September 27, 2007

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS RE: REQUIRED ACTION END STATES CONSISTENT WITH TECHNICAL SPECIFICATION TASK FORCE TSTF-423 (TAC NOS. MD3581 AND MD3582)

Dear Mr. Crane:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 184 to Facility Operating License No. NPF-11 and Amendment No. 171 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively. The amendments are in response to your application dated October 18, 2006, as supplemented by letter dated, March 26, 2007.

The amendments modify the technical specifications (TS) to risk-inform requirements regarding selected Required Action End States consistent with the Nuclear Regulatory Commission (NRC) approved industry and TS Task Force No. 423 (TSTF-423), Revision 0, "Technical Specifications End States, NEDC-32988-A." TSTF-423 was published in the *Federal Register* on March 23, 2006, as part of the consolidated line item improvement process.

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/

Stephen P. Sands, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

Enclosures:
1. Amendment No. 184 to NPF-11
2. Amendment No. 171 to NPF-18
3. Safety Evaluation

cc w/encls: See next page

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LaSalle County Station, Units 1 and 2

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via e-mail

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EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 184
License No. NPF-11

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

   A. The application for amendment filed by the Exelon Generation Company, LLC (the licensee), dated October 18, 2006, as supplemented by letter dated, March 26, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 184, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: September 27, 2007

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 171
License No. NPF-18

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

   A. The application for amendment filed by the Exelon Generation Company, LLC (the licensee), dated October 18, 2006, as supplemented by letter dated, March 26, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 171, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: September 27, 2007

ATTACHMENT TO LICENSE AMENDMENT NOS. 184 AND 171

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

DOCKET NOS. 50-373 AND 50-374

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

Remove	Insert
License NPF-11	License NPF-11
Page 3	Page 3
License NPF-18	License NPF-18
Page 3	Page 3
TSs	TSs
3.3.8.2-2	3.3.8.2-2
3.5.1-2	3.5.1-2
3.5.1-3	3.5.1-3
3.6.1.1-1	3.6.1.1-1
3.6.1.6-1	3.6.1.6-1
3.6.1.6-2	3.6.1.6-2
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
3.7.1-2	3.7.1-2
3.7.1-3	3.7.1-3
3.7.4-1	3.7.4-1
3.7.4-2	3.7.4-2
3.7.5-1	3.7.5-1
3.7.5-2	3.7.5-2
3.7.6-1	3.7.6-1
3.8.1-7	3.8.1-7
3.8.1-8	3.8.1-8
3.8.4-3	3.8.4-3
3.8.7-2	3.8.7-2
3.8.7-3	3.8.7-3

(4) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

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

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

(1) Maximum Power Level

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 184, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

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

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

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

(1) Maximum Power Level

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 171, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 184 TO FACILITY OPERATING LICENSE NO. NPF-11

AND AMENDMENT NO. 171 TO FACILITY OPERATING LICENSE NO. NPF-18

EXELON GENERATION COMPANY, LLC

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated October 18, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML062980149) as supplemented by letter dated March 26, 2007 (ADAMS Accession No. ML070860068), Exelon Generation Company, LLC (the licensee), requested changes to the technical specifications (TSs), for LaSalle County Station, Units 1 and 2. The requested changes are the adoption of TS Task Force No. 423 (TSTF-423), Revision 0, “Technical Specification End States, NEDC-32988-A,” to the boiling-water reactor (BWR) standard technical specifications (STS) (NUREG 1433 and NUREG 1434), which was proposed by the Nuclear Energy Institute Risk Informed TSTF on August 12, 2003, on behalf of the industry. TSTF-423, Revision 0, incorporates the BWR Owners Group (BWROG) approved Topical Report NEDC-32988, Revision 2, “Technical Justification to Support Risk Informed Modification to Selected Required Action End States for BWR Plants” (Reference 1), into the BWR STS (NOTE: The changes are made with respect to Revision 2 of the STS NUREGs). The supplement provided the licensee’s regulatory commitment to follow the industry on implementation guidance associated with TSTF-423 (Reference 11). This TSTF was published in the *Federal Register* on March 23, 2006, as part of the consolidated line item improvement process.

TSTF-423 is one of the industry’s initiatives developed under the risk management TS program. These initiatives are intended to maintain or improve safety through the incorporation of risk assessment and management techniques in TS, while reducing unnecessary burden and making TS requirements consistent with the Commission’s other risk-informed regulatory requirements, in particular the maintenance rule.

Title 10 of the *Code of Federal Regulations*, (10 CFR) Section 50.36(c)(2)(i), “Technical Specifications,” states: “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.” The STS and many plant TS provide a completion time (CT) for the plant to meet the limiting condition for operation (LCO). If the LCO or the remedial action cannot be met, then the reactor is required to be shut down. When the STS and individual plant TSs were written, the shutdown condition or end state specified was usually cold shutdown.

Topical Report NEDC-32988, Revision 2, provides the technical basis to change certain required end states when the TS Actions for remaining in power operation cannot be met within the CTs. Most of the requested 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 TS. The request was 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 STS for BWR plants define five operational modes. In general, they are:

- Mode 1 - Power Operation: The reactor mode switch is in 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 200 °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 200 °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 shutdown or refuel position, and one or more reactor vessel head closure bolts are less than fully tensioned. Criticality is not allowed in Modes 3 through 5.

TSTF-423 generally allows a Mode 3 end state rather than a Mode 4 end state for selected initiating conditions in order to perform short-duration repairs which necessitate exiting the original Mode of operation. Short duration repairs are on the order of 2 to 3 days, but not more than a week.

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 five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TS. As stated in 10 CFR 50.36(c)(2)(i), the "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications ... ." Reference 1 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 steam-driven 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 here, a problem with a component or train has or will result in a failure to meet a TS, and a controlled shutdown has begun because a TS Action requirement cannot be met within the TS CT.

Most of today's TS 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, STS were developed and many licensees improved their TS. Since enactment of a shutdown rule was expected, almost all TS changes involving power operation, including a revised end state requirement, were postponed (see, for example the Final Policy Statement on TS Improvements, Reference 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 10 CFR 50.65 (Reference 3), the maintenance rule. Section 50.65(a)(4) states: "Before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventative maintenance), 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 (Reference 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. The revised Section 11 of NUMARC 93-01, Revision 2, was subsequently incorporated into Revision 3 of NUMARC 93-01 (Reference 5). However, Revision 3 has not yet been formally endorsed by the NRC. The changes in TSTF-423 are consistent with the rules, regulations and associated regulatory guidance, as noted above.

3.0 TECHNICAL EVALUATION

The changes proposed in TSTF-423 are consistent with the changes proposed and justified in Topical Report GE NEDC-32988-A, Revision 2, and approved by the associated NRC safety evaluation (SE) (Reference 6). The evaluation included in Reference 6, as appropriate and applicable to the changes of TSTF-423 (Reference 7), is reiterated here and differences from the SE are justified. In its application, the licensee commits to TSTF-IG-05-02, Implementation Guidance for TSTF-423, Revision 0, (Reference 8), 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 Reference 1.

3.1 Risk Assessment

The objective of the BWROG topical report (Reference 1) 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 topical report 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 (Reference 9) and RG 1.177 (Reference 10). The three-tiered approach documented in RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications," was followed. The first tier of the three-tiered approach 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 entitled "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." In addition, the first tier aims at ensuring that there are no unacceptable temporary risk increases during the implementation of the proposed 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 implementation of the proposed TS change. The third tier addresses the application of 10 CFR 50.65(a)(4) of the maintenance rule for identifying risk-significant configurations resulting from maintenance related activities and taking appropriate compensatory measures to avoid such configurations. Unless invoked, by this or another TS application, 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.

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

<u>Risk Impact of the Proposed Change (Tier 1)</u>:

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.

<u>Avoidance of Risk-Significant Configurations (Tier 2)</u>:

The performed risk analyses, which are based on single LCOs, shows 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 the maintenance rule 10 CFR 50.65(a)(4), shall ensure that high-risk configurations are avoided. As part of the implementation of TSTF-423, the licensee commits 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. The NRC staff finds that such guidance is adequate for preventing risk-significant plant configurations.

Configuration Risk Management (Tier 3):

The licensee has a program in place to comply with 10 CFR 50.65(a)(4) to assess and manage the risk from proposed maintenance activities. This program can support a licensee 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 proposed end state mode change was evaluated subject to the following assumptions:

1. The entry into the proposed 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.

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

3. 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, in general, 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.

3.2 Assessment of TS Changes

The changes proposed by the licensee and in TSTF-423 are consistent with the changes proposed in topical report GE NEDC-32988, Revision 2, and approved by the NRC SE of September 27, 2002. Following are the proposed changes, including a synopsis of the STS LCO, the change, and a brief conclusion of acceptability.

3.2.1 TS 3.5, "Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System"

The ECCS systems provide cooling water to the core in the event of a loss-of-coolant accident (LOCA). This set of ECCS TS provide 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 six safety relief valves must be operable.

Conditions requiring entry into end state:

If the LCO cannot be met, the following actions must be taken for the listed conditions:

- Condition A: If one low-pressure ECCS injection/spray subsystem is inoperable, the subsystem must be restored to operable status in 7 days.

- Condition B: If the High-Pressure Core Spray (HPCS) system is inoperable, the Reactor Core Isolation Cooling (RCIC) system must be verified to be operable by administrative means immediately, and the HPCS system restored to operable status within 14 days.

- Condition C: If two low-pressure ECCS injection/spray subsystem are inoperable, one low pressure ECCS injection/spray subsystem must be restored to operable status within 72 hours.

- Condition D: If ADS accumulator backup compressed gas system bottle pressure < 500 psig, ADS accumulator backup compressed gas system bottle pressure must be restored ≥ 500 psig within 72 hours or associated ADS valves must be declared inoperable within 72 hours.

- Condition E: If the Required Action and associated Completion Time of Condition A, B or C is not met, the plant must be placed in Mode 3 within 12 hours and Mode 4 within 36 hours.

- Condition F: If one ADS valve is inoperable, it must be restored to operable status within 14 days.

- Condition G: If Required Action and associated Completion Time of Condition F 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.

Proposed modification for end state required actions:

- Condition A: No change.

- Condition B: No change.

- Condition C: No change.

- Condition D: No change.

- Condition E: If the Required Action and associated Completion Time of Condition A, B or C is not met, the plant must be placed in Mode 3 within 12 hours. The plant is not taken into Mode 4 (cold shutdown).

- Condition F: No change.

- Condition G: Add a new Condition G as follows: If Required Action and associated Completion Time of Condition F is not met, the plant must be placed in Mode 3 within 12 hours. The reactor is not depressurized and not taken to Mode 4.

- Condition H: Renumber original Condition G to H and revise original Condition G by deleting statement "Required Action and associated Completion Time of Condition F not met."

- Condition I: Renumber original Condition H to I.

Assessment:

The BWROG topical report did a comparative PRA evaluation of the core damage risks of operation in the current end state and the proposed Mode 3 end state. The evaluation indicates that the core damage risks are lower in Mode 3 than in the current end state Mode 4. Going to Mode 4 for one ECCS subsystem or one ADS valve would cause loss of the high pressure steam-driven injection system (RCIC/HPCS), and loss of the power conversion system (condenser/feedwater), and require activating the residual heat removal (RHR) system. In addition, plant emergency operating procedures (EOPs) direct the operator to take control of the depressurization function if low-pressure injection/spray systems are needed for reactor pressure vessel (RPV) water makeup and cooling. Based on the low probability of loss of the reactor coolant inventory and the number of systems available in Mode 3, the NRC staff concludes in the SE to the BWR topical report that the risks of staying in Mode 3 are approximately the same as, and in some cases lower than, the risks of going to the Mode 4 end state.

Finding:

Based on the above assessment, and because the time spent in Mode 3 to perform the repair is infrequent and limited, and in light of defense-in-depth considerations, the proposed change is acceptable.

3.2.2 TS 3.3.8.2, "Reactor Protection System (RPS) Electric Power Monitoring"

RPS Electric Power Monitoring System is provided to isolate the RPS bus from the motor generator set or an alternate power supply in the event of over voltage, under voltage, or under frequency. This system protects the load connected to the RPS bus against unacceptable voltage and frequency conditions and forms an important part of the primary success path of the essential safety circuits. Some of the essential equipment powered from the RPS buses includes the RPS logic, scram solenoids, and various valve isolation logic. The TS change allows the plant to remain in Mode 3 until the repairs are completed.

LCO: For (a) Modes 1, 2, 3, (b) Modes 4 and 5 with RHR shutdown cooling isolation valves open, and (c) Mode 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, two RPS electric power monitoring assemblies shall be operable for each in-service RPS motor generator set or alternate power supply.

Condition Requiring Entry into End State:

If the LCO cannot be met, the associated in-service power supply(ies) must be removed from service within 72 hours for one electric power assembly (EPA) inoperable or within 1 hour for both EPAs inoperable. In Modes 1, 2, or 3, if the in-service power supply(ies) cannot be removed from service within the allotted time, the plant must be placed in Mode 3 within 12 hours and Mode 4 within 36 hours.

Proposed Modification:

The proposed change is to keep the plant in Mode 3 until the repair actions are completed. Delete required action in C.2 which required the plant to be in Mode 4.

Assessment:

To reach Mode 3 per the TSs, there must be a functioning power supply with degraded protective circuitry in operation. However, the over voltage, under voltage, or under frequency condition must exist for an extended time period to cause damage. There is a low probability of this occurring in the short period of time that the plant would remain in Mode 3 without this protection. The specific failure condition of interest is not risk significant for BWR PRAs. If the required restoration actions cannot be completed within the specified time, going into Mode 4 would cause loss of the high-pressure steam-driven injection system (RCIC/HPCS) and loss of the power conversion system (condenser/feedwater), and would require activating the RHR system. In addition, 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. Based on the low probability of loss of the RPS power monitoring system during the infrequent and limited time in Mode 3, and the number of systems available in Mode 3, the NRC staff concludes in the SE to the BWR topical report that the risks of staying in Mode 3 are approximately the same as and in some cases lower than the risks of going to the Mode 4 end state.

Finding:

Based upon the above assessment, and because the time spent in Mode 3 to perform the repair is infrequent and limited, and in light of defense-in-depth considerations, the proposed change is acceptable.

3.2.3 TS 3.8.1, "AC Sources (Operating)"

The purpose of the alternate current (AC) electrical system is to provide during all situations the power required to put and maintain the plant in a safe condition and prevent the release of radioactivity to the environment.

The Class 1E electrical power distribution system AC sources consist of the offsite power source (preferred power sources, normal and alternate(s)), and the onsite standby power sources (e.g., emergency diesel generators (EDGs)). In addition, many sites provide a crosstie capability between units. As required by General Design Criterion 17 of 10 CFR Part 50, Appendix A, the design of the AC electrical system provides independence and redundancy.

The onsite Class 1E AC distribution system is divided into redundant divisions so that the loss of any one division does not prevent the minimum safety functions from being performed. Each division has connections to two preferred offsite power sources and a single EDG or other Class 1E Standby AC power source.

Offsite power is supplied to the unit switchyard(s) from the transmission network by two transmission lines. From the switchyard(s), two electrically and physically separated circuits provide AC power through a stepdown transformer(s) to the 4.16-kV emergency buses. In the event of a loss of offsite power, the emergency electrical loads are automatically connected to the EDGs in sufficient time to provide for a safe reactor shutdown and to mitigate the consequence of a design-basis accident (DBA) such as a LOCA.

<u>LCO</u>: The following AC electrical power sources shall be operable in Modes 1, 2, and 3:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electric Power Distribution System;

b. Three diesel generators (DGs); and

c. The opposite unit's Division 2 DG capable of supporting the associated equipment required to be OPERABLE by LCO 3.6.4.3, “Standby Gas Treatment (SGT) System,” LCO 3.7.4, “Control Room Area Filtration (CRAF) System,” and LCO 3.7.5, “Control Room Area Ventilation Air Conditioning (AC) System.”

<u>Condition requiring entry into end state</u>:

Plant operators must bring the plant to Mode 4 within 36 hours following the sustained inoperability of either or both required off site circuits; either one, two or three required EDGs; or one required off site circuit and one, two or three required EDGs.

<u>Proposed modification for end state require actions</u>:

The proposed change is to delete required action G.2 to go to Mode 4 (cold shutdown). The plant will remain in Mode 3 (hot shutdown).

<u>Assessment</u>:

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.

The BWROG topical report did a comparative PRA evaluation of the core damage risks of operation in the current end state and in the proposed Mode 3 end state. Events initiated by the loss of off-site power are dominant contributors to CDF in most BWR PRAs, and the steam-driven core cooling systems, RCIC and HPCS, play a major role in mitigating these events. The evaluation indicates that the core damage risks are lower in Mode 3 than in Mode 4 for one inoperable AC power source. Going to Mode 4 for one inoperable AC power source would cause loss of the high-pressure steam-driven injection system (RCIC/HPCS), and loss of the power conversion system (condenser/feedwater), and require activating the RHR system.

In addition, 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. Based on the low probability of loss of the AC power and the number of steam-driven systems available in Mode 3, the NRC staff concludes in the SE to the BWR topical report that the risks of staying in Mode 3 are lower than going to the Mode 4 end state.

Finding:

Based upon the above assessment, and because the time spent in Mode 3 to perform the repair is infrequent and limited, and in light of defense-in-depth considerations, the proposed change is acceptable.

3.2.4 TS 3.8.4, "DC Sources (Operating)"

The purpose of the direct current (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 Division 1 125 VDC and 250 VDC, Division 2 125 VDC, Division 3 125 VDC, and the opposite unit Division 2 125 VDC electrical power subsystems shall be operable.

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 DC electrical power subsystem for a period of 2 hours.

Proposed modification for end state required actions:

The proposed TS change is to remove the requirement to place the plant in Mode 4 if Required Action and associated Completion Time for Condition A is not met.

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. The BWROG topical report did a comparative PRA evaluation of the core damage risks of operation in the current end state and in the proposed Mode 3 end state, with one DC system inoperable. Events initiated by the loss of off site power are dominant contributors to CDF in most BWR PRAs, and the steam-driven core cooling systems, RCIC and HPCS, play a major role in mitigating these events. The evaluation indicates that the core damage risks are lower in Mode 3 than in Mode 4. Going to Mode 4 for one inoperable DC power source would cause loss of the high-pressure steam-driven injection system (RCIC/HPCS), and loss of the power conversion system (condenser/feedwater), and require activating the RHR system. In addition, 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. Based on the low probability of loss of the DC power and the number of systems available in Mode 3, the NRC staff concludes in the SE to the BWR topical report that the risks of staying in Mode 3 are approximately the same as and in some cases lower than the risks of going to the Mode 4 end state.

Finding:

Based upon the above assessment, and because the time spent in Mode 3 to perform the repair is infrequent and limited, and in light of defense-in-depth considerations, the proposed change is acceptable.

3.2.5 TS Section 3.8.7, "Distribution Systems - Operating"

The onsite Class 1E AC and DC electrical power distribution system is divided into redundant and independent AC, DC, and AC vital bus electrical power distribution systems. The primary AC electrical power distribution subsystem for each division consists of a 4.16-kV engineered safety feature (ESF) bus having an offsite source of power as well as a dedicated onsite EDG source. The secondary plant distribution subsystems include 600-VAC emergency buses and associated load centers, motor control centers, distribution panels and transformers.

The 120-VAC vital buses are arranged in four load groups and normally powered from DC via the inverters. There are two independent 125/250-VDC station service electrical power distribution systems and three independent 125-VDC DG electrical power distribution subsystems that support the necessary power for ESF functions. Each subsystem consists of a 125-VDC and 250-VDC bus and associated distribution panels.

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

a. Division 1 and Division 2 AC and 125 V DC distribution subsystems;

b. Division 3 AC and 125 V DC distribution subsystems;

c. Division 1 250 V DC distribution subsystem; and

d. The portions of the opposite unit's Division 2 AC and 125 V DC electrical power distribution subsystems capable of supporting the equipment required to be operable by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," LCO 3.7.4, "Control Room Area Filtration (CRAF) System," LCO 3.7.5, "Control Room Area Ventilation Air Conditioning (AC) System," and LCO 3.8.1, "AC Sources-Operating."

Condition requiring entry into end state:

The plant operator must bring the plant to Mode 3 within 12 hours and Mode 4 within 36 hours if required actions and associated completion times of Condition A or B not met.

Proposed modification for end state required actions:

The proposed TS change is to remove the requirement to place the plant in Mode 4 if Required Action and associated CT for Condition A or B is not met.

Assessment:

If one of the AC/DC subsystems is inoperable, the remaining AC/DC subsystems have the capacity to support a safe shutdown and to mitigate an accident condition. The BWROG topical report did a comparative PRA evaluation of the core damage risks of operation in the current end state and in the proposed Mode 3 end state, with one of the AC/DC subsystems inoperable. Events initiated by the loss of off site power are dominant contributors to core damage frequency in most BWR PRAs, and the steam-driven core cooling systems, RCIC and HPCS, play a major role in mitigating these events. The evaluation indicates that the core damage risks are lower in Mode 3 than in Mode 4. Going to Mode 4 for one inoperable AC/DC subsystem would cause loss of the high-pressure steam-driven injection system (RCIC/HPCS), and loss of the power conversion system (condenser/feedwater), and require activating the RHR system. In addition, 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. Based on the low probability of loss of the AC/DC electrical subsystems during the infrequent and limited time in Mode 3 and the number of systems available in Mode 3, the NRC staff concludes in the SE to the BWR topical report that the risks of staying in Mode 3 are approximately the same as and in some cases lower than the risks of going to the Mode 4 end state.

Finding:

Based upon the above assessment, and because the time spent in Mode 3 to perform the repair is infrequent and limited, and in light of defense-in-depth considerations, the proposed change is acceptable.

3.2.6 TS 3.6.1.1, "Primary Containment"

The function of the primary containment is to isolate and contain fission products released from the RPS following a design-basis LOCA and to confine the postulated release of radioactivity. The primary containment consists of a steel-lined, reinforced concrete vessel, which surrounds the RPS and provides an essentially leak-tight barrier against an uncontrolled release of radioactivity to the environment. Additionally, this structure provides shielding from the fission products that may be present in the primary containment atmosphere following accident conditions.

LCO: The primary containment shall be operable in Modes 1, 2, and 3.

Condition Requiring Entry into End State:

If the LCO cannot be met, the primary containment must be returned to operability within 1 hour (Required Action A.1). If the primary containment cannot be returned 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).

Proposed Modification for End State Required Actions:

The proposed TS change is to delete Required Action B.2.

Assessment:

The primary containment is one of the three primary boundaries to the release of radioactivity. (The other two are the fuel cladding and the RPS pressure boundary.) Compliance with this LCO ensures that a primary containment configuration exists, including equipment hatches and penetrations, that is structurally sound and will limit leakage to those leakage rates assumed in the safety analyses. This LCO entry condition does not include leakage through an unisolated release path. The BWROG topical report has determined that previous generic PRA work related to Appendix J requirements has shown that containment leakage is not risk significant. Should a fission product release from the primary containment occur, the secondary containment and related functions would remain operable to contain the release, and the standby gas treatment system would remain available to filter fission products from being released to the environment. By remaining in Mode 3, HPCS, RCIC, and the power conversion system (condensate/feedwater) remain available for water makeup and decay heat removal. Additionally, the EOPs direct the operators to take control of the depressurization function if low pressure injection/spray are needed for reactor coolant makeup and cooling. Therefore, defense-in-depth is maintained with respect to water makeup and decay heat removal by remaining in Mode 3.

Finding:

The requested change is acceptable. Note that the NRC staff's approval relies upon the secondary containment and the standby gas treatment system for maintaining defense-in-depth while in this reduced end state which the licensee commits to in TSTF-IG-05-02, Revision 1, "Implementation Guidance for TSTF-423, Revision 0."

3.2.7 TS 3.6.1.6, "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: Each suppression chamber-to-drywell vacuum breaker shall be operable in Modes 1, 2, and 3.

Condition Requiring Entry into End State:

If one suppression chamber-to-drywell vacuum breaker is inoperable for opening, the 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).

Proposed Modification for End State Required Actions:

The proposed TS change is to modify the Required Actions so that if vacuum breaker(s) cannot be returned to operable status within the required completion times, the plant is placed in hot shutdown. That is, Condition C is modified to relate only to Condition A, and Required Action C.2 is deleted, and Condition D is added with Required Actions D.1 and D.2, shutting down the plant to Mode 3 and then Mode 4 respectively, to address an inability to comply with the required actions related to Condition B, to close the vacuum breaker.

Assessment:

The BWROG topical report has determined that the specific failure of interest is not risk significant in BWR PRAs. The reduced end state would only be applicable to the situation where one suppression chamber-to-drywell vacuum breaker is inoperable for opening, with the remaining operable vacuum breakers capable of providing the necessary vacuum relief function, since they are required in Modes 1, 2, and 3. By remaining in Mode 3, HPCS, RCIC, and the power conversion system (condensate/feedwater) remain available for water makeup and decay heat removal. 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 maintained with respect to water makeup and decay heat removal by remaining in Mode 3. The existing end state remains unchanged for conditions involving any suppression chamber-to-drywell vacuum breakers that are stuck open, as established by new Condition D.

Finding:

Based upon the above assessment, and because the time spent in Mode 3 to perform the repair is infrequent and limited, and in light of defense-in-depth considerations, the proposed change is acceptable.

3.2.8 TS 3.6.2.4, “Residual Heat Removal (RHR) Suppression Pool Spray”

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

LCO: Two RHR suppression pool spray subsystems shall be operable in Modes 1, 2, and 3.

Condition Requiring Entry into End State:

If one RHR suppression pool spray subsystem is inoperable (Condition A), it must be restored to operable status within seven days (Required Action A.1). If both RHR suppression pool spray subsystems are inoperable (Condition B), one of them must be restored to operable status within 8 hours (Required Action B.1). If the RHR 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).

Proposed Modification for End State Required Actions:

The proposed TS change is to delete Required Action C.2.

Assessment:

The main function of the RHR suppression spray system is to remove heat from the suppression chamber so that the pressure and temperature inside primary containment remain within analyzed design limits. The RHR suppression 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 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, and the RHR suppression spray system could be needed to condense steam so that the pressure and temperature inside primary containment remain within analyzed design-basis 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 RHR suppression 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 RHR suppression 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 RHR suppression spray system. Section 6 of reference 6 summarizes the staff 's risk argument for approval of TS 4.5.1.11 and LCO 3.6.2.4, "Residual Heat Removal (RHR) Suppression Pool Spray." The argument for staying in Mode 3 instead of going to Mode 4 to repair the RHR Suppression Pool Spray system (one or both trains) is also supported by defense-in-depth considerations. Section 6.2 makes a comparison between the current (Mode 3) and the proposed (Mode 4) end state, 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 precluding 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 lower initial energy level and 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 (if not safer) for repairing an inoperable RHR suppression spray system.

Finding:

Based upon the above assessment, and because the time spent in Mode 3 to perform the repair is infrequent and limited, and in light of defense-in-depth considerations, the proposed change is acceptable.

3.2.9 TS 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. The associated leakage rates assumed in the accident analysis and the fission products entrapped within the secondary containment structure will be treated by the 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).

Proposed Modification for End State Required Actions:

The proposed TS change is to delete Required Action B.2.

Assessment:

This LCO entry condition does not include gross leakage through an unisolable release path. The BWROG topical report has determined that previous generic PRA work related to Appendix J 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. By remaining in Mode 3, HPCS, RCIC, and the power conversion system (condensate/feedwater) remain available for water makeup and decay heat removal. 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.

Finding:

The requested change is acceptable. Note that the NRC staff 's approval relies upon the primary containment and all other primary and secondary containment-related functions to still be operable, including the SGT system, for maintaining defense-in-depth while in this end state which the licensee commits to in TSTF-IG-05-02, Revision 1, "Implementation Guidance for TSTF-423, Revision 0."

3.2.10 TS 3.6.4.3, "Standby Gas Treatment (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, LCO 3.0.3 must be entered immediately (Required Action D.1).

Proposed Modification for End State Required Actions:

The proposed TS change is to delete Required Action B.2. The change Required Action D.1 to "Be in Mode 3" with a Completion Time of "12 hours."

Assessment:

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 off site 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 Regulatory Guide 1.177 for much higher consequence risks, such as large early release.

Section 6 of Reference 6 summarizes the NRC staff's risk argument for approval of TS 4.5.1.13, 4.5.2.11, and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System." The argument for 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. Section 6.2 makes a comparison between the current (Mode 3) and the proposed (Mode 4) end state, 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. 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 (if not safer) for repairing an inoperable SGT system.

Finding:

Based upon the above assessment, and because the time spent in Mode 3 to perform the repair is infrequent and limited, and in light of defense-in-depth considerations, the proposed change is acceptable.

3.2.11 TS 3.7.1, "Residual Heat Removal Service Water (RHRSW) System"

The RHRSW system is designed to provide cooling water for the RHR system heat exchangers, which are required for safe shutdown following a normal shutdown or DBA or transient.

LCO: Two RHRSW subsystems shall be operable in Modes 1, 2, and 3.

Condition Requiring Entry into End State:

If the LCO cannot be met, the following actions must be taken for the listed conditions:

- If one RHRSW subsystem is inoperable, (not applicable to Unit 2 during replacement of the Division 1 CSCS isolation valves during Unit 1 Refueling 11, while Unit 1 is in Mode 4, 5, or defueled), RHRSW subsystem must be restored to operable status within 7 days (Required Action A.1).
- If one RHRSW subsystem is inoperable, (only applicable to Unit 2 during replacement of the Division 1 CSCS isolation valves during Unit 1 Refueling 11 while Unit 1 is in Mode 4,5, or defueled), RHRSW subsystem must be restored to operable status within 10 days (Required Action B .1).
- If the required action and associated completion time cannot be met within the allotted time (Condition D), 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).

Proposed Modification for End State Required Actions:

The proposed TS change renumbers Conditions C (and Required Action C.1), and D (and Required Actions D.1 and D.2), to Conditions D (and Required Action D.1) and E (and Required Actions E.1 and E.2), respectively. The change modifies new Condition E to address new Condition D, which maintains the existing requirements with respect to both RHR subsystems being inoperable.

The change adds a new Condition C, which establishes requirements for existing Conditions A, and B, that are similar to existing Condition D but without Required Action D.2.

Assessment:

The BWROG topical report performed a comparative PRA evaluation of the core damage risks when operating in the current end state versus the proposed Mode 3 end state. The results 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. By remaining in Mode 3, HPCS, RCIC, and the power conversion system (condensate/feedwater) remain available for water makeup and decay heat removal. 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 RHRSW subsystem components that are still operable.

Finding:

Based upon the above assessment, and because the time spent in Mode 3 to perform the repair is infrequent and limited, and in light of defense-in-depth considerations, the proposed change is acceptable.

3.2.12 TS 3.7.4, “Control Room Area Filtration (CRAF) System”

The CRAF system provides a radiologically controlled environment from which the plant can be safely operated following a DBA.

LCO: Two CRAF subsystems shall be operable.

Condition Requiring Entry into End State:

If one CRAF subsystem is inoperable, it must be restored to operable status within 7 days (Required Action A.1). If the CRAF 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). If two CRAF subsystems are inoperable in Modes 1, 2, or 3, LCO 3.0.3 must be entered immediately (Required Action D.1).

Proposed Modification for End State Required Actions:

The proposed TS change is to delete Required Action B.2 and change Required Action D.1 to “Be in Mode 3" with a Completion Time of “12 hours.”

Assessment:

The unavailability of one or both CRAF subsystems 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 CRAF 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 6 of Reference 6 summarizes the NRC staff ’s risk argument for approval of TS 4.5.1.16, and LCO 3.7.4, “Control Room Area Filtration (CRAF) System.” The argument for staying in Mode 3 instead of going to Mode 4 to repair the CRAF system (one or both trains) is also supported by defense-in-depth considerations. Section 6.2 makes a comparison between the current (Mode 3) and the proposed (Mode 4) end state, 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. 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 (if not safer) for repairing an inoperable CRAF system.

Finding:

Based upon the above assessment, and because the time spent in Mode 3 to perform the repair is infrequent and limited, and in light of defense-in-depth considerations, the proposed change is acceptable.

3.2.13 TS 3.7.5, “Control Room Area Ventilation Air Conditioning (AC) System”

The Control Room AC system provides temperature control for the control room following control room isolation during accident conditions.

LCO: Two control room area ventilation AC subsystems shall be operable.

Condition Requiring Entry into End State:

If one control room area ventilation AC subsystem is inoperable, the subsystem must be restored to operable status within 30 days (Required Action A.1). If the required actions and associated completion times 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). If two control room area ventilation AC subsystems are inoperable, LCO 3.0.3 must be entered immediately (Required Action D.1).

Proposed Modification for End State Required Actions:

The proposed TS change is to delete Required Action B.2, and change Required Action D.1 to “Be in Mode 3” with a Completion Time of “12 hours.”

Assessment:

The unavailability of one or both area ventilation AC subsystems 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 area ventilation AC 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 6 of Reference 6 summarizes the NRC staff's risk argument for approval of TS 4.5.1.17, and LCO 3.7.5, "Control Room Air Conditioning (AC) System." The argument for staying in Mode 3 instead of going to Mode 4 to repair the area ventilation AC system (one or both trains) is also supported by defense-in-depth considerations. Section 6.2 makes a comparison between the current (Mode 3) and the proposed (Mode 4) end state, 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. The risk and defense-in-depth arguments, used according to the "integrated decision-making" process of RG 1.174 and 1.177, support the conclusion that Mode 3 is as safe as Mode 4 (if not safer) for repairing an inoperable AC system.

Finding:

Based upon the above assessment, and because the time spent in Mode 3 to perform the repair is infrequent and limited, and in light of defense-in-depth considerations, the proposed change is acceptable.

3.2.14 TS 3.7.6, "Main Condenser Offgas"

The off-gas 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 off site 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.

LCO: The gross gamma activity rate of the noble gases measured at the main condenser evacuation system pretreatment monitor station shall be ≤ 340,000 μCi/second after decay of 30 minutes.

Condition Requiring Entry into End State:

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 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:

- All steamlines must be isolated within 12 hours (Required Action B.1). The steam jet air ejector must be isolated within 12 hours (Required Action B.2).

- 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).

Proposed Modification for End State Required Actions:

The proposed TS change is to delete Required Action B.3.2.

Assessment:

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 6 of Reference 6 summarizes the NRC staff’s risk argument for approval of TS 4.5.1.18 and LCO 3.7.6, “Main Condenser Offgas.” The argument for staying in Mode 3 instead of going to Mode 4 to repair the MCOG system (one or both trains) is also supported by defense in-depth considerations. Section 6.2 makes a comparison between the current (Mode 3) and the proposed (Mode 4) end state, 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. 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 (if not safer) for repairing an inoperable MCOG system.

Finding:

Based upon the above assessment, and because the time spent in Mode 3 to perform the repair is infrequent and limited, and in light of defense-in-depth considerations, the proposed change is acceptable.

3.2.15 TS 3.6.2.3, “Residual Heat Removal (RHR) 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 redundant RHR suppression pool cooling subsystems.

LCO: Two RHR suppression pool cooling subsystems shall be operable.

Condition Requiring Entry into End State:

If one RHR suppression pool cooling subsystem is inoperable (Condition A), it must be restored to operable status within 7 days (Required Action A.1). If the RHR suppression pool spray 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).

Proposed Modification for End State Required Actions:

The proposed TS change is to delete Required Action C.2, and retain Condition C and Required Action C.1 for one RHR suppression pool spray subsystem inoperable. The change adds Condition D, with Required Actions D.1 and D.2, identical to existing Condition C, with Required Actions C.1 and C.2, to maintain existing requirements unchanged for two RHR suppression pool subsystems inoperable.

Assessment:

The BWROG topical report has completed a comparative PRA evaluation of the core damage risks of operation in the current end state versus operation in the proposed Mode 3 end state. The results 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. One loop of the RHR suppression pool cooling system is sufficient to accomplish the required safety function. By remaining in Mode 3, HPCS, RCIC, and the power conversion system (condensate/feedwater) remain available for water makeup and decay heat removal. 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.

Finding:

Based upon the above assessment, and because the time spent in Mode 3 to perform the repair is infrequent and limited, and in light of defense-in-depth considerations, the proposed change is acceptable.

4.0 REGULATORY COMMITMENTS

The PRA branch of the NRC staff reviewed the proposed changes to assure consistency with the requirements and conditions of the NRC safety evaluation preformed for the topical report NEDC-32988-A, which is the basis for TSTF-423. On the basis of the review, the NRC staff identified a concern with the industry implementation guidance document TSTF-IG-05-02, “Implementation Guidance for TSTF-423, Rev. 0.” These concerns were communicated to the BWROG TSTF. By letter dated March, 9, 2007, Revision 1 of TSTF-IG-05-02 was submitted, which resolved the NRC staff’s concerns. The licensee subsequently revised its commitment to follow the guidance of TSTF-IG-05-02, Revision 1, in its supplemental letter dated March 26, 2007.

The following table identifies those actions committed to by the licensee in their submittal.

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	Implement with amendment	No	Yes

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of the facilities components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (72 FR 26177; May 8, 2007). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, on the basis of 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.

8.0 REFERENCES

1. NEDC-32988-A, Revision 2, "Technical Justification to Support Risk-Informed Modification to Selected Required Action End States for BWR Plants," September 2005.

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

3. Title 10 of the *Code of Federal Regulations* (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. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision 3, July 2000.

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

7. TSTF-423, Revision 0, "Technical Specifications End States, NEDC-32988-A."

8. TSTF-IG-05-02, Revision 1, Implementation Guidance for TSTF-423, Revision 0, "Technical Specifications End States, NEDC-32988-A," dated March, 2007.

9. 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).

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

11. Exelon Nuclear letter RS-07-041, dated March 26, 2007, supplement to Application for Technical Specification Change TSTF-423, "Risk Informed Modification to selected Required Action End States for BWR Plants, Using the Consolidated Line Item Improvement Process."

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