NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

November 8, 2002

10 CFR Part 50,
Section 50.90

US Nuclear Regulatory Commission
Document Control Desk
Washington DC, 20555-0001

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

Response to NRC Questions Related To License Amendment Request
Regarding Drywell Leakage and Sump Monitoring System (TAC # MB6493)

Reference 1: Nuclear Management Company, LLC, License Amendment Request for Drywell Leakage and Sump Monitoring System Technical Specification Changes, dated October 8, 2002

Reference 1 proposed Technical Specifications changes to Appendix A of Operating License DPR-22, for the Monticello Nuclear Generating Plant. The purpose of the License Amendment Request was to revise the Monticello Technical Specifications (TS) to incorporate a change to the Drywell Leakage and Sump Monitoring System.

In a Conference Call between Darl Hood, NRC Senior Project Manager for Monticello, and Doug Neve, Licensing Manager for Nuclear Management Company, LLC (NMC) Monticello Nuclear Generating Plant, the need to provide additional information was discussed.

Exhibit A provides NMC's response to the NRC's questions regarding the previously submitted License Amendment Request. Exhibit B provides a marked-up revision to the requested changes to the Technical Specification pages provided in Reference 1. Exhibit C provides retyped Technical Specification pages that include these revisions.

Reference 1 proposed changes were evaluated by the NRC in their Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for Hearing, published in the *Federal Register* by letter dated October 17, 2002. The NRC determined that the NMC submittal dated October 8, 2002 did not involve any significant hazards consideration. The attached information does not impact that determination, therefore, the Determination of No Significant Hazards Consideration as noticed in the *Federal Register* by letter dated October 17, 2002, is also applicable to this submittal.

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Additionally, the Reference 1 proposed changes were evaluated and determined to meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9) and pursuant to 10 CFR 51.22(b) an Environmental Assessment was not required. The attached information does not impact that determination, therefore, the Environmental Assessment submitted by the original letter dated October 8, 2002, is also applicable to this submittal.

If you have any questions regarding this additional information related to our previously submitted License Amendment Request, please contact John Fields, Senior Licensing Engineer, at (763) 295-1663.

Jeffrey S. Forbes
Site Vice President
Monticello Nuclear Generating Plant

Subscribed to and sworn before me this 8th day of November, 2002

Notary

Lila A. Imholte
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2005

Attachments: Exhibit A – Response to NRC Questions Related to License Amendment Request Regarding Drywell Leakage and Sump Monitoring system
Exhibit B - Marked-Up Technical Specification Pages
Exhibit C - Revised Technical Specification Pages

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

Exhibit A

Response to NRC Questions Related to License Amendment Request,
Dated October 8, 2002, Regarding Drywell Leakage and
Sump Monitoring System Technical Specification (TS) Changes

NRC Question #1:

Monticello's SR 4.D.1 specifies a 12-hour surveillance interval for monitoring increases in unidentified leakage rate, while the STS specifies a surveillance interval of 8 hours. However, regardless of the surveillance interval, the licensee should be trending the results so that it will have some idea when it is about to exceed the limit and take action as required, even if this action may be required before the next 12-hour surveillance interval. The NRC staff requests that the licensee confirm that this is consistent with its understanding and intended operating procedures.

NMC Response:

Drywell floor and equipment drain sump levels are recorded on Level Recorder (LR-7409). Every 12 hours, operators are required to perform a sensor check on LR-7409 by observing a correct, normal, and expected trace for each pen. Also, every 12 hours the operators are required to perform a sensor check on the flow recorder (FR-2544). This flow recorder is verified to have correct and normal pen traces. The leakage rates for unidentified and identified leakage, and unidentified leakage rate increase, are also verified to be less than the Technical Specification (TS) limits every 12 hours utilizing the sump volume rate of change computer points. If the computer is unavailable, the leakage rates are determined from LR-7409. There are alarms on both the computer and on the control board panel C-04, for the rate of change that would alert the operators that leakage rates are approaching TS limits. Additionally, operators monitor sump fill rates via LR-7409.

Therefore, Monticello Nuclear Generating Plant (MNGP) operators would be aware of drywell leakage changes prior to TS limits being exceeded so that appropriate actions could be taken.

NRC Question #2:

The licensee's discussion for the proposed change to TS 3.D.5 does not adequately address the situations when either the drywell floor drain sump (DFDS) or drywell equipment drain sump (DEDS) monitoring system is inoperable, with its respective sump not yet at the point of overflowing and registering in the sump with the operable level monitoring system. During this period, the total leakage rate and/or the unidentified leakage rate are/is indeterminate. The NRC staff requests that the licensee address these situations.

NMC Response:

After further review we are proposing further revisions to TS 3.6.D.5 and providing further clarification in the corresponding TS Bases. The revised TS Bases further define the instrumentation used to meet the TS description of a sump monitoring system. See Exhibit B for the marked-up changes to the TS pages. See Exhibit C for the retyped TS pages.

Two leakage collection sumps are located inside primary containment. The drywell equipment drain sump collects identified leakage and the drywell floor drain sump collects unidentified leakage. Each sump contains a subsystem of the Sump Monitoring System, the Drywell Equipment Drain sump monitoring system and the Drywell Floor Drain Sump Monitoring System. Each subsystem consists of three channels of leak/leak rate detection instrumentation. These channels are the sump discharge flow integrator, sump level recorder and the sump fill rate computer point (rate of change). Because of the physical size of the sumps, it is possible to determine the required leakage limit (5 gpm or 25 gpm) and leakage rate limit (2 gpm/24 hours) are being met during the period of time it takes to actually overflow from one sump to the other.

There is no 12-hour period when the limits contained in TS 3.6.D.1 cannot be verified as being met. The worst case scenario, with regard to the length of time required to achieve sump volume overflow, would be the time required for a completely dry sump to fill and reach the overflow level. Using the TS limit of 2 gpm, the time to overflow from a dry sump condition is equal to 559 minutes (9.32 hours). This is a conservative time estimate considering that the sump pump configuration does not allow a sump to be pumped dry. In addition, this length of time is bounding for both the period of time between the high computer alarm and the overflow to the operable sump. The difference between the time that the high computer alarm is received and the overflow level is reached, using the TS limit of 2 gpm, is equal to 281 minutes. Thus, during the time that either sump is filling, it will be verified every 12 hours that the leakage is less than the TS limit by verifying that the one sump does not overflow into the other within a specified time limit. This methodology would also bound the higher TS limits of 5 gpm and 25 gpm.

For example, if the rate of change of unidentified leakage started at 0 gpm and steadily increased at a rate consistent with the 2 gpm rate of change limit (i.e., 083 gpm per hour over a 24-hour period) from a completely dry sump, the sump would overflow to the other sump in 21.2 hours, well within the 24 hour time limit of the TS. Again this would be equivalent to a leakage increase from 0 gpm to 2 gpm over a 24 hour period.

The overflow leakage rates and times, as discussed above, bound the overflow leakage from the Drywell Equipment Drain sump to the Drywell Floor Drain sump and vice versa.

Therefore, as explained above, the total leakage rate and/or the unidentified leakage rate and the unidentified leakage rate of change can be determined to be within TS limits when a sump monitoring subsystem is inoperable, but the associated sump has not overflowed into the sump with an operable sump monitoring subsystem.

Additionally, the primary containment atmosphere monitor continuously monitors the primary containment atmosphere and would detect a sudden increase in radioactivity that may be attributable to Reactor Coolant Pressure Boundary leakage. This monitor is not capable of quantifying leakage rates, but is sensitive enough to indicate increased leakage rates.

MNGP is also proposing a revision to TS 3.6.D.5.a to delete the requirement to "Perform manual leak rate measurements once per 12 hours and restore a measurement instrument to operable status within 30 days," and is replacing it with a requirement to restore a channel of the Sump Monitoring System to operable status within 24 hours. This change is acceptable because the Sump Monitoring System contains redundant instrumentation that is used to monitor drywell leakage. Current TS 3.6.D.5 allows only one measurement system per sump to be inoperable for a period of 30 days as long as manual leak rate measurements are performed once per 12 hours. The proposed TS 3.6.D.5 requires that one Sump Monitoring System channel be operable in one sump or the other. This recognizes that the sumps are interconnected such that one sump can overflow into the other and the ability to measure leakage parameters is restored once the spillover into the other sump occurs. Upon a complete loss of the Sump Monitoring System, the ability to monitor potential leakage from the Reactor Coolant Pressure Boundary using TS instrumentation would be lost. The 24 hours to restore a system to operable status is a reasonable time based on the capability of non-TS instrumentation available to determine leakage to satisfy the requirements of TS 4.6.D.1, operating experience and the importance of the Sump Monitoring System. If a Sump Monitoring System channel cannot be restored to operable status within 24 hours then an orderly shutdown of the reactor is required to place it in a condition in which the LCO is no longer applicable.

The deletion of the current Monticello TS 3.6.D.5.a is also acceptable because the proposed changes to the Surveillance Requirements (SR) of 4.6.D are being revised to verify that total leakage, unidentified leakage and unidentified leakage rate change in the drywell are maintained within TS limits, but will no longer require leakage measurement. Therefore, since the Surveillance Requirement no longer requires measurement of the leakage there would no longer be a requirement to perform a manual leak rate measurement.

As discussed above Monticello will continue to measure, monitor and trend Reactor Coolant System leakage as a reasonable and prudent operating practice to detect the unlikely event of a degrading pressure boundary.

NRC Question #3:

The licensee's proposed changes to required action completion times represent a mixture of custom features from the current Monticello TS and completion times from the STS. This mixture is not consistently applied to all TSs of relevance to reactor coolant system operational leakage. The proposed shutdown actions of TS 3.D.2, 3.D.3, 3.D.4, and 3.D.5 (to "[b]e in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours") are based upon the STS. However, no similar change is proposed if the drywell particulate radioactivity monitoring system should be inoperable (i.e., TS 3.D.6.b maintains the Monticello language "Otherwise, initiate an orderly shutdown of the reactor and reduce water temperature to less than 212°F within 24 hours"). The licensee should confirm that this inconsistency is intentional. The licensee should also explain/justify the proposed relaxation in the shutdown requirement.

NMC Response:

MNGP recognizes the inconsistency in shutdown times addressed by this question. Due to the current conditions of the facility, it was our intention not to address the radiation monitors at this time.

MNGP believes the proposed relaxation in the shutdown requirement of TS 3.6.D.2, 3.6.D.3, 3.6.D.4 and 3.6.D.5 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 would cause less stress in the piping and a corresponding reduction in the leakage rate during that period.

Therefore it is believed that the requested change in shutdown requirements (i.e., "12 hours to be in hot shutdown and 24 hours to be in cold shutdown") is reasonable.

Additional Revisions:

MNGP is also proposing the addition of a clarifying statement to SR 4.6.D.2.b to state that the proposed monthly channel functional test will be performed on the flow instruments only. This is acceptable because these flow instruments are the only instruments that can be functionally tested monthly. Level instrumentation, with the exception of the recorders are located inside the drywell, which makes them inaccessible during power operation. This level instrumentation is required to have a sensor check performed every 12 hours and a Channel Calibration performed once per operating cycle. This testing ensures that significant portions of the circuitry are operating properly and will detect significant failure of the instrument channel.

Additionally, changes to the TS Bases are also being made to provide greater detail regarding the instrumentation that makes up a Sump Monitoring System and how that instrumentation provides operators with the necessary data to monitor leakage inside the drywell.

Additional Marked-Up Monticello Technical Specification Pages

This Exhibit consists of additional marked-up Technical Specification pages that supercede the marked-up Technical Specification pages submitted in our original License Amendment Request dated October 8, 2002.

The following remove and insert instructions provide a list of those pages provided by our submittal dated October 8, 2002 that should be replaced with the following:

Pages 5 and 5a in our original submittal will remain the same.

Remove	Insert
126	126
126a	126a
150	150
151	151
152	152
-	152a
-	153

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 System - ~~Primary containment sump leakage measurement system~~ perform~~ance 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.

   * 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 channel of the Sump Monitoring System~~leakage measurement instruments associated with each sump~~ shall be operable. If no channels of the Sump Monitoring System~~leak rate measurement instruments associated with a sump~~ are operable, then:
   a. ~~Perform manual leak rate measurements once per 12 hours and r~~Restore a Sump Monitoring System channel~~measurement instrument~~ to operable status within ~~30 days~~24 hours.
   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.

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~~s~~ 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.

The primary means of quantifying leakage in the drywell is the Sump Monitoring System. The Sump Monitoring System monitors the leakage collected in the floor drain sump and the equipment drain sump. The Unidentified Leakage is collected in the floor drain sump and consists of leakage from control rod drives, valve flanges or packings, floor drains, the Closed Cooling Water System, and drywell air cooling unit condensate drains, and any leakage not collected in the drywell equipment drain sump. The Identified Leakage is collected in the equipment drain sump and consists of leakage from various expected leakage sources.

Both the drywell floor drain sump and the drywell equipment drain sump have level transmitters that supply level indications in the control room. Both of these readings are also recorded.

These level transmitters also feed calculated computer points which monitor the sump fill rate by measuring the interval between level changes. An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached.

Bases 3.6/4.6 (Continued):

The flow transmitters in the discharge lines of the drywell floor drain sump pumps and drywell equipment drain sump pumps provide flow indications on a recorder in the control room and to the flow integrators Controls for the sump pumps are located in the control room.

The equipment drain sump subsystem instrumentation consists of one channel of equipment drain sump pump discharge flow integrator, sump level recorder, and the fill rate computer point (rate of change). The floor drain sump subsystem instrumentation likewise consists of one channel of floor drain sump pump discharge flow integrator, sump level recorder, and the fill rate computer point (rate of change). The Sump Monitoring System is operable when any one of these six channels is operable.

The Drywell Floor Drain Sump Subsystem is required to quantify the Unidentified Leakage from the reactor coolant system (RCS). Thus, for the subsystem to be considered operable, either the flow monitoring channel or the sump level monitoring channels of the system must be operable. Some common failures are direct pump failure, loss of power to the circuit, failure of the sump level transmitter or inability to open one of the sump isolation valves that is interlocked with the sump pump. Any failure associated with the pump subsystem should be evaluated for its impact on the ability of the associated equipment to measure leakage. The loss of flow through the flow element prevents the flow integrator from performing its intended function of measuring leakage and it should be considered inoperable.

An alternate to Drywell Floor Drain Sump Subsystem is the Drywell Equipment Drain Sump Subsystem. Because of the known capacity of the sumps, it is possible to verify the leakage below the required limits (5 or 25 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 subsystem can be used to quantify leakage. However, the alarm settings for the equipment drain sump instruments must be reset to annunciate when the lower limit Unidentified Leakage in reached. In this condition, all leakage measured by the Drywell Equipment Drain Sump Subsystem is assumed to be Unidentified Leakage unless it can be identified and quantified as Identified Leakage. 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 since the pre-existing alarm set point is conservative with respect to identified leakage.

The total loss of the Sump Monitoring System results from the loss of all sump level and flow indicators (either directly or indirectly). The Sump Monitoring System remains operable when any one of the six channels is operable.

Bases 3.6/4.6 (Continued):

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. 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 Sump Monitoring System equipment can detect within a reasonable time period.

The Total Leakage limit is based on a reasonable minimum detectable amount. The limit also accounts for leakage from known sources (Identified Leakage).

An Unidentified Leakage increase of >2 gpm within the previous 24 hour period indicates a potential flaw in the RCPB and must be promptly 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 rate of increase limit provides an early warning of such deterioration. Although the increase does not necessarily exceed the 5 gpm 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 that it is not the source of the increased leakage. This type of piping is very susceptible to IGSCC.

Once 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. *] moved from page 150*

[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. *] moved from page 151*

Bases 3.6/4.6 (Continued):

[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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G. Jet Pumps

By monitoring jet pump performance on a prescribed schedule, significant degradation in performance that would precede jet pump failure can be detected. An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it may present a hazard in the event of a large break accident by reducing the capability of reflooding the core; thus, the requirement for shutdown of the reactor with an inoperable jet pump.

The jet pump performance monitoring procedures are comprised of the following tests:

1. Core Flow versus Square Root of Core Plate Differential Pressure: change in core resistance is the main contributor to recirculation system performance changes. If core resistance increases, it requires more energy (pump speed) to produce rated core flow. If resistance decreases, less speed is needed.

2. Recirculation Pump Flow/Speed Ratio: the pump operating characteristic is determined by the flow resistance from the loop suction through the jet pump nozzle. Since this resistance is essentially independent of core power, the flow is linearly proportional to pump speed, making their ratio a constant (flow/RPM is constant). A decrease in the ratio indicates a plug, flow restriction, or loss in pump hydraulic performance. An increase indicates a leak or new flow path between the recirculation pump discharge and jet pump nozzle.

3. Jet Pump Loop Flow/Recirculation Pump Speed Ratio: this relationship is an indication of overall system performance.

4. Jet Pump Differential Pressure Relationships: if a potential problem is indicated, the individual jet pump differential pressures are used to determine if a problem exists since this is the most sensitive indicator of significant jet pump performance degradation.

The data base used to determine the normal operating range for (2) and (3) above is verified during the startup following each refueling outage. Surveillance tests are performed as soon as practical after reaching a pump speed of 60%.

Additional Revised Monticello Technical Specification Pages

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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 System - 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

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 channel of the Sump Monitoring System shall be operable. If no channels of the Sump Monitoring System are operable, then:

   a. Restore a Sump Monitoring System channel to operable status within 24 hours.

   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. 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 System 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.

The primary means of quantifying leakage in the drywell is the Sump Monitoring System. The Sump Monitoring System monitors the leakage collected in the floor drain sump and the equipment drain sump. The Unidentified Leakage is collected in the floor drain sump and consists of leakage from control rod drives, valve flanges or packings, floor drains, the Closed Cooling Water System, and drywell air cooling unit condensate drains, and any leakage not collected in the drywell equipment drain sump. The Identified Leakage is collected in the equipment drain sump and consists of leakage from various expected leakage sources.

Both the drywell floor drain sump and the drywell equipment drain sump have level transmitters that supply level indications in the control room. Both of these readings are also recorded.

These level transmitters also feed calculated computer points which monitor the sump fill rate by measuring the interval between level changes. An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached.

The flow transmitters in the discharge lines of the drywell floor drain sump pumps and drywell equipment drain sump pumps provide flow indications on a recorder in the control room and to the flow integrators. Controls for the sump pumps are located in the control room.

The equipment drain sump subsystem instrumentation consists of one channel of equipment drain sump pump discharge flow integrator, sump level recorder, and the fill rate computer point (rate of change). The floor drain sump subsystem instrumentation likewise consists of one channel of floor drain sump pump discharge flow integrator, sump level recorder, and the fill rate computer point (rate of change). The Sump Monitoring System is operable when any one of these six channels is operable.

The Drywell Floor Drain Sump Subsystem is required to quantify the Unidentified Leakage from the reactor coolant system (RCS). Thus, for the subsystem to be considered operable, either the flow monitoring channel or the sump level monitoring channels of the system must be operable. Some common failures are direct pump failure, loss of power to the circuit, failure of the sump level transmitter or inability to open one of the sump isolation valves that is interlocked with the sump pump. Any failure associated with the pump subsystem should be evaluated for its impact on the ability of the associated equipment to measure leakage. The loss of flow through the flow element prevents the flow integrator from performing its intended function of measuring leakage and it should be considered inoperable.

An alternate to Drywell Floor Drain Sump Subsystem is the Drywell Equipment Drain Sump Subsystem. Because of the known capacity of the sumps, it is possible to verify the leakage below the required limits (5 or 25 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 subsystem can be used to quantify leakage. However, the alarm settings for the equipment drain sump instruments must be reset to annunciate when the lower limit Unidentified Leakage in reached. In this condition, all leakage measured by the Drywell Equipment Drain Sump Subsystem is assumed to be Unidentified Leakage unless it can be identified and quantified as Identified Leakage. 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 since the pre-existing alarm set point is conservative with respect to identified leakage.

The total loss of the Sump Monitoring System results from the loss of all sump level and flow indicators (either directly or indirectly). The Sump Monitoring System remains operable when any one of the six channels is operable.

Bases 3.6/4.6 (Continued):

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. 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 Sump Monitoring System equipment can detect within a reasonable time period.

The Total Leakage limit is based on a reasonable minimum detectable amount. The limit also accounts for leakage from known sources (Identified Leakage).

An Unidentified Leakage increase of >2 gpm within the previous 24 hour period indicates a potential flaw in the RCPB and must be promptly 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 rate of increase limit provides an early warning of such deterioration. Although the increase does not necessarily exceed the 5 gpm 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 that it is not the source of the increased leakage. This type of piping is very susceptible to IGSCC.

Once 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.

Bases 3.6/4.6 (Continued):

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.

F. Deleted

G. Jet Pumps

By monitoring jet pump performance on a prescribed schedule, significant degradation in performance that would precede jet pump failure can be detected. An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it may present a hazard in the event of a large break accident by reducing the capability of reflooding the core; thus, the requirement for shutdown of the reactor with an inoperable jet pump.

The jet pump performance monitoring procedures are comprised of the following tests:

1. Core Flow versus Square Root of Core Plate Differential Pressure: change in core resistance is the main contributor to recirculation system performance changes. If core resistance increases, it requires more energy (pump speed) to produce rated core flow. If resistance decreases, less speed is needed.

2. Recirculation Pump Flow/Speed Ratio: the pump operating characteristic is determined by the flow resistance from the loop suction through the jet pump nozzle. Since this resistance is essentially independent of core power, the flow is linearly proportional to pump speed, making their ratio a constant (flow/RPM is constant). A decrease in the ratio indicates a plug, flow restriction, or loss in pump hydraulic performance. An increase indicates a leak or new flow path between the recirculation pump discharge and jet pump nozzle.

3. Jet Pump Loop Flow/Recirculation Pump Speed Ratio: this relationship is an indication of overall system performance.

4. Jet Pump Differential Pressure Relationships: if a potential problem is indicated, the individual jet pump differential pressures are used to determine if a problem exists since this is the most sensitive indicator of significant jet pump performance degradation.

The data base used to determine the normal operating range for (2) and (3) above is verified during the startup following each refueling outage. Surveillance tests are performed as soon as practical after reaching a pump speed of 60%.