July 12, 2007

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENT RE: RISK-INFORMED MODIFICATION TO SELECTED REQUIRED END STATES (TAC NOS. MD2631 AND MD2632)

Dear Mr. Crane:

The Commission has issued the enclosed Amendments Nos. 261 and 265 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station, Units 2 and 3, respectively. These amendments modify Technical Specification (TS) requirements related to required end states for TS action statements in response to your application dated July 14, 2006, as supplemented by letter dated June 5, 2007. The proposed changes are consistent with the NRC-approved Revision 0 to Technical Specification Task Force (TSTF) Change Traveler, TSTF-423, “Risk Informed Modification to Selected Required Action End States for BWR [boiling-water reactor] Plants.”

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

Sincerely,

/ra/

John Hughey, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-277 and 50-278

Enclosures:
1. Amendment No. 261 to Renewed DPR-44
2. Amendment No. 265 to Renewed DPR-56
3. Safety Evaluation

cc w/encls: See next page

July 12, 2007

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OFFICE	LPLI-2/PE	LPLI-1/PM	LPL1-2/PM	LPL3-2/LA	ITSB/BC	OGC	LPLI-2/BC
NAME	JKim	RGuzman	JHughey	EWhitt	TKobetz	LSubin *	HChernoff
DATE	1/10/07	1/10/07	7/11/07	06/21/07	3/14/07	07/03/07	7/12/07

Official Record Copy

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EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 261
Renewed License No. DPR-44

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), and PSEG Nuclear LLC (the licensees), dated July 14, 2006, as supplemented by letter dated June 5, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Renewed Facility Operating License No. DPR-44 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 261, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/ra/

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

Attachment:
Changes to the Technical Specifications
and Facility Operating License

Date of Issuance: July 12, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 261

RENEWED FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following page of Facility Operating License No. DPR-44 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

Remove	Insert
Page 3	Page 3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised page is identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3.5-1	3.5-1
3.5-2	3.5-2
3.5-3	3.5-3
3.5-12	3.5-12
3.6-1	3.6-1
3.6-19	3.6-19
3.6-21	3.6-21
3.6-27	3.6-27
3.6-29	3.6-29
3.6-34	3.6-34
3.6-40	3.6-40
3.6-41	3.6-41
3.7-1	3.7-1
3.7-2	3.7-2
3.7-7	3.7-7
3.7-8	3.7-8
3.7-10	3.7-10
3.8-5	3.8-5
3.8-29	3.8-29
3.8-43	3.8-43

EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 265
Renewed License No. DPR-56

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

    A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), and PSEG Nuclear LLC (the licensees), dated July 14, 2006, as supplemented by letter dated June 5, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Renewed Facility Operating License No. DPR-56 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 265, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/ra/

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

Attachment:
Changes to the Technical Specifications
and Facility Operating License

Date of Issuance: July 12, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 265

RENEWED FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following page of Facility Operating License No. DPR-56 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

Remove	Insert
Page 3	Page 3

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

Remove	Insert
3.5-1	3.5-1
3.5-2	3.5-2
3.5-3	3.5-3
3.5-12	3.5-12
3.6-1	3.6-1
3.6-19	3.6-19
3.6-21	3.6-21
3.6-27	3.6-27
3.6-29	3.6-29
3.6-34	3.6-34
3.6-40	3.6-40
3.6-41	3.6-41
3.7-1	3.7-1
3.7-2	3.7-2
3.7-7	3.7-7
3.7-8	3.7-8
3.7-10	3.7-10
3.8-5	3.8-5
3.8-29	3.8-29
3.8-43	3.8-43

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 261 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-44 AND

AMENDMENT NO. 265TO RENEWED FACILITY OPERATING LICENSE NO. DPR-56

EXELON GENERATION COMPANY, LLC

PSEG NUCLEAR LLC

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-277 AND 50-278

1.0 INTRODUCTION

By application dated July 14, 2006, as supplemented by letter dated June 5, 2007, Exelon Generation Company, LLC (the licensee) proposed changes to the Technical Specifications (TSs) for Peach Bottom Atomic Power Station, Units 2 and 3. The requested changes are the adoption of Technical Specification Task Force (TSTF) Change Traveler TSTF-423, Revision 0, to the Boiling Water Reactor (BWR) Standard Technical Specifications (STS) (NUREG 1433 and NUREG 1434), which was proposed by the Owners Groups 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-A, Revision 2, "Technical Justification to Support Risk Informed Modification to Selected Required Action End States for BWR Plants," December 2002, into the BWR STS. (Note: The changes in TSTF-423 are made with respect to Revision 2 of the BWR STS NUREGs.)

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 management techniques in TSs, while reducing unnecessary burden and making TS requirements consistent with the Nuclear Regulatory Commission's (NRC's or 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, "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 most plant TSs 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 TSs. 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 degrees 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 degrees 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 1 week.

The supplement dated June 5, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on December 19, 2006 (71 FR 75994).

2.0 <u>REGULATORY EVALUATION</u>

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; (4) design features; and (5) administrative controls. 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), 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 ... ."

Topical Report NEDC-32988-A, Revision 2, 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 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, STSs were developed and many licensees improved 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 (see, for example, Federal Register, Vol. 58, No. 139, p. 39136, "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Plants," July 22, 1993). 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, the maintenance rule. Section 50.65(a)(4) of 10 CFR states: "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, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," May 2000 (ML003699426), 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, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision 3, July 2000. 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 the Amendment are consistent with the changes proposed and justified in Topical Report GE NEDC-32988-A, Revision 2, (see NRC Safety Evaluation for Topical Report NEDC-32988, Revision 2, September 27, 2002, ML022700603) and which have already been approved by the associated NRC Safety Evaluation (SE) for TSTF-423. The evaluation included in TSTF-423, Revision 0, “Technical Specifications End States, NEDC-32988-A,” as appropriate and applicable to the changes of TSTF-423, is reiterated here and differences from the SE are justified. In its letter dated June 5, 2007, the licensee commits to TSTF-IG-05-02, Revision 1, Implementation Guidance for TSTF-423, Revision 0, “Technical Specifications End States, NEDC-32988-A,” March 2007, 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 NEDC-32988-A, Revision 2.

3.1 Risk Assessment

The objective of the BWROG topical report NEDC-32988-A, Revision 2, risk assessment was to show that any risk increases associated with the 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, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decision Making on Plant Specific Changes to the Licensing Basis,” USNRC, August 1998 (ML003740133) and RG 1.177, “An Approach for Plant Specific Risk-Informed Decision Making: Technical Specifications,” USNRC, August 1998 (ML003740176). The three-tiered approach documented in RG 1.177 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.

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) of the Maintenance Rule for identifying risk-significant configurations resulting from maintenance related activities and taking appropriate compensatory measures to avoid such configurations. This TS 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.

The 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:

(a) 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.

(b) Avoidance of Risk-Significant Configurations (Tier 2): The performed risk analyses, which are based on a single LCO, show 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 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. The NRC staff finds that such guidance is adequate for preventing risk-significant plant configurations.

(c) Configuration Risk Management (Tier 3): The licensee has a program, as described above, in place to comply with 10 CFR 50.65(a)(4) to assess and manage the risk from 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 end state mode change was evaluated subject to the following assumptions which are incorporated into the TSs, TS Bases, and TSTF-IG-05-02, Revision 1, March 2007:

1. 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.

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 in Topical Report GE NEDC-32988-A, Revision 2, and approved by the NRC SE (ML022700603). The following are the changes, including a synopsis of the STS LCO, and a conclusion of acceptability.

3.2.1 LCO 3.5.1 (Units 2 and 3): Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling System (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 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 five safety/relief valves (SRVs) 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:

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

b. If the inoperable ECCS injection/core spray cannot be restored to operable status, the plant must be placed in Mode 3 within 12 hours, and Mode 4 within 36 hours.

Modification for end state required actions:

a. No change.

b. If the ECCS injection or spray system is inoperable, the plant must be restored to operable status within 12 hours. The plant is not taken into Mode 4 (cold shutdown).

c. If the required actions described in Conditions C, D, E or F cannot be met, the plant must be placed in Mode 3 within 12 hours. The reactor is not depressurized and not taken to Mode 4.

Assessment: The BWROG topical report did a comparative PRA evaluation of the core damage risks of operation in the current end state and the 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 (reactor core isolation cooling (RCIC)/high-pressure coolant injection (HPCI)), 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 concluded 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 (discussed in NEDC-32988-A, Revision 2), the change is acceptable.

3.2.2 LCO 3.5.3 (Units 2 and 3): RCIC System

The function of the RCIC system is to provide reactor coolant makeup during loss of feedwater and other transient events. This TS provides the operability requirements for the RCIC system as described below. The TS change allows the plant to remain in Mode 3 until the repairs are completed.

LCO: The RCIC system must be operable during Modes 1, 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 by administrative means within 1 hour that the HPCI system is operable, and (b) restore the RCIC system to operable status within 14 days. If either or both actions cannot be completed within the allotted time, the plant must be in placed in Mode 3 within 12 hours and the reactor steam dome pressure reduced to less than 150 psig within 36 hours.

Modification for end state required actions: This TS change keeps the plant in Mode 3 (hot shutdown) until the required repairs are completed. The reactor steam dome pressure is not reduced to less than 150 psig.

Assessment: This change would allow the inoperable RCIC system to be repaired in a plant operating mode with lower risk, and without challenging the normal shutdown systems. The BWROG Topical Report did a comparative PRA evaluation of the core damage risks of operation in the current end state and in the Mode 3 end state. The evaluation indicates that the core damage risks are lower in Mode 3 than in Mode 4. Going to Mode 3 with reactor steam dome pressure less than 150 psig for inoperability of RCIC would also cause loss of the high-pressure steam-driven injection system HPCI and loss of the power conversion system (condenser/feedwater), 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. Based on the low probability of loss of the necessary overpressure protection function and the number of systems available in Mode 3, the NRC staff concluded 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 (discussed in NEDC-32988-A, Revision 2), the change is acceptable.

3.2.3 LCO 3.8.1 (Units 2 and 3): AC Sources (Operating)

The purpose of the AC electrical system is to provide, during all situations, the power required to establish 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.

LCO: 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 Class1E AC electric power distribution system,

b. Four EDGs.

Condition requiring entry into end state: Plant operators must bring the plant to Mode 4 within 36 hours following the sustained inoperability of either, or both, required offsite circuits; either one, two or more required EDGs; or one required offsite circuit and one required EDG.

Modification for end state require actions: Delete required action G.2 to go to Mode 4 (cold shutdown). The plant will remain in Mode 3 (hot shutdown).

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.

The BWROG topical report did a comparative PRA evaluation of the core damage risks of operation in the current end state and in the Mode 3 end state. Events initiated by the loss of offsite power are dominant contributors to CDF in most BWR PRAs, and the steam-driven core cooling systems, RCIC and HPCI, 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/HPCI), 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 concluded 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 (discussed in NEDC-32988-A, Revision 2), the change is acceptable.

3.2.4 LCO 3.8.4 (Units 2 and 3): 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.

<u>LCO</u>: For Modes 1, 2 and 3, Divisions I and II of Units 2 and 3's DC electrical power subsystems shall be operable.

<u>Modification for end state required actions</u>: The TS change is to remove the requirement to place the plant in Mode 4, Required Actions in D.2 are deleted.

<u>Assessment</u>: 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 Mode 3 end state, with one DC system inoperable. Events initiated by the loss of offsite power are dominant contributors to CDF in most BWR PRAs, and the steam-driven core cooling systems, RCIC and HPCI, 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/HPCI), 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 concluded 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.

<u>Finding</u>: 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 (discussed in NEDC-32988-A, Revision 2), the change is acceptable.

3.2.5 LCO 3.8.7 (Units 2 and 3): 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.

<u>LCO</u>: For Unit 2, Modes 1, 2, and 3, the following electrical power distribution subsystems shall be operable:

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

b. Unit 3 AC and DC electrical power distribution subsystems needed to support equipment required to be operable by LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System-Hot Shutdown;" LCO 3.5.1, “ECCS-Operating;" LCO 3.6.2.3, “RHR

Suppression Pool Cooling;" LCO 3.6.2.4, "RHR Suppression Pool Spray;" LCO 3.6.3.1, "Containment Atmospheric Dilution (CAD) System;" LCO 3.6.4.3, "Standby Gas Treatment (SGT) System;" LCO 3.7.1, "High Pressure Service Water (HPSW) System;" LCO 3.7.2, "Emergency Service Water (ESW) System and Normal Heat Sink;" LCO 3.7.3,"Emergency Heat Sink;" and LCO 3.8.1, "AC Sources-Operating."

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

a. Unit 3 Division I and Division II AC and DC electrical power distribution subsystems.

b. Unit 2 AC and DC electrical power distribution subsystems needed to support equipment required to be operable by LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System-Hot Shutdown;" LCO 3.5.1, "ECCS-Operating;" LCO 3.6.2.3, "RHR Suppression Pool Cooling;" LCO 3.6.2.4, "RHR Suppression Pool Spray;" LCO 3.6.3.1, "Containment Atmospheric Dilution (CAD) System;" LCO 3.6.4.3, "Standby Gas Treatment (SGT) System;" LCO 3.7.1, "High Pressure Service Water (HPSW) System;" LCO 3.7.2, "Emergency Service Water (ESW) System and Normal Heat Sink;" LCO 3.7.3, "Emergency Heat Sink;" LCO 3.7.4, "Main Control Room Emergency Ventilation (MCREV) System;" 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 and Mode 4 within 36 hours following the sustained inoperability of one AC (Condition A), or one AC vital bus (Condition C), or one DC (Condition D) electrical power subsystem for a period of 7 days, 8 hours, and 2 hours, respectively (with a maximum 16-hour CT limit for Conditions C and D, from initial of failure to meet Condition A of 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, and the Required Action in E.2 is deleted.

Assessment: If one of the AC/DC/AC vital subsystems is inoperable, the remaining AC/DC/AC vital 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 Mode 3 end state, with one of the AC/DC/AC vital subsystems inoperable. Events initiated by the LOOP are dominant contributors to CDF in most BWR PRAs, and the steam-driven core cooling systems, RCIC and HPCI, 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/AC vital subsystem would cause loss of the high-pressure steam-driven injection system (RCIC/HPCI), 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/AC vital electrical subsystems during the infrequent and limited time in Mode 3 and the number of systems available in Mode 3, the NRC staff concluded 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.

<u>Finding</u>: 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 (discussed in NEDC-32988-A, Revision 2), the change is acceptable.

3.2.6 LCO 3.6.1.1 (Units 2 and 3): Primary Containment

The function of the primary containment is to isolate and contain fission products released from the reactor primary system 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 reactor primary system 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.

<u>LCO</u>: The primary containment shall be operable.

<u>Condition requiring entry into end state</u>: If the LCO cannot be met, the primary containment must be returned to operability within one 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).

<u>Modification for end state required actions</u>: Delete Required Action B.2.

<u>Assessment</u>: The primary containment is one of the three primary boundaries to the release of radioactivity. (The other two are the fuel cladding and the reactor primary system 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 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 SGT system would remain available to filter fission products from being released to the environment. By remaining in Mode 3, HPCI, 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.

<u>Finding</u>: The requested change is acceptable. Note that the NRC staff's approval relies upon the secondary containment and the SGT system for maintaining defense-in-depth while in this reduced end state.

3.2.7 LCO 3.6.1.5 (Units 2 and 3): 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 serving to preserve 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 72 hours (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: Modify the Required Actions so that if vacuum breaker(s) cannot be returned to operable status within the required CTs, the plant is placed in hot shutdown. That is, insert new Condition D to relate only to Condition C, renumber existing Conditions D and E as Conditions E and F, respectively, and revise Condition F to be applicable to Conditions A, B, or E. Therefore, if the CTs of Conditions A, B, or E are not met, the plant is placed in MODE 3 (Action F.1) within 12 hours and MODE 4 (Action F.2) within 36 hours.

Assessment: The BWROG Topical Report has 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. In Mode 3, for other accident considerations, HPCI, 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 is 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: 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 (discussed in NEDC-32988-A, Revision 2), the change is acceptable.

3.2.8 LCO 3.6.1.6 (Units 2 and 3): 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.

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).

Modification for end state required actions: Modify the Required Actions so that if vacuum breaker(s) cannot be returned to operable status within the required CTs, the plant is placed in hot shutdown. That is, insert new Condition B to relate only to Condition A, renumber existing Conditions B and C as Conditions C and D, respectively, and revise Condition D to be applicable to only Condition C. Therefore, if the CT of Condition C is not met, the plant is placed in MODE 3 (Action D.1) within 12 hours and MODE 4 (Action D.2) within 36 hours.

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, HPCI, 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 is 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 (discussed in NEDC-32988-A, Revision 2), the change is acceptable.

3.2.9 LCO 3.6.2.4 (Units 2 and 3): 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.

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 7 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).

Modification for end state required actions: 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 such 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 the NRC staff's Safety Evaluation for Topical Report NEDC-32988, Revision 2, September 27, 2002 (ML022700603), summarizes the NRC staff's risk argument for approval of TRS 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 Mode 3 and the 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 (discussed in NEDC-32988-A, Revision 2), the change is acceptable.

3.2.10 LCO 3.6.4.1 (Units 2 and 3): 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 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: 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, HPCI, 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 is 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 remain operable, including the SGT system, for maintaining defense-in-depth while in this end state.

3.2.11 LCO 3.6.4.3 (Units 2 and 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, LCO 3.0.3 must be entered immediately (Required Action D.1).

Modification for end state required actions: Delete Required Action B.2. Change Required Action D.1 to "Be in Mode 3" with a CT 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 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 6 of the NRC staff's Safety Evaluation for Topical Report NEDC-32988, Revision 2, September 27, 2002 (ML022700603), summarizes the NRC staff’s risk argument for approval of TRS 4.5.1.13, TRS 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 Mode 3 and the 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 (discussed in NEDC-32988-A, Revision 2, the change is acceptable.

3.2.12 LCO 3.7.1 (Units 2 and 3): HPSW system

The HPSW 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 HPSW subsystems 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 HPSW subsystem is inoperable (Condition A), the HPSW subsystem must be restored to operable status within 7 days (Required Action A.1).

B. If the required action and associated completion time cannot be met 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: Renumber Conditions B (and Required Action B.1), and C (and Required Actions C.1 and C.2), to Conditions C (and Required Action C.1) and D (and Required Actions D.1 and D.2), respectively. Modify new Condition D to address new Condition C, which maintains the existing requirements with respect to both HPSW subsystems being inoperable. Add a new Condition B, which establishes requirements for existing Conditions A 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 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, HPCI, 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 is 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 HPSW 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 (discussed in NEDC-32988-A, Revision 2, the change is acceptable.

3.2.13 LCO 3.7.4 (Units 2 and 3): Main Control Room Emergency Ventilation (MCREV) System

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

LCO: Two MCREV subsystems shall be operable.

Condition requiring entry into end state: If one MCREV subsystem is inoperable, it must be restored to operable status within 7 days (Required Action A.1). If the MCREV 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 MCREV subsystems are inoperable in Mode 1, 2, or 3, LCO 3.0.3 must be entered immediately (Required Action D.1).

Modification for end state required actions: Delete Required Action B.2, and change Required Action D.1 to “Be in Mode 3" with a CT of “12 hours.”

Assessment: The unavailability of one or both MCREV 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 MCREV 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 the NRC staff's Safety Evaluation for Topical Report NEDC-32988, Revision 2, September 27, 2002 (ML022700603), summarizes the NRC staff’s risk argument for approval of TRS 4.5.1.16, and LCO 3.7.4, “Main Control Room Emergency Ventilation (MCREV) System.” The argument for staying in Mode 3 instead of going to Mode 4 to repair the MCREV system (one or both trains) is also supported by defense-in-depth considerations. Section 6.2 makes a comparison between the Mode 3 and the 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 MCREV 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 (discussed in NEDC-32988-A, Revision 2), the change is acceptable.

3.2.14 LCO 3.7.5 (Units 2 and 3): 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.

LCO: The gross gamma activity rate of the noble gases measured at the steam jet air ejector (SJAE) discharge at the offgas sample rack shall be ≤320,000 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 CT cannot be met, one of the following must occur:

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

b. The SJAE 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).

Modification for end state required actions: 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 the NRC staff's Safety Evaluation for Topical Report NEDC-32988, Revision 2, September 27, 2002 (ML022700603), summarizes the NRC staff's risk argument for approval of TRS 4.5.1.18 and LCO 3.7.5, "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 Mode 3 and the 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.

<u>Finding</u>: 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 (discussed in NEDC-32988-A, Revision 2), the change is acceptable.

3.2.15 LCO 3.6.2.3 (Units 2 and 3): 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.

<u>LCO</u>: Two RHR suppression pool cooling subsystems shall be operable.

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

<u>Modification for end state required actions</u>: Insert new Condition B so that if the RHR suppression pool cooling system cannot be returned to operable status within the required CT, the plant is placed in hot shutdown. That is, insert new Condition B to relate only to Condition A, renumbering existing Conditions B and C as Conditions C and D, respectively, and revise Condition D to be applicable to only Condition C. Therefore, if the CT of Condition C is not met, the plant is placed in MODE 3 (Action D.1) within 12 hours, and MODE 4 (Action D.2) within 36 hours.

<u>Assessment</u>: 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 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.

<u>Finding</u>: 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 (discussed in NEDC-32988-A, Revision 2), the change is acceptable.

The licensee's application dated July 14, 2006, provided revised TS Bases pages to be implemented with the associated TS changes. The NRC staff notes that the Peach Bottom TS Bases Control Program is the appropriate process for updating the TS Bases.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. The State official provided no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 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 considerations, and there has been no public comment on such finding (71 FR 75994). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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