

The Note "LCO 3.0.4.a is not applicable when entering Mode 3" in TS Required Action D.1 prevents an inappropriate use of the LCO 3.0.4.a allowance to go up in Mode with inoperable systems or equipment.

3.4. Overall Assessment of Proposed Technical changes:

Based upon the above assessments, and because the time spent in Mode 3 to perform the repair on any of the systems described above would be infrequent and limited, and in light of defense-in-depth considerations (discussed above and in the BWROG TR, and as evaluated by the NRC staff's TR SE), the NRC staff concludes the changes to the DNPS TSs described above are acceptable.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves 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 adopting TSTF-423, Rev 0, involves no significant hazards considerations, and there has been no public comment on the finding in *Federal Register* Notice 70 FR 74037, December 14, 2005. 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 amendment.

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.

7.0 REFERENCES

1. NEDC-32988-A, Revision 2, "Technical Justification to Support Risk-Informed Modification to Selected Required Action End States for BWR Plants," December 2002. (ADAMS Accession No. ML030170084).

2. Federal Register, Vol. 58, No. 139, p. 39136, "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Plants," July 22, 1993 (58 FR 39132).

3. 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

4. Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," May 2000 (ADAMS Accession No. ML003699426).

5. Regulatory Guide 1.177, "An Approach for Plant Specific, Risk-Informed Decision Making: Technical Specifications," USNRC, August 1998. (ADAMS Accession No. ML003740176).

6. NRC Safety Evaluation for Topical Report NEDC-32988, Revision 2, September 27, 2002. (ADAMS Accession No. ML022700603).

7. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision 2, April 1996.

8. TSTF-423, Revision 0, "Technical Specifications End States, NEDC-32988-A." (ADAMS Accession No. ML032270250).

9. TSTF-IG-05-02, "Implementation Guidance for TSTF-423, Revision 0, 'Technical Specifications End States,' NEDC-32988-A," September 2005. (ADAMS Accession No. ML052700156).

10. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decision Making on Plant Specific Changes to the Licensing Basis," USNRC, August 1998. (ADAMS Accession No. ML003740133).

Principal Contributors: R. P. Grover, NRR
K. Bucholtz, NRR

Date: October 20, 2009

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

October 20, 2009

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS (TAC NOS. MD7202 AND MD7203)

Dear Mr. Pardee:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 233 to Renewed Facility Operating License No. DPR-19 and Amendment No. 226 to Renewed Facility Operating License No. DPR-25 for Dresden Nuclear Power Station, Units 2 and 3. The amendments are in response to your application dated October 9, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072830088), as supplemented by letter dated January 30, 2009 (ADAMS Accession No. ML090350151).

The amendments would modify the technical specifications to risk-inform requirements regarding selected required action end states as provided in Technical Specification Task Force (TSTF) change traveler TSTF-423, Revision 0, "Technical Specifications End States, NEDC-32988-A, Revision 2."

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. 50-237 and 50-249

Enclosures:
1. Amendment No. 233 to DPR-19
2. Amendment No. 226 to DPR-25
3. Safety Evaluation

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R.P. Grover, NRR
K. Bucholtz, NRR

ADAMS Accession No. : ML092510152 *SE dated

OFFICE	LPL3-2/PM	LPL3-2/LA	DSS/SRXB	DRA/APOB (A)	DE/EEEB	ITSB	OGC	LPL3-2/BC
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