FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TECHNICAL SPECIFICATIONS TASK FORCE TRAVELER

TSTF-568, REVISION 2

"REVISE APPLICABILITY OF BWR/4 TS 3.6.2.5 AND TS 3.6.3.2"

USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS

(EPID L-2017-PMP-0024)

1.0 INTRODUCTION

By letter dated May 21, 2019 (Reference 1), the Technical Specifications Task Force (TSTF) submitted Traveler TSTF-568, Revision 2, "Revise Applicability of BWR/4 TS [Technical Specification] 3.6.2.5 and TS 3.6.3.2," to the U.S. Nuclear Regulatory Commission (NRC). Traveler TSTF-568, Revision 2, proposes changes to the Standard Technical Specification (STS) for boiling-water reactor type four (BWR/4) design, NUREG-1433 (Reference 2). These changes would be incorporated into future revisions of NUREG-1433. This traveler would be made available to licensees for adoption through the consolidated line item improvement process (CLIIP).

The proposed changes would revise STS 3.6.2.5, "Drywell-to-Suppression Chamber Differential Pressure," and STS 3.6.3.2, "Primary Containment Oxygen Concentration." The proposed changes simplify and clarify the applicability statements, which if misapplied, could conflict with the corresponding required actions. The proposed changes also remove the undefined term "scheduled plant shutdown" and provide adequate terminal actions.

The TSTF submitted Revision 0 to TSTF-568 by letter dated December 19, 2017 (Reference 3). The NRC staff transmitted requests for additional information (RAIs) and follow-up comments to the TSTF on April 27, 2018, and August 2, 2018 (References 4 and 5, respectively). Responses to these RAIs and additional comments resulted in two revisions to the traveler: TSTF-568, Revision 1 (Reference 6) and Revision 2.

2.0 REGULATORY EVALUATION

2.1 Description of Structures, Systems, Components and STS Sections

2.1.1 Current Mark I Drywell-to-Suppression Chamber Differential Pressure Control STS Requirement

The drywell-to-suppression chamber differential pressure control is a safety-related operational feature of Mark I containment designs. The STS 3.6.2.5 requires a minimum differential pressure of [1.5] pounds per square inch differential (psid) to reduce the loss-of-coolant accident (LOCA) hydrodynamic loads during the Mark I containment load definition short- and long-term programs (NUREG-0661, "Safety Evaluation Report Mark I Containment Long-term Program Resolution of Generic Technical Activity A-7," Reference 7). The LOCA pool swell loads are significantly reduced because the differential pressure control reduces the length of water leg in the downcomer. The LOCA vent clearing and pool swell due to bubble formation would occur

earlier (i.e., at a lower drywell pressure resulting in lesser forces on the suppression chamber thereby increasing the safety margin for containment integrity, containment internal structures, and pressure boundary). Decreasing the allowable suppression chamber water level has a similar effect.

It is difficult to control the differential pressure during startup and shutdown transients. This is because of the variation of the drywell heat loads from the primary and auxiliary systems and because the inerting (during startup) or the de-inerting (during shutdown) of containment. Inerting the containment during startup involves the addition of large volumes of nitrogen. De-inerting containment during shutdown involves the addition of large volumes of air. In order to allow operation during the time differential pressure control is difficult, the current STS 3.6.2.5 is applicable from [24] hours following startup after the reactor thermal power exceeds [15] percent to [24] hours prior to reducing thermal power less than [15] percent reactor thermal power (RTP) during a scheduled shutdown.

2.1.2 Current Mark I and II Containment Oxygen Concentration Requirement

The regulation at Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.44, “Combustible gas control for nuclear power reactors,” states that for a plant with an inerted containment atmosphere, the oxygen concentration in the primary containment is required to be maintained below 4 percent by volume during normal plant operation. This requirement ensures that an accident that produces hydrogen does not result in a combustible mixture inside the primary containment. The current STS 3.6.3.2 requires primary containment oxygen concentration to be less than 4 percent by volume when in Mode 1 during the period from [24] hours after the thermal power exceeds [15] percent RTP following startup, and to [24] hours prior to reducing the RTP to less than [15] percent RTP during next scheduled shutdown. The TSTF stated that the [24]-hour allowance above [15] percent 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 allowance is provided so that plant personnel can safely enter the primary containment without breathing apparatus to perform the needed inspections and maintenance adjustments.

2.1.3 Mark I 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 containment atmosphere is inerted with nitrogen gas during normal operation to prevent a combustible 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. This containment design is used in BWR/2, BWR/3, and some BWR/4 reactors.

2.1.4 Mark II Containment

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 containment atmosphere is inerted with nitrogen gas during normal operation to prevent a combustible 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. This containment design is used in some BWR/4s and all BWR/5 reactors.

2.1.5 Pressure Suppression Following a LOCA

In Mark I and II containments, 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. Non-condensable 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 and on the structures in the drywell, the pressure decreases until the suppression chamber pressure exceeds the drywell pressure and the suppression chamber-drywell vacuum breakers open and vent non-condensable gases back into the drywell.

2.1.6 STS 3.6.2.5, “Drywell-to-Suppression Chamber Differential Pressure” (Mark I Containments)

A drywell-to-suppression chamber differential pressure limit is required to ensure the containment conditions assumed in the safety analyses are met. 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 during a LOCA. Drywell-to-suppression chamber differential pressure must be controlled when the primary containment is inert. The TS requires that the drywell pressure be maintained ≥ [1.5] psid above the pressure of the suppression chamber.

2.1.7 STS 3.6.3.2, “Primary Containment Oxygen Concentration”

The primary containment oxygen concentration is maintained to ensure that a LOCA, a postulated event that produces hydrogen, does not result in a combustible mixture inside primary containment. The TS requires that the primary containment oxygen concentration be maintained below 4 volume percent. Below this concentration, the primary containment is inerted and no combustion can occur.

2.2 Proposed Changes to the Standard Technical Specifications

2.2.1 Proposed Changes to STS 3.6.2.5, "Drywell-to-Suppression Chamber Differential Pressure" (Mark I Containments Only)

The Applicability of STS 3.6.2.5, "Drywell-to-Suppression Chamber Differential Pressure," would be revised as shown below.

Current TS Applicability	Proposed TS Applicability
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.	MODE 1 with THERMAL POWER > [15]% RTP.

Required Action A.1 and the completion time (CT) would be revised as shown below.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell-to-suppression chamber differential pressure not within limit.	A.1 ---------- *NOTE* ---------- *LCO 3.0.4.c is applicable.* ---------- Restore differential pressure to within limit.	*72* ~~8~~ hours

The NRC staff understands the overall purpose of the proposed changes is to simplify the applicability statement by adding a new note and revising the CT. This change provides similar operational flexibility but more closely follows established TS conventions.

2.2.2 Proposed Changes to STS 3.6.3.2, "Primary Containment Oxygen Concentration"

The Applicability of STS 3.6.3.2, "Primary Containment Oxygen Concentration," would be revised as shown below.

Current TS Applicability	Proposed TS Applicability
MODE 1 during the time period: c. From [24] hours after THERMAL POWER is > [15]% RTP following startup, to d. [24] hours prior to reducing THERMAL POWER to < [15]% RTP prior to the next scheduled reactor shutdown.	MODES 1 and 2.

Required Actions A.1 and B.1 and their associated CTs would be revised as shown below.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Primary containment oxygen concentration not within limit.	A.1 ---------- *NOTE* ---------- *LCO 3.0.4.c is applicable.* ---------- Restore oxygen concentration to within limit.	*72* ~~24~~ hours
B. Required Action and associated Completion Time not met.	B.1 *Be in MODE 3*~~Reduce THERMAL POWER to ≤ [15]% RTP~~.	*12*~~8~~ hours

The NRC staff understands the overall purpose of the proposed changes is to simplify the applicability statement by adding a new note and revising the CT. This change provides operational flexibility but more closely follows established TS conventions and requires that the plant be in Mode 3 if oxygen concentration cannot be restored to within limits.

2.3 Applicable Regulatory Requirements and Guidance

As described in the Commission's "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," published in the *Federal Register* on July 22, 1993 (58 FR 39132-39139), the NRC and industry task groups for new STSs recommended that improvements include greater emphasis on human factors principles in order to add clarity and understanding to the text of the STSs, and provide improvements to the Bases of the STSs, which provides the purpose for each requirement in the specification. The improved vendor-specific STSs were developed and issued by the NRC in September 1992.

Section IV, "The Commission Policy," of the Final Policy Statement on TSs 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[s] [(e.g., TSTF-568)], 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 each NSSS [nuclear steam supply system] owners group. 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[s] be used …to the extent practicable.

The Summary section of the Final Policy Statement on TS states, in part:

> Implementation of the Policy Statement through implementation of the improved STS[s] is expected to produce an improvement in the safety of nuclear power plants through the use of more operator-oriented Technical Specifications, Improved Technical Specification Bases, reduced action statement induced plant transients, and more efficient use of NRC and industry resources.

The regulation, 10 CFR 50.36(a)(1), requires:

> Each applicant for a license authorizing operation of a …utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section. 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.

The regulation, 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 listed in 10 CFR 50.36(c).

In accordance with 10 CFR 50.36(c)(2), limiting conditions for operation (LCOs) are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When LCOs are not met, the licensee must shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met. In addition, 10 CFR 50.36(c)(2)(ii)(B) requires a TS LCO of a nuclear reactor must be established for a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The regulation, 10 CFR 50.44(b)(2)(i), states that "All boiling water reactors with Mark I or Mark II type containments must have an inerted atmosphere." Section 50.44(a)(1) defines "[i]nerted atmosphere" as a containment atmosphere with less than 4 percent of oxygen by volume.

Chapter 6.2.1.1.C, Revision 7, "Pressure-Suppression Type BWR Containments" of NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear

Power Plants: LWR [Light-Water Reactor] Edition" (SRP), March 2007 (Reference 8), states: "The acceptability of LOCA pool dynamic loads for plants with Mark I containments is based on conformance with NRC acceptance criteria found in NUREG-0661" (Reference 7).

The NRC staff's guidance for the review of TSs is in Chapter 16.0, Revision 3, "Technical Specifications," of the SRP, March 2010 (Reference 9). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared STSs for each of the LWR nuclear designs. Accordingly, the NRC staff's review includes consideration of whether the proposed changes are consistent with the applicable reference STSs (i.e., the current STSs), as modified by NRC-approved travelers. In addition, the guidance states that comparing the change to previous STS can help clarify the TS intent. The STSs for BWR/4 plants are published as NUREG-1433 (Reference 2).

3.0 TECHNICAL EVALUATION

3.1 PROPOSED CHANGES TO STS 3.6.2.5 (MARK I CONTAINMENTS)

3.1.1 Proposed Changes in the Applicability

In TSTF-568, Revision 2, the TSTF proposed to delete the time periods, dependent on startup and shutdown times, from the applicability section and to replace them with a thermal power value. These time periods are "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." These time periods would be replaced by flexibilities and requirements in the revised completion times and the inserted note referencing LCO 3.0.4.c. This would result in requiring the drywell pressure during Mode 1 to be maintained above the specified limit whenever the thermal power is above [15] percent. The current limitations of applicability, dependent on startup and shutdown, were established to allow licensees operational flexibilities, such as containment entry to perform maintenance and surveillances while at power.

In TSTF-568, Revision 2, Attachment, General Electric Safety Communication (SC) 02-10, page 4, under the heading "Corrective/Preventive Actions," item 2, it is recommended that Mark I plants that use STS 3.6.2.5 should "confirm that their containment is structurally designed for pool swell loads with a zero drywell-to-suppression chamber differential pressure." For these plants, the Mark I containment load definition program has defined the pool swell loads associated with zero drywell-to-suppression chamber differential pressure. NUREG-0661 (Reference 7), Appendix A, Section 2.3, states that each plant with a differential pressure control (i.e., STS 3.6.2.5) perform a structural assessment to demonstrate that the containment can maintain its functional capability when the differential pressure control is out-of-service (i.e., the differential pressure is zero).

The TSTF stated that presently the following Mark I containment plants, Browns Ferry, Units 1, 2, and 3; Dresden Units 2 and 3; Quad Cities, Units 1 and 2; and FitzPatrick, are applying the drywell-to-suppression chamber differential pressure control STS 3.6.2.5. The licensees of these plants submitted their respective plant-specific analysis reports called a Plant Unique Analysis Report (PUAR) for NRC review. The NRC approved these reports for the Browns Ferry Units 1, 2, and 3 in Reference 10; for the Dresden Units 2 and 3 in References 11, 12, and 13; for the Quad Cities Units 1 and 2 in Reference 14, and for FitzPatrick in Reference 15. As stated in SC 02-10, page 3, structural assessment based on zero drywell-to-suppression chamber differential pressure pool swell load definition was used to confirm the functional

capability of the suppression chamber against the Service Level D limit. The SC02-10 also identifies the following two major conservatisms in the pool swell load definitions based on the Mark I Quarter Scale tests:

(a) The drywell pressurization test transient was based on the predicted drywell pressure from the NRC approved conservative General Electric (GE) code M3CPT. This code predicts about 50% higher drywell pressurization than a realistic analysis using the GE-Hitachi code TRACG.

(b) The break was simulated by air to pressurize the drywell, which produces a more severe pool swell response than a realistic nitrogen/steam mixture and enhances the bubble growth.

Each NRC approval confirmed that the licensees of these plants met the acceptance criteria specified in NUREG-0661, Appendix A, and reviewed and approved any exceptions the licensees took from the acceptance criteria. Therefore, for the Mark I plants applying the STS 3.6.2.5, the NRC staff approval of the PUARs confirmed that with the drywell-to-suppression chamber differential pressure out-of-service, the containment is structurally designed for the pool swell loads during a large-break LOCA.

Based on the PUARs, the NRC staff finds it acceptable for the reactor to not be depressurized when the differential pressure is out-of-service at ≤ [15] percent RTP. Further, NUREG-0661 (Reference 7), Section 3.12.7 concluded that if the differential pressure is out-of-service, the probability of occurrence of a large-break LOCA, is less than 10E-7 per reactor-year, which is sufficiently small. This minimal probability of occurrence paired with the short period during which plants are in the transition state of less than [15] percent rated thermal power, support the adequacy of this change because the LOCA dynamic loads are not adversely affected. The NRC staff determined the proposed deletion of the time periods is acceptable because they are now included in the note insertion (discussed in Section 3.1.2) and change in the CT (discussed in Section 3.1.3). In addition, the proposed change is acceptable since it simplifies and clarifies the applicability statement and continues to provide the lowest functional capability of equipment required for safe operation of the facility as required by 10 CFR 50.36(c)(2) by protecting containment integrity.

3.1.2 Proposed Changes in Required Action A.1

In TSTF-568, Revision 2, the TSTF proposed to add the following note to Required Action A.1: “LCO 3.0.4.c is applicable.” STS LCO 3.0.4 states:

> When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:
>
> a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;
>
> b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The criteria applicable to STS LCO 3.6.2.5 is LCO 3.0.4.c since this LCO establishes an individual value or parameter (i.e., drywell pressure maintained above a certain value). The new note will allow entry into the mode of applicability of STS LCO 3.6.2.5 with the drywell pressure outside of the required limit. This note allows the licensee operational flexibility as it permits entry into Mode 1 at greater than [15] percent RTP when drywell pressure is outside of the required limit during startup configurations. The NRC staff concludes the addition of the note is acceptable because it clarifies and simplifies the intent of the current STS LCO 3.6.2.5 applicability statement “a.” of allowing startup operation with the LCO not met. In addition, inserting the note for this purpose is consistent with STS format and provides operational flexibility similar to that in the previous applicability statement “a.”.

3.1.3 Proposed Changes in the CT of Condition A

In TSTF-568, Revision 2, the TSTF proposed to change the CT for Required Action A.1 from 8 hours to 72 hours. The TSTF stated the proposed change will permit safe entry of personnel into the containment in Modes 1 and 2. The 72 hours provides: [24] hours to de-inert the containment to permit safe personnel access, [24] hours to perform the required work, and [24] hours to re-inert containment. The NRC staff finds that the extended CT incorporates the time currently allowed through the applicability statement in Section 3.1.1 of this SE. The NRC staff finds that 72 hours is reasonable to conduct these activities based on operating experience and the requested completion time does not present a significant change in risk given the low probability that a large line break would occur during this period. Therefore, NRC staff finds this change acceptable.

3.1.4 Conclusion for Proposed Changes to STS 3.6.2.5

The NRC staff finds the changes proposed in STS 3.6.2.5 acceptable and continue to meet 10 CFR 50.36(c)(2) since the revised LCO provides the lowest functional capability of equipment required for safe operation of the facility by protecting containment integrity.

3.2 PROPOSED CHANGES TO STS 3.6.3.2

3.2.1 Proposed Changes in the Applicability

The regulation at 10 CFR 50.44(b)(2)(i) states, “All boiling water reactors with Mark I or Mark II type containments must have an inerted atmosphere.” After the publication of the Final Rule, “Interim Requirements Related to Hydrogen Control” in the *Federal Register* on December 2, 1981 (46 FR 58484-58486) (Reference 16), all BWR plants with Mark I or II containments TS were amended to include an LCO requiring primary containment oxygen concentration be maintained below the 4.0 volume percent limit.

The TSTF proposed to expand the applicability of this LCO to Modes 1 and 2 without exception. The NRC staff finds the proposed change acceptable because it is more restrictive since an

unlikely LOCA event leading to a degraded core that could produce hydrogen has the highest probability of occurrence during Modes 1 and 2 conditions.

3.2.2 Proposed Changes in Required Action A.1

In TSTF-568, Revision 2, the TSTF proposed to add the following note to Required Action A.1: "LCO 3.0.4.c is applicable." As stated in Section 3.1.2 of this safety evaluation, STS LCO 3.0.4.c allows entering the mode of applicability of STS LCO 3.6.3.2 with the LCO not met. Therefore, the proposed change would permit entry into Modes 1 and 2 with primary containment oxygen concentration higher than the required limit. The NRC staff concludes the addition of the note is acceptable because it clarifies and simplifies the intent of the current STS LCO 3.6.3.2 applicability statement "a." of allowing startup operation with the LCO not met. In addition, inserting the note for this purpose is consistent with STS format and provides flexibility similar to that in the previous applicability statement (a.).

3.2.3 Proposed Changes in the CT of Condition A

The TSTF proposed changing the CT from 24 hours to 72 based on the following sequence of operations: allow [24] hours to de-inert the containment to permit safe personnel access, allow [24] hours to perform the required maintenance or repair work, and allow [24] hours to inert the containment. The NRC staff determined that the presence of a higher oxygen concentration for the 72-hour CT is appropriate considering the low safety significance of the change for potential accidents and the additional restrictions and conservatisms in the revised applicability.

3.2.4 Proposed Changes in Required Action B.1

In TSTF-568, Revision 2, the TSTF proposed to change the applicability statement of STS LCO 3.6.3.2 to Modes 1 and 2. If the oxygen concentration cannot be restored within the required limit and CT of Required Action A.1, the reactor should be brought to Mode 3. In this mode, the reactor would be in a hot shutdown condition (control rods fully inserted) with all reactor vessel head bolts fully tensioned.

The NRC staff recognizes that on entering Mode 3, the decay heat is rapidly decreasing. Steam is initially available for operating the reactor core isolation cooling/high pressure coolant injection steam turbine-driven pumps until the reactor pressure and thus water temperature is substantially reduced. As the decay heat continues to decrease, operators have increased time and options for achieving adequate water injection using the low-pressure emergency core cooling system to avoid core damage and associated generation of combustible gas. Therefore, the occurrence of a LOCA leading to degraded core is highly unlikely in Mode 3.

The NRC staff finds the proposed change in Required Action B.1 acceptable because:

- It provides a more appropriate terminal action since it requires the plant to be placed in a mode in which the LCO does not apply and the oxygen concentration limit is no longer required. The previous terminal action allowed an indefinite period of operation at ≤ [15] percent RTP.

- The revised required action is consistent with STS 3.6.3.1 in BWR/6 STS (Reference 17), in which Required Action C.1 brings the reactor to Mode 3 (hot shutdown) when the hydrogen control system (igniter system) is not available. An available hydrogen control system, in a BWR/6 Mark III containment, performs a function

(reduces the potential for an explosive mixture) similar to an inerted Mark I/II containment.

Due to similar low potential for hydrogen generation when the reactor is in Mode 3, inerting of containment in Mode 3 is not needed in BWR/6 Mark III and Mark I/II containments. Therefore, the NRC staff concluded the proposed change is acceptable because it continues to protect containment integrity and meets 10 CFR 50.36(c)(2) by providing the lowest functional capability of equipment required for safe operation of the plant.

3.2.5 Proposed Changes in the CT of Condition B

The TSTF also proposed to change the Condition B CT from 8 hours to 12 hours, stating that 12 hours is a reasonable time to reduce reactor power from full power conditions to Mode 3 in an orderly manner and without challenging plant systems. The proposed change from 8 hours to 12 hours for bringing the reactor to a hot shutdown condition from full power is acceptable to NRC staff because it is not a significant change and is based on industry operating experience.

3.2.6 Conclusion for Proposed Changes to STS 3.6.3.2

The NRC staff concludes the proposed changes in the applicability statement for STS 3.6.3.2 are acceptable since they are more restrictive as the applicability now extends to Modes 1 and 2 without exception. In addition, for Mark I and II containment BWRs, the occurrence of a LOCA that could lead to degraded core conditions with containment de-inerted, while in Mode3, is unlikely. Therefore, the changes proposed in STS 3.6.3.2 are acceptable and continue to meet 10 CFR 50.36(c)(2).

4.0 CONCLUSION

The NRC staff reviewed Traveler TSTF-568, Revision 2, which proposed changes to the STS in NUREG-1433. The NRC staff determined that, with the proposed changes, the STS will continue to meet the Commission's "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" because the requirements of 10 CFR 50.36(c)(2) continue to be met. Additionally, the changes to each STS were reviewed and found to be technically clear and consistent with customary terminology and format in accordance with SRP Chapter 16.0.

The NRC staff finds that the proposed traveler meets or is consistent with applicable regulations and associated guidance. Therefore, the NRC staff concludes that the proposed STS changes are acceptable.

5.0 REFERENCES

1. Technical Specifications Task Force letter to U.S. Nuclear Regulatory Commission, "Transmittal of TSTF-568, Revision 2, "Revise Applicability of BWR/4 TS 3.6.2.5 and TS 3.6.3.2," dated May 21, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19141A122).

2. U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, General Electric Plants BWR/4," NUREG-1433, Volume 1, "Specifications," and Volume 2, "Bases," Revision 4.0, April 2012 (ADAMS Accession Nos. ML12104A192 and ML12104A193).

3. Technical Specifications Task Force letter to U.S. Nuclear Regulatory Commission, “Transmittal of TSTF-568, Revision 0, “Clarify Applicability of BWR/4 TS 3.6.2.5 and TS 3.6.3.2,” dated December 19, 2017 (ADAMS Accession No. ML17353A437).

4. U.S. Nuclear Regulatory Commission, “Request for Additional Information For TSTF-568, Clarify Applicability of BWR/4 Technical Specification 3.6.2.5 and 3.6.3.2,” dated April 27, 2018 (ADAMS Accession No. ML18122A166).

5. U.S. Nuclear Regulatory Commission, “NRC Staff Feedback Re: Draft TSTF-568, Revision 1, “Clarify Applicability of BWR/4 TS 3.6.2.5 and TS 3.6.3.2” dated August 2, 2018 (ADAMS Accession No. ML18204A024).

6. Technical Specifications Task Force, “Draft TSTF-568, Revision 1, ‘Clarify Applicability of BWR/4 TS 3.6.2.5 and TS 3.6.3.2,’” dated May 31, 2018 (ADAMS Accession No. ML18211A437).

7. U.S. Nuclear Regulatory Commission, NUREG-0661, “Safety Evaluation Report Mark I Containment Long-term Program Resolution of Generic Technical Activity A-7,” July 1980 (ADAMS Accession No. ML072710452).

8. U.S. Nuclear Regulatory Commission, Chapter 6.2.1.1.C, Revision 7, “Pressure-Suppression Type BWR Containments” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” March 2007 (ADAMS Accession No. ML063600403).

9. U.S. Nuclear Regulatory Commission, Chapter 16.0, Revision 3, “Technical Specifications,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” March 2010 (ADAMS Accession No. ML100351425).

10. Vassallo, Domenic B., U.S. Nuclear Regulatory Commission, letter to Hugh G. Parris, Tennessee Valley Authority, “Mark I Containment Long-Term Program, Re: Browns Ferry Nuclear Plant, Units 1, 2, and 3,” dated May 6, 1985 (ADAMS Package Accession No. ML18029A537).

11. Zwolinski, John A., U.S. Nuclear Regulatory Commission, letter to Dennis L. Farrer, Commonwealth Edison Company, “Mark I Containment Long-Term Program, Re: Dresden Nuclear Power Station, Unit Nos. 2, and 3,” dated September 18, 1985 (ADAMS Accession No. ML17195A950).

12. U.S. Nuclear Regulatory Commission, “Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Mark I Containment Long-Term Program Pool Dynamic Loads Review, Commonwealth Edison Company, Docket Nos. 50-237/249,” dated September 18, 1985 (ADAMS Accession No. ML17195A952).

13. U.S. Nuclear Regulatory Commission, “Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Mark I Containment Long-Term Program Structural Review, Commonwealth Edison Company, Docket Nos. 50-237/249,” dated September 18, 1985 (ADAMS Accession No. ML17195A953).

14. Zwolinski, John A., U.S. Nuclear Regulatory Commission, letter to Dennis L. Farrar, Commonwealth Edison Company, “Mark I Containment Long-Term Program, Re: Quad Cities Nuclear Power Station, Units 1 and 2,” dated February 15, 1986 (ADAMS Accession No. ML19199A123).

15. Vassallo, Domenic B., U.S. Nuclear Regulatory Commission, letter to C. A. McNeill, Jr., Power Authority of the State of New York, “Mark I Containment Long-Term Program Re: James A. Fitzpatrick Nuclear Power Plant,” dated December 12, 1984 (ADAMS Accession No. ML19203A093).

16. U.S. Nuclear Regulatory Commission, Final Rule, “Interim Requirements Related to Hydrogen Control,” *Federal Register*, Vol. 46, No. 231, December 2, 1981; pp. 58484-58486.

17. U.S. Nuclear Regulatory Commission, NUREG-1434, Volume 1, Revision 4, “Standard Technical Specifications, General Electric BWR/6 Plants,” April 2012 (ADAMS Accession No. ML12104A195).

Principal Contributors: A. Sallman, NRR/DSS/SNSB
C. Tilton, NRR/DSS/STSB

Date: December 17, 2019