Technical Specifications Task Force
Improved Standard Technical Specifications Change Traveler

Clarify Applicability of BWR/4 TS 3.6.2.5 and TS 3.6.3.2

NUREGs Affected: ☐ 1430 ☐ 1431 ☐ 1432 ☑ 1433 ☐ 1434 ☐ 2194

Classification: 1) Technical Change

Recommended for CLIIP?: Yes

Correction or Improvement: Correction

NRC Fee Status: Not Exempt

Changes Marked on ISTS Rev 4.0

See attached.

Revision History

OG Revision 0 Revision Status: Closed

Revision Proposed by: NRC

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 18-May-17

Owners Group Comments
(No Comments)

Owners Group Resolution: Superceeded Date: 29-Sep-17

OG Revision 1 Revision Status: Closed

Revision Proposed by: BWROG

Revision Description:
Replaced with Revision 1 based on BWROG comments.

Owners Group Review Information

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(No Comments)

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TSTF Review Information

TSTF Received Date: 02-Oct-17 Date Distributed for Review 02-Oct-17

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 19-Dec-17

OG Revision 1 **Revision Status: Closed**

NRC Review Information

NRC Received Date: 19-Dec-17

NRC Comments:

Presubmittal discussion held with NRC on November 9, 2017. No changes identified.

TSTF Revision 1 **Revision Status: Active**

Revision Proposed by: NRC

Revision Description:
Revised to address NRC comments during a May 2, 2018 teleconference.

Includes the GE Safety Communication SC02-10, referenced in the traveler, as an attachment.

Clarifies that the change to TS 3.6.2.5, " Drywell-to-Suppression Chamber Differential Pressure," is only applicable to BWRs with a Mark 1 containment.

Clarifies that the traveler is not a technical change. Indefinite plant operation in Mode 1 with power less than or equal to 15% RTP is currently allowed by Required Action B.1.

Revise the justification to the change to TS 3.6.3.2, " Primary Containment Oxygen Concentration" to state:
oThe traveler is not a technical change. Indefinite plant operation in Mode 1 with power less than or equal to 15% RTP is currently allowed by Required Action B.1
oThe technical basis for the existing 15% allowance is not being changed. The proposed change is a clarification to the presentation of the existing 15% allowance, not a change to that limit.
oThe 24 hour startup/shutdown allowance is only applicable when power is > 15%.

Owners Group Review Information

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Owners Group Comments
(No Comments)

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TSTF Review Information

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TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 31-May-18

NRC Review Information

NRC Received Date: 31-May-18

TSTF Revision 1 **Revision Status: Active**

NRC Comments:

Draft provided to NRC on 5/31/18.

Affected Technical Specifications

Appl. 3.6.2.5	Drywell-to-Suppression Chamber Differential Pressure
Appl. 3.6.2.5 Bases	Drywell-to-Suppression Chamber Differential Pressure
SR 3.6.2.5.1	Drywell-to-Suppression Chamber Differential Pressure
SR 3.6.2.5.1 Bases	Drywell-to-Suppression Chamber Differential Pressure
Appl. 3.6.3.2	Primary Containment Oxygen Concentration
Appl. 3.6.3.2 Bases	Primary Containment Oxygen Concentration
SR 3.6.3.2.1	Primary Containment Oxygen Concentration
SR 3.6.3.2.1 Bases	Primary Containment Oxygen Concentration

1. SUMMARY DESCRIPTION

The Applicability of Technical Specification (TS) 3.6.2.5, "Drywell-to-Suppression Chamber Differential Pressure," and TS 3.6.3.2, "Primary Containment Oxygen Concentration," requires the associated limiting conditions for operation (LCO) to be met when the unit is in Mode 1 during the time period: a. from [24] hours after Thermal Power is > [15]% Rated Thermal Power (RTP) following startup, to [24] hours prior to reducing Thermal Power to < [15]% RTP prior to the next scheduled reactor shutdown. This change clarifies the Applicability presentation of these Specifications in NUREG-1433, "Standard Technical Specifications General Electric BWR/4 Plants,[1]" to by stating that these Specifications are applicable in MODE 1 with Thermal Power > [15]% RTP, while maintaining the [24]-hour allowance prior to power exceeding [15]% RTP, and prior to reducing power to below [15]% RTP as Notes in the respective Surveillance Requirements.

2. DETAILED DESCRIPTION

2.1. System Description and Operation

There are three containment designs used in the various boiling water reactor (BWR) plants. BWR/2, BWR/3, and early model BWR/4 plants have the Mark I containment. Later model BWR/4 and BWR/5 plants have the Mark II containment. BWR/6 plants have the Mark III containment.

The Mark I containment consists of a drywell (in the shape of an inverted light bulb), a suppression chamber (in the shape of a toroid), and a network of vents which extend radially outward and downward from the drywell to the suppression chamber. The Mark II containment consists of a drywell (in the shape of a truncated cone), a suppression chamber directly below the drywell (in the shape of a right circular cylinder), and a network of vertical vents extending downward from the drywell to the suppression chamber. The Mark III containment is cylindrical with a domed head, and surrounds the drywell and suppression pool.

The Mark I and II containment designs are inerted with nitrogen gas during normal operation to prevent an explosive mixture of hydrogen and oxygen from forming during accident conditions. Long term control of post LOCA hydrogen gas concentration is accomplished by adding additional nitrogen gas and then venting the primary containment through the standby gas treatment system. The Mark III containment atmosphere is not inerted with nitrogen due to its large volume, and hydrogen ignitors are typically used for long-term combustible gas control following a postulated design basis event. The proposed change in only applicable to plants with Mark I and Mark II containment designs.

The Mark I and II primary containment inerting system consists of a nitrogen (N2) purge supply and an N2 makeup supply. The N2 purge supply is used to initially inert the atmosphere in the

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[1] NUREG-1433 is based on the BWR/4 plant design, but is also applicable of the BWR/2, BWR/3, and, for some requirements, to the BWR/5 plant designs.

primary containment and is typically provided simultaneously to the suppression chamber air space and the drywell. At times, the suppression chamber pressure may increase at a rate faster than drywell pressure resulting in a differential pressure between the two volumes that is less than the TS 3.6.2.5 limit. The inerting process continues until primary containment oxygen concentration is less than 4% (or a plant-specific limit), as required by TS 3.6.3.2. The inerting process takes approximately eight to twelve hours.

In a Mark I or II containment, the drywell is immediately pressurized when a postulated line break occurs within the primary containment. As drywell pressure increases, drywell atmosphere (primarily nitrogen gas) and steam are blown down through the vents into the suppression pool via the downcomers. The steam condenses in the suppression pool which suppresses the peak pressure in the drywell. Noncondensible gases discharged into the suppression pool collect in the free air volume of the suppression chamber, increasing the suppression chamber pressure. As steam is condensed in the suppression pool, drywell pressure decreases until the suppression chamber pressure exceeds the drywell pressure and the suppression chamber-drywell vacuum breakers open and vent noncondensible gases back into the drywell.

3. CURRENT TECHNICAL SPECIFICATIONS REQUIREMENTS

The Applicability of NUREG-1433 TS 3.6.2.5 and TS 3.6.3.2 both state:

> MODE 1 during the time period:
>
> a. From [24] hours after THERMAL POWER is > [15]% RTP following startup, to
>
> b. [24] hours prior to reducing THERMAL POWER to < [15]% RTP prior to the next scheduled reactor shutdown.

Only four BWR plants (Browns Ferry, Dresden, Fitzpatrick, and Quad Cities) contain a Drywell-to-Suppression Chamber Differential Pressure Specification (ISTS 3.6.2.5). All four of these have Mark I containments.

All BWR/2, BWR/3, BWR/4, and BWR/5 units contain a Primary Containment Oxygen Concentration specification (ISTS 3.6.3.2).

The 24-hour allowance above 15% RTP is provided in the Primary Containment Oxygen Concentration specification to delay inerting the primary containment in a plant startup and to accelerate de-inerting for a plant shutdown. This is for personnel safety considerations so that plant personnel can access the primary containment without breathing apparatus. This allowance is also needed for the Drywell-to-Suppression Chamber Differential Pressure specification due to the pressure fluctuations in the drywell and suppression chamber during the primary containment inerting process that make it difficult to maintain the differential pressure within the required limit.

4. REASON FOR THE PROPOSED CHANGE

The TS requires primary containment oxygen concentration to be less than 4.0 volume percent (or a plant-specific limit) and drywell pressure to be at least [1.5] psid above the suppression

chamber pressure when in Mode 1 during the time period from [24] hours after Thermal Power is > 15% RTP following startup to [24] hours prior to reducing Thermal Power to < 15% RTP prior to the next scheduled reactor shutdown.

The current convoluted presentation in the Applicability can be misunderstood to mean that the requirements are applicable in all of Mode 1, including when reactor power is below 15% RTP, except when the 24-hour allowance is being utilized during a startup or a scheduled plant shutdown. Also, the term "scheduled plant shutdown" is ambiguous and could be misinterpreted to mean only a scheduled refueling outage instead of any planned outage, such as a mid-cycle shutdown for maintenance, and only a power reduction to less than Mode 1. The specification is intended to be applicable in MODE 1 with Thermal Power > [15]% RTP, except during the 24 hours after exceeding [15]% RTP following a startup and prior to reducing power to less than or equal to [15]% RTP. The proposed change clarifies the intent of the Applicability.

4.1. Description of the Proposed Change

The Applicability of both Drywell-to-Suppression Chamber Differential Pressure and Primary Containment Oxygen Concentration Specifications are revised as shown. Deleted text is identified with ~~strikethrough~~ and inserted text is identified in *italics*.

MODE 1 ~~during the time period:~~ *with THERMAL POWER > [15]% RTP*

~~a. From [24] hours after THERMAL POWER is > [15]% RTP following startup, to~~

~~b. [24] hours prior to reducing THERMAL POWER to < [15]% RTP prior to the next scheduled reactor shutdown.~~

Surveillance Requirement (SR) 3.6.2.5.1 and 3.6.3.2.1 are revised to incorporate the following Notes:

1. *Not required to be met until 24 hours after THERMAL POWER > [15]% RTP.*

2. *Not required to be met 24 hours prior to THERMAL POWER being reduced ≤ [15]% RTP.*

The proposed change is supported by changes to the TS Bases. The Bases for these specifications are revised to clarify the Applicability is Mode 1 with Thermal Power > [15]% RTP and to describe the SR Notes. The regulation at Title 10 of the Code of Federal Regulations (10 CFR), Part 50.36, states, "A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications." A licensee may make changes to the TS Bases without prior NRC review and approval in accordance with the Technical Specifications Bases Control Program. The proposed TS Bases changes are consistent with the proposed TS changes and provide the purpose for each requirement in the specification consistent with the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 2, 1993 (58 FR 39132). Therefore, the Bases changes are provided for information and approval of the Bases is not requested.

A model application is attached. The model may be used by licensees desiring to adopt the traveler following NRC approval.

5. TECHNICAL EVALUATION

The drywell-to-suppression chamber differential pressure limit ensures the containment conditions assumed in the safety analyses for Mark I containments are met by limiting the length of the suppression chamber downcomer water leg. High water level in the downcomers could result in excessive forces on the suppression chamber from the downcomer vents and higher pressure buildup in the drywell.

The primary containment oxygen concentration must be maintained below the limit to ensure that an accident that produces hydrogen does not result in a combustible mixture inside primary containment.

Revising the presentation of Applicability of the Drywell-to-Suppression Chamber Differential Pressure and Primary Containment Oxygen Concentration Specifications to state that these requirements are applicable in MODE 1 with thermal power above [15]% RTP clarifies the existing intent of the specification. In both of these Specifications, the existing Action to follow if the LCO is not met and compliance is not restored within the Completion Time is to reduce Thermal Power to less than or equal to [15]% RTP. The Bases for these Actions state, "the plant must be placed in a MODE in which the LCO does not apply. This is done by reducing power to ≤ [15]% RTP." Therefore, the current specifications allow indefinite operation in Mode 1 ≤ [15]% RTP with the LCO not met. The proposed change does not alter this existing allowance but clarifies the Applicability to reflect the current TS requirements. .

Allowing the plant to remain in Mode 1 ≤ [15]% RTP with drywell-to-suppression chamber differential pressure limit not met has been evaluated and determined to be acceptable. General Electric Nuclear Energy Safety Communication 02-10 (Reference 1 and attached), discussed the ISTS LCO 3.6.2.5 Action when drywell-to-wetwell differential pressure is not within limit and the allowance for extended plant operation with reactor power below 15% RTP. The Safety Communication concluded that operation below 15% RTP with the differential pressure limit not met is acceptable for BWRs with Mark II containments and for BWRs with Mark I containments that have demonstrated acceptable loads with zero differential pressure. Only four BWR plants, all of which have Mark I containments, have a specification on drywell-to-suppression chamber differential pressure: Browns Ferry, Dresden, Fitzpatrick, and Quad Cities. Fitzpatrick, Dresden, and Quad Cities performed containment analyses, approved by the NRC in Safety Evaluations dated December 12, 1984, September 18, 1985, and February 15, 1986, respectively, that demonstrated that operation at less than 15% RTP without meeting the differential pressure limit was acceptable. A similar evaluation for Browns Ferry could not be identified and the proposed change to TS 3.6.2.5 is not applicable to that plant.

All BWR/2, BWR/3, BWR/4, and BWR/5 plants with Mark I or II containments have a specification on primary containment oxygen concentration. The Bases for the Primary Containment Oxygen Concentration states, "As long as reactor power is < 15% RTP, the potential for an event that generates significant hydrogen is low and the primary containment need not be inert." The allowance to not meet the limit when < 15% RTP has appeared in the

Standard Technical Specifications since 1978 (NUREG-0123, Revision 1, "Standard Technical Specifications for General Electric Boiling Water Reactors.") This allowance is not being changed. The proposed change only clarifies the presentation of the existing requirement. Therefore, this change is applicable to all BWR/2, BWR/3, BWR/4, and BWR/5 plants with this specification.

Therefore, revising the specification to clarify via Surveillance Requirement Notes that allowing operation at less than [15]% RTP without meeting the differential pressure or primary oxygen concentration specifications is acceptable for the applicable plants.

Moving the 24-hour exceptions from the Applicability to Notes in the SRs is functionally equivalent. Example 1.4-4 in the Use and Application section of NUREG-1433 states that such Notes constitute an "otherwise stated" exception to the Applicability of this Surveillance. A more direct example of this type of Note being used is with TS 3.4.8 where SR 3.4.8.1 allows deferral of part of the LCO requirements for a two-hour period.

Additionally, the existing Applicability statements refer to "following startup" and "prior to the next scheduled reactor shutdown." These phrases are ambiguous as the terms "startup" and "next scheduled reactor shutdown" are undefined. The use of these ambiguous terms is unnecessary. The term "after startup" is implied and not needed in a Note that states "after THERMAL POWER > [15]% RTP," as power will only transition from less than 15% RTP to greater than 15% RTP during a startup. The need to de-inert the containment to permit safe personnel access is applicable to any power reduction to less than 15% RTP that permits planning such an activity. For unplanned shutdowns or power reductions, such as a reactor trip or rapid power-down, Thermal Power will be below [15]% RTP so TS 3.6.2.5 and TS 3.6.3.2, as revised, will not be applicable. Therefore, the proposed SR Notes do not include the phrases "after startup" and "prior to the next scheduled reactor shutdown."

In summary, moving the Applicability exceptions of TS 3.6.2.5 and TS 3.6.3.2 to Notes in SR 3.6.2.5.1 and SR 3.6.3.2.1 and simplifying the presentation does not alter the existing TS requirements. The TS will continue to apply during the period from 24 hours after entry into Mode 1 and power > 15% RTP, and until 24 hours prior to a power reduction to below 15% RTP. The basis for this conclusion is that the proposed change is consistent with the current TS Actions and associated TS Bases.

6. REGULATORY EVALUATION

Section IV, "The Commission Policy," of the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (58 Federal Register 39132), dated July 22, 1993, states in part:

> The purpose of Technical Specifications is to impose those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety by identifying those features that are of controlling importance to safety and establishing on them certain conditions of operation which cannot be changed without prior Commission approval.

> …[T]he Commission will also entertain requests to adopt portions of the improved STS, even if the licensee does not adopt all STS improvements.
>
> …The Commission encourages all licensees who submit Technical Specification related submittals based on this Policy Statement to emphasize human factors principles.
>
> …In accordance with this Policy Statement, improved STS have been developed and will be maintained for [BWR designs]. The Commission encourages licensees to use the improved STS as the basis for plant-specific Technical Specifications.
>
> …[I]t is the Commission intent that the wording and Bases of the improved STS be used … to the extent practicable.

As described in the Commission's "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," recommendations were made by NRC and industry task groups for new STS that include greater emphasis on human factors principles in order to add clarity and understanding to the text of the STS, and provide improvements to the Bases of STS, which provides the purpose for each requirement in the specification. Improved vendor-specific STS were developed and issued by the NRC in September 1992.

The regulation at Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36(a)(1) requires an applicant for an operating license to include in the application proposed TS in accordance with the requirements of 10 CFR 50.36. The applicant must include in the application a "summary statement of the bases or reasons for such specifications, other than those covering administrative controls…." However, per 10 CFR 50.36(a)(1), these technical specification bases "shall not become part of the technical specifications." The Final Policy Statement provides the following description of the scope and the purpose of the Technical Specification Bases:

> Appropriate Surveillance Requirements and Actions should be retained for each LCO [limiting condition for operation] which remains or is included in the Technical Specifications. Each LCO, Action, and Surveillance Requirement should have supporting Bases. The Bases should at a minimum address the following questions and cite references to appropriate licensing documentation (e.g., FSAR, Topical Report) to support the Bases.
>
> 1. What is the justification for the Technical Specification, i.e., which Policy Statement criterion requires it to be in the Technical Specifications?
>
> 2. What are the Bases for each LCO, i.e., why was it determined to be the lowest functional capability or performance level for the system or component in question necessary for safe operation of the facility and, what are the reasons for the Applicability of the LCO?
>
> 3. What are the Bases for each Action, i.e., why should this remedial action be taken if the associated LCO cannot be met; how does this Action relate to other Actions associated with the LCO; and what justifies continued operation of the system or

component at the reduced state from the state specified in the LCO for the allowed time period?

4. What are the Bases for each Safety Limit?

5. What are the Bases for each Surveillance Requirement and Surveillance Frequency; i.e., what specific functional requirement is the surveillance designed to verify? Why is this surveillance necessary at the specified frequency to assure that the system or component function is maintained, that facility operation will be within the Safety Limits, and that the LCO will be met?

   Note: In answering these questions the Bases for each number (e.g., Allowable Value, Response Time, Completion Time, Surveillance Frequency), state, condition, and definition (e.g., operability) should be clearly specified. As an example, a number might be based on engineering judgment, past experience, or PSA [probabilistic safety assessment] insights; but this should be clearly stated.

Additionally, 10 CFR 50.36(b) requires:

> Each license authorizing operation of a … utilization facility … will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to [10 CFR] 50.34 ["Contents of applications; technical information"]. The Commission may include such additional technical specifications as the Commission finds appropriate.

The categories of items required to be in the TSs are provided in 10 CFR 50.36(c). As required by 10 CFR 50.36(c)(2)(i), the TSs will include LCOs, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Per 10 CFR 50.36(c)(2)(i), when an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

The regulation at 10 CFR 50.36(c)(3) requires TSs to include items in the category of SRs, which are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

Per 10 CFR 50.90, whenever a holder of a license desires to amend the license, application for an amendment must be filed with the Commission, fully describing the changes desired, and following as far as applicable, the form prescribed for original applications.

Per 10 CFR 50.92(a), in determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses to the extent applicable and appropriate.

The NRC staff's guidance for the review of TSs is in Chapter 16, "Technical Specifications," of NUREG-0800, Revision 3, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), dated March 2010 (ADAMS Accession No. ML100351425). As

described therein, as part of the regulatory standardization effort, the NRC staff has prepared STS for each of the light-water reactor nuclear designs.

In conclusion, based on the considerations discussed above, the proposed revision does not alter the current manner of operation and (1) there is reasonable assurance that the health and safety of the public will not be endangered by continued operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

7. REFERENCES

1. Safety Communication SC02-10, "Drywell-to-Wetwell Differential Pressure Control Technical Specification for some Mark I Containments."

Attachment

General Electric Safety Communication SC02-10, "Drywell-to-Wetwell Differential Pressure Control Technical Specification for some Mark I Containments."

GE Nuclear Energy

10 CFR Part 21 Notification

SC02-10 ***July 26, 2002***

To: ***BWRs with Mark I Containments***

Subject: ***Drywell-to-Wetwell Differential Pressure Control Technical Specification for Some Mark I Containments***

[ ]	***Reportable Condition [21.21(d)]***	[ ]	***60 Day Interim Report [21.21(a)(2)]***
[ ]	***Transfer of Information [21.21(b)]***	[X]	***Safety Information Communication***

Summary:

The required action given in the Improved Technical Specifications (ITS) for Mark I plants with operating drywell-to-wetwell differential pressure control (LCO 3.6.2.5) may not be consistent with the intended purpose of mitigating the pool swell load in the suppression pool from a postulated design basis accident loss-of-coolant-accident (DBA-LOCA). Some plants with Mark I containment use drywell-wetwell differential pressure (ΔP) control to maintain a positive ΔP (drywell pressure above wetwell pressure). This reduces the water leg length in the downcomer lines and reduces the magnitude of the pool swell loads. If the ΔP is less than the Technical Specification required minimum value, it is necessary to reduce reactor pressure to effectively mitigate pool swell loads from a postulated DBA-LOCA. ITS LCO 3.6.2.5 allows extended plant operation with reactor power reduced to below 15% of Rated Thermal Power, without reducing reactor pressure. This is acceptable for Mark I plants that have demonstrated acceptable loads with zero ΔP, but may be unacceptable for plants that require a positive ΔP to maintain pool swell loads within acceptable design limits.

This concern does not produce a significant safety hazard or violate a technical specification safety limit. Therefore, it is not reportable under 10 CFR Part 21. However, Mark I plants should confirm that LCO 3.6.2.5 is consistent with the plant's design basis for pool swell loads.

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SC02-10

Background

During the Mark I Containment Long-Term Program (LTP), which defined the hydrodynamic loads for Mark I plants, it was determined that pool swell loads were mitigated by maintaining a positive drywell-to-wetwell pressure difference. Based on this determination, a Technical Specification (TS) Limiting Condition for Operation (LCO) was specified to maintain a positive drywell-to-wetwell pressure differential. As required by Reference 1, plants with Mark I containments had the option of either confirming their design was adequate for the higher pool swell loads with a zero drywell-to-wetwell pressure difference (zero ΔP), or of maintaining an operating drywell-to-wetwell pressure difference (operating ΔP) and demonstrating that their containment design was adequate for the lower pool swell loads.

The applicable generic TS LCO included action items to restore the operating ΔP within a specified time period. If the operating ΔP could not be restored, subsequent actions were described that required the plant to be in the hot shutdown condition and ultimately in the cold shutdown condition within specified times. Since pool swell loads are driven by the reactor pressure, the action to be in cold shutdown (which requires vessel depressurization) is adequate to eliminate pool swell loads during a postulated LOCA.

In the course of reviewing plant-specific TS changes for a proposed plant modification, it was discovered that the existing TS LCO for drywell-to-suppression chamber differential pressure, based on the ITS, did not include an action to depressurize the reactor. The ITS maintains the requirement to restore the pressure differential, when lost. However, if the ΔP is not restored, the subsequent action required by LCO 3.6.2.5 is to reduce reactor power to 15% rated thermal power (RTP), and does not include an action to reduce reactor pressure. This action would not significantly mitigate pool swell loads during a postulated DBA-LOCA. This action is also inconsistent with requirements of Reference 1.

Safety Basis

The impact of adhering to ITS LCO 3.6.2.5 was evaluated to determine if the pool swell load for a postulated DBA-LOCA during operation with zero ΔP could produce a significant safety hazard or lead to violation of a TS safety limit. This affects Mark I plants that have not demonstrated adequate structural capability with zero ΔP and have implemented the ITS recommendations for LCO 3.6.2.5.

The safety evaluation focuses on the NRC requirement in Reference 1, which states that Mark I plants with drywell-to-wetwell differential pressure control are required to demonstrate (by structural evaluations) that a DBA-LOCA with the differential pressure control out-of-service would not result in unacceptable consequences. An additional consideration used in the safety evaluation is the significant conservatisms in the Mark I pool swell load definition.

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Structural Evaluation Requirements

Mark I plants had a choice to either design for the higher loads associated with a zero drywell-to-wetwell operating ΔP, or implement an operating ΔP and design for the corresponding lower [redacted] these plants with a zero ΔP, it could not produce a significant safety hazard.

Conservatism in the Mark I Pool Swell Load

The Mark I pool swell loads are based on the Reference 2, Mark I Quarter Scale Test Facility (QSTF) tests. These tests were designed to develop conservative pool swell loads. Two key conservatisms are identified here.

- Pressurization Rate Conservatism: The drywell pressurization transient used for the test was based on the predicted drywell pressure from the approved containment licensing evaluation model M3CPT. Comparisons to the drywell pressurization rate obtained with a more realistic vessel blowdown model (TRACG) have shown that the initial Mark I drywell pressurization rate using the M3CPT model is about 50% higher than realistically expected. Sensitivity tests (Reference 3) shows that the pool swell vertical upforce is approximately proportional to the pressurization rate. They show that the pool swell velocity and vent header impact loads also increase linearly with pressurization rate.

- Air Test Conservatism: A second major conservatism in the tests is the use of air to simulate the break flow to pressurize the test drywell. The use of air instead of steam produces a more severe response. This is because: 1) air tests introduce non-condensible gas into the drywell, which enhances the bubble growth in the suppression pool and 2) the effects of steam condensation, which would occur with steam tests, are not accounted for. Comparison of air versus steam pool swell tests were not available for the Mark I containment. However, a comparison of air and steam tests performed in the Mark III 1/3 area scale Partial Scale Test Facility (PSTF) provides a measure of the conservatism in the use of air tests. Per Reference 4, pool swell air tests produce consistently higher pool surface velocities (by approximately 10 feet per second) than steam tests. According to Reference 4, typical pool swell velocities for the steam tests were 33 ft/sec. Therefore a 10 ft/sec increase represents an increase in the 1/3 scale PSTF pool swell response of 30% due to the use of air as the simulated break flow.

The large (~50%) conservatism in the pressurization rate used for the Mark I QSTF test and the application of other conservatisms in the QSTF tests, such as the use of air to pressurize the

drywell, introduce a large margin of conservatism in the Mark I pool swell load definition. This provides additional assurance that this concern could not produce a significant safety hazard.

Therefore, it is concluded that a possible failure to comply with appropriate design requirements in the application of ITS LCO does not constitute a Reportable Condition within the context of 10 CFR Part 21 because it does not create a substantial safety hazard or contribute to the exceeding of a Technical Specification Safety Limit (i.e., reactor pressure, reactor water level or minimum critical power ratio).

Corrective/Preventive Actions

The BWR Owners' Group Potential Issues Resolution Team (PIRT) and ITS Committee have been informed of this issue, and have been provided a copy of this safety communication. It is recommended that:

1. The ITS Committee modify LCO 3.6.2.5 to require reactor depressurization when ΔP control is lost for plants that require a positive drywell-to-wetwell ΔP to be consistent with design basis pool swell loads

2. Plants with Mark I containments confirm that if they are using ITS LCO 3.6.2.5, their containment is structurally designed for pool swell loads associated with a zero drywell-to-wetwell ΔP. If not, then LCO 3.6.2.5 should be modified to require reactor depressurization if the required drywell-to-wetwell ΔP cannot be restored within the allowable time.

References

1. US-NRC NUREG-0661, Safety Evaluation Report, Mark I Containment Long-Term Program, July 1980.

2. NEDE-21944-P. "Mark I Containment Program, Quarter Scale Plant Unique Tests, April 1979.

3. NEDE-23545-P, "Mark I Containment Program, 1/4 Scale Pressure Suppression Pool Swell Test Program: LDR Load Tests- Generic Sensitivity, December 1978.

4. 22A7007, Rev. 0, GESSAR II, Appendix 3B, Attachment O, Response to NRC Question 3B.3.

SC02-10

Attachment 1 - Affected Plants

	Utility	Plant
	AmerGen Energy Co.	Clinton
X	AmerGen Energy Co.	Oyster Creek
X	Constellation Generation Group	Nine Mile Point 1
	Constellation Generation Group	Nine Mile Point 2
X	Carolina Power & Light Co.	Brunswick 1
X	Carolina Power & Light Co.	Brunswick 2
X	Detroit Edison Co.	Fermi 2
X	Dominion Generation	Millstone 1
	Energy Northwest	Columbia
X	Entergy Nuclear Northeast	FitzPatrick
X	Entergy Nuclear Northeast	Pilgrim
	Entergy Operations, Inc.	Grand Gulf
	Entergy Operations, Inc.	River Bend
	Exelon Generation Co.	CRIT Facility
X	Exelon Generation Co.	Dresden 2
X	Exelon Generation Co.	Dresden 3
	Exelon Generation Co.	LaSalle 1
	Exelon Generation Co.	LaSalle 2
	Exelon Generation Co.	Limerick 1
	Exelon Generation Co.	Limerick 2
X	Exelon Generation Co.	Peach Bottom 2
X	Exelon Generation Co.	Peach Bottom 3
X	Exelon Generation Co.	Quad Cities 1
X	Exelon Generation Co.	Quad Cities 2
	FirstEnergy Nuclear Operating Co.	Perry 1
X	Nebraska Public Power District	Cooper
X	Nuclear Management Co.	Duane Arnold
X	Nuclear Management Co.	Monticello
	Pooled Equipment Inventory Co.	PIM
	PPL Inc.	Susquehanna 1
	PPL Inc.	Susquehanna 2
X	Public Service Electric & Gas Co.	Hope Creek
X	Southern Nuclear Operating Co.	Hatch 1
X	Southern Nuclear Operating Co.	Hatch 2
X	Tennessee Valley Authority	Browns Ferry 1
X	Tennessee Valley Authority	Browns Ferry 2
X	Tennessee Valley Authority	Browns Ferry 3
X	Vermont Yankee Nuclear Power Corp.	Vermont Yankee
	Non-US Plants Affected	
X	Bernische Kraftwerke AG	Muehleberg
X	Japan Atomic Power Corporation	Tsuruga
X	Nuclenor SA	Santa Maria de Garona
X	Taiwan Power Company	Chinshan 1
X	Taiwain Power Company	Chinshan 2
X	Tokyo Electric Power Corporation	Fukushima Daiichi 1
X	Tokyo Electric Power Corporation	Fukushima Daiichi 2

Model Application

[DATE] 10 CFR 50.90

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

DOCKET NO.PLANT NAME
50-[xxx]
SUBJECT: Application to Revise Technical Specifications to Adopt TSTF-568, "Clarify Applicability of BWR TS 3.6.2.5 and TS 3.6.3.2"

Pursuant to 10 CFR 50.90, [LICENSEE] is submitting a request for an amendment to the Technical Specifications (TS) for [PLANT NAME, UNIT NOS.].

[LICENSEE] requests adoption of TSTF 568, "Clarify Applicability of BWR TS 3.6.2.5 and TS 3.6.3.2." TSTF-568 clarifies the Applicability presentation of Technical Specification (TS) [applicable PLANT TS numbers and titles] to state that the[se] Specification[s are][is] applicable in MODE 1 with Thermal Power > [15]% RTP while maintaining the [24]-hour exception after power exceeds [15]% RTP and prior to reducing power to below [15]% RTP as a Surveillance Requirement Note.

The enclosure provides a description and assessment of the proposed changes. Attachment 1 provides the existing TS pages marked to show the proposed changes. Attachment 2 provides revised (clean) TS pages. Attachment 3 provides the existing TS Bases pages marked to show revised text associated with the proposed TS changes and is provided for information only.

Approval of the proposed amendment is requested by [date]. Once approved, the amendment shall be implemented within [ ] days.

There are no regulatory commitments made in this submittal.

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

[In accordance with 10 CFR 50.30(b), a license amendment request must be executed in a signed original under oath or affirmation. This can be accomplished by attaching a notarized affidavit confirming the signature authority of the signatory, or by including the following statement in the cover letter: "I declare under penalty of perjury that the foregoing is true and correct. Executed on (date)." The alternative statement is pursuant to 28 USC 1746. It does not require notarization.]

If you should have any questions regarding this submittal, please contact [NAME, TELEPHONE NUMBER].

Sincerely,

[Name, Title]

Enclosure: Description and Assessment

Attachments:
1. Proposed Technical Specification Changes (Mark-Up)
2. Revised Technical Specification Pages
3. Proposed Technical Specification Bases Changes (Mark-Up) – For Information Only

[The attachments are to be provided by the licensee and are not included in the model application.]

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

ENCLOSURE

DESCRIPTION AND ASSESSMENT

1.0 DESCRIPTION

[LICENSEE] requests adoption of TSTF-568, "Clarify Applicability of BWR TS 3.6.2.5 and TS 3.6.3.2." TSTF-568 clarifies the Applicability presentation of Technical Specification (TS) [applicable PLANT TS numbers and titles] to state that the[se] Specification[s are][is] applicable in MODE 1 with Thermal Power > [15]% RTP, while maintaining the [24]-hour allowance after power exceeds [15]% RTP and prior to reducing power to below [15]% RTP as Notes to SR 3.6.2.5.1 and SR 3.6.3.2.1.

2.0 ASSESSMENT

2.1 Applicability of Safety Evaluation

[LICENSEE] has reviewed the safety evaluation for TSTF-568 provided to the Technical Specifications Task Force in a letter dated [DATE]. This review included a review of the NRC staff's evaluation, as well as the information provided in TSTF-568. [As described herein,] [LICENSEE] has concluded that the justifications presented in TSTF-568 and 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.

2.2 Optional Changes and Variations

[LICENSEE is not proposing any variations from the TS changes described in the TSTF-568 or the applicable parts of the NRC staff's safety evaluation dated [DATE].] [LICENSEE is proposing the following variations from the TS changes described in the TSTF-568 or the applicable parts of the NRC staff's safety evaluation: describe the variations]

[The [PLANT] TS utilize different [numbering][and][titles] than the Standard Technical Specifications on which TSTF-568 was based. Specifically, [describe differences between the plant-specific TS numbering and/or titles and the TSTF-568 numbering and titles.] These differences are administrative and do not affect the applicability of TSTF-568 to the [PLANT] TS.]

[The [PLANT] TS provide a different limit on primary containment oxygen concentration than the 4.0 volume percent limit shown in NUREG-1433. This difference does not affect the applicability of the proposed change.]

[The proposed change to TS 3.6.2.5, "Drywell-to-Suppression Chamber Differential Pressure," is not applicable to [PLANT] and is not included.]

[The [PLANT] TS contain requirements that differ from the Standard Technical Specifications on which TSTF-568 was based, but are encompassed in the TSTF-568 justification. [Describe differences and why TSTF-568 is still applicable.]]

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Analysis

[LICENSEE] requests adoption of TSTF-568, "Clarify Applicability of BWR TS 3.6.2.5 and TS 3.6.3.2." TSTF-568 clarifies the Applicability presentation of Technical Specification (TS) [3.6.2.5, "Drywell-to-Suppression Chamber Differential Pressure," and TS] 3.6.3.2, "Primary Containment Oxygen Concentration," to state that the[se] Specification[s are][is] applicable in MODE 1 with Thermal Power > [15]% Rated Thermal Power (RTP), while maintaining the [24]-hour allowance after power exceeds [15]% RTP and prior to reducing power to below [15]% RTP as Notes to SR 3.6.2.5.1 and SR 3.6.3.2.1.

[LICENSEE] has evaluated if a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change clarifies the Applicability presentation of the [Drywell-to-Suppression Chamber Differential Pressure and] Primary Containment Oxygen Concentration Technical Specification[s]. [Drywell-to-Suppression Chamber Differential Pressure and] Primary Containment Oxygen Concentration are not initiators to any accident previously evaluated. [Drywell-to-Suppression Chamber Differential Pressure and] Primary Containment Oxygen Concentration are assumptions in the mitigation of some accidents previously evaluated. [Analysis has demonstrated that the Drywell-to-Suppression Chamber differential pressure limit is not required to be met at less than 15% power.] [When in Mode 1, but at less than 15% RTP, the potential for an event that generates significant hydrogen is low and the primary containment need not be inert.] Therefore, the consequences of an accident previously evaluated are not significantly increased.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change clarifies the Applicability presentation of the [Drywell-to-Suppression Chamber Differential Pressure and] Primary Containment Oxygen Concentration Technical Specification[s]. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). No credible new failure mechanisms, malfunctions, or accident initiators that would have been considered a design basis accident in the UFSAR are created.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The proposed change clarifies the Applicability presentation of the [Drywell-to-Suppression Chamber Differential Pressure and] Primary Containment Oxygen Concentration Technical Specification[s]. No safety limits are affected. No Limiting Conditions for Operation or Surveillance limits are affected. The [Drywell-to-Suppression Chamber Differential Pressure and] Primary Containment Oxygen Concentration Technical Specification requirements assure sufficient safety margins are maintained, and that the design, operation, surveillance methods, and acceptance criteria specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plants' licensing basis. The proposed change does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits, or limiting safety system settings that would adversely affect plant safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, [LICENSEE] concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

3.2 Conclusion

In conclusion, based on the considerations discussed above, (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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 ENVIRONMENTAL EVALUATION

The proposed change does not change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or does not change an inspection or surveillance requirement. The proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

Technical Specifications and Bases Changes

3.6 CONTAINMENT SYSTEMS

3.6.2.5 Drywell-to-Suppression Chamber Differential Pressure

LCO 3.6.2.5 The drywell pressure shall be maintained ≥ [1.5] psid above the pressure of the suppression chamber.

APPLICABILITY: MODE 1 ~~during the time period:~~ with THERMAL POWER > [15]% RTP

~~a. From [24] hours after THERMAL POWER is > [15]% RTP following startup, to~~

~~b. [24] hours prior to reducing THERMAL POWER to < [15]% RTP prior to the next scheduled reactor shutdown.~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell-to-suppression chamber differential pressure not within limit.	A.1 Restore differential pressure to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ [15]% RTP.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.2.5.1	---------------------------------- NOTES --------------------- 1. Not required to be met until 24 hours after THERMAL POWER > [15]% RTP. 2. Not required to be met 24 hours prior to THERMAL POWER being reduced to ≤ [15]% RTP. --- Verify drywell-to-suppression chamber differential	[12 hours

SURVEILLANCE	FREQUENCY
pressure is within limit.	<u>OR</u> In accordance with the Surveillance Frequency Control Program]

3.6 CONTAINMENT SYSTEMS

3.6.3.2 Primary Containment Oxygen Concentration

LCO 3.6.3.2 The primary containment oxygen concentration shall be < 4.0 volume percent.

APPLICABILITY: MODE 1 with THERMAL POWER is > [15]% RTP ~~during the time period:~~

~~a. From [24] hours after THERMAL POWER is > [15]% RTP following startup, to~~

~~b. [24] hours prior to reducing THERMAL POWER to < [15]% RTP prior to the next scheduled reactor shutdown.~~

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Primary containment oxygen concentration not within limit.	A.1	Restore oxygen concentration to within limit.	24 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ [15]% RTP.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.2.1	---------------------------------- NOTES --------------------- 1. Not required to be met until 24 hours after THERMAL POWER > [15]% RTP. 2. Not required to be met 24 hours prior to THERMAL POWER being reduced to ≤ [15]% RTP. --- Verify primary containment oxygen concentration is	[7 days

SURVEILLANCE	FREQUENCY
within limits.	<u>OR</u> In accordance with the Surveillance Frequency Control Program]

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.5 Drywell-to-Suppression Chamber Differential Pressure

BASES

BACKGROUND

The toroidal shaped suppression chamber, which contains the suppression pool, is connected to the drywell (part of the primary containment) by [eight] main vent pipes. The main vent pipes exhaust into a continuous vent header, from which [96] downcomer pipes extend into the suppression pool. The pipe exit is [4] ft below the minimum suppression pool water level required by LCO 3.6.2.2, "Suppression Pool Water Level." During a loss of coolant accident (LOCA), the increasing drywell pressure will force the waterleg in the downcomer pipes into the suppression pool at substantial velocities as the "blowdown" phase of the event begins. The length of the waterleg has a significant effect on the resultant primary containment pressures and loads.

APPLICABLE SAFETY ANALYSES

The purpose of maintaining the drywell at a slightly higher pressure with respect to the suppression chamber is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the LOCA blowdown. The required differential pressure results in a downcomer waterleg of [3.06 to 3.58] ft.

Initial drywell-to-suppression chamber differential pressure affects both the dynamic pool loads on the suppression chamber and the peak drywell pressure during downcomer pipe clearing during a Design Basis Accident LOCA. Drywell-to-suppression chamber differential pressure must be maintained within the specified limits so that the safety analysis remains valid.

Drywell-to-suppression chamber differential pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

A drywell-to-suppression chamber differential pressure limit of [1.5] psid is required to ensure that the containment conditions assumed in the safety analyses are met. A drywell-to-suppression chamber differential pressure of < [1.5] psid corresponds to a downcomer water leg of > [3.58] ft. Failure to maintain the required differential pressure could result in excessive forces on the suppression chamber due to higher water clearing loads from downcomer vents and higher pressure buildup in the drywell.

BASES

APPLICABILITY

Drywell-to-suppression chamber differential pressure must be controlled when the primary containment is inert. The primary containment must be inert in MODE 1 with THERMAL POWER > [15]% RTP, since this is the condition with the highest probability for an event that could produce hydrogen. It is also the condition with the highest probability of an event that could impose large loads on the primary containment.

~~Inerting primary containment is an operational problem because it prevents primary containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the unit startup and is de-inerted as soon as possible in the unit shutdown. As long as reactor power is < [15]% RTP, the probability of an event that generates hydrogen or excessive loads on primary containment occurring within the first [24] hours following a startup or within the last [24] hours prior to a shutdown is low enough that these "windows," with the primary containment not inerted, are also justified. The [24] hour time period is a reasonable amount time to allow plant personnel to perform inerting or de-inerting.~~

ACTIONS

A.1

If drywell-to-suppression chamber differential pressure is not within the limit, the conditions assumed in the safety analyses are not met and the differential pressure must be restored to within the limit within 8 hours. The 8 hour Completion Time provides sufficient time to restore differential pressure to within limit and takes into account the low probability of an event that would create excessive suppression chamber loads occurring during this time period.

B.1

If the differential pressure cannot be restored to within limits within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by reducing power to ≤ [15]% RTP within 12 hours. The 12 hour Completion Time is reasonable, based on operating experience, to reduce reactor power from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.5.1

The drywell-to-suppression chamber differential pressure is regularly monitored to ensure that the required limits are satisfied. [The 12 hour Frequency of this SR was developed based on operating experience relative to differential pressure variations and pressure instrument drift during applicable MODES and by assessing the proximity to the specified LCO differential pressure limit. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal pressure condition.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
---]

This Surveillance is modified by two Notes. The first Note allows 24 hours to inert the primary containment after increasing THERMAL POWER above [15]% RTP. The second Note allows 24 hours to de-inert the primary containment prior to decreasing THERMAL POWER to less than or equal to [15]% RTP. The Notes take exception to the requirements of the Surveillance being met (i.e., establishing the drywell-to-suppression chamber differential pressure limits for this 24 hour period). Inerting primary containment is an unnecessary operational personnel safety hazard because it prevents primary containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible after exceeding > [15]% RTP, and is de-inerted as soon as possible prior to reducing power to less than [15]% RTP. As long as reactor power is ≤ [15]% RTP, the potential for an event that generates significant hydrogen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen or excessive loads on primary containment occurring within the [24] hours following exceeding [15]% RTP, or within [24] hours prior to reducing THERMAL POWER to less than or equal to [15]% RTP, is low enough that these "windows," with the primary containment not inerted, are also justified. The [24] hour time period is a reasonable amount time to allow plant personnel to perform inerting or de-inerting.

REFERENCES

None.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.2 Primary Containment Oxygen Concentration

BASES

BACKGROUND	All nuclear reactors must be designed to withstand events that generate hydrogen either due to the zirconium metal water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inert, that is, oxygen concentration < 4.0 volume percent (v/o), a combustible mixture cannot be present in the primary containment for any hydrogen concentration. An event that rapidly generates hydrogen from zirconium metal water reaction will result in excessive hydrogen in primary containment, but oxygen concentration will remain < 4.0 v/o and no combustion can occur. This LCO ensures that oxygen concentration does not exceed 4.0 v/o during operation in the applicable conditions.
APPLICABLE SAFETY ANALYSES	The Reference 1 calculations assume that the primary containment is inerted when a Design Basis Accident loss of coolant accident occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of metal water reaction in the reactor core will not produce combustible gas mixtures in the primary containment. Primary containment oxygen concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The primary containment oxygen concentration is maintained < 4.0 v/o to ensure that an event that produces any amount of hydrogen does not result in a combustible mixture inside primary containment.
APPLICABILITY	The primary containment oxygen concentration must be within the specified limit when primary containment is inerted~~, except as allowed by the relaxations during startup and shutdown addressed below~~. The primary containment must be inert in MODE 1 with THERMAL POWER > [15]% RTP, since this is the condition with the highest probability of an event that could produce hydrogen. ~~Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is < 15% RTP, the potential for an event that generates significant hydrogen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first [24] hours of a startup, or within the last [24] hours before a shutdown, is low enough that these "windows," when the primary containment is not inerted, are also justified. The~~

~~[24] hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.~~

BASES

ACTIONS

A.1

If oxygen concentration is ≥ 4.0 v/o ~~at any time~~ while operating above [15]% RTP, ~~in MODE 1, with the exception of the relaxations allowed during startup and shutdown,~~ oxygen concentration must be restored to < 4.0 v/o within 24 hours. The 24 hour Completion Time is allowed when oxygen concentration is ≥ 4.0 v/o because of the low probability and long duration of an event that would generate significant amounts of hydrogen occurring during this period.

B.1

If oxygen concentration cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, power must be reduced to ≤ [15]% RTP within 8 hours. The 8 hour Completion Time is reasonable, based on operating experience, to reduce reactor power from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.2.1

The primary containment must be determined to be inert by verifying that oxygen concentration is < 4.0 v/o. [The 7 day Frequency is based on the slow rate at which oxygen concentration can change and on other indications of abnormal conditions (which would lead to more frequent checking by operators in accordance with plant procedures). Also, this Frequency has been shown to be acceptable through operating experience.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------------------------------REVIEWER'S NOTE---------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
--]

This Surveillance is modified by two Notes. The first Note allows 24 hours to inert the primary containment after increasing THERMAL POWER above [15]% RTP. The second Note allows 24 hours to de-inert the primary containment prior to decreasing THERMAL POWER to less than or equal to [15]% RTP. The Notes take exception to the requirements of

the Surveillance being met (i.e., establishing the primary containment oxygen concentration limits for this 24 hour period). Inerting the primary containment is an unnecessary operational personnel safety hazard because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible after exceeding > [15]% RTP, and is de-inerted as soon as possible prior to reducing power to less than [15]% RTP. As long as reactor power is ≤ [15]% RTP, the potential for an event that generates significant hydrogen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen occurring within [24] hours following exceeding [15]% RTP, or within [24] hours prior to reducing THERMAL POWER to less than or equal to [15]% RTP, is low enough that these "windows," when the primary containment is not inerted, are also justified. The [24] hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

REFERENCES

1. FSAR, Section [6.2.5].