NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

January 29, 2003

L-MT-03-006

10 CFR Part 50,
Section 50.90

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

License Amendment Request for Drywell Leakage and
Sump Monitoring System Technical Specification Changes (TAC No. MB6493)

Nuclear Management Company, LLC (NMC) hereby requests a change to the Technical Specifications (TS), Appendix A of Operating License DPR-22, for the Monticello Nuclear Generating Plant (MNGP). This request is submitted pursuant to and in accordance with the provisions of 10 CFR Part 50, Section 50.90. This request supercedes, in its entirety, the License Amendment Request submitted by letter dated October 8, 2002, and supplemented by letter dated November 8, 2002.

The purpose of this License Amendment Request is to propose changes to the Drywell Leakage and Sump Monitoring Detection Section of the Technical Specification (TS). These proposed changes clarify the definitions and restructure the Coolant Leakage Section of the TS by dividing it into two subsections. One subsection provides criteria for Reactor Coolant System (RCS) Operational Leakage and the other subsection provides criteria for RCS Leakage Detection Instrumentation. The proposed revisions to the Monticello TS also revise the TS by focusing on Unidentified Leakage and Total Leakage requirements. The revisions add a TS Limiting Condition for Operation (LCO) requirement in the event the required leakage detection instrumentation is inoperable. Additionally, a TS LCO is being revised for Unidentified Leakage Rate Increase that focuses the attention of the operator on Intergranular Stress Corrosion Cracking (IGSCC) susceptible piping when an increase in leakage occurs.

Exhibit A contains the Proposed Changes, Reasons for Change, a Safety Evaluation, a Determination of No Significant Hazards Consideration and an Environmental Assessment. Exhibit B contains current Monticello Technical Specification pages marked up with the proposed changes. Exhibit C contains revised Monticello Technical Specification pages. Exhibit D contains current Monticello Technical Specification Bases pages marked up with supporting changes. Exhibit E contains revised Monticello Technical Specification Bases pages.

This submittal does not contain any new NRC commitments and does not modify any prior commitments.

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The Monticello Operations Committee has reviewed this application. A copy of this submittal, along with the Determination of No Significant Hazards Consideration, is being forwarded to our appointed state official pursuant to 10 CFR 50.91(b)(1).

Nuclear Management Company, LLC requests a period of up to 60 days following receipt of this license amendment to implement the changes.

Monticello staff will contact the NRC Project Manager to discuss a review schedule for this License Amendment Request.

If you have any questions regarding this License Amendment Request please contact John Fields, Senior Licensing Engineer, at 763-295-1663.

David L. Wilson
Site Vice President
Monticello Nuclear Generating Plant

Subscribed to and sworn before me this 29th day of January, 2003.

Notary

STEPHEN R. BLEGEN
NOTARY PUBLIC-MINNESOTA
My Comm. Exp. Jan. 31, 2005

Attachments:
- Exhibit A – Evaluation of Proposed Changes to the Monticello Technical Specifications
- Exhibit B – Current Monticello Technical Specification Pages Marked Up With Proposed Changes
- Exhibit C – Revised Monticello Technical Specification Pages
- Exhibit D – Current Monticello Technical Specification Bases Pages Marked Up With Supporting Changes
- Exhibit E – Revised Monticello Technical Specification Bases Pages

cc: Regional Administrator-III, NRC
NRR Project Manager, NRC
Sr. Resident Inspector, NRC
Minnesota Department of Commerce
J Silberg, Esq.

License Amendment Request for Drywell Leakage and
Sump Monitoring Detection Technical Specification Changes (TAC No. MB6493)

Evaluation of Proposed Changes to the Monticello Technical Specifications

Background

On September 7, 2002, during a routine Drywell Equipment Drain Sump (DEDS) pumping operation, it was discovered that #12 DEDS pump (P-20B) was not performing as designed. Due to the location of the equipment (inside the drywell), troubleshooting and research into the failure have been limited. Monticello Nuclear Generating Plant (MNGP) has verified that electrically, the motor is working and receiving a start command. However, indications are that the pump is spinning but the shaft appears to have decoupled, sheared, or has internal blockage as no discernable flow can be measured. The problem cannot be fixed without a drywell entry. Therefore, a complete analysis of this pump failure cannot be performed at this time.

Current MNGP Technical Specifications require the following for Coolant Leakage:

1) TS 4.6.D.1 states:
   Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212 °F, the following surveillance program shall be carried out:

   a. *Unidentified and Identified Leakage rates shall be recorded once per 12 hours using primary containment floor and equipment drain sump monitoring equipment.*

2) TS 4.6.D.2 states:
   The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

   a. *Primary containment sump leakage measurement system-performance of a sensor check once per 12 hours and a channel calibration test at least once per cycle.*

3) TS 3.6.D.5 states:
   Any time irradiated fuel is in the reactor vessel and reactor water temperature is above 212 °F at least one of the leakage measurement instruments associated with each sump shall be operable. If no leak rate measurement instruments associated with a sump are operable, then:

   a. *Perform manual leak rate measurements once per 12 hours and restore a measurement instrument to operable status within 30 days.*

b. *Otherwise, initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212 °F within 24 hours.*

The NMC is concerned that if the companion sump pump, #11 Drywell Equipment Drain Sump pump (P-20A) were to fail, MNGP would be required to shut down. With both pumps inoperable in the DEDS, there is no capability to remove water from the sump. Therefore, within a short period of time, the water from the DEDS would overflow and spill into the Drywell Floor Drain Sump (DFDS). Thus Identified Leakage would become indistinguishable from Unidentified Leakage.

With the inability to remove water from the DEDS and the configuration of the instrumentation on the sump, the ability to record Identified Leakage rate once per 12 hours using the equipment drain sump monitoring equipment (TS 4.6.D.1) would be impossible. Further, performance of sensor checks for DEDS instrumentation (TS 4.6.D.2) would be impossible. Finally, a manual calculation of leak rate (TS 3.6.D.5) could not be performed. Therefore, TS 3.6.D.5.b would require a unit shutdown within 24 hours.

Proposed Changes

The purpose of this License Amendment Request is to propose changes to the Drywell Leakage and Sump Monitoring Detection Section of the TS. These proposed changes clarify the definitions and restructure the Coolant Leakage Section of TS by dividing it into two subsections. One subsection provides criteria for Reactor Coolant System (RCS) Operational Leakage and the other subsection provides criteria for RCS Leakage Detection Instrumentation. The proposed revisions to the Monticello TS also revise the TS by focusing on Unidentified Leakage and Total Leakage requirements. The revisions add a TS Limiting Condition for Operation (LCO) requirement in the event the required leakage detection instrumentation is inoperable. Additionally, a TS LCO is being revised for Unidentified Leakage Rate Increase that focuses the attention of the operator on Intergranular Stress Corrosion Cracking (IGSCC) susceptible piping when an increase in leakage occurs.

The proposed changes to the Monticello TS were developed considering requirements similar to those contained in NUREG-1433 as a means of incorporating best industry practices. The proposed changes to Appendix A of the Monticello Operating License, Technical Specifications, are described below. Marked-up changes to the Monticello TS are included in Exhibit B. Revised changes to the Monticello TS are included in Exhibit C. The following provides a description of the changes, the reason for the changes, and a safety evaluation for each of the changes:

Proposed
Change 1 - Revise TS 1.0.AC, 1.0.AD and add a new TS 1.0.AE to the Monticello TS Definitions Section for the Definitions of Identified Leakage and Unidentified Leakage, and the addition of a Definition for Total Leakage (page 5).

Reason for Changes

The proposed changes to the Definitions for Identified Leakage and Unidentified Leakage and the addition of a definition for Total Leakage are being made so that the Monticello TS Definitions for Leakage match the proposed TS changes and are generally consistent with those in NUREG-1433 (Reference 1).

The proposed revisions to the Definitions for Identified Leakage and Unidentified Leakage and the addition of a definition for Total Leakage are needed to be consistent with the application of the definitions in TS Section 3.6.D/4.6.D (as revised below).

Safety Evaluation

The changes to the Identified and Unidentified Leakage definitions are acceptable because they clarify that the leakage is not limited to reactor coolant and includes all leakage in the drywell, not just to the collection systems. Essentially, there is no change to the actual intent or meaning of the words as they are to be used in the TS. Removing the "reactor coolant" portion of these definitions is more conservative and is consistent with guidance provided in NUREG-1433 (Reference 1). Use of the term "leakage detection systems" is being made to be consistent with the wording to be used in proposed TS 3.6.D/4.6.D. The changes also provide clearer and more concise definitions that are consistent with industry standards.

Adding the Definition for Total Leakage as the sum of Identified and Unidentified Leakage is required to support the changes proposed to TS 3.6.D/4.6.D in that the Total Leakage is the sum of Identified and Unidentified Leakage.

In summary, these changes are also considered acceptable because they provide consistency between the definitions and the requirements of the proposed changes to the Monticello TS. They are also consistent with industry standards.

Proposed
Change 2 - Revise TS 3.6.D by deleting the name of "Coolant Leakage" and renaming this Section of TS as "Reactor Coolant System (RCS)." Also divide the TS Section into two TS Subsections, Subsection 1 titled "Operational Leakage," and Subsection 2 titled "RCS Leakage Detection Instrumentation." Revised TS Table of Contents and renumber subparagraphs accordingly. (pages ii, 126 and 127)

Reason for Changes

The proposed changes to the TS are needed for clarification to make the TS more usable and understandable. By dividing the TS Section 3.6/4.6.D into two subsections it is clearer which portion is dedicated to RCS operational leakage and which part is dedicated to the instrumentation that detects the RCS leakage. Additionally, revising the TS Table of Contents (TOC) supports the proposed changes.

Safety Evaluation

This revision is consistent with best industry practices. These changes are acceptable because they provide consistency between the requirements of the changes to the Monticello TS proposed below. This change will provide a more easily understood TS in that it divides the TS dealing with RCS leakage into two separate subsections, one that focuses on RCS operational leakage and another that focuses on the required instrumentation to monitor the RCS leakage. Changing the TS TOC page provides editorial consistency for the change.

Proposed
Change 3 - Revise Current TS (CTS) 3.6.D.1 to Proposed TS (PTS) 3.6.D.1.a to delete reference to "based on sump monitoring."

Reason for Changes

This change is needed to support the transfer from the current method of complying with the current Monticello Coolant Leakage TS to the proposed method of complying with the Reactor Coolant Leakage requirements (i.e., changes from compliance by measuring/monitoring leakage to compliance by verifying leakage to be within TS limits). The proposed TS change requires Monticello to continue to meet TS Leakage limits regardless of how the leakage is monitored.

This change provides continued verification of being within TS limits. This TS deals with the protection of the Reactor Coolant Pressure Boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded.

Safety Evaluation

The TS and TS Bases indicate that the safety significant concern with leakage in the drywell is pressure boundary leakage.

This change is acceptable because the allowable RCS leakage limits are based on the predicted and experimentally observed behavior of pipe cracks. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage has also been considered. Evidence from experiments suggests that, for leakage even greater than the specified unidentified leakage limits, the probability is small that the imperfection or crack associated with such leakage would grow rapidly, based on information contained in the Updated Safety Analysis Report (USAR) and NUREG-1061. USAR section 10.3.6.3.1 states:

> *"The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USNRC-sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study) (Reference 56). Analysis, utilizing the data obtained in this study, has shown that there is a high probability that a leaking crack can be detected before it grows to a dangerous or 'critical' size. 'Critical' size is considered to be the size that would result in self propagation at the stress level existing. Mechanically or thermally induced cyclic loading, stress corrosion cracking, earthquake and normal vibration stresses are considered in the determination of the critical crack size. The critical crack size results in water leakage of about 150 gpm. Identified leakage (equipment drain sump) originates predominantly from pump seals and valve packing leakoffs. Background leakage is normally 1 to 3 gpm. It is estimated that a detection capability of 5 gpm is achievable. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. From the crack size a leakage rate can be determined. For a crack size which gives a leakage of 5 gpm, the probability of rapid propagation is less than 10^{-5}. Thus, an unidentified leak of 5 gpm when assumed to be from the primary system had less than one chance in 100,000 of propagating, which provides adequate margin."*

Further, evidence of the slow rate of growth of pipe cracks comes from NUREG-1061, Volume 3, Figure 6.1, which shows the rate of growth of the most IGSCC susceptible piping to be less than 1 in./year.

Clearly these documents demonstrate that a crack will not propagate significantly within the proposed Technical Specification 12 hour period prior to being in hot shutdown (See Change 6). After hot shutdown is achieved, the plant will begin to depressurize. This will cause less stress in the piping and a corresponding reduction in the leakage rate during that period.

Proposed
Change 4 - Revise CTS 4.6.D.1 to PTS 4.6.D.1 to delete the requirement to carry out a surveillance program to record unidentified and identified leakage rates once per 12 hours and replace it with a statement to every 12 hours verify drywell Unidentified Leakage, Total Leakage and Unidentified Leakage increase are within limits. (page 126)

Reason for Changes

This proposed change revises the TS to provide for a new methodology for determining compliance with PTS 3.6.D.1.a. This new methodology verifies that the drywell unidentified leakage and unidentified leakage increase and total leakage are all within the acceptable limits of PTS 3.6.D.1.a, instead of requiring the unidentified and identified leakage rates to be recorded once per 12 hours using primary containment floor and equipment drain sump monitoring equipment. This change is needed to support the transfer from the present method of complying with the current Monticello TS to the proposed method of complying with the proposed Monticello TS (i.e., from a compliance by measuring/monitoring leakage to a compliance by verifying leakage to be within TS limits).

The verification of leakage being within TS limits deals with the protection of the RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded.

Safety Evaluation

This change is acceptable because the RCS leakage is monitored by a variety of instruments designed to provide alarms when leakage is indicated and to quantify the various types of leakage. Sump level and flow rate are typically monitored to determine actual leakage rates; however, other methods may be used to quantify leakage. It is permissible to use pre-existing information, in conjunction with secondary measurements (e.g., Drywell pressure and temperature), to verify that leakage remains within TS limits by looking for step changes in conditions or performing calculations to determine leakage. The complete failure to demonstrate that RCS leakage is within limits, within the required frequency, constitutes a failure to meet this Surveillance Requirement, notwithstanding entrance into conditions and required actions of PTS LCO 3.6.D.2. In conjunction with alarms and other administrative controls, a 12-hour frequency for this surveillance is appropriate for identifying leakage and for tracking required trends. The 12-hour frequency is also consistent with other existing Monticello TS (e.g., 4.3.D).

Additionally, a control room alarm allows the operators to evaluate the significance of the indicated leakage and, if necessary, shut down the reactor for further investigation and corrective action. The allowed leakage rates are well below the rates predicted for critical crack sizes. Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

This change is applicable to Monticello and incorporates best industry practices into the Monticello TS, and provides wording and requirements similar to those found in NUREG-1433 (Reference 1).

Proposed
<u>Change 5</u> - Revise CTS 3.6.D.1.a, 3.6.D.1.b, 3.6.D.1.c and 3.6.D.1.d to PTS 3.6.D.1.a.1), 3.6.D.1.a.2), 3.6.D.1.a.3) and 3.6.D.1.a.4) respectively, to add "≤ " to PTS 3.6.D.1.a.1), 3.6.D.1.a.2) and 3.6.D.1.a.3). Revise PTS 3.6.D.1.a.2) to delete "any," replace it with "the previous" and add "while in the run mode." Also, revise PTS 3.6.D.1.a.3) to delete reference to "20 gpm Identified Leakage" and replace it with "≤ 25 gpm Total Leakage averaged over the previous 24 hour period," (page 126).

Reason for Changes

The proposed changes to CTS 3.6.D.1.a, 3.6.D.1.b and 3.6.D.1.c to PTS 3.6.D.1.a.1), 3.6.D.1.a.2) and 3.6.D.1.a.3) to add "≤" in front of leakage limits are needed to make it clear that the leakage must be less than or equal to the specified limits. The TS and TS Bases indicate that a safety significant concern with RCS leakage is with RCPB leakage. This would appear as Unidentified Leakage.

This proposed change combines leakage limits for the existing CTS "Unidentified Leakage" and "Identified Leakage" for the new category of "Total Leakage," which is based on a reasonable minimum detectable amount of leakage that also accounts for leakage from known sources (Identified Leakage). This change is needed to align the Monticello TS with industry standards.

A proposed change to CTS 3.6.D.1.b to PTS 3.6.D.1.a.2) will delete the word "any" and replace it with "the previous" and add the words "in the run mode." This will create a Leakage limit that reads "≤ 2 gpm increase in Unidentified Leakage within the previous 24 hour period while in the run mode." This change is needed to make the proposed wording consistent with industry standards.

Safety Evaluation

These changes are acceptable because they maintain the existing leakage limit values of the Monticello TS. They provide clarifying wording that the Unidentified Leakage increase is limited to the previous 24-hour period in the run mode, which is acceptable because this leakage limit is a very small fraction of the calculated flow from a critical crack in the primary system piping. The increase is measured relative to the steady state value; temporary changes in leakage rate as a result of transient conditions (e.g., startup) are not considered. As such, the 2 gpm increase limit is only applicable during the run mode when operating pressures and temperatures have stabilized. The "previous 24-hour period" is acceptable because it reflects the time period of interest, since Monticello is always required to be within limits, the term "any" could be misleading and create confusion as to exactly which 24 hour period the Leakage rate increase shall be verified.

Additionally, crack behavior from experimental programs shows that significantly higher leakage rates will precede crack instability and this flow increase limit is capable of providing an early warning of Intergranular Stress Corrosion Cracking (IGSCC) produced deterioration.

The summation of the limits for Unidentified Leakage (≤ 5 gpm) and Identified Leakage (20 gpm) to create the category of Total Leakage (≤ 25 gpm) is based on a reasonable minimum detectable amount and is acceptable because it is based on values already approved for Monticello and merely combines them to account for both "Identified" and "Unidentified" Leakage. The revision to average the Total Leakage over the previous 24-hour period is acceptable because it is recognized that during normal operation there may be occasions when Total Leakage could spike above the ≤ 25 gpm limit on a momentary basis. Averaging the Total Leakage limit requirement over the previous 24-hour period normalizes these spikes and is therefore acceptable.

Proposed
Change 6 - Revise CTS 3.6.D.2, 3.6.D.3, 3.6.D.4, 3.6.D.5.b and 3.6.D.6.b to PTS 3.6.D.1.b, 3.6.D.1.c, 3.6.D.1.d, 3.6.D.2.a.2) and 3.6.D.2.b.2) to change the unit shutdown statements to state that the unit should be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours (pages 126 and 126a).

Reason for Changes

These proposed changes provide for consistency between these shutdown requirements and other similar shutdown requirements found elsewhere in the Monticello TS. The time frames of 12 hours to be in Hot Shutdown and an additional 24 hours to be in Cold Shutdown are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging the plant safety systems.

Safety Evaluation

This proposed change modifies the shutdown language for these Monticello TS to add a Hot Shutdown requirement and lengthen the total time to achieve Cold Shutdown. The change is justified based on information contained in the Updated Safety Analysis Report (USAR) and NUREG-1061. USAR section 10.3.6.3.1 states:

> *"The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USNRC-sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study) (Reference 56). Analysis, utilizing the data obtained in this study, has shown that there is a high probability that a leaking crack can be detected before it grows to a dangerous or 'critical' size. 'Critical' size is considered to be the size that would result in self propagation at the stress level existing. Mechanically or thermally induced cyclic loading, stress corrosion cracking, earthquake and normal vibration stresses are considered in the determination of the critical crack size. The critical crack size results in water leakage of about 150 gpm.*

Identified leakage (equipment drain sump) originates predominantly from pump seals and valve packing leakoffs. Background leakage is normally 1 to 3 gpm. It is estimated that a detection capability of 5 gpm is achievable. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. From the crack size a leakage rate can be determined. For a crack size which gives a leakage of 5 gpm, the probability of rapid propagation is less than 10^{-5}. Thus, an unidentified leak of 5 gpm when assumed to be from the primary system had less than one chance in 100,000 of propagating, which provides adequate margin."

Further, evidence of the slow rate of growth of pipe cracks comes from NUREG-1061, Volume 3, Figure 6.1, which shows the rate of growth of the most IGSCC susceptible piping to be less than 1 in./year.

Clearly these documents demonstrate that a crack will not propagate significantly within the proposed Technical Specification 12 hour period prior to being in hot shutdown. After hot shutdown is achieved, the plant will begin to depressurize. This will cause less stress in the piping and a corresponding reduction in the leakage rate during that period.

Therefore, the requested change in shutdown requirements (i.e., from "reduce reactor water temperature to less than 212°F within 24 hours" to "12 hours to be in hot shutdown and an additional 24 hours to be in cold shutdown") is safe and reasonable.

Proposed
<u>Change 7</u> - Revise CTS 3.6.D.2 and 3.6.D.3 to PTS 3.6.D.1.b and 3.6.D.1.c to revise wording and add a clarifying statement to PTS 3.6.D.1.c to state, "reduce leakage to within limits within 4 hours or identify that the source of increased leakage is not service sensitive type 304 or type 316 austenitic stainless steel." (page 126)

<u>Reason for Changes</u>

The proposed changes provide time limits to reduce Unidentified Leakage or Total Leakage to within limits and investigate and identify the source of increased Unidentified Leakage and, more importantly, reduce the increase in leakage to within limits or verify the source of the leakage. This change is needed because the current Monticello TS is unclear as to what the expectation is once a leakage source is identified. This change is being proposed to enhance the readability and usability of the TS. Additionally, the proposed change focuses the investigation of increased unidentified leakage to determine if the leakage originates from service sensitive type 304 or type 316, austenitic stainless steel piping which is susceptible to IGSCC. This change is consistent with industry standards.

Safety Evaluation

The proposed changes are acceptable because they enhance the readability and usability of the TS. Requiring a reduction in Unidentified Leakage and Total Leakage to be within TS limits within 4 hours is acceptable because it allows time to identify the source of the leakage and, more importantly, reduce the leakage within limits. Additionally, if there is an increase in Unidentified Leakage above the limits then these changes focus the investigation to determine if the leakage originates from piping that is susceptible to IGSCC. Industry experience has shown that type 304 and type 316 austenitic stainless steel piping that is subject to high stress or that contains relatively stagnant or intermittent flow fluids is particularly susceptible to IGSCC. IGSCC produces tight cracks and the small flow increase limit is capable of providing an early warning of such deterioration. Verification that the source of the leakage is not type 304 or type 316, austenitic stainless steel eliminates IGSCC as a possible cause of the increased leakage. This significantly reduces concerns about crack instability, crack growth and a failure of the RCS boundary. Also, the unidentified leakage limit is still being maintained and will continue to limit the maximum unidentified leakage allowed.

Proposed
Change 8 - Revise CTS 3.6.D.4 to PTS 3.6.D.1.d to delete "is detected when the corrective actions outlined in 3.6.D.2 and 3.6.D.3 above are taken," and replace it with "exists" (page 126a).

Reason for Changes

This proposed change enhances the readability, usability and understanding of the TS. The proposed deletion of the referenced statement is needed because it has been made redundant by proposed changes in PTS 3.6.D.1.b and 3.6.D.1.c. The proposed changes identified in Changes 6 and 7 provide each of the referenced TS subsections with their own evaluation criteria and action statements. Therefore, there is no longer any need to reference these TS Sections in PTS 3.6.D.1.d. The addition of the word "exists" is administrative and is used to make the wording grammatically correct.

Safety Evaluation

This change is acceptable because deletion of the referenced statement is needed to eliminate redundant changes to CTS 3.6.D.2 and 3.6.D.3. The proposed changes identified in Changes 6 and 7 provide each TS subsection with its own evaluation criteria and action statements.

Based on the explanation above there is no need for proposed TS 3.6.D.1.d to reference the current TS 3.6.D.2 and 3.6.D.3 because proposed revisions to each of these TS will provide each subsection with there own evaluation criteria and action statements. This change eliminates redundant statements from the Monticello Proposed TS. The insertion of the word "exists" is administrative and is used to make the wording grammatically correct and is therefore acceptable. Additionally, this change clarifies that PTS 3.6.D.1.d specifically addresses the requirement that no pressure boundary leakage is allowed..

Proposed
Change 9 - Revise CTS 3.6.D.5 to PTS 3.6.D.2.a by deleting the statements "at least one of the leakage measurement instruments associated with each sump," and "If no leak rate measurement instruments associated with a sump" and replacing them with "the Drywell Floor Drain Sump Monitoring System," and "If the Drywell Floor Drain Sump Monitoring System is not operable," respectively. (page 126a)

Reason for Changes

These proposed changes delete the requirement to have leakage and leak rate measurement instruments, associated with each sump, operable and replaces it with a requirement for the Drywell Floor Drain Sump Monitoring System to be operable. These changes require the Drywell Floor Drain Sump Monitoring System to be operable to quantify the unidentified leakage from the RCS.

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. Leakage rate limits are set low enough to detect the leakage emitted from a single crack in the RCPB. Each of the leakage detection systems inside the drywell is designed with the capability of detecting leakage less than the established leakage rate limits and providing an appropriate alarm of excess leakage in the control room. The equipment drain sump monitoring system verifies that Total Leakage is within limits and is not required to monitor Unidentified Leakage. Therefore, because this installed instrumentation is not normally used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary, it is not required to be included in the TS.

Safety Evaluation

This change is acceptable because for the Drywell Floor Drain Sump Monitoring System to be considered operable, either the flow or level monitoring portion of the system must be operable. The Drywell Floor Drain Sump Monitoring System consists of a sump discharge flow integrator, one sump level recorder and one sump fill rate computer point (rate of change).

The total loss of the Drywell Floor Drain Sump Monitoring System results from the loss of all flow and level instrumentation (either directly or indirectly). The Drywell Floor Drain Sump Monitoring System remains operable when any one of the three channels is operable.

An alternate to the drywell floor drain sump system is the drywell equipment drain sump system. Because of the physical size of the sumps, it is possible to verify through detection (by level indication and the drywell particulate radioactivity monitor) and/or calculation that the required Unidentified Leakage limit ($\leq$ 5 gpm) and the increase in Unidentified Leakage rate limit ($\leq$ 2 gpm/24 hours) are within limits during the period of time it takes to actually overflow from one sump to the other.

During the period of time when the Drywell Floor Drain Sump level and flow indications are not capable of being monitored the Drywell Floor Drain Sump Monitoring System will be declared inoperable. With the Drywell Floor Drain Sump Monitoring System inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the drywell particulate radioactivity monitoring system will provide indication of changes in leakage. While unable to quantify the amount of leakage, the sensitivity is such that indications of leakage and changes in leakage allow verification that leakage is within limits.

Once the drywell floor drain sump is overflowing to the drywell equipment drain sump, the drywell equipment drain sump system instrumentation can be used to quantify floor drain sump leakage. However, the alarm settings for the equipment drain sump instruments must be reset to detect the lower limit for unidentified leakage. In this condition, any additional leakage measured by the drywell equipment drain sump system is assumed to be unidentified leakage unless the leakage has been identified and quantified. The opposite situation is also allowed, where the equipment drain sump is allowed to overflow into the floor drain sump. In this configuration, the alarm settings need not be reset, as they would conservatively quantify all additional leakage as unidentified, unless the leakage has been identified and quantified, and alarm at the appropriate limit. The other monitoring systems provide additional indication to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the sump monitoring systems inoperable, monitoring for leakage is degraded.

Normal operation of the equipment drain sump monitoring systems installed instrumentation is not used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. Operation of this system to monitor floor drain sump leakage is described in the TS Bases. Therefore, because this instrumentation does not meet the requirements of 10 CFR 50.36(c)(2)(ii)(A), it is not required to be included separately in the TS.

Proposed
Change 10 - Revise CTS 3.6.D.5.a to PTS 3.6.D.2.a.1) by deleting the statement "Perform manual leak rate measurements once per 12 hours and restore a measurement instrument" and replace it with "Restore the Drywell Floor Drain Sump Monitoring System."

Reason for Changes

This proposed change deletes an existing requirement to perform manual leak rate measurements because Proposed Change 9 has redefined the equipment for the Drywell Floor Drain Sump Monitoring System to incorporate the equipment that would be used to perform a manual leak rate measurement. This change provides consistency between the proposed revised Monticello TS. With the revised description of the Drywell Floor Drain Sump Monitoring System, a total loss of the Drywell Floor Drain Sump Monitoring System results from the loss of the sump discharge flow integrator, the level recorder and the sump fill rate computer points (either directly or indirectly). In other words, the Drywell Floor Drain Sump Monitoring System remains operable when any of the three channels are operable.

Safety Evaluation

This change is acceptable because it recognizes the importance of the identification of RCPB leakage so that appropriate action can be taken before the integrity of the RCPB is impaired. Sump monitoring systems for the RCS are provided to alert the operators when leakage rates above normal background levels are detected.

Leakage from the RCPB inside the drywell is detected by various independently monitored variables, such as sump level changes and drywell particulate radioactivity levels. The drywell particulate radioactivity monitoring system provides a backup to the Drywell Floor Drain Sump Monitoring System and is capable of monitoring leakage at least as low as 10^{-9} μCi/cc radioactivity for air particulate monitoring. However, the primary means of quantifying leakage in the drywell is the Drywell Floor Drain Sump Monitoring System.

With the Drywell Floor Drain Sump Monitoring System inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the drywell particulate radioactivity monitoring system will provide indication of changes in leakage.

With the Drywell Floor Drain Sump Monitoring System inoperable, operation may continue for only the next 30 days. The 30-day completion time is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available. In particular, these changes, along with proposed change 14 below, require that either the sump monitoring system or the drywell particulate radioactivity monitoring system always be operable.

The current Monticello TS require the performance of manual leak rate measurements once per 12 hours if the leakage measurement instruments associated with each sump are not operable. There is no current TS requirement for the drywell particulate radioactivity monitor to be operable during this period. The proposed TS requires the drywell particulate radioactivity monitoring system to be operable if the Drywell Floor Drain Sump Monitoring System is inoperable. As stated in other proposed TS changes, the drywell particulate radioactivity monitoring system provides a backup for the Drywell Floor Drain Sump Monitoring System. And, while it cannot quantify leakage in the drywell, it can monitor leakage increases.

As discussed above, once the sump is overflowing, the Floor Drain Sump Monitoring System is considered operable as long as the equipment drain sump system instrumentation is operable.

Proposed
<u>Change 11</u> - Add a new TS Limiting Condition for Operation (LCO) Section as PTS 3.6.D.2.c to state, "Any time irradiated fuel is in the reactor vessel and reactor water temperature is above 212°F at least one channel of the required leakage detection instrumentation shall be operable. If all channels of both systems (Drywell Floor Drain Sump Monitoring System and drywell particulate radioactivity monitoring system) are inoperable, restore at least one channel of the required leakage detection instrumentation to operable status within 1 hour, or be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours." (page 127)

<u>Reason for Changes</u>

This change is needed to provide direction if all the required leakage detection instrumentation is inoperable. If all channels of both systems (Drywell Floor Drain Sump Monitoring System and the drywell particulate radioactivity monitoring system) are inoperable, no TS means of detecting leakage are available.

<u>Safety Evaluation</u>

This change is acceptable because it adds a new LCO requirement to the Monticello TS to require at least one channel of the required leakage detection instrumentation to be available to monitor for potential RCS leakage. With all the required leakage detection instrumentation inoperable, no TS means of detecting leakage is available per this specification. This condition does not provide the required means of leakage detection. The required action is to restore at least one channel of the required leakage detection instrumentation system (Drywell Floor Drain Sump Monitoring System or drywell particulate radioactivity monitoring system) to operable status within 1 hour, to regain the intended leakage detection capability, or place the reactor in a condition in which the Limiting Condition for Operation is no longer applicable. The 1-hour completion time ensures that the plant will not operate in a degraded configuration for a lengthy time period, although drywell pressure and temperature will be monitored, which will provide an indication of change in RCS leakage. As justified above (Proposed Change 6), the 12 hours to Hot Shutdown and the additional 24 hours to Cold Shutdown are reasonable and acceptable.

Proposed
<u>Change 12</u> - Add a footnote to PTS Sections 3.6.D.2.a and 3.6.D.2.b to state, "A mode change is allowed when this system is inoperable." (page 127)

<u>Reason for Changes</u>

This change is needed to provide clarification that a mode change is allowed if either the Drywell Floor Drain Sump Monitoring System or the drywell particulate radioactivity monitoring system is inoperable. This change is applicable to Monticello and is being made to be consistent with the wording in NUREG-1433.

<u>Safety Evaluation</u>

This change is acceptable because, if the Drywell Floor Drain Sump Monitoring System is inoperable, other instrumentation is available to monitor RCS leakage. Instrumentation such as the drywell particulate radioactivity monitoring system is available, although it cannot quantify leakage, it will provide indication of changes in RCS leakage. Additionally, if the drywell particulate radioactivity monitoring system is inoperable, other instrumentation is available to monitor RCS leakage. Instrumentation such as the Drywell Floor Drain Sump Monitoring System will be able to quantify the RCS leakage. Also, drywell pressure and temperature are monitored, which will provide another indication of change in RCS leakage. Therefore, with other instrumentation available to monitor RCS leakage, a reactor mode change should be allowed.

Proposed
<u>Change 13</u> - Revise CTS 4.6.D.2 to PTS 4.6.D.2 to replace "The reactor coolant system" with "RCS Leakage Detection Instrumentation" and revise CTS 4.6.D.2.b to PTS 4.6.D.2.b by deleting "Primary containment sump leakage measurement system," and replacing it with "Required leakage detection instrumentation." (page 126a)

<u>Reason for Changes</u>

These proposed changes delete existing wording and replace them with wording more consistent with the other proposed TS changes. They also delete an existing requirement to perform a sensor check on the primary containment sump leakage measurement system and replace it with a requirement to perform a sensor check on the required leakage detection instrumentation. These changes are needed to provide consistency within the proposed Monticello TS. With these revisions to the TS, additional editorial changes are provided for consistency. A revised description for the required leakage detection instrumentation replaces the previously titled primary containment sump leakage measurement system.

Safety Evaluation

This change is acceptable because they are editorial and provide wording consistent with other proposed TS changes that are being made for this section. Deleting the reactor coolant system and replacing it with RCS Leakage Detection Instrumentation is acceptable because it provides consistency between the proposed LCO requirements of 3.6.D.2 and the proposed Surveillance Requirements of 4.6.D.2.

Additionally, the revision to CTS 4.6.D.2.b to delete "Primary containment sump leakage measurement system" and replacing it with PTS 4.6.D.2.b "Required leakage detection instrumentation," is acceptable because it is consistent with other proposed TS changes. These changes are editorial and administrative and are needed to revise wording consistent with other TS changes proposed by this license amendment request.

Proposed
Change 14 - Add a new TS SR for PTS 4.6.D.2.b by requiring a channel functional test** (flow instruments only) at least monthly. And add a new footnote to state, "** A functional test of this instrument means injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action."

Reason for Changes

This change adds a monthly functional test for the required leakage detection instrumentation. This functional test along with the sensor check and channel calibration will ensure that the required leakage detection instrumentation will operate with a high degree of reliability. This test is to be performed on the flow instruments only because they are the only instruments that can be functionally tested monthly. Level instrumentation, with the exception of the recorders, is inaccessible during power operation.

In addition, a footnote is added to this test requirement for clarification and to eliminate a potential conflict with TS Definition 1.0.E, "Instrument Functional Test." The footnote is needed to provide clarification that a functional test at the flow instruments sensor cannot be performed while at power because of the location of the instrumentation. The note is consistent with similar notes elsewhere in the Monticello TS (e.g., TS Table 4.2.1, note 5).

Safety Evaluation

The addition of a monthly functional test for the required leakage detection instrumentation flow instruments is acceptable because it provides an additional means of ensuring that the instrumentation is performing properly. In addition to the existing sensor check once per 12 hours, and the channel calibration test at least once per cycle, this test provides added assurance that at least part of the circuitry of the required leakage detection instrumentation is performing as designed.

Adding a footnote to this test is acceptable because it provides clarification and eliminates a potential conflict with an existing Monticello TS Definition. The footnote is also acceptable because of the physical configuration of the required leakage detection instrumentation. The injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action is acceptable because the signal cannot be injected into the primary sensor due to its location. This note is worded to be consistent with similar notes in the Monticello TS.

TS Bases

The applicable TS Bases have been revised, consistent with the above changes, to document the proposed revisions and provide supporting information. Current Monticello TS Bases pages marked up with supporting changes are provided in Exhibit D and revised Monticello TS Bases pages are provided in Exhibit E.

No Significant Hazards Consideration Determination

Nuclear Management Company, LLC (NMC) proposes the changes to Appendix A of the Operating License for the Monticello Nuclear Generating Plant to revise the Monticello TS to provide changes to the TS Definitions Section and the Coolant Leakage Section. These proposed changes clarify the definitions and restructure the Coolant Leakage Section of TS by retitling it as Reactor Coolant System (RCS) and dividing it into two subsections. One subsection provides criteria for Reactor Coolant System (RCS) Operational Leakage and the other subsection provides criteria for RCS Leakage Detection Instrumentation. The proposed revisions to the Monticello TS also revise the TS by focusing on Unidentified Leakage and Total Leakage requirements. The revisions add a TS Limiting Condition for Operation (LCO) requirement in the event the required leakage detection instrumentation is inoperable. Additionally, a TS LCO is being revised for Unidentified Leakage Rate Increase that focuses the attention of the operator on Intergranular Stress Corrosion Cracking (IGSCC) susceptible piping when an increase in leakage occurs.

The proposed amendment has been evaluated to determine whether it constitutes a significant hazards consideration as required by 10 CFR Part 50, Section 50.91, using standards provided in Section 50.92. This analysis is provided below:

1. *The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.*

The proposed Technical Specification changes do not introduce new equipment or new equipment operating modes, nor do the proposed changes alter existing system relationships. Additionally, the proposed changes do not affect any accident previously evaluated in the Monticello Updated Safety Analysis Report (USAR). The changes simply redefine the parameters for evaluation of leakage in the drywell. The evaluation criteria for drywell leakage have been refocused into the areas that are most susceptible to IGSCC. Consequently, the probability of an accident previously evaluated is not significantly increased.

The equipment referenced in the proposed changes is still required to monitor the reactor coolant system operational leakage to ensure appropriate action is taken before the integrity of the reactor coolant pressure boundary is impaired. As a result, operation of the facility with the proposed changes will continue to meet the licensing basis and applicable guidelines. As such, the consequences of any accident previously evaluated are not significantly affected.

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

2. *The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.*

The proposed changes do not involve physical alterations of the plant; no new or different type of equipment will be installed; nor are there significant changes in the methods governing normal plant operation. The changes simply redefine the parameters for evaluation of leakage in the drywell. The evaluation criteria for drywell leakage have been refocused into the areas that are most susceptible to IGSCC. Additionally, the changes do not create any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases.

Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

3. *The proposed amendment will not involve a significant reduction in the margin of safety.*

The proposed amendment redefines the parameters for evaluation of leakage in the drywell. There are no physical alterations of the plant; no new or different type of equipment will be installed; nor are there significant changes in the methods governing normal plant operation. Additionally, the proposed changes do not exceed or alter a design basis or safety limit as established in the Monticello licensing basis.

Therefore, these proposed changes will not involve a significant reduction in the margin of safety.

Based on the evaluation described above and pursuant to 10 CFR Part 50, Section 50.91, NMC has determined that operation of the Monticello Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined in 10 CFR Part 50, Section 50.92.

Environmental Assessment

Nuclear Management Company, LLC has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration.
2. The changes do not involve a significant change in the type or significant increase in the amounts of any effluent that may be released offsite.
3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR Part 51, Section 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51, Section 51.22(b), an environmental assessment of the proposed changes is not required.

References:

1. NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," Revision 2, April, 2001.

License Amendment Request for Drywell Leakage and
Sump Monitoring Detection Technical Specification Changes (TAC No. MB6493)

Current Monticello Technical Specification Pages Marked Up With Proposed Changes

This exhibit consists of marked up Technical Specification pages that incorporate the proposed change. The pages included in this exhibit are as listed below:

Pages

ii
5
126
126a
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TABLE OF CONTENTS (Cont'd)

Y. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. In this condition, a reactor scram is initiated and a rod block is inserted directly from the mode switch. The scram can be reset after a short time delay.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.
2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

Z. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

AA. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling, also referred to as partial nucleate boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

AB. Pressure Boundary Leakage - Pressure boundary leakage shall be leakage through a non-isolable fault in the reactor coolant system pressure boundary.

AC. Identified Leakage - Identified leakage shall be:

CHANGE 1

1. ~~Reactor coolant l~~ **L**eakage into **the** drywell ~~collection systems~~, such as **that from** pump seals or valve packing leaks, that is captured and conducted to a sump or collecting tank, or
2. ~~Reactor coolant l~~ **L**eakage into the drywell atmosphere from sources ~~which~~ **that** are **both** specifically located and known **either** not to ~~be~~ **interfere with the operation of leakage detection systems or not to be** Pressure Boundary Leakage. ~~or which do not significantly impair the methods used to detect reactor coolant leakage~~.

AD. Unidentified Leakage - ~~Unidentified leakage shall be all reactor coolant~~ **All** leakage **into the drywell that** ~~which~~ is not Identified Leakage.

AE. Total Leakage – Sum of the Identified and Unidentified Leakage.

AF ~~E~~. through AH. (Deleted)

AI. Purging - Purging is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

AJ. Venting - Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required.

CHANGE 2

D. ~~Coolant Leakage~~ **Reactor Coolant System (RCS)**

1. Operational Leakage

~~1.~~ a. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, reactor coolant system leakage, ~~based on sump monitoring,~~ shall be limited to:

CHANGE 3

CHANGE 5

CHANGE 2

~~a.~~ 1) ≤ 5 gpm Unidentified Leakage,

~~b.~~ 2) ≤ 2 gpm increase in Unidentified Leakage within ~~Any~~ **the previous** 24 hour period **while in the Run Mode,**

~~c.~~ 3) ≤ 25 ~~0~~ gpm ~~Identified~~ **Total** Leakage **averaged over the previous 24 hour period, and**

~~d.~~ 4) no pressure boundary leakage

CHANGE 2 ~~2.~~ b. With reactor coolant system leakage greater than 3.6.D.1.**a.1)** or 3.6.D.1.~~c~~ **a.3)** above, reduce the leakage ~~rate~~ to within ~~acceptable~~ limits within four hours or ~~initiate an orderly shutdown of the reactor. and reduce reactor water temperature to less than 212°F within 24 hours.~~ **be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.**

CHANGE 7

CHANGE 6

CHANGE 2 ~~3.~~ c. With an increase in Unidentified Leakage in excess of the rate specified in 3.6.D.1.~~b~~ **a.2) reduce leakage to within limits within four hours or** identify **that** the source of increased leakage **is not service sensitive type 304 or type 316 austenitic stainless steel** within four hours or ~~initiate an orderly shutdown of the reactor. and reduce reactor water temperature to less than 212°F within 24 hours.~~ **be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.**

CHANGE 7

CHANGE 6

D. ~~Coolant Leakage~~ **Reactor Coolant System (RCS)**

1. Operational Leakage

~~1.~~ Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, ~~the following surveillance program shall be carried out:~~ **every 12 hours verify the following:**

CHANGE 4

~~a. Unidentified and Identified Leakage rates shall be recorded once per 12 hours using primary containment floor and equipment drain sump monitoring equipment.~~

a. Unidentified Leakage is within limits,

b. Unidentified Leakage increase is within limits, and

c. Total Leakage is within limits.

CHANGE 8

CHANGE 2 ~~4.~~ d. If any Pressure Boundary Leakage **exists, be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours** ~~is detected when the corrective actions outlined in 3.6.D.2 and 3.6.D.3 above are taken, initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.~~ CHANGE 6 CHANGE 8 CHANGE 6

CHANGE 2 **~~2.~~ 5. RCS Leakage Detection Instrumentation**

a. Any time irradiated fuel is in the reactor vessel and reactor water temperature is above 212°F ~~at least one of~~ the **Drywell Floor Drain Sump Monitoring System** ~~leakage measurement instruments associated with each sump~~ shall be operable*. If ~~no~~ **the Drywell Floor Drain Sump Monitoring System** ~~leak rate measurement instruments associated with a sump are~~ **is not** operable, then: CHANGE 9

CHANGE 2 ~~a.~~ 1) ~~Perform manual leak rate measurements once per 12 hours and r~~ Restore ~~a measurement~~ **the Drywell Floor Drain Sump Monitoring System** ~~instrument~~ to operable status within 30 days. CHANGE 10

CHANGE 2 ~~b.~~ 2) Otherwise, ~~initiate an orderly shutdown of the reactor. and reduce reactor water temperature to less than 212°F within 24 hours~~ **be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.** CHANGE 6

CHANGE 2 ~~6.~~ b. Any time irradiated fuel is in the reactor vessel and reactor water temperature is above 212°F the drywell particulate radioactivity monitoring system shall be operable*. If the drywell particulate radioactivity monitoring system is not operable, then: CHANGE 12

CHANGE 2 ~~a.~~ 1) Analyze grab samples of the primary containment atmosphere once per 12 hours.

~~b.~~ 2) Otherwise, ~~initiate an orderly shutdown of the~~ **be in Hot Shutdown within the next** ~~reactor.~~ **12 hours and in** ~~and reduce reactor water temperature~~ **Cold Shutdown within the following 24 hours.** ~~to less than 212°F within 24 hours.~~ CHANGE 6

2. RCS Leakage Detection Instrumentation

~~The reactor coolant system~~ **RCS** leakage detection ~~systems~~ **instrumentation** shall be demonstrated OPERABLE by:

CHANGE 13

a. Primary containment atmosphere particulate monitoring systems - perform~~ance of~~ a sensor check once per 12 hours, a channel functional test at least monthly and a channel calibration at least once per cycle.

b. **Required leakage detection instrumentation** ~~Primary containment sump leakage measurement system-performance of a~~ sensor check once per 12 hours, **a channel functional test** (flow instruments only) at least monthly,** and a channel calibration test at least once per cycle.

CHANGE 14 ** **A functional test of this instrument means injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action.**

CHANGE 2 C. Any time irradiated fuel is in the reactor vessel and reactor water temperature is above 212°F at least one channel of the required leakage detection instrumentation shall be operable. If all channels of both systems (Drywell Floor Drain Sump Monitoring System and drywell particulate radioactivity monitoring system) are inoperable, restore at least one channel of the required leakage detection instrumentation to operable status within 1 hour, or be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

CHANGE 11

CHANGE 12 * A mode change is allowed when this system is inoperable.

E. Safety/Relief Valves

1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F the safety valve function (self actuation) of seven safety/relief valves shall be operable (note: Low-Low Set and ADS requirements are located in Specification 3.2.H. and 3.5.A, respectively).

   Valves shall be set as follows:

   8 valves at ≤1120 psig

2. If Specification 3.6.E.1 is not met, initiate an orderly shutdown and have reactor coolant pressure and temperature reduced to 110 psig or less and 345°F or less within 24 hours.

E. Safety/Relief Valves

1. a. Safety/relief valves shall be tested or replaced each refueling outage in accordance with the Inservice Testing Program.

   b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage.

   c. The integrity of the safety/relief valve bellows shall be continuously monitored.

   d. The operability of the bellows monitoring system shall be demonstrated each operating cycle.

2. Low-Low Set Logic surveillance shall be performed in accordance with Table 4.2.1.

License Amendment Request for Drywell Leakage and
Sump Monitoring Detection Technical Specification Changes (TAC No. MB6493)

Revised Monticello Technical Specification Pages

This exhibit consists of revised Technical Specification pages that incorporate the proposed change. The pages included in this exhibit are as listed below:

Pages

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5
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126a
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TABLE OF CONTENTS (Cont'd)

Y. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. In this condition, a reactor scram is initiated and a rod block is inserted directly from the mode switch. The scram can be reset after a short time delay.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.
2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

Z. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

AA. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling, also referred to as partial nucleate boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

AB. Pressure Boundary Leakage - Pressure boundary leakage shall be leakage through a non-isolable fault in the reactor coolant system pressure boundary.

AC. Identified Leakage - Identified leakage shall be:

1. Leakage into the drywell, such as that from pump seals or valve packing leaks, that is captured and conducted to a sump or collecting tank, or
2. Leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be Pressure Boundary Leakage.

AD. Unidentified Leakage - All leakage into the drywell that is not Identified Leakage.

AE. Total Leakage - Sum of the Identified and Unidentified Leakage.

AF. through AH. (Deleted)

AI. Purging - Purging is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

AJ. Venting - Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

D. Reactor Coolant System (RCS)

1. Operational Leakage

a. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, reactor coolant system (RCS) leakage, shall be limited to:

1) ≤5 gpm Unidentified Leakage

2) ≤2 gpm increase in Unidentified Leakage within the previous 24 hour period while in the run mode,

3) ≤25 gpm Total Leakage averaged over the previous 24 hour period, and

4) no pressure boundary leakage

b. With reactor coolant system leakage greater than 3.6.D.1.a.1) or 3.6.D.1.a.3) above, reduce the leakage to within limits within four hours, or be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

c. With an increase in Unidentified Leakage in excess of the rate specified in 3.6.D.1.a.2) reduce leakage to within limits within four hours, or verify that the source of increased leakage is not service sensitive type 304 or type 316 austenitic stainless steel within four hours, or be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

D. Reactor Coolant System (RCS)

1. Operational Leakage

Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, every 12 hours verify the following:

a. Unidentified Leakage is within limits,

b. Unidentified Leakage increase is within limits, and

c. Total Leakage is within limits.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

d. If any Pressure Boundary Leakage exists, be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

2. RCS Leakage Detection Instrumentation

   a. Any time irradiated fuel is in the reactor vessel and reactor water temperature is above 212°F the Drywell Floor Drain Sump Monitoring System shall be operable.* If the Drywell Floor Drain Sump Monitoring System is not operable, then:

      1) Restore the Drywell Floor Drain Sump Monitoring System to operable status within 30 days.

      2) Otherwise, be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

   b. Any time irradiated fuel is in the reactor vessel and reactor water temperature is above 212°F the drywell particulate radioactivity monitoring system shall be operable.* If the drywell particulate radioactivity monitoring system is not operable, then:

      1) Analyze grab samples of the primary containment atmosphere once per 12 hours.

* A mode change is allowed when this system is inoperable.

4.0 SURVEILLANCE REQUIREMENTS

2. RCS Leakage Detection Instrumentation

   RCS leakage detection instrumentation shall be demonstrated OPERABLE by:

   a. Primary containment atmosphere particulate monitoring system - perform a sensor check once per 12 hours, a channel functional test at least monthly and a channel calibration at least once per cycle.

   b. Required leakage detection instrumentation - perform a sensor check once per 12 hours, a channel functional test** (flow instruments only) at least monthly, and a channel calibration test at least once per cycle.

** A functional test of this instrument means injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

2) Otherwise, be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

c. Any time irradiated fuel is in the reactor vessel and reactor water temperature is above 212°F at least one channel of the required leakage detection instrumentation shall be operable. If all channels of both systems (Drywell Floor Drain Sump Monitoring System and drywell particulate radioactivity monitoring system) are inoperable, restore at least one channel of the required leakage detection instrumentation to operable status within 1 hour, or be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

E. Safety/Relief Valves

1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F the safety valve function (self actuation) of seven safety/relief valves shall be operable (note: Low-Low Set and ADS requirements are located in Specification 3.2.H. and 3.5.A, respectively).

   Valves shall be set as follows:

   8 valves at $\leq$1120 psig

2. If Specification 3.6.E.1 is not met, initiate an orderly shutdown and have reactor coolant pressure and temperature reduced to 110 psig or less and 345°F or less within 24 hours.

E. Safety/Relief Valves

1. a. Safety/relief valves shall be tested or replaced each refueling outage in accordance with the Inservice Testing Program.

   b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage.

   c. The integrity of the safety/relief valve bellows shall be continuously monitored.

   d. The operability of the bellows monitoring system shall be demonstrated each operating cycle.

2. Low-Low Set Logic surveillance shall be performed in accordance with Table 4.2.1.

License Amendment Request for Drywell Leakage and
Sump Monitoring Detection Technical Specification Changes (TAC No. MB6493)

Current Monticello Technical Specification Bases Pages
Marked Up With Supporting Changes

This exhibit consists of marked up Technical Specification Bases pages, consistent with the TS changes, to document the proposed revisions and provide supporting information. The TS Bases are revised in accordance with Monticello TS 6.8.K, TS Bases Control Program. The pages included in this exhibit are as listed below:

Pages

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152a

D. ~~Coolant Leakage~~ **Reactor Coolant System (RCS)**

1. **RCS Operational Leakage**

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be Pressure Boundary Leakage and they cannot be reduced within the allowed times, the reactor will be shutdown to allow further investigation and corrective action.

The low limit on increase in unidentified leakage assumes a failure mechanism of Intergranular Stress Corrosion Cracking (IGSCC) that produces tight cracks. This flow increase limit is capable of providing an early warning of such deterioration.

No applicable safety analysis assumes the total leakage limit. The total leakage limit considers RCS inventory makeup capability and drywell sump capacity. Drywell Equipment Drain Sump instrumentation is required to support verification of the Total Leakage limit.

With RCS unidentified or total leakage greater than the limits, actions must be taken to reduce the leak. Because the leakage limits are conservatively below the leakage that would constitute a critical crack size, 4 hours is allowed to reduce the leakage rates before the reactor must be shut down. If unidentified leakage has been identified and quantified, it may be reclassified and considered as identified leakage; however, the total leakage limit would remain unchanged.

An unidentified leakage increase of > 2 gpm within a 24 hour period is an indication of a potential flaw in the Reactor Coolant Pressure Boundary (RCPB) and must be quickly evaluated. The increase does not necessarily violate the absolute unidentified leakage limit, therefore, an option exists to allow continued reactor operation if certain susceptible components are determined not to be the source of the leakage increase within the required completion time. For an unidentified leakage increase greater than required limits, an alternative to reducing leakage increase to within limits (i.e., reducing the leakage rate such that the current rate is less than the "2 gpm increase in the previous 24 hours" limit; either by isolating the source or other possible methods) is to evaluate service sensitive type 304 and type 316 austentic stainless steel piping that is subject to high stress or that contains relatively stagnant or intermittent flow fluids and determine it is not the source of the increased leakage. This type of piping is very susceptible to IGSCC. Note also that once leakage is attributed to a specific source, that leakage can be considered to be identified and can be applied against the identified limit, rather than the unidentified limit. The 4 hour completion time is reasonable to properly reduce the unidentified leakage increase or verify the source before the reactor must be shut down without unduly jeopardizing plant safety.

The Surveillance Requirement (SR) associated with RCS leakage is acceptable because RCS leakage is monitored by a variety of instruments designed to provide alarms when leakage is indicated and to quantify the various types of leakage. Sump level and flow rate are typically monitored to determine actual leakage rates; however, other methods may be used to verify leakage. It is permissible to use pre-existing information, in conjunction with secondary measurements (e.g., Drywell pressure and temperature), to verify that leakage remains within limits by looking for step changes in conditions or to perform calculations to estimate leakage. The complete failure to demonstrate that RCS leakage is within limits, on the required frequency, constitutes a failure to meet this SR, notwithstanding entrance into conditions and required actions of TS 3.6.D.2.

2. **RCS Leakage Detection Instrumentation**

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from the recirculation pump seals, valve stem leak-offs, reactor vessel flange leak-off, bulkhead and bellows drains, and vent cooler drains to the drywell equipment drain sump. All other leakage is collected in the drywell floor drain sump. Both sumps are equipped with level and flow transmitters connected to recorders in the control room. **The Drywell Floor Drain Sump Monitoring System instrumentation consists of one floor drain sump flow integrator, one sump level recorder and one sump fill rate computer point (rate of change). The Drywell Floor Drain Sump Monitoring System is operable when any one of these three channels is operable.** An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached.

Drywell airborne particulate radioactivity is continuously monitored as well as drywell atmospheric temperature and pressure. **The drywell particulate radioactivity monitoring system monitors the drywell for airborne particulate radioactivity. A sudden increase in radioactivity may be attributed to RCPB steam or reactor water leakage. The drywell particulate radioactivity monitoring system is not capable of quantifying leakage rates, but is sensitive enough to indicate increased leakage rates. The drywell particulate radioactivity monitoring system provides a backup to the Drywell Floor Drain Sump Monitoring System and is capable of monitoring leakage at least as low as 10^{-9} µCi/cc radioactivity for air particulate monitoring.** Systems connected to the reactor coolant systems boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

The Drywell Floor Drain Sump Monitoring System is required to quantify the unidentified leakage from the RCS. Thus, for the system to be considered operable, either the flow monitoring or the sump level monitoring portion of the system must be operable. Any failure of a sump monitoring subsystem should be evaluated for its impact on the ability of the associated instrumentation to measure leakage.

Since the flow integrator for each sump is not directly tied to the sump for its input signals, they are not affected in the same way as other instrumentation. However, the loss of flow through the flow integrator prevents the flow integrator from performing its intended safety function of measuring leakage, and even though its associated SRs continue to be met, it should be declared inoperable.

It should be noted that system isolation in response to Required Actions of LCO 3.7.D.2, would not render these instruments inoperable, provided the system could be unisolated as allowed by the footnote of LCO 3.7.D.2, as manual operation is allowed.

The total loss of the Drywell Floor Drain Sump Monitoring System results from the loss of all flow and level instrumentation (either directly or indirectly).

An alternate to the Drywell Floor Drain Sump Monitoring System is the drywell equipment drain sump system. Because of the physical size of the sumps, it is possible through detection or calculation to verify the required leakage limit (5 gpm) and rate limit (2 gpm/24 hrs) during the period of time it takes to actually overflow from one sump to the other. Once the drywell floor drain sump is overflowing to the drywell equipment drain sump, the drywell equipment drain sump system can be used to quantify leakage. However, the alarm settings for the equipment drain sump instruments must be reset to detect the lower limit for unidentified leakage. In this condition, all additional leakage measured by the drywell equipment drain sump system is assumed to be unidentified leakage unless the leakage has been identified and

quantified. The opposite situation is also allowed, where the equipment drain sump is allowed to overflow into the floor drain sump. In this configuration, the alarm settings need not be reset, as they would conservatively quantify all additional leakage as unidentified, unless the leakage has been identified and quantified, and alarm at the appropriate limit. The other monitoring systems provide additional indication to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for leakage is degraded.

With the Drywell Floor Drain Sump Monitoring System inoperable, no other form of sampling can provide the equivalent information to quantify unidentified leakage. However, the drywell particulate radioactivity monitoring system will provide indication of changes in leakage.

With the Drywell Floor Drain Sump Monitoring System inoperable, operation may continue for 30 days. The 30 days is acceptable, based on operating experience, considering other methods of detecting leakage are available. The action requirements are modified by a footnote that allows a Mode change when the Drywell Floor Drain Sump Monitoring System is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

With the drywell particulate radioactivity monitoring system inoperable, operation may continue as long as grab samples are taken every 12 hours to analyze the drywell atmosphere. The action requirements are modified by a footnote that allows a Mode change when the drywell particulate radioactivity monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

With the required leakage detection instrumentation inoperable, no means of detecting leakage is available. This condition does not provide the required means of leakage detection. The required action is to restore one channel of the inoperable monitoring systems (Drywell Floor Drain Sump Monitoring System or drywell particulate radioactivity monitoring system) to operable status within 1 hour to regain the intended leakage detection capability. The 1-hour completion time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.

The sensitivity of the sump leakage detection systems for detection of leak rate changes is better than one gpm in a one hour period.
Other leakage detection methods provide warning of abnormal leakage and are not directly calibrated to provide leak rate measurements.

E. Safety/Relief Valves

The reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, with at least one safety/relief valve set to open at a pressure no greater than design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the Code for a condition of MSIV closure while operating at 1775 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety/relief valves set as specified herein, the maximum vessel pressure remains below the 1375 psig ASME Code limit. Only five of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 3% above their setpoint of 1109 psig with a 0.4 second delay. The upper limit on safety/relief valve setpoint is established by the operating limit of the HPCI and RCIC systems of 1120 psig. The design capability of the HPCI and RCIC systems has been conservatively demonstrated to be acceptable at pressures 3% greater than the safety/relief valve setpoint of 1109 psig. HPCI and RCIC pressures required for system operation are limited by the Low-Low Set SRV System to well below these values.

The safety/relief valves have two functions; 1) over-pressure relief (self-actuation by high pressure), and 2) Depressurization/ Pressure Control (using air actuators to open the valves via ADS, Low-Low Set system, or manual operation).

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9, Section N-911.4(a)(4) of the ASME Pressure Vessel Code Section III Nuclear Vessels (1965 and 1968 editions) requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

Low-Low Set Logic has been provided on three non-Automatic Pressure Relief System valves. This logic is discussed in detail in the Section 3.2 Bases. This logic, through pressure sensing instrumentation, reduces the opening setpoint and increases the blowdown range of the three selected valves following a scram to eliminate the discharge line water leg clearing loads resulting from multiple valve openings.

Testing of the safety/relief valves in accordance with ANSI/ASME OM-1-1981 each refueling outage ensures that any valve deterioration is detected. An as-found tolerance value of 3% for safety/relief valve setpoints is specified in ANSI/ASME OM-1-1981. Analyses have been performed with the valves assumed to open at 3% above their setpoint of 1109 psig. The 1375 psig Code limit is not exceeded in any case. When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system once per cycle provides assurance of bellows integrity.

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License Amendment Request for Drywell Leakage and
Sump Monitoring Detection Technical Specification Changes (TAC No. MB6493)

Revised Monticello Technical Specification Bases Pages

This exhibit consists of revised Technical Specification Bases pages, consistent with the TS changes, to document the proposed revisions and provide supporting information. The TS Bases are revised in accordance with Monticello TS 6.8.K, TS Bases Control Program. The pages included in this exhibit are as listed below:

Pages

150
151
152
152a
152b

Bases 3.6/4.6 (Continued):

D. Reactor Coolant System (RCS)

1. RCS Operational Leakage

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be Pressure Boundary Leakage and they cannot be reduced within the allowed times, the reactor will be shutdown to allow further investigation and corrective action.

The low limit on increase in Unidentified Leakage assumes a failure mechanism of Intergranular Stress Corrosion Cracking (IGSCC) that produces tight cracks. This flow increase limit is capable of providing an early warning of such deterioration.

No applicable safety analysis assumes the Total Leakage limit. The Total Leakage limit considers RCS inventory makeup capability and drywell sump capacity. Drywell Equipment Drain Sump instrumentation is required to support verification of the Total Leakage limit.

With RCS Unidentified or Total Leakage greater than the limits, actions must be taken to reduce the leak. Because the leakage limits are conservatively below the leakage that would constitute a critical crack size, 4 hours is allowed to reduce the leakage rates before the reactor must be shut down. If Unidentified Leakage has been identified and quantified, it may be reclassified and considered as Identified Leakage; however, the Total Leakage limit would remain unchanged.

An Unidentified Leakage increase of >2 gpm within a 24 hour period is an indication of a potential flaw in the Reactor Coolant Pressure Boundary (RCPB) and must be quickly evaluated. The increase does not necessarily violate the absolute Unidentified Leakage limit, therefore, an option exists to allow continued reactor operation if certain susceptible components are determined not to be the source of the leakage increase within the required completion time. For an Unidentified Leakage increase greater than required limits, an alternative to reducing leakage increase to within limits (i.e., reducing the leakage rate such that the current rate is less than the "2 gpm increase in the previous 24 hours" limit; either by isolating the source or other possible methods) is to evaluate service sensitive type 304 and type 316 austenitic stainless steel piping that is subject to high stress or that contains relatively stagnant or intermittent flow fluids and determine it is not the source of the increased leakage. This type of piping is very susceptible to IGSCC. Note also that once leakage is attributed to a specific source, that leakage can be considered to be identified and can be applied against the identified limit, rather than the unidentified limit. The 4 hour completion time is reasonable to properly reduce the Unidentified Leakage increase or verify the source before the reactor must be shut down without unduly jeopardizing plant safety.

The Surveillance Requirement (SR) associated with RCS leakage is acceptable because RCS leakage is monitored by a variety of instruments designed to provide alarms when leakage is indicated and to quantify the various types of leakage. Sump level and flow rate are typically monitored to determine actual leakage rates; however, other methods may be used to verify leakage. It is permissible to use pre-existing information, in conjunction with secondary measurements (e.g., drywell pressure and temperature), to verify that leakage remains within limits by looking for step changes in conditions or to perform calculations to estimate leakage. The complete failure to demonstrate that RCS leakage is within limits, on the required frequency, constitutes a failure to meet this SR, notwithstanding entrance into conditions and required actions of TS 3.6.D.2.

2. RCS Leakage Detection Instrumentation

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from the recirculation pump seals, valve stem leak-offs, reactor vessel flange leak-off, bulkhead and bellows drains, and vent cooler drains to the drywell equipment drain sump. All other leakage is collected in the drywell floor drain sump. Both sumps are equipped with level and flow transmitters connected to recorders in the control room. The Drywell Floor Drain Sump Monitoring System instrumentation consists of one floor drain sump flow integrator, one sump level recorder and one sump fill rate computer point (rate of change). The Drywell Floor Drain Sump Monitoring System is operable when any one of these three channels is operable. An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached.

Drywell airborne particulate radioactivity is continuously monitored as well as drywell atmospheric temperature and pressure. The drywell particulate radioactivity monitoring system monitors the drywell for airborne particulate radioactivity. A sudden increase in radioactivity may be attributed to RCPB steam or reactor water leakage. The drywell particulate radioactivity monitoring system is not capable of quantifying leakage rates, but is sensitive enough to indicate increased leakage rates. The drywell particulate radioactivity monitoring system provides a backup to the Drywell Floor Drain Sump Monitoring System and is capable of monitoring leakage at least as low as 10^{-9} μCi/cc radioactivity for air particulate monitoring. Systems connected to the reactor coolant systems boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

The Drywell Floor Drain Sump Monitoring System is required to quantify the unidentified leakage from the RCS. Thus, for the system to be considered operable, either the flow monitoring or the sump level monitoring portion of the system must be operable. Any failure of a sump monitoring system should be evaluated for its impact on the ability of the associated instrumentation to measure leakage.

Since the flow integrator for each sump is not directly tied to the sump for its input signals, they are not affected in the same way as other instrumentation. However, the loss of flow through the flow integrator prevents the flow integrator from performing its intended safety function of measuring leakage, and even though its associated SRs continue to be met, it should be declared inoperable.

It should be noted that system isolation in response to Required Actions of LCO 3.7.D.2, would not render these instruments inoperable, provided the system could be unisolated as allowed by the footnote of LCO 3.7.D.2, as manual operation is allowed.

The total loss of the Drywell Floor Drain Sump Monitoring System results from the loss of all flow and level instrumentation (either directly or indirectly).

An alternate to the Drywell Floor Drain Sump Monitoring System is the drywell equipment drain sump system. Because of the physical size of the sumps, it is possible through detection or calculation to verify the required leakage limit (5 gpm) and rate limit (2 gpm/24 hours) during the period of time it takes to actually overflow from one sump to the other. Once the drywell floor drain sump is overflowing to the drywell equipment drain sump, the drywell equipment drain sump system can be used to quantify leakage. However, the alarm settings for the equipment drain sump instruments must be reset to detect the lower limit for unidentified leakage. In this condition, all additional leakage measured by the drywell equipment drain sump system is assumed to be Unidentified Leakage unless the leakage has been identified and quantified. The opposite situation is also allowed, where the equipment drain sump is allowed to overflow into the floor drain sump. In this configuration, the alarm settings need not be reset, as they would conservatively quantify all additional leakage as unidentified, unless the leakage has been identified and quantified, and alarm at the appropriate limit. The other monitoring systems provide additional indication to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. The drywell particulate radioactivity monitoring system provides a backup system to the Drywell Floor Drain Sump Monitoring System. With the leakage detection systems inoperable, monitoring for leakage is degraded.

With the Drywell Floor Drain Sump Monitoring System inoperable, no other form of sampling can provide the equivalent information to quantify Unidentified Leakage. However, the drywell particulate radioactivity monitoring system will provide indication of changes in leakage.

Bases 3.6/4.6 (Continued):

With the Drywell Floor Drain Sump Monitoring System inoperable, operation may continue for 30 days. The 30 days is acceptable, based on operating experience, considering other methods of detecting leakage are available. The action requirements are modified by a footnote that allows a Mode change when the Drywell Floor Drain Sump Monitoring System is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

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The sensitivity of the sump leakage detection systems for detection of leak rate changes is better than one gpm in a one hour period. Other leakage detection methods provide warning of abnormal leakage and are not directly calibrated to provide leak rate measurements.

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The reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, with at least one safety/relief valve set to open at a pressure no greater than design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the Code for a condition of MSIV closure while operating at 1775 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety/relief valves set as specified herein, the maximum vessel pressure remains below the 1375 psig ASME Code limit. Only five of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 3% above their setpoint of 1109 psig with a 0.4 second delay. The upper limit on safety/relief

Bases 3.6/4.6 (Continued):

valve setpoint is established by the operating limit of the HPCI and RCIC systems of 1120 psig. The design capability of the HPCI and RCIC systems has been conservatively demonstrated to be acceptable at pressures 3% greater than the safety/relief valve setpoint of 1109 psig. HPCI and RCIC pressures required for system operation are limited by the Low-Low Set SRV System to well below these values.

The safety/relief valves have two functions; 1) over-pressure relief (self-actuation by high pressure), and 2) Depressurization/ Pressure Control (using air actuators to open the valves via ADS, Low-Low Set system, or manual operation).

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9, Section N-911.4(a)(4) of the ASME Pressure Vessel Code Section III Nuclear Vessels (1965 and 1968 editions) requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

Low-Low Set Logic has been provided on three non-Automatic Pressure Relief System valves. This logic is discussed in detail in the Section 3.2 Bases. This logic, through pressure sensing instrumentation, reduces the opening setpoint and increases the blowdown range of the three selected valves following a scram to eliminate the discharge line water leg clearing loads resulting from multiple valve openings.

Testing of the safety/relief valves in accordance with ANSI/ASME OM-1-1981 each refueling outage ensures that any valve deterioration is detected. An as-found tolerance value of 3% for safety/relief valve setpoints is specified in ANSI/ASME OM-1-1981. Analyses have been performed with the valves assumed to open at 3% above their setpoint of 1109 psig. The 1375 psig Code limit is not exceeded in any case. When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system once per cycle provides assurance of bellows integrity.

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