NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

October 8, 2002

10 CFR Part 50
Section 50.90
Section 50.91

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

Attached is a request for a change to the Technical Specifications (TS), Appendix A of Operating License DPR-22, for the Monticello Nuclear Generating Plant. This request is submitted pursuant to and in accordance with the provisions of 10 CFR Part 50, Sections 50.90 and 50.91.

The purpose of this License Amendment Request is to revise the Drywell Leakage and Sump Monitoring Detection Section of the Technical Specification (TS) to clarify existing requirements, make wording improvements, revise existing limiting condition for operations (LCO) and surveillance requirements (SR), and add an additional TS LCO to the Monticello TS.

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.

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

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).

A001

On September 7, 2002, during routine Drywell Equipment Drain Sump (DEDS) pumping operation, it was discovered that #12 DEDS pump (P-20B) was not performing as designed (i.e. no discernable flow can be measured). Due to the location of the equipment (in the drywell), troubleshooting and research into the failure has been limited at this point. However, our investigations seem to indicate that the pump shaft has decoupled or sheared, or the pump has internal blockage. The pump cannot be fixed without a unit shutdown. Therefore, a complete analysis of this pump failure cannot be performed at this time.

The NMC is concerned that if the companion sump pump, #11 Drywell Equipment Drain Sump pump (P-20A) were to fail at this time, MNGP would be required to shut down for this condition. With both pumps inoperable in the DEDS, there is no ability 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 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.b) would be impossible. Finally, a manual calculation of leak rate (TS 3.6.D.5) could not be performed either. Therefore, TS 3.6.D.5.b would require a unit shutdown within 24 hours.

Based on discussions with NRC Project Manager, John Stang and given the current plant configuration, NMC requests the amendment be approved in a timely manner.

By approval of the proposed TS changes, MNGP would be required to monitor for only Unidentified Leakage and Total Leakage. In addition, the TS would change such that only one Sump Monitoring System would need to be operable to avoid a unit shutdown. This would give the flexibility needed to be able to monitor only one sump when the other sump is inoperable. However, in this condition, all leakage would be classified as Unidentified, unless it can be identified and quantified in accordance with the proposed Technical Specification definition for Identified Leakage (1.0.AC).

Nuclear Management Company, LLC expects the amendment to be effective immediately upon issuance. We request a period of 45 days to fully implement the change.

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

Jeffrey S. Forbes
Vice President
Monticello Nuclear Generating Plant

Subscribed to and sworn before me this 08 day of October, 2002.

KATHRYN I. KLEINE
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2005

Kathryn I Kleine
Notary

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

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

Evaluation of Proposed Changes to the Monticello Technical Specifications

Pursuant to 10 CFR 50.90 and 10 CFR 50.91, Nuclear Management Company, LLC (NMC) hereby proposes the following changes to Appendix A, of Facility Operating License DPR-22, Technical Specifications (TS) and Bases for Monticello Nuclear Generating Plant.

Background

A. Monticello TS include definitions for Identified Leakage and Unidentified Leakage. These definitions are being amended. In addition, a new definition for Total Leakage is being proposed. This wording is similar to the wording in NUREG-1433 (Reference 1). Change 1 provides the proposed changes to the TS section 1.0.

B. Changes to the LCO Statements of TS Section 3.6.D.1 through 3.6.D.5 are being proposed. These changes incorporate Total Leakage into the TS and eliminate Identified Leakage requirements. The change also modifies the shutdown requirements for inoperability. This wording is similar to the wording in NUREG-1433 (Reference 1). Change 2 of this license amendment request provides the proposed changes of TS 3.6.D.

C. Changes to the Action Statements of TS Section 4.6.D.1 and 4.6.D.2.b are being proposed. These changes modify the current language to eliminate the Identified Leakage surveillance and replace it with a Total Leakage surveillance requirement. This wording is similar to the wording in NUREG-1433 (Reference 1). Change 3 provides this proposed change.

MNGP does not view this application as a partial conversion to NUREG-1433, "Improved Standard Technical Specifications (ISTS)", but rather, as a means to incorporate best industry practices, while maintaining the custom features of the Monticello Technical Specifications. Development of the propose changes considered similar requirements contained in NUREG-1433 in order to be generally consistent with industry practices. The proposed changes to Appendix A of the Monticello Operating License are described below. Specific wording changes are shown in Exhibits B and C. The following provides a description of the changes referenced above, the reason for the change, and a safety evaluation for each of the changes.

A. Change 1 – Revision to Definitions for Identified Leakage, Unidentified Leakage and Addition of the Definition for Total Leakage (page 5)

Proposed Changes and Reasons for Changes

The changes to the TS definition for Identified Leakage and Unidentified Leakage are generally consistent with NUREG-1433 (Reference 1). The changes are made to be consistent with the application of the definitions in TS section 3.6.D/4.6.D.

The addition of the definition for Total Leakage is considered required for the application of the definitions in TS section 3.6.D/4.6.D.

Safety Evaluation

Making the changes to Identified and Unidentified Leakage is acceptable because removing the "reactor coolant" portion of these definitions is more conservative. Any and all liquid in the drywell sumps would be considered as leakage in accordance with the revised definitions. Use of the term "leakage detection systems" is being made to be consistent with the understanding of the wording to be used later in TS 3.6.D/4.6.D.

Adding the Definition for Total Leakage as the sum of Identified and Unidentified Leakage is also considered necessary to support the revisions to TS 3.6.D and 4.6.D as discussed below.

B. Change 2 – Changes to the LCO Statements of TS Section 3.6.D.1 through 3.6.D.5 (pages 126 and 126a)

Proposed Changes and Reasons for Changes

Change to the following LCO statements for TS Sections 3.6.D.1, 3.6.D.2, 3.6.D.3, 3.6.D.4 and 3.6.D.5 are proposed:

1) For TS 3.6.D.1.c, changed the leakage value from 20 gpm to 25 gpm, changed "Identified Leakage" to "Total Leakage". These changes were required to eliminate the need to separately measure Unidentified and Identified Leakage.

2) For TS 3.6.D.2, 3.6.D.3, 3.6.D.4 and 3.6.D.5.b, changed 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. Change made to be consistent with other similar shutdown requirements found elsewhere in the Monticello Technical Specifications.

3) For TS 3.6.D.1 and 3.6.D.2, removed "reactor coolant system" and replaced with "drywell". Removed "based on sump monitoring" from 3.6.D.1. Changes are made for consistency with leakage definitions.

4) For TS 3.6.D.3, added the phrase, "reduce leakage to within limits within 4 hours or..." Change made for consistency between 3.6.D.2 and 3.6.D.3.

5) For TS 3.6.D.3, added the phrase, "is not service sensitive type 304 or type 316 austenitic stainless steel". Change made to focus the investigation of leakages to piping that is highly susceptible to Intergranular Stress Corrosion Cracking (IGSCC).

6) For TS 3.6.D.4 removed the reference to corrective actions outlined in TS 3.6.D.2 and TS 3.6.D.3. These statements were made redundant based on changes 2, 4 and 5 above. The changes also enhance the readability and understanding of the TS.

7) For TS 3.6.D.5 replaced "leakage measurement instruments associated with each sump" with "sump monitoring system". Clarifies the requirement for an operable system and reduces the burden to have both sumps monitored when only one sump needs to be monitored if all leakage is classified as Unidentified Leakage.

The TS Bases have been revised, consistent with the changes described above.

Safety Evaluation

Change B.1 modifies the TS leakage from Identified Leakage to Total Leakage. The TS and TS Bases indicate that the safety significant concern with leakage in the drywell is pressure boundary leakage. This would appear as Unidentified Leakage. This statement is changed to sum the Unidentified Leakage rate (5 gpm) and the Identified Leakage rate (20 gpm) to create the category of Total Leakage (25 gpm). The change is acceptable because the total leakage limit is based on a reasonable minimum detectable amount of leakage and our previously approved values of Identified and Unidentified Leakage. The limit also accounts for leakage from known sources (Identified leakage). Violation of this LCO indicates an unexpected amount of leakage and, therefore, could indicate new or additional degradation in a Reactor Coolant Pressure Boundary (RCPB) component or system. The proposed change does not modify the currently approved leakage rates. Therefore, the change does not involve any change in the type or increase the amount of any effluent that may be released offsite. In addition, change does not involve any increase in individual or cumulative occupational radiation exposure.

Change B.2 modifies the shutdown language for TS 3.6.D.2, 3.6.D.3, 3.6.D.4 and 3.6.D.5.b to that which is consistent with other similar shutdown requirements found elsewhere in our current technical specifications. These changes clarify the Monticello shutdown requirements. These changes are acceptable because they are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.

Change B.3 is conservative in nature and is therefore acceptable. As discussed previously in Change 1 (Section A. above), the TS are being revised to not only include RCS leakage, but all drywell leakage to be consistent with the revised definitions. Further, drywell leakage is monitored by more than the sump monitoring system. The drywell continuous air monitor (CAM) is also a TS required means for measuring drywell leakage (TS 3.6.D.6). This change provides for a Technical Specification that is more consistent with the current Monticello Technical Specifications.

Changes B.4 and B.5 provide the plant time to investigate the source of Unidentified Leakage and, more importantly, reduce it. The current TS were unclear as to what the expectation was once a leakage source was identified, unless the user read further into TS 3.6.D.4. TS 3.6.D.3 only required identification of the increase in leakage. This change is made to enhance the readability of the TS. Further, the interest in investigation is 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 and type 316 austenitic stainless steel eliminates IGSCC as a cause of a leak. This significantly reduces concerns about crack instability and the rapid failure in the RCS boundary. As described in the TS bases, if it is determined that the increase in Unidentified Leakage is not from type 304 or 316 austenitic stainless steel sources and is quantified, then the leakage may be reclassified as Identified Leakage. The Unidentified and Identified Leakage rate limits will not be changed. This change is consistent with the shutdown requirements of proposed 3.6.D.2.

Change B.6 removed the phrase "when corrective actions outlined in 3.6.D.2 and 3.6.D.3 above are taken" from TS 3.6.D.4. This statement became redundant based on changes proposed to 3.6.D.2 and 3.6.D.3. With the proposed changes outlined in changes B.2, B.4 and B.5, each subsection has its own evaluation criteria and shutdown statement. Therefore, there is no need to reference sections 3.6.D.2 and 3.6.D.3 in terms of 3.6.D.4. The changes enhance the readability and understanding of the TS.

Change B.7 changes the requirement to have a sump monitoring system operable in each sump, to only require a sump monitoring system operable in only one of the sumps. Each sump contains 2 sump monitoring systems. The DEDS and DFDS are designed to not only collect leakage, but also to overflow into each other. If one of the drywell collection sumps or both of the sump monitoring systems for a given sump are inoperable, the opposite sump monitoring system is available to monitor leakage rates, once overflow to the opposite pump begins. This is because the overall ability of the leakage detection system has not been compromised in this configuration. Other components in the containment leakage detection system are not compromised based on this configuration. If this condition were to occur, all leakage measured would be treated as Unidentified Leakage, unless it can be identified and quantified in accordance with the proposed Technical Specification definition for Identified Leakage (1.0.AC). Historically, leakage from Identified sources is much greater than leakage from unidentified sources. Therefore, this is a conservative change and is considered acceptable.

C. Change 3 – Changes to the Surveillance Requirements of TS Section 4.6.D.1 and 4.6.D.2.b (pages 126 and 126a)

Proposed Changes and Reasons for Changes

This change requires a surveillance to demonstrate that the three limits for leakage in TS 3.6.D.1 (Unidentified Leakage, the Total Leakage and the Unidentified Leakage Rate Increase) have a corresponding surveillance requirement. This provides a timely verification that all the leakage rates are within their respective TS requirements. The change deletes the Identified Leakage requirement and replaces it with the Total Leakage requirement. This was deleted to be consistent with the changes in 3.6.D.1.c. In addition, the Unidentified Leakage Rate Increase was added to the TS to be consistent with verification of the rest of the Leakage requirements. The term "Sump monitoring system" replaces the more cumbersome term "Primary containment sump leakage measurement system". This term is used to define the instrumentation used in the sumps.

This change also adds a monthly functional test for the Sump Monitoring Systems. This functional test along with the sensor check and calibration will ensure that the Sump Monitoring Systems will operate with a high degree of reliability. A note is added to eliminate a conflict with Definition 1.0.E., "Instrument Functional Test".

The TS Bases have been revised, consistent with the changes described above.

Safety Evaluation

The Unidentified Leakage surveillance requirement has not changed. The safety significance of leakage from the RCPB varies widely depending on the source, rate and duration. Therefore, detection of the leakage from sources within primary containment is necessary. Methods for quickly separating Identified from Unidentified leakage are necessary to provide the operators with quantitative information to permit them to take corrective action in the event of a leak that is detrimental to the safety of the facility or the public. TS 3.6.D.5 provides actions to be taken if all the Sump Monitoring Systems become inoperable. Therefore, these changes are acceptable because they require measurements of leakage rates to be taken on a continuous basis and the increased channel functional testing ensures that the systems in place to detect leakage conditions are functioning properly. The note is required because a functional test at the sensor cannot be performed while at power (a drywell entry would be required). The note is consistent with similar notes elsewhere in the Monticello Technical Specifications (TS Table 4.2.1 note 5).

These changes do not involve equipment modifications or program changes and therefore do not adversely affect the public health and safety.

No Significant Hazards Consideration Determination

Nuclear Management Company, LLC (NMC), proposes to revise the Monticello Technical Specifications (TS) for Drywell Leakage and Sump Monitoring Detection system. These proposed changes restructure the Drywell leakage criteria by focusing on Unidentified Leakage and Total Leakage requirements. It adds a requirement for surveillance of Unidentified Leakage Rate Increase and focuses the operator to look at 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. The changes simply redefine the parameters for evaluation of leakage in the drywell. Changes in the time required to perform shutdown actions proposed are acceptable because they are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems. 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 be operable. As a result, 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. Changes in the time required to perform shutdown actions proposed are acceptable because they are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems. The evaluation criteria for drywell leakage have been refocused into the areas that are most susceptible to IGSCC.

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. Changes in the time required to perform shutdown actions proposed are acceptable because they are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems. 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.

<u>Environmental Assessment</u>

Based upon the above assessment of the proposed changes, the Nuclear Management Company, LLC has determined that:

1. The changes do not involve a significant hazards consideration.

2. The change does not involve a significant change in the type or significant increase in the amounts of any effluent that may be released offsite.

3. The change does 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 change is not required.

<u>References:</u>

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

Exhibit B

License Amendment Request for
Drywell Leakage and Sump Monitoring Detection Technical Specification Changes

Current Monticello Technical Specification Pages Marked Up
With Proposed Change

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

Pages

5
5a
126
126a
150
151
152

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

AD. Unidentified Leakage - ~~Unidentified leakage shall be all reactor coolant~~ All drywell leakage 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.

AK. Dose Equivalent I-131 - Dose Equivalent I-131 is the concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or in NRC Regulatory Guide 1.109, Rev 1, October, 1977.

AL. through AP. (Deleted)

AQ. Core Operating Limits Report The Core Operating Limits Report is the unit specific document that provides core operating limits for the current operating reload cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.7.A.7. Plant operation within these operating limits is addressed in individual specifications.

AR. Allowable Value - The Allowable Value is the limiting value of the sensed process variable at which the trip setpoint may be found during instrument surveillance.

3.0 LIMITING CONDITIONS FOR OPERATION

D. Coolant Leakage

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

   a. 5 gpm Unidentified Leakage,

   b. 2 gpm increase in Unidentified Leakage within any 24 hour period,

   c. 25~~0~~ gpm ~~Identified~~ Total Leakage and

   d. no pressure boundary leakage

2. With ~~reactor coolant system~~ drywell leakage greater than 3.6.D.1.a or 3.6.D.1.c 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.

3. With an increase in Unidentified Leakage in excess of the rate specified in 3.6.D.1.b, 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 within 4~~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.

4.0 SURVEILLANCE REQUIREMENTS

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, every 12 hours verify drywell Unidentified Leakage and Total Leakage and Unidentified Leakage Increase are within limits.~~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. The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

   a. Primary containment atmosphere particulate monitoring systems-performance 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. Sump Monitoring Systems - ~~Primary containment sump leakage measurement system~~ performan~~ce of~~ a sensor check once per 12 hours, a channel functional test* 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

4. If any Pressure Boundary Leakage exists, initiate an orderly shutdown of the reactor. 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.~~

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

   a. Perform manual leak rate measurements once per 12 hours and restore a Sump Monitoring System~~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.~~ Be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

6. 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:

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

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

Bases 3.6/4.6 (Continued):

D. Coolant 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.

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. 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. Systems connected to the reactor coolant systems boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

The sensitivity of the Sump Monitoring~~sump leakage detection s~~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.

If one of the Drywell Collection Sumps or both Sump Monitoring Systems for a given sump are inoperable, the opposite Sump Monitoring System will be used to monitor leakage rates. Each sump not only collects all leakage directed to the sump but also the overflow from the opposite sump. In this condition, all leakage measured by the Sump Monitoring System is assumed to be Unidentified Leakage unless quantified and identified in accordance with the TS definition for Identified Leakage (1.0.AC).

No Pressure Boundary Leakage is allowed, being indicative of material degradation. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the reactor coolant pressure boundary (RCPB). Leakage past seals and gaskets is not Pressure Boundary Leakage.

The 5 gpm of Unidentified Leakage is allowed as a reasonable minimum detectable amount that the containment air monitoring and drywell sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB.

The Total Leakage limit is based on a reasonable minimum detectable amount. The limit also accounts for leakage from known sources (Identified Leakage). Violation of this LCO indicates an unexpected amount of leakage and, therefore, could indicate new or additional degradation in an RCPB component or system.

An Unidentified Leakage increase of >2 gpm within the previous 24 hour period indicates a potential flaw in the RCPB and must be quickly evaluated to determine the source and extent of the leakage. 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. Although the increase does not necessarily violate the absolute Unidentified Leakage limit, certain susceptible components must be 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 within any 24 hour period" 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. Violation of this LCO could result in continued degradation of the RCPB.

If an Unidentified Leakage has been identified and quantified, it may be reclassified and considered as Identified Leakage; however, the Total Leakage limit would remain unchanged.

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.

FI. Deleted

Exhibit C

License Amendment Request for
Drywell Leakage and Sump Monitoring Detection Technical Specification Changes

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

5
5a
126
126a
150
151
152

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 pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

2) Leakage into the drywell atmosphere from sources which are specifically located and known not to be Pressure Boundary Leakage or which do not interfere with the operation of leakage detection systems.

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

AE. Total Leakage - Sum of 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.

AK. Dose Equivalent I-131 - Dose Equivalent I-131 is the concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or in NRC Regulatory Guide 1.109, Rev 1, October, 1977.

AL. through AP. (Deleted)

AQ. Core Operating Limits Report The Core Operating Limits Report is the unit specific document that provides core operating limits for the current operating reload cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.7.A.7. Plant operation within these operating limits is addressed in individual specifications.

AR. Allowable Value - The Allowable Value is the limiting value of the sensed process variable at which the trip setpoint may be found during instrument surveillance.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, drywell leakage shall be limited to:

   a. 5 gpm Unidentified Leakage,

   b. 2 gpm increase in Unidentified Leakage within any 24 hour period,

   c. 25 gpm Total Leakage, and

   d. no pressure boundary leakage

2. With drywell leakage greater than 3.6.D.1.a or 3.6.D.1.c above, reduce the leakage rate to within acceptable limits within four hours or initiate an orderly shutdown of the reactor. Be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

3. With an increase in Unidentified Leakage in excess of the rate specified in 3.6.D.1.b, 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 within 4 hours or initiate an orderly shutdown of the reactor. Be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, every 12 hours verify drywell Unidentified Leakage and Total Leakage and Unidentified Leakage Increase are within limits.

2. The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

   a. Primary containment atmosphere particulate monitoring systems-performance 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. Sump Monitoring Systems - perform a sensor check once per 12 hours, a channel functional test* 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

4. If any Pressure Boundary Leakage exists, initiate an orderly shutdown of the reactor. Be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

5. Any time irradiated fuel is in the reactor vessel and reactor water temperature is above 212°F at least one of the Sump Monitoring Systems shall be operable. If no Sump Monitoring Systems are operable, then:

   a. Perform manual leak rate measurements once per 12 hours and restore a Sump Monitoring System to operable status within 30 days.

   b. Otherwise, initiate an orderly shutdown of the reactor. Be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

6. 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:

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

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

Bases 3.6/4.6 (Continued):

D. Coolant 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.

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. 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. Systems connected to the reactor coolant systems boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

The sensitivity of the Sump Monitoring 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.

If one of the Drywell Collection Sumps or both Sump Monitoring Systems for a given sump are inoperable, the opposite Sump Monitoring System will be used to monitor leakage rates. Each sump not only collects all leakage directed to the sump but also the overflow from the opposite sump. In this condition, all leakage measured by the Sump Monitoring System is assumed to be Unidentified Leakage unless quantified and identified in accordance with the TS definition for Identified Leakage (1.0.AC).

No Pressure Boundary Leakage is allowed, being indicative of material degradation. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the reactor coolant pressure boundary (RCPB). Leakage past seals and gaskets is not Pressure Boundary Leakage.

The 5 gpm of Unidentified Leakage is allowed as a reasonable minimum detectable amount that the containment air monitoring and drywell sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB.

The Total Leakage limit is based on a reasonable minimum detectable amount. The limit also accounts for leakage from known sources (Identified Leakage). Violation of this LCO indicates an unexpected amount of leakage and, therefore, could indicate new or additional degradation in an RCPB component or system.

An Unidentified Leakage increase of >2 gpm within the previous 24 hour period indicates a potential flaw in the RCPB and must be quickly evaluated to determine the source and extent of the leakage. 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. Although the increase does not necessarily violate the absolute Unidentified Leakage limit, certain susceptible components must be 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 within any 24 hour period" 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. Violation of this LCO could result in continued degradation of the RCPB.

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Bases 3.6/4.6 (Continued):

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