Dated: March 16, 2006.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E6–4193 Filed 3–22–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Meeting of the Subcommittee on Plant License Renewal; Notice of Meeting

The ACRS Subcommittee on Plant License Renewal will hold a meeting on April 5, 2006, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland. The entire meeting will be open to public attendance. The agenda for the subject meeting shall be as follows:

Wednesday, April 5, 2006—8:30 a.m.–12 Noon

The purpose of this meeting is to discuss the License Renewal Application for Nine Mile Point and the related Safety Evaluation Report (SER) with open items prepared by the NRR staff. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff, Constellation Energy Group, and other interested persons regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. John G. Lamb (telephone 301/415–6855) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: March 16, 2006.

Michael R. Snodderly,

Acting Branch Chief, ACRS/ACNW.

[FR Doc. E6–4194 Filed 3–22–06; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Notice of Availability of Model Application Concerning Technical Specifications for Boiling Water Reactor Plants to Risk-Inform Requirements Regarding Selected Required Action End States Using the Consolidated Line Item Improvement Process

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

SUMMARY: Notice is hereby given that the staff of the Nuclear Regulatory Commission (NRC) has prepared a model application related to the revision of Boiling Water Reactor (BWR) plant required action end state requirements in technical specifications (TS). The purpose of this model is to permit the NRC to efficiently process amendments that propose to revise BWR TS required action end state requirements. Licensees of nuclear power reactors to which the model applies may request amendments utilizing the model application.

DATES: The NRC staff issued a **Federal Register** Notice (70 FR 74037, December 14, 2005) that provided a model safety evaluation (SE) and a model no significant hazards consideration (NSHC) determination relating to changing BWR TS required action end state requirements. The NRC staff hereby announces that the model SE and NSHC determination may be referenced in plant-specific applications to adopt the changes. The staff has posted a model application on the NRC Web site to assist licensees in using the consolidated line item improvement process (CLIIP) to revise the BWR TS required action end state requirements. The NRC staff can most efficiently consider applications based upon the model application if the application is submitted within a year of this **Federal Register** Notice.

FOR FURTHER INFORMATION CONTACT: T. R. Tjader, Mail Stop: O12H2, Division of Inspection and Regional Support, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone 301–415–1187.

SUPPLEMENTARY INFORMATION:

Background

Regulatory Issue Summary 2000–06, ''Consolidated Line Item Improvement Process for Adopting Standard Technical Specification Changes for Power Reactors,'' was issued on March 20, 2000. The Consolidated Line Item Improvement Process (CLIIP) is intended to improve the efficiency of NRC licensing processes. This is accomplished by processing changes to the standard TS (STS) in a manner that supports subsequent license amendment applications. The CLIIP includes an opportunity for the public to comment on proposed changes to the STS following a preliminary safety assessment by the NRC staff and finding that the change will likely be offered for adoption by licensees. The CLIIP includes NRC staff evaluation of any comments received for a proposed change to the STS, and either a reconsideration of the change or an announcement of the availability of the change for adoption by licensees. Those licensees opting to apply for the subject change to their TS are responsible for reviewing the staff's evaluation, referencing the applicable technical justifications, and providing any necessary plant-specific information. Each amendment application made in response to the notice of availability will be processed and noticed in accordance with applicable rules and NRC procedures.

This notice involves the revision of BWR TS required action end state requirements. This change was proposed for incorporation into the STS by participants in the Owners Groups Technical Specification Task Force (TSTF) and is designated TSTF–423, Revision 0. TSTF–423, as well as the NRC staff's safety evaluation and model application, may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records are accessible electronically from the ADAMS Public Library component on the NRC Web site, (the Electronic Reading Room). TSTF–423, the NRC staff's safety evaluation, and the model application, can be viewed on the NRC Web site at: (*http://www.nrc.gov/reactors/operating/licensing/techspecs.html*).

Applicability

This proposal to modify technical specification requirements by the adoption of TSTF–423 is applicable to all licensees of BWR plants who have adopted or will adopt, in conjunction with the change, technical specification requirements for a Bases control program consistent with the TS Bases Control Program described in Section 5.5 of the BWR STS. Licensees that have not adopted requirements for a Bases control program by converting to the improved STS or by other means, are requested to include the requirements for a Bases control program consistent with the STS in their application for the change. The need for a Bases control program stems from the need for adequate regulatory control of some key elements of the proposal that are contained in the Bases in TSTF–423. The staff is requesting that the Bases changes be included with the proposed license amendments consistent with the Bases in TSTF–423, prior to implementing TSTF–423. To ensure that the overall change, including the Bases, includes appropriate regulatory controls, the staff plans to condition the issuance of each license amendment on the licensee's incorporation of the changes into the Bases document and on requiring the licensee to control the changes in accordance with the Bases Control Program. The CLIIP does not prevent licensees from requesting an alternative approach or proposing the changes without the requested Bases and Bases control program. However, deviations from the approach recommended in this notice may require additional review by the NRC staff and may increase the time and resources needed for the review. Significant variations from the approach, or inclusion of additional changes to the license, will result in staff rejection of the submittal. Instead, licensees desiring significant variations and/or additional changes should submit a LAR that does not claim to adopt TSTF–423.

Public Notices

In a notice in the **Federal Register** dated December 14, 2005 (70 FR 74037), the staff requested comment on the use of the CLIIP to process requests to revise the BWR TS regarding required action end state requirements, designated as TSTF–423.

In response to the notice soliciting comments from interested members of the public about modifying the TS requirements regarding revising required action end state requirements, the staff received one set of comments (from the Owners Groups TSTF, representing licensees). Specific comments on the model SE were offered, and are summarized and discussed below:

1. *Comment:* We commend the staff for adopting a draft Safety Evaluation format that simplifies the application of the model Safety Evaluation for TSTF–423 to individual licensees. For example, providing blanks for plant name, operating license number, etc. We encourage the staff to follow this example for future CLIIP model Safety Evaluations.

Response: The NRC staff acknowledges the comment; no action taken.

2. *Comment:* The ''Applicability'' portion of the notice states that each licensee applying for the changes proposed in TSTF–423 should include Bases for the proposed Technical Specifications (TS) consistent with the Bases proposed in TSTF–423. We request that the section be revised to not require licensees to submit Bases changes. The Bases changes in TSTF–423 are not integral to the change and both licensee and NRC resources could be saved by allowing licensees to adopt the necessary Bases changes using the Technical Specifications Bases Control Program. As a precedent, the CLIIP for TSTF–460 (Notice of Availability published in the **Federal Register** on August 23, 2004) allowed licensees to commit to updating their TS Bases under the TS Bases Control Program instead of requiring licensees to submit their Bases changes for NRC review. We propose that the TSTF–423 CLIIP take a similar approach.

Response: The NRC staff does not agree with the comment. The associated TS Bases changes are an essential and integral element of the change, must be consistent with the Bases in TSTF–423, and should be submitted by the licensees with the license amendment request for adoption of TSTF–423.

3. *Comment:* Section 1.0 of the model Safety Evaluation, in the first paragraph, states that TSTF–423 was proposed by the Nuclear Energy Institute Risk Informed Technical Specification Task Force. That is incorrect. TSTF–423, Revision 0, was submitted by the Owners Group Technical Specifications Task Force in a letter to the NRC dated August 12, 2003.

Response: The NRC staff agrees with the comment. The correction has been made.

4. *Comment:* Section 1.0 of the model Safety Evaluation, the last sentence of section, states, ''Short duration repairs are on the order of 2-to-3 days, but not more than a week.'' We recommend replacing this sentence with the statement from the Implementation Guidance (TSTF–IG–05–02), which states, ''A 'short duration' is envisioned to be the duration that boiling water reactors (BWRs) are most physically and practicably able to remain in the hot shutdown condition (*i.e.*, from a few days to approximately one week).'' This clarifies that the time frames are a statement of fact rather than a restriction which must be incorporated in plant operating controls.

Response: The NRC staff does not agree with the comment. This issue was discussed thoroughly and the one week limit was determined appropriate. The one week limit is explicitly stated in the Implementation Guidance (Reference 8 to the SE) submitted by industry, agreed to by the NRC staff, and to which the licensees must commit. Section 1 paragraph 6 of the Implementation Guidance (TSTF–IG–05–02) states, ''Any entry into Mode 3 using this TS allowance must be limited to no more than seven days.'' No action has been taken.

5. *Comment:* In Section 3.2, ''Assessment of TS Changes,'' of the model Safety Evaluation, each subsection is titled with the applicable Topical Report section number and the ITS LCO number. The abbreviation ''TS'' is used to indicate the Topical Report section number (*e.g.*, ''TS 4.5.1.2 and LCO 3.4.3 (BWR/4); TS 4.5.2.2 and LCO 3.4.4 (BWR/6), Safety/Relief Valves (SRVs).'' The labels ''4.5.1.2'' and ''4.5.2.2'' are the Topical Report sections associated with these LCO changes. These references also appear in the text of Section 3.2. This presentation is confusing as ''TS'' is defined in the model Safety Evaluation as ''Technical Specifications'' and non-ITS plants have Technical Specification requirements with numbers similar to the Topical Report numbers. We recommend replacing this use of the abbreviation ''TS'' with either ''Topical Report section'' or defining another acronym, such as ''TR.''

Response: The NRC staff agrees with the comment. The abbreviation for the Topical Report Section was poorly chosen, in that it was easily confused with the abbreviation for Technical Specification. The Topical Report Section abbreviation has been changed to ''TRS.''

6. *Comment:* In Section 3.2.4 of the model Safety Evaluation, in the title and in the first paragraph, the LCO name ''Low-Low Set Logic (LLS) Valves'' is used. The word ''logic'' should not appear in the LCO name. The document should be revised to state ''Low-Low Set (LLS) Valves.''

Response: The NRC staff agrees with the comment. The correction has been made.

7. *Comment:* Section 5.0, ''Environmental Consideration,'' of the model Safety Evaluation states that the amendments meet the eligibility criteria for categorical exclusion set forth in ''10 CFR 51.22(c)(9) [and (c)(10)].'' 10 CFR 51.22(c)(10) pertains to issuance of an amendment pursuant to ''parts 30, 31, 32, 33, 34, 35, 36, 39, 40, 50, 60, 61, 70 or part 72 of this chapter which (i) changes surety, insurance and/or indemnity requirements, or (ii) changes

recordkeeping, reporting, or administrative procedures or requirements.'' Paragraph (c)(10) is not applicable to this change and the reference should be deleted.

Response: The NRC staff agrees with the comment. The correction has been made.

8. *Comment:* Section 7.0, ''References,'' of the model Safety Evaluation, Reference 1, states the date of NEDC–32988–A, Revision 2, as September 2005. The correct date of the document is December 2002.

Response: The NRC staff agrees with the comment. The correction has been made.

The NRC staff has made editorial changes to the previously published model SE related to TSTF–423 resulting from the disposition of comments 3, 5, 6, 7, and 8. The staff finds that technically the previously published SE remains unaltered. Below are the republished model SE and model NSHC determination (previously published in the **Federal Register**; 70 FR 23238, December 14, 2005), and the model application prepared by the staff that licensees may reference in their plant-specific applications.

Dated at Rockville, Maryland, this 16th day of March 2006.

For the Nuclear Regulatory Commission.

Thomas H. Boyce,

Branch Chief, Technical Specifications Branch, Division of Inspection and Regional Support, Office of Nuclear Reactor Regulation.

Model Plant Specific Safety Evaluation for Technical Specification Task Force (TSTF) Change TSTF–423, Risk Informed Modification to Selected Required Action End States, a Consolidated Line Item Improvement

U.S. Nuclear Regulatory Commission, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. [______] to Facility Operating License NFP–[______ [Utility Name] [Plant Name], [Unit ______] Docket No. –[____]

1.0 Introduction

By letter dated __________, 20__, [Utility Name] (the licensee) proposed changes to the technical specifications (TS) for [plant name]. The requested changes are the adoption of 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 Technical Specifications Task Force (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 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 (RMTS) 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.

The Code of Federal Regulations, 10 CFR 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 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 technical specifications 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 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 a week.

2.0 Regulatory Evaluation

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TS. Pursuant to 10 CFR 50.36(c), TS 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) limiting conditions for operation (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, standard TS 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 * * * 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, (Reference 1) and approved by the associated NRC 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, ''Technical Specifications End States, NEDC–32988–A,'' (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 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. 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.

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:

- *Risk Impact of the Proposed Change (Tier 1).* The risk changes associated with the TS changes in TSTF–423, in terms of mean yearly increases in core damage frequency (CDF) and large early release frequency (LERF), are risk neutral or risk beneficial. In addition, there are no significant temporary risk increases, as defined by RG 1.177 criteria, associated with the implementation of the TS end state changes.
- *Avoidance of Risk-Significant Configurations (Tier 2).* The 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 (CRMP), 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 staff finds that such guidance is adequate for preventing risk-significant plant configurations.
- *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 TS, TS Bases, and TSTF–IG–05–02 (Reference 8):

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 technical specification.

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 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 (Reference 1), and approved by the NRC SE (Reference 6). The following are the changes, including a synopsis of the STS LCO, and a conclusion of acceptability.

3.2.1 Topical Report Section (TRS) 4.5.1.2 and LCO 3.4.3 (BWR/4); TRS 4.5.2.2 and LCO 3.4.4 (BWR/6), Safety/Relief Valves (SRVs).

The function of the SRVs is to protect the plant against severe overpressurization events. These TS provide the operability requirements for the SRVs as described below. The TS change allows the plant to remain in Mode 3 until the repairs are completed.

[**Note:** Plant Applicability, BWR4/6]

LCO: The safety function of 11 SRVs must be operable (BWR/4 plants). The safety function of seven SRVs must be operable and the relief function of seven additional SRVs must be operable (BWR/6 plants).

Condition requiring entry into end state: If the LCO cannot be met with one or two SRVs inoperable, the inoperable valves must be returned to operability within 14 days. If the SRVs cannot be returned to operable status within that time, the plant must be placed in Mode 3 within 12 hours and in Mode 4 within 36 hours.

Modification for end state required actions: If the LCO cannot be met with one or two SRVs inoperable, the inoperable valves must be returned to operability within 14 days. If the one or two inoperable SRVs cannot be returned to operable status within 14 days, the plant must be placed in Mode 3 within 12 hours. If three or more SRVs become inoperable, the plant must be placed in Mode 4 within 36 hours.

Assessment: 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 4 for one inoperable SRV 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, 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 necessary overpressure protection function and the number of systems available in Mode 3, the staff concluded in the SE (reference 6) for the BWROG 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. The change allows the inoperable SRV to be repaired in a plant operating mode with lower risks. After repairs are made, the plant can be brought to full-power operation with less potential for transients and errors. The plant is taken into cold shutdown only when three or more SRVs are inoperable.

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 Reference 1), the staff finds that the requested change to allow operation in Mode 3 with a minimum number of SRVs inoperable, after plant risk has been assessed and managed, is acceptable.

3.2.2 TRS 4.5.1.3 and LCO 3.5.1(BWR/4); TRS 4.5.2.3 and LCO 3.5.1 (BWR/6), Emergency Core Cooling Systems (ECCS) (Operating).

The ECCS systems provide cooling water to the core in the event of a loss-of-coolant accident (LOCA). This set of ECCS 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.

[**Note:** Plant Applicability, BWR4/6]

LCO: Each ECCS injection/spray subsystem and the automatic depressurization system (ADS) function of seven BWR/4, or eight BWR/6, 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 (BWR/4 plants only).

c. If two ECCS injection subsystems are inoperable or one ECCS injection subsystem and one ECCS spray system are inoperable, one ECCS injection/spray subsystem must be restored to operable status within 72 hours. If this required action cannot be met, the plant must be placed in Mode 3 within 12 hours and in Mode 4 within 36 hours (BWR/6 plants only).

d. If the HPCI/High Pressure Core Spray (HPCS) system is inoperable, the RCIC system must be verified to be operable by administrative means within 1 hour and the HPCI/HPCS system restored to operable status within 14 days.

e. If one ADS valve is inoperable, it must be restored to operable status within 14 days.

f. If one ADS valve is inoperable and one low-pressure ECCS injection/spray subsystem is inoperable, the ADS valve must be restored to operable status within 72 hours or the low-pressure ECCS injection/spray subsystem must be restored to operable status within 72 hours.

g. If two or more ADS valves become inoperable, or the required actions described in items e and/or f cannot be met, 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.

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 two ECCS injection subsystems are inoperable or one ECCS injection subsystem and one ECCS spray system are inoperable, one ECCS injection/spray subsystem must be restored to operable status within 72 hours. If this required action cannot be met, the plant must be placed in Mode 3 within 12 hours. The plant is not taken into Mode 4 (BWR/6 plants only).

d. No change

e. No change

f. No change

g. If two or more ADS valves become inoperable or the required actions described in item e and/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 (RCIC/HPCI), and loss of the power conversion system (condenser/feedwater), and require activating the 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 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 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 (discussed in Reference 1), the change is acceptable.

3.2.3 TRS 4.5.1.4 and LCO 3.5.3 (BWR/4 only), Reactor Core Isolation Cooling (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.

[**Note:** Plant Applicability, BWR/4]

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, (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 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 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 (discussed in Reference 1), the change is acceptable.

3.2.4 TRS 4.5.1.6 and LCO 3.6.1.6 (BWR/4); TRS 5.5.2.5 and LCO 3.6.1.6 (BWR/6), Low-Low Set (LLS) Valves.

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

[**Note:** Plant Applicability, BWR 4/6]

Conditions requiring entry into end state: If one LLS valve is inoperable, it must be returned to operability within 14 days. If the LLS valve cannot be returned to operable status within the allotted time, the plant must be placed in Mode 3 within 12 hours and in Mode 4 within 36 hours.

Modification for end state required actions: The TS change would keep the plant in Mode 3 until the required repair actions are completed. The plant would not be taken into Mode 4 (cold shutdown).

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 Mode 4, the current end state. Going to Mode 4 for one LLS inoperable SRV would cause loss of the high-pressure steam-driven injection system (RCIC/HPCI), and loss of the power conversion system (condenser/feedwater), and would require activating the RHR system. With one LLS valve inoperable, the remaining valves are adequate to perform the required function. 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 during the infrequent and limited time in Mode 3 and the number of systems available in Mode 3, the 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. The change allows repairs of the inoperable SRV to be performed in a plant operating mode with lower risks.

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 Reference 1), the change is acceptable.

3.2.5 TRS 4.5.1.1, TRS 4.5.2.1 and LCO 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 (MG) 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.

[**Note:** Plant Applicability, BWR 4/6]

LCO: For Modes 1, 2, 3 and Modes 4 and 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(s) must be removed from service within 72 hours for one Electric Power Assembly (EPA) inoperable or within one hour for both EPAs inoperable. In Modes 1, 2, and 3, if the in-service power supply(s) 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.

Modification: The 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 TS, 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/HPCI) 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 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 (discussed in Reference 1), the change is acceptable.

3.2.6 TRS 4.5.1.19 and LCO 3.8.1(BWR/4); TRS 4.5.2.17 and LCO 3.8.1(BWR/6), AC Sources (Operating).

The purpose of the 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 (GDC) 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.

[**Note:** Plant Applicability, BWR 4/6]

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. Three EDGs,

c. Automatic Load Sequencers.

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

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 core damage frequency 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 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 (discussed in Reference 1), the change is acceptable.

3.2.7 TRS 4.5.1.20 and LCO 3.8.4 (BWR/4); TRS 4.5.2.18 and LCO 3.8.4 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.

The DC electrical system:

a. Provides the AC emergency power system with control power,

b. Provides motive and control power to selected safety related equipment, and

c. Provides power to preferred AC vital buses (via inverters).

[**Note:** Plant Applicability, BWR 4/6]

LCO: For Modes 1, 2 and 3, the following DC sources are required to be operable: BWR/4: The (Division 1 and Division 2 station service, and DG 1B, 2A, and 2C) DC electrical power systems shall be operable.

BWR/6: The (Divisions 1, 2, and 3) DC 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.

Modification for end state required actions: The TS change is to remove the requirement to place the plant in Mode 4, Required Actions in D.2 (BWR/4) and E.2 (BWR/6) are deleted.

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 Mode 3 end state, with one DC system inoperable. Events initiated by the loss of offsite power are dominant contributors to core damage frequency 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 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 (discussed in Reference 1), the change is acceptable.

3.2.8 TRS 4.5.1.21 and LCO 3.8.7 (BWR/4); TRS 4.5.2.19 and 3.8.7 (BWR/6), Inverters (Operating).

In Modes 1,2,and 3, the inverters provide the preferred source of power for the 120-VAC vital buses which power the reactor protection system (RPS) and the Emergency Core Cooling Systems (ECCS) initiation. The inverter can be powered from an internal AC source/rectifier or from the station battery.

[**Note:** Plant Applicability, BWR 4/6]

LCO: For Modes 1, 2, and 3 the following Inverters shall be operable:

BWR/4: The (Division 1 and Division 2) shall be operable.

BWR/6: The (Divisions 1, 2, and 3) 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 the required inverter for a period of 24 hours.

Modification for end state required actions: The TS change is to remove the requirement to place the plant in Mode 4. Required Actions in B.2 (BWR/4) and C.2 (BWR/6) are deleted.

Assessment: If one of the Inverters is inoperable, the remaining Inverters 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 an inoperable Inverter. Events initiated by the loss of offsite power are dominant contributors to core damage frequency 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 Inverter 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 Inverters during the infrequent and limited time in Mode 3 and the number of systems available in Mode 3, the 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 (discussed in Reference 1), the change is acceptable.

3.2.9 TRS 4.5.1.22 and LCO 3.8.9 (BWR/4); TRS 4.5.2.20 and LCO 3.8.9 (BWR/6), 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.

[**Note:** Plant Applicability, BWR 4/6]

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

BWR/4: The Division 1 and Division 2 AC, DC, and AC vital buses shall be operable.

BWR/6: The Divisions 1, 2, and 3 AC, DC, and AC vital buses 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 AC or one DC or one AC vital bus electrical power subsystem for a period of 8 hours, 2 hours and 2 hours, respectively (with a maximum 16 hour Completion Time limit from initial discovery of failure to meet the LCO, to preclude being in the LCO indefinitely).

Modification for end state required actions: The TS change is to remove the requirement to place the plant in Mode 4, Required Action in D.2 (BWR/4) and D.2 (BWR/6) are 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 loss of offsite power are dominant contributors to core damage frequency 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 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 (discussed in Reference 1), the change is acceptable.

3.2.10 TRS 4.5.1.5 and LCO 3.6.1.1, 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.

[**Note:** Plant Applicability, BWR 4/6]

LCO: The primary containment shall be operable.

Condition Requiring Entry into End State: 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).

Modification for End State Required Actions: 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 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 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, 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.

Finding: The requested change is acceptable. Note that the 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.

3.2.11 TRS 4.5.1.7 and LCO 3.6.1.7, Reactor Building-to-Suppression Chamber Vacuum Breakers(BWR/4 only).

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.

[**Note:** Plant Applicability, BWR/4]

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 Completion Times, the plant is place in hot shutdown. That is, modify Condition E to relate only to Condition C, delete Required Action E.2, and add Condition F, with Required Actions F.1 and F.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 the other Conditions (i.e., Conditions A, B, and D).

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 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: 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 Reference 1), the change is acceptable.

3.2.12 TRS 4.5.1.8 and LCO 3.6.1.8, Suppression Chamber-to-Drywell Vacuum Breakers(BWR/4 only).

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.

[**Note:** Plant Applicability, BWR/4]

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 Completion Times, the plant is placed in hot shutdown. That is, modify Condition C to relate only to Condition A, and delete Required Action C.2, and add Condition D, 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, 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 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 Reference 1), the change is acceptable.

3.2.13 TRS 4.5.1.9, TRS 4.5.2.8, and LCO 3.6.1.9, Main Steam Isolation Valve (MSIV) Leakage Control System (LCS).

The MSIV LCS supplements the isolation function of the MSIVs by processing the fission products that could leak through the closed MSIVs after core damage, assuming leakage rate limits which are based on a large LOCA.

[**Note:** Plant Applicability, BWR 4/6]

LCO: Two MSIV LCS subsystems shall be operable.

Condition Requiring Entry into End State: If one MSIV LCS subsystem is inoperable, it must be restored to operable status within 30 days (Required Action A.1). If both MSIV LCS subsystems are inoperable, one of the MSIV LCS subsystems must be restored to operable status within seven days (Required Action B.1). If the MSIV LCS subsystems 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 BWROG topical report has determined that this system is not significant in BWR PRAs and, based on a BWROG program, many plants have eliminated the system altogether. The unavailability of one or both MSIV LCS 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 MSIV LCS system (*i.e.*, the frequency with which the system is expected to be challenged to mitigate offsite radiation releases resulting from MSIV leaks 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 staff's risk argument for approval of TRS 4.5.1.9, TRS 4.5.2.8, and LCO 3.6.1.9, ''Main Steam Isolation Valve (MSIV) Leakage Control System (LCS).'' The argument for staying in Mode 3 instead of going to Mode 4 to repair the MSIV LCS 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 Regulatory Guides 1.174 and 1.177, support the conclusion that the plant in Mode 3 is as safe as Mode 4 (if not safer) for repairing an inoperable MSIV LCS system. Personnel safety must be considered separately.

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 Reference 1), the change is acceptable.

3.2.14 TRS 4.5.1.11 and LCO 3.6.2.4, Residual Heat Removal (RHR) Suppression Pool Spray(BWR/4 only).

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.

[**Note:** Plant Applicability, BWR/4]

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 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 eight 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 three 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 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 Regulatory Guides 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 Reference 1), the change is acceptable.

3.2.15 TRS 4.5.1.12, TRS 4.5.2.10, and LCO 3.6.4.1, Secondary Containment.

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

[**Note:** Plant Applicability, BWR 4/6]

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 four 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 standby gas treatment 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 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 staff's approval relies upon the primary containment, and all other primary and secondary containment-related functions, to still be operable, including the standby gas treatment system, for maintaining defense-in-depth while in this end state.

3.2.16 TRS 4.5.1.13, TRS 4.5.2.11, and LCO 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.

Applicability: BWR4/6.

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

Section 6 of reference 6 summarizes the 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 Regulatory Guides 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 Reference 1), the change is acceptable.

3.2.17 TRS 4.5.1.14 and LCO 3.7.1, Residual Heat Removal Service Water (RHRSW) System (BWR/4 only).

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.

[**Note:** Plant Applicability, BWR/4]

LCO: Two RHRSW 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 RHRSW pump is inoperable (Condition A), it must be restored to operable status within 30 days (Required Action A.1).

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

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

d. If the required action and associated completion time cannot be met within the allotted time (Condition E), the plant must be placed in Mode 3 within 12 hours (Required Action E.1) and in Mode 4 within 36 hours (Required Action E.2). {Note: Condition D addresses both RHRSW subsytems inoperable for reason other than Condition B, and its Required Action D.1 is not affected by this change.

Modification for End State Required Actions: Renumber Conditions D (and Required Action D.1), and E (and Required Actions E.1 and E.2), to Conditions E (and Required Action E.1) and F (and Required Actions F.1 and F.2), respectively. Modify new Condition F to address new Condition E, which maintains the existing requirements with respect to both RHR subsystems being inoperable for reasons other than Condition B. Add a new Condition D, which establishes requirements for existing Conditions A, B, and C, that are similar to existing Condition E but without Required Action E.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 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 (discussed in Reference 1), the change is acceptable.

3.2.18 TRS 4.5.1.15 and LCO 3.7.2, Plant Service Water (PSW) System and Ultimate Heat Sink (UHS) (BWR/4 only).

The PSW system (in conjunction with the UHS) is designed to provide cooling water for the removal of heat from certain safe shutdown-related equipment heat exchangers following a DBA or transient.

[**Note:** Plant Applicability, BWR/4]

LCO: Two PSW subsystems and UHS 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 PSW pump is inoperable (Condition A), it must be restored to operable status within 30 days (Required Action A.1).

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

c. If the required action and associated completion time cannot be met 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: Renumber unaffected Conditions C, D, E, and F to Conditions D, E, F, and G respectively, and renumber associated Required Actions accordingly. Add a new Condition C, for the Required Actions and associated Completion Time of Conditions A and B not met, with a Required Action C.1, to be in Mode 3 in a Completion Time of 12 hours. Change the new Condition G to read, ''Required Action and associated Completion Time of Condition E not met, OR Both [PSW subsystems inoperable for reasons other than Condition(s) B [and D], [OR [UHS] inoperable for reasons other than Conditions D [or E].''

Assessment: The BWROG topical report performed a comparative PRA evaluation of the core damage risks associated with 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. With one pump inoperable in one or more subsystems, the remaining pumps are adequate to perform the PSW heat removal function. 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 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 (discussed in Reference 1), the change is acceptable.

3.2.19 TRS 4.5.1.16 and LCO 3.7.4, Main Control Room Environmental Control (MCREC) System (BWR/4 only).

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

[**Note:** Plant Applicability, BWR/4]

LCO: Two MCREC subsystems shall be operable.

Condition Requiring Entry into End State: If one MCREC subsystem is inoperable, it must be restored to operable status within seven days (Required Action A.1). If the MCREC 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 MCREC 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 Completion Time of ''12 hours.''

Assessment: The unavailability of one or both MCREC 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 MCREC 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 Regulatory Guide 1.177 for much higher consequence risks, such as large early release.

Section 6 of reference 6 summarizes the staff's risk argument for approval of TRS 4.5.1.16, and LCO 3.7.4, ''Main Control Room Environmental Control (MCREC) System.'' The argument for staying in Mode 3 instead of going to Mode 4 to repair the MCREC 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 Regulatory Guides 1.174 and 1.177, support the conclusion that Mode 3 is as safe as Mode 4 (if not safer) for repairing an inoperable MCREC 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 Reference 1), the change is acceptable.

3.2.20 TRS 4.5.1.17 and LCO 3.7.5, Control Room Air Conditioning (AC) System (BWR/4 only).

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

[**Note:** Plant Applicability, BWR/4]

LCO: Two control room AC subsystems shall be operable.

Condition Requiring Entry into End State: If one control room 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 AC subsystems are inoperable, 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 Completion Time of ''12 hours.''

Assessment: The unavailability of one or both 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 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 Regulatory Guide 1.177 for much higher consequence risks, such as large early release.

Section 6 of reference 6 summarizes the staff's risk argument for approval of TRS 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 AC 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 Regulatory Guides 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 (discussed in Reference 1), the change is acceptable.

3.2.21 TRS 4.5.1.18 and LCO 3.7.6, Main Condenser Off gas (BWR/4 only).

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 offsite dose limits will not be exceeded during postulated accidents. The main condenser Off gas (MCOG) gross gamma activity rate is an initial condition of a DBA which assumes a gross failure of the MCOG system pressure boundary.

[**Note:** Plant Applicability, BWR/4]

LCO: The gross gamma activity rate of the noble gases measured at the main condenser evacuation system pretreatment monitor station shall be ≤240 mCi/second after decay of 30 minutes.

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

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

b. The steam jet air ejector (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 main condenser Off gas (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 Regulatory Guide 1.177 for much higher consequence risks, such as large early release.

Section 6 of reference 6 summarizes the staff's risk argument for approval of TRS 4.5.1.18 and LCO 3.7.6, ''Main Condenser Off gas.'' 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 Regulatory Guides 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 (discussed in Reference 1), the change is acceptable.

3.2.22 TRS 4.5.2.6 and LCO 3.6.1.7, Residual Heat Removal (RHR) Containment Spray System (BWR/6 only).

The primary containment must be able to withstand a postulated bypass leakage pathway that allows the passage of steam from the drywell directly into the primary containment airspace, bypassing the suppression pool. The primary containment also must be able to withstand a low energy steam release into the primary containment airspace. The RHR Containment Spray System is designed to mitigate the effects of bypass leakage and low energy line breaks.

[**Note:** Plant Applicability, BWR/6]

LCO: Two RHR containment spray subsystems shall be operable.

Condition Requiring Entry into End State: If one RHR Containment Spray Subsystem is inoperable, it must be restored to operable status within seven days (Required Action A.1). If two RHR Containment Spray Subsystems are inoperable, one of them must be restored to operable status within eight hours (Required Action B.1). If the RHR Containment Spray System 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 primary containment is designed with a suppression pool so that, in the event of a LOCA, steam released from the primary system is channeled through the suppression pool water and condensed without producing significant pressurization of the primary containment. The primary containment is designed so that with the pool initially at the minimum water level and the worst single failure of the primary containment heat removal systems, suppression pool energy absorption combined with subsequent operator controlled pool cooling will prevent the primary containment pressure from exceeding its design value. However, the primary containment must also withstand a postulated bypass leakage pathway that allows the passage of steam from the drywell directly into the primary containment airspace, bypassing the suppression pool. The primary containment also must withstand a postulated low energy steam release into the primary containment airspace. The main function of the RHR containment spray system is to suppress steam, which is postulated to be released into the primary containment airspace through a bypass leakage pathway and a low energy line break under DBA conditions, without producing significant pressurization of the primary containment (*i.e.*, ensure that the pressure inside primary containment remains within analyzed design limits).

Under the conditions assumed in the DBA, steam blown down from the break could find its way into the primary containment through a bypass leakage pathway. In addition to the DBA, a postulated low energy pipe break could add more steam into the primary containment airspace. Under such an extremely unlikely scenario (very small frequency of a DBA combined with the likelihood of a bypass pathway and a concurrent low energy pipe brake inside the primary containment), the RHR containment spray system could be needed to condense steam so that the pressure inside the primary containment remains within analyzed design limits. Furthermore, containments have considerable margin to failure above the design limit (it is very likely that the containment will be able to withstand pressures as much as three times the design limit). For these reasons, the unavailability of one or both RHR containment 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 containment 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 three 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 containment spray system.

Section 6 of reference 6 summarizes the staff's risk argument for approval of TRS 4.5.2.6 and LCO 3.6.1.7, ''Residual Heat Removal (RHR) Containment Spray

System.'' The argument for staying in Mode 3 instead of going to Mode 4 to repair the RHR containment 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. The risk and defense-in-depth arguments, used according to the ''integrated decision-making'' process of Regulatory Guides 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 containment 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 Reference 1), the change is acceptable.

3.2.23 TRS 4.5.2.7 and LCO 3.6.1.8, Penetration Valve Leakage Control System (PVLCS) (BWR/6 only).

The PVLCS supplements the isolation function of primary containment isolation valves (PCIVs) in process lines that also penetrate the secondary containment. These penetrations are sealed by air from the PVLCS to prevent fission products leaking past the isolation valves and bypassing the secondary containment after a design basis loss-of-coolant accident (LOCA).

[**Note:** Plant Applicability, BWR/6]

LCO: Two PVLCS subsystems shall be operable.

Condition Requiring Entry into End State: If one PVLCS subsystem is inoperable, it must be restored to operable status within 30 days (Required Action A.1). If two PVLCS subsystems are inoperable, one of the PVLCS subsystems must be restored to operable status within seven days (Required Action B.1). If the PVLCS 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).

Assessment: The BWROG topical report has determined that this system is not significant in BWR PRAs. The unavailability of one or both PVLCS 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 PVLCS system (*i.e.*, the frequency with which the system is expected to be challenged to prevent fission products leaking past the isolation valves and bypassing the secondary containment) 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 staff's risk argument for approval of TRS 4.5.2.7 and LCO 3.6.1.8, ''Penetration Valve Leakage Control System (PVLCS).'' The argument for staying in Mode 3 instead of going to Mode 4 to repair the PVLCS 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 Regulatory Guides 1.174 and 1.177, support the conclusion that Mode 3 is as safe as Mode 4 (if not safer) for repairing an inoperable PVLCS 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 Reference 1), the change is acceptable.

3.2.24 TRS 4.5.1.10, TRS 4.5.2.9 and LCO 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.

[**Note:** Plant Applicability, BWR 4/6]

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 seven days (Required Action A.1). If the RHR suppression pool spray subsystem cannot be restored to operable status within the allotted time (Condition B), 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, and retain Condition B and Required Action B.1 for one RHR suppression pool spray subsystem inoperable. Add Condition C, with Required Actions C.1 and C.2, identical to existing Condition B, with Required Actions B.1 and B.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 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 (discussed in Reference 1), the change is acceptable.

3.2.25 TRS 4.5.2.12 and LCO 3.6.5.6, Drywell Vacuum Relief System (BWR/6 only).

The Mark III pressure suppression containment is designed to condense, in the suppression pool, the steam released into the drywell in the event of a loss-of-coolant accident (LOCA). The steam discharging to the pool carries the non-condensibles from the drywell. Therefore, the drywell atmosphere changes from low humidity air to nearly 100% steam (no air) as the event progresses. When the drywell subsequently cools and depressurizes, non-condensibles in the drywell must be replaced to avoid excessive weir wall overflow into the drywell. Rapid weir wall overflow must be controlled in a large break LOCA, so that essential equipment and systems located above the weir wall in the drywell are not subjected to excessive drag and impact loads. The drywell post-LOCA and the drywell purge vacuum relief subsystems are the means by which non-condensibles are transferred from the primary containment back to the drywell.

[**Note:** Plant Applicability, BWR/6]

LCO: Two drywell post-LOCA and two drywell purge vacuum relief subsystems shall be operable.

Condition Requiring Entry into End State: If one or two drywell post-LOCA vacuum relief subsystems are inoperable (Condition A), or if one drywell purge vacuum relief subsystem is inoperable (Condition B), for reasons other than being not closed, the subsystem(s) must be restored to operable status within 30 days (Required Actions B.1 and C.1, respectively). If the required actions cannot be completed within the allotted time, the plant must be placed in Mode 3 within 12 hours and in Mode 4 within 36 hours.

Modification for End State Required Actions: Renumber Conditions D, E, F and G, to Conditions E, F, G, and H respectively, and renumber associated Required Actions accordingly. Add a new Condition D for when Required Action and associated Completion Time of Condition B or C not met, with Required Action D.1 to be in Mode 3 in a Completion Time of 12 hours. Change new Condition G to read, ''Required Action and associated Completion Time of Condition A, E or F not met.''

Assessment: The BWROG topical report has determined that the specific failure conditions of interest are not risk significant in BWR PRAs. With one or two drywell post-LOCA vacuum relief subsystems inoperable or one drywell purge vacuum relief subsystem inoperable, for reasons other than not being closed, the remaining operable vacuum relief subsystems are adequate to perform the depressurization mitigation 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 (discussed in Reference 1), the change is acceptable.

3.2.26 TRS 4.5.2.13 and LCO 3.7.1, Standby Service Water (SSW) System and Ultimate Heat Sink (UHS) (BWR/6 only).

The SSW system (in conjunction with the UHS) is designed to provide cooling water for the removal of heat from certain safe shutdown-related equipment heat exchangers following a DBA or transient.

[**Note:** Plant Applicability, BWR/6]

LCO: Division 1 and 2 SSW subsystems and UHS shall be operable.

Condition Requiring Entry into End State: If one or more cooling towers with one cooling tower fan is inoperable (Condition A), the cooling tower fan(s) must be restored to operable status within seven days (Required Action A.1). If one SSW subsystem is inoperable for reasons other than Condition A (Condition C), the SSW subsystem must be restored to operable status within 72 hours (Required Action C.1). If the required action(s) and associated completion time(s) (of Conditions A or C) cannot be met (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).

Modification: The existing second and third conditions of existing Condition D have been transferred to a new Condition E in an unchanged form (with Required Actions E.1 and E.2 identical to existing Required Actions D.1 and D.2). Existing Condition B with its associated Required Actions and Associated Completion Times, has been transferred to a new Condition D in an unchanged form. Existing Condition C, with its associated Required Action and Associated Completion Time, has been moved to a new Condition B in unchanged form. A new Condition C has been created. If the Required Actions and Associated Completion Times for new Condition A or B are not met (new Condition C), then the plant must be placed in Mode 3 in 12 hours (new Required Action C.1).

Assessment: The BWROG topical report determined that the specific failure condition of interest is not risk significant in BWR PRAs. With the specified inoperable components/subsystems, a sufficient number of operable components/subsystems are still available to perform the heat removal 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 (discussed in Reference 1), the change is acceptable.

3.2.27 TRS 4.5.2.14 and LCO 3.7.3, Control Room Fresh Air (CRFA) System (BWR/6 only).

The CRFA system provides a radiologically controlled environment from which the unit can be safely operated following a DBA. The CRFA system consists of two independent and redundant high efficiency air filtration subsystems for treatment of recirculated air or outside supply air. Each subsystem consists of a demister, an electric heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, a fan, and the associated ductwork and dampers. Demisters remove water droplets from the airstream. Prefilters and HEPA filters remove particulate matter that may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

[**Note:** Plant Applicability, BWR/6]

LCO: Two CRFA subsystems shall be operable.

Condition Requiring Entry into End State: If one CRFA subsystem is inoperable (Condition A), it must be restored to operable status within seven days (Required Action A.1). If two CRFA subsystems are inoperable (Condition B for control room boundary and Condition E for reasons for inoperability), one CRFA subsystem must be restored to operable status in 24 hours (Required Action B.1) or enter LCO 3.0.3 (Required Action E.1). If Conditions A or B, and associated Required Actions A.1 and B.1) cannot be met in the required Completion 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: Delete Required Action C.2, and change Required Action E.1 to ''Be in Mode 3'' within a Completion Time of ''12 hours.''

Assessment: The unavailability of one or both CRFA 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 CRFA 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 Regulatory Guide 1.177 for much higher consequence risks, such as large early release.

Section 6 of reference 6 summarizes the staff's risk argument for approval of TRS 4.5.2.14 and LCO 3.7.3, ''Control Room Fresh Air (CRFA) System.'' The argument for staying in Mode 3 instead of going to Mode 4 to repair the CRFA 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 Regulatory Guides 1.174 and 1.177, support the conclusion that Mode 3 is as safe as Mode 4 (if not safer) for repairing an inoperable CRFA 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 Reference 1), the change is acceptable.

3.2.28 TRS 4.5.2.15 and LCO 3.7.4, Control Room Air Conditioning (CRAC) System (BWR/6 only).

The control room AC system provides temperature control for the control room following control room isolation. The control room AC system consists of two independent, redundant subsystems that provide cooling and heating of recirculated control room air. Each subsystem consists of heating coils, cooling coils, fans, chillers, compressors, ductwork, dampers, and instrumentation and controls to provide for control room temperature control. The control room AC system is designed to provide a controlled environment under both normal and accident conditions. A single subsystem provides the required temperature control to maintain a suitable control room environment for a sustained occupancy of 12 persons.

[Note: Plant Applicability, BWR/6]

LCO: Two control room AC subsystems shall be operable.

Condition Requiring Entry into End State: If one control room AC subsystem is inoperable, it 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 AC subsystems are inoperable, LCO 3.0.3 must be entered immediately (Condition D).

Modification for End State Required Actions: 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 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 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 which leads to core damage) 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 staff's risk argument for approval of TRS 4.5.2.15 and LCO 3.7.4, ''Control Room Air Conditioning (AC) System.'' The argument for staying in Mode 3 instead of going to Mode 4 to repair the CRAC 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 Regulatory Guides 1.174 and 1.177, support the conclusion that Mode 3 is as safe as Mode 4 (if not safer) for repairing an inoperable CRAC 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 Reference 1), the change is acceptable.

3.2.29 TRS 4.5.2.16 and LCO 3.7.5, Main Condenser Off gas (BWR/6 only).

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 offsite dose limits will not be exceeded during postulated accidents.

[Note: Plant Applicability, BWR/6]

LCO: The gross gamma activity rate of the noble gases measured at the Off gas recombiner effluent shall be ≤380 mCi/second after decay of 30 minutes.

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

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

b. The steam jet air ejector (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 main condenser Off gas (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 Regulatory Guide 1.177 for much higher consequence risks, such as large early release.

Section 6 of reference 6 summarizes the staff's risk argument for approval of TRS 4.5.2.16 and LCO 3.7.5, ''Main Condenser Off gas.'' 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 Regulatory Guides 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 (discussed in Reference 1), the change is acceptable.

4.0 State Consultation

In accordance with the Commission's regulations, the [________] State official was notified of the proposed issuance of the amendment. The State official had [(1) no comments or (2) the following comments—with subsequent disposition by the staff].

5.0 Environmental Consideration

The amendment changes requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that adopting TSTF–423, Rev 0, involves no significant hazards considerations, and there has been no public comment on the finding in **Federal Register** Notice 70 FR 74037, December 14, 2005. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 Conclusion

The Commission has concluded, 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.

7.0 References

1. NEDC–32988–A, Revision 2, ''Technical Justification to Support Risk-Informed Modification to Selected Required Action End States for BWR Plants,'' December 2002.

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

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

4. Regulatory Guide 1.182, ''Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants,'' May 2000. (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, September 27, 2002. (ML022700603)

7. TSTF–423, Revision 0, ''Technical Specifications End States, NEDC–32988–A.''

8. TSTF–IG–05–02, Implementation Guidance for TSTF–423, Revision 0, ''Technical Specifications End States, NEDC–32988–A,'' September 2005.

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. (ML003740133)

10. Regulatory Guide 1.177, ''An Approach for Pant Specific Risk-Informed Decision Making: Technical Specifications,'' USNRC, August 1998. (ML003740176)

Proposed No Significant Hazards Consideration Determination

Description of Amendment Request: A change is proposed to the technical specifications (TS) of [plant name], consistent with Technical Specifications Task Force (TSTF) change TSTF–423 to the standard technical specifications (STS) for BWR Plants (NUREG 1433 and NUREG 1434) to allow, for some systems, entry into hot shutdown rather than cold shutdown to repair equipment, if risk is assessed and managed consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4). Changes proposed in will be made to the [plant name] TS for selected Required Action end states providing this allowance.

Basis for proposed no-significant-hazards-consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of no-significant-hazards-consideration is presented below:

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change allows a change to certain required end states when the TS Completion Times for remaining in power operation will be exceeded. Most of the requested technical specification (TS) changes are to permit an end state of hot shutdown (Mode 3) rather than an end state of cold shutdown (Mode 4) contained in the current TS. The request was limited to: (1) Those end states where 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 technical specification, and (3) the primary purpose is to correct the initiating condition and return to power operation as soon as is practical. Risk insights from both the qualitative and quantitative risk assessments were used in specific TS assessments. Such assessments are documented in Section 6 of GE NEDC–32988, Revision 2, ''Technical Justification to Support Risk Informed Modification to Selected Required Action End States for BWR Plants.'' They provide an integrated discussion of deterministic and probabilistic issues, focusing on specific technical specifications, which are used to support the proposed TS end state and associated restrictions. The staff finds that the risk insights support the conclusions of the specific TS assessments. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident after adopting proposed TSTF–423, are no different than the consequences of an accident prior to adopting TSTF–423. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). If risk is assessed and managed, allowing a change to certain required end states when the TS Completion Times for remaining in power operation are exceeded, i.e., entry into hot shutdown rather than cold shutdown to repair equipment, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change and the commitment by the licensee to adhere to the guidance in TSTF–IG–05–02, Implementation Guidance for TSTF–423, Revision 0, ''Technical Specifications End States, NEDC–32988–A,'' will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The proposed change allows, for some systems, entry into hot shutdown rather than cold shutdown to repair equipment, if risk is assessed and managed. The BWROG's risk assessment approach is comprehensive and follows staff guidance as documented in RGs 1.174 and 1.177. In addition, the analyses show that the criteria of the three-tiered approach for allowing TS changes are met. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in RG 1.177. A risk assessment was performed to justify the proposed TS changes. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

The Following Example of an Application Was Prepared by the NRC Staff To Facilitate Use of the Consolidated Line Item Improvement Process (CLIIP). The Model Provides the Expected Level of Detail and Content for an Application To Adopt TSTF–423, Revisions 0, ''Risk-Informed Modifications to Selected Required Action End States,'' for BWR Plants (and Adoption of a Technical Specification Bases Control Program)* Using CLIIP. Licensees Remain Responsible for Ensuring That Their Actual Application Fulfills Their Administrative Requirements as Well as Nuclear Regulatory Commission Regulations.

U.S. Nuclear Regular Commission,
Document Control Desk,
Washington, DC 20555.

Subject: Plant Name, Docket No. 50—Application for Technical Specification Change TSTF–423, Risk Informed Modification To Selected Required Action End States for BWR Plants, (and Adoption of a Technical Specifications Bases Control Program)* Using the Consolidated Line Item Improvement Process

Gentleman: In accordance with the provisions of 10 CFR 50.90 [Licensee] is submitting a request for an amendment to the technical specifications (TS) for [Plant Name, Unit Nos.].

The proposed amendment would modify TS to risk-inform requirements regarding selected Required Action End States, (and, in conjunction with the proposed change, TS requirements for a Bases control program consistent with TS Bases Control Program described in Section 5.5 of the BWR Standard Technical Specifications.)*

Enclosure 1 provides a description of the proposed change, the requested confirmation of applicability, and plant-specific verifications. Enclosure 2 provides the existing TS pages marked up to show the proposed change. Enclosure 3 provides revised (clean) TS pages. Enclosure 4 provides a summary of the regulatory commitments made in this submittal. Enclosure 5 provides the existing TS Bases pages marked up to show the proposed change (*for information only).*)

[Licensee] requests approval of the proposed license amendment by [Date], with the amendment being implemented [by Date or Within X Days].

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated [State] Official.

* If not already in the facility Technical Specifications.

I declare under penalty of perjury under the laws of the United States of America that I am authorized by [Licensee] to make this request and that the foregoing is true and correct. (Note that request may be notarized in lieu of using this oath or affirmation statement).

If you should have any questions regarding this submittal, please contact [Name, Telephone Number]

Sincerely,
[Name, Title]

Enclosures:
1. Description and Assessment
2. Proposed Technical Specification Changes
3. Revised Technical Specification Pages
4. Regulatory Commitments
5. Proposed Technical Specification Bases Changes

cc: NRC Project Manager
NRC Regional Office
NRC Resident Inspector
State Contact

ENCLOSURE 1

Description and Assessment

1.0 Description

The proposed amendment would modify technical specifications to risk-inform requirements regarding selected Required Action End States.[1]

The changes are consistent with Nuclear Regulatory Commission (NRC) approved Industry/Technical Specification Task Force (TSTF) TSTF–423 Revision 0. The availability of this TS improvement was published in the **Federal Register** on [Date] as part of the consolidated line item improvement process (CLIIP).

2.0 Assessment

2.1 Applicability of Topical Report, TSTF–423, and Published Safety Evaluation

[Licensee] has reviewed GE topical report (Reference 1), TSTF–423 (Reference 2), and the NRC model safety evaluation (Reference 3) as part of the CLIIP. [Licensee] has concluded that the information in the GE topical report and TSTF–423, as well as the safety evaluation prepared by the NRC staff are applicable to [Plant, Unit Nos.] and justify this amendment for the incorporation of the changes to the [Plant] TS. [Note: Only those changes proposed in TSTF–423 are addressed in the model SE. The model SE and associated topical report address the entire fleet of BWR plants, and the plants adopting TSTF–423 must confirm the applicability of the changes to their plant.]

2.2 Optional Changes and Variations

[Licensee] is not proposing any variations or deviations from the GE topical report and the TS changes described in the TSTF–423 Revision 0 or the NRC staff's model safety evaluation dated [Date]. [Note: The CLIIP does not prevent licensees from requesting an alternate approach or proposing changes without the requested Bases or Bases control program. However, deviations from the approach recommended in this notice may require additional review by the NRC staff and may increase the time and resources needed for the review. Significant variations

[1] [In conjunction with the proposed change, technical specifications (TS) requirements for a Bases Control Program, consistent with the TS Bases Control Program described in Section 5.5 of the applicable vendor's standard TS (STS), shall be incorporated into the licensee's TS, if not already in the TS.]

from the approach, or inclusion of additional changes to the license, will result in staff rejection of the submittal. Instead, licensees desiring significant variations and/or additional changes should submit a LAR that does not claim to adopt TSTF–423.]

3.0 Regulatory Analysis

3.1 No Significant Hazards Consideration Determination

[Licensee] has reviewed the proposed no significant hazards consideration determination (NSHCD) published in the **Federal Register** as part of the CLIIP. [Licensee] has concluded that the proposed NSHCD presented in the **Federal Register** notice is applicable to [Plant] and is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

3.2 Verification and Commitments

As discussed in the notice of availability published in the **Federal Register** on [Date] for this TS improvement, plant-specific verifications were performed as follows:

[Licensee] commits to the regulatory commitments in Enclosure 4. In addition, [Licensee] has proposed TS Bases consistent with the GE topical report and TSTF–423, which provide guidance and details on how to implement the new requirements. Implementation of TSTF–423 requires that risk be managed and assessed, and the licensee's configuration risk management program is adequate to satisfy this requirement. The risk assessment need not be quantified, but may be a qualitative assessment of the vulnerability of systems and components when one or more systems are not able to perform their associated function. Finally, [Licensee] has a Bases Control Program consistent with Section 5.5 of the Standard Technical Specifications (STS).

4.0 Environmental Evaluation

The amendment changes requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment adopting TSTF–423, Rev 0, involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that TSTF–423, Rev 0, involves no significant hazards considerations, and there has been no public comment on the finding in **Federal Register** Notice 70 FR 74037, December 14, 2005. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 References

1. NEDC–32988–A, Revision 2, ''Technical Justification to Support Risk-Informed Modification to Selected Required Action End States for BWR Plants,'' December 2002.

2. TSTF–423, Revision 0, ''Technical Specifications End States, NEDC–32988–A.''

3. **Federal Register**, Vol. XX, No. XX, p. XXXXX, ''Notice of Availability of Model Application Concerning Technical Specifications for Boiling Water Reactor Plants to Risk-Inform Requirements Regarding Selected Required Action End States Using the Consolidated Line Item Improvement Process, and NRC Model Safety Evaluation,'' [Date].

Enclosure 2

Proposed Technical Specification Changes (Mark-up)

Enclosure 3

Proposed Technical Specification Pages

[Clean copies of Licensee specific Technical Specification (TS) pages, corresponding to the TS pages changed by TSTF–423, Rev 0, are to be included in Enclosure 3]

Enclosure 4

List of Regulatory Commitments

The following table identifies those actions committed to by [Licensee] in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to [Contact Name].

Regulatory committments	Due date/event
[Licensee] 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, or implement with amendment].
[Licensee] will follow the guidance established in TSTF–IG–05–02, Implementation Guidance for TSTF–423, Revision 0, ''Technical Specifications End States, NEDC–32988–A,'' September 2005.	[Implement with amendment, when TS Required Action End State remains within the Applicability of TS].

Enclsoure 5

Proposed Changes to Technical Specification Bases Pages

[FR Doc. 06–2803 Filed 3–22–06; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549.

Extension:

Rule 15c2–8, SEC File No. 270–421, OMB Control No. 3235–0481.

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*), the Securities and Exchange Commission (''Commission'') has submitted to the Office of Management and Budget (''OMB'') a request for extension of the previously approved collection of information discussed below.

- Rule 15c2–8 Delivery of Prospectus

Rule 15c2–8 under the Securities Exchange Act of 1934 requires broker-dealers to deliver preliminary or final prospectuses to specified persons in association with securities offerings. This requirement ensures that information concerning issuers flows to purchasers of the issuers' securities in a timely fashion. It is estimated that there are approximately 8,000 broker-dealers, any of which potentially may participate in an offering subject to Rule 15c2–8. The Commission estimates that Rule 15c2–8 creates approximately 10,600 burden hours with respect to 120 initial public offerings and 460 other offerings. Please note that an agency may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid control number.

Written comments regarding the above information should be directed to the following persons: (i) Desk Officer for the Securities and Exchange Commission, Office of Information and Regulatory Affairs, Office of Management and Budget, Room 10102, New Executive Office Building, Washington, DC 20503 or by sending an e-mail to *David_Rostker@omb.eop.gov*; and (ii) R. Corey Booth, Director/Chief Information Officer, Securities and Exchange Commission, C/O Shirley Martinson, 6432 General Green Way, Alexandria, Virginia 22312; or send an e-mail to: *PRA_Mailbox@sec.gov*.