ENCLOSURE 5

MONTICELLO NUCLEAR GENERATING PLANT LICENSE AMENDMENT REQUEST TO SUPPORT 24-MONTH FUEL CYCLES

BASES FOR CHANGE REQUEST FOR TECHNICAL SPECIFICATIONS AND SURVEILLANCE REQUIREMENTS INTERVAL EXTENSIONS

The purpose of this submittal is to request an amendment to Appendix A of Operating License DPR-22 for the Monticello Nuclear Generating Plant (MNGP). This license amendment request proposes revisions to the Monticello Technical Specification (TS) and Surveillance Requirements (SR) intervals, as identified, to support the implementation of a 24-month fuel cycle. A description of, and the justification for, each proposed TS change is provided below:

TS 1.0.U Definition – Refueling Operation and Refueling Outage

This definition is proposed to be changed based on the guidance provided in NRC Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle." The new definition will be for Refueling Operation and Refueling Interval, where the definition for Refueling Operation remains as it is currently and the definition for Refueling Interval is revised to meet the guidance provided in GL 91-04.

The revised TS Definition will read as follows:

U. *Refueling Operation and Refueling Interval – Refueling Operation is any operation when water temperature is less than 212°F and movement of fuel or core components is in progress. Refueling Interval is a designated frequency for performing surveillances of once per 24 months.*

This proposed change to the definitions section of the TS supports the request for changes in TS Surveillance Intervals to accommodate a 24-month fuel cycle. The bounding time interval for these surveillances will be 30 months under the provisions of Monticello TS 4.0.B that allows a surveillance to be extended by 25 percent of the specified interval.

GL 91-04 also allowed the omission of the TS qualification that an 18-month surveillance be performed "during shutdown" when specifying the surveillance interval as "…at least once per Refueling Interval," because the added restriction of performing certain surveillances during shutdown may be misinterpreted. This restriction ensures that surveillances would only be performed when it is consistent with safe plant operation. However, this consideration is valid for other

surveillances that are performed during plant operation, plant startup, or shutdown, but is not addressed by restricting the conduct of these surveillances.

The NRC Staff concluded, in GL 91-04 that the TS need not restrict surveillances as only being performed during shutdown. Nevertheless, safety dictates that when refueling interval surveillances are performed during power operation proper regard shall be given for their effect on the safe operation of the plant. If the performance of refueling interval surveillances during plant operation would adversely affect safety, the surveillances should be postponed until the unit is shutdown for refueling or is in a condition or mode that is consistent with the safe conduct of the surveillances.

TS Bases 4.0.B - Surveillance Requirements

This proposed change to the TS Bases is based on the guidance provided in NRC Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," and is being provided for information only. The new TS Bases 4.0.B will be revised to meet the guidance provided in GL 91-04.

The revised TS Bases will read as follows:

B. *A tolerance for performing surveillance activities beyond the nominal interval is provided to allow operational flexibility because of scheduling and performance considerations. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are specified to be performed at least once each Refueling Interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed once each Refueling Interval. Likewise, it is not the intent that Refueling Interval surveillances be performed during power operation unless it is consistent with safe plant operation. Each surveillance test is completed within 25% of each scheduled date. Scheduled dates are based on dividing each calendar year into four 13-week "surveillance" quarters consisting of 3 4-week "surveillance" months and one "catch-up" week. This method of scheduling permits certain tests to always be scheduled on certain days of the week.*

The changes to the Monticello TS Bases 4.0.B, which are controlled by MNGP TS 6.8.K, "Technical Specification Bases Contol Program," as shown above incorporate changes provided by the NRC in GL 87-09, "Sections 3.0 and 4.0 of Standard Technical Specifications on the Applicability of Limiting Conditions for Operation and Surveillance Requirements," and GL 89-14, "Line-Item Improvements in Technical Specifications – Removal of the 3.25 Limit on Extending Surveillance Intervals." The TS changes to accommodate a longer fuel cycle will alter the basis for Monticello TS 4.0.B, therefore, the above changes are

provided for information only to be consistent with these TS changes and particularly with respect to the safe conduct of refueling interval surveillances. The draft revision to Monticello TS Bases Section 4.0.B reflects the guidance provided in GL 91-04.

TS 3.1\4.1 Reactor Protection System (RPS)

The MonticelloTS Bases states that the reactor protection system initiates a reactor scram to preserve the integrity of the fuel cladding, preserve the integrity of the primary system barrier and minimize the energy which must be absorbed, and prevent criticality following a loss-of-coolant-accident (LOCA). The instrumentation in this section will be functionally tested and calibrated at regularly scheduled intervals. Specific surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P, "Technical Specification Improvement Analysis for BWR Reactor Protection System," as approved by the NRC and documented in the Safety Evaluation (SE) dated July 15, 1987 (Reference 2). TS 4.1.A requires that instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.

The following SRs for the RPS functions were evaluated relative to extending their respective testing intervals. These SRs ensure the RPS will function as designed during an analyzed event. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are discussed below:

Table 3.1.1 Reactor Protection System (RPS)
Function 1 Reactor Mode Switch in Shutdown

The Monticello Updated Safety Analysis Report (USAR) Section 7.6.1, states that the Reactor Mode Switch provides appropriate protective functions for the condition in which the reactor is to be operated. The reactor is to be shutdown with all control rods inserted when the mode switch is in Shutdown. To enforce the condition defined for the Shutdown position, placing the mode switch in the Shutdown position initiates a reactor scram. This scram is not required to protect the fuel or primary system process barrier, and it bears no relationship to minimizing the release of radioactive material from any barrier. The scram signal is removed after a short time delay, permitting a scram reset, which restores the normal valve lineup in the control rod drive hydraulic system. This Function was not specifically credited in the safety analysis, but is retained for the overall redundancy and diversity of the RPS as required by the NRC-approved licensing basis. There is no TS Trip Setting for this Function, since the channels are mechanically actuated based solely on reactor mode switch position.

SR Table 4.1.1 Perform FUNCTIONAL TEST

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The SR ensures the channel will perform the intended function. No instrumentation is associated with this function; therefore, drift has no effect when increasing the surveillance interval. Extending the surveillance interval for this Functional Test is acceptable because major portions of the circuits required to shut down the reactor are verified by more frequent Functional Tests of other RPS components per Functional Test of Manual Scram (Weekly).

A multi-position key-lock mode switch is provided which selects the necessary scram functions for various plant operating modes. In addition to selecting scram functions from the proper sensors the mode switch also interlocks such functions as control rod blocks and refueling equipment restrictions, which are not considered here as part of the Reactor Protection System. The switch itself is designed to provide separation between trip systems.

Based on the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS Table 3.1.1 Reactor Protection System (RPS)
Function 7 Reactor Vessel Water Level – Low

The Reactor Pressure Vessel (RPV) low water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage can result. Therefore, the low reactor water level instrumentation is set to trip when water level is ≥ 7” on the instrument. This corresponds to a lower water level inside the shroud at 100% power due to the pressure drop across the dryer/separator. This has been accounted for in the affected safety analyses. The reactor scram reduces the amount of energy required to be absorbed and, along with the Emergency Core Cooling System (ECCS), ensures the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Pressure Vessel low water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the RPV. This Function is required in Refuel, Startup and Run modes where energy exists in the Reactor Coolant System resulting in the limiting

events analyzed. The ECCS initiations at low-low reactor vessel water level provide sufficient protection for level transients in all other operating conditions.

SR Table 4.1.2 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months including the 25% grace period. An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of Electric Power Research Institute (EPRI) Technical Report (TR)-103335. The drift for the extended surveillance interval was appropriately considered in the development of the associated plant setpoints.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS Table 3.1.1 Reactor Protection System (RPS)
Function 10 Main Steamline Isolation Valve – Closure

A Main Steamline Isolation Valve (MSIV) closure results in loss of the main turbine and the condenser as a heat sink for the nuclear steam supply system, and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steamline Isolation Valve – Closure signal before the MSIVs are completely closed, in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. The main steamline isolation valve closure scram is set to actuate when the isolation valves are ≤ 10% from full open. This scram anticipates the pressure and flux transient, which would occur when the valves close. Initiating a scram at this setting causes the resultant transient to be insignificant. Specific analyses have generated limits that allow this scram to be bypassed below 600 psig.

SR Table 4.1.2 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. Limit switches perform this Function. Limit switches are mechanical devices that require mechanical setting only; drift is not applicable to these devices therefore an increase in this surveillance interval to accommodate a 24-month fuel cycle does not affect these limit switches relative to

drift. Proper operation of the Trip Channel and Alarm is verified quarterly by the performance of the Functional Tests.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS Table 3.1.1 Reactor Protection System (RPS)
Function 12 Turbine Stop Valve – Closure

The Monticello TS Bases states that the turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of #10% of valve closure from full open, the resultant increase in surface heat flux is limited such that Minimum Critical Power Ratio (MCPR) remains above the Safety Limit even during the worst case transient that assumes the turbine bypass is closed. Specific analyses have generated specific limits that allow this scram to be bypassed below 45% Rated Thermal Power (RTP). To ensure the availability of this scram above 45% RTP, this scram is only bypassed below 30% RTP as indicated by turbine first stage pressure. This takes into account the possibility of 14% power being passed directly to the condenser through the bypass valves.

SR Table 4.1.2 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months including the 25% grace period. Local limit switches perform this Function. Limit switches are mechanical devices that require mechanical adjustment only; drift is not applicable to these devices therefore an increase in this surveillance interval to accommodate a 24-month fuel cycle does not affect these limit switches relative to drift. Proper operation of the limit switch is verified by the quarterly performance of a Functional Test.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 3.1\4.1 RPS Power Monitoring System

Two motor generator (MG) sets provide AC power for operation of the Reactor Protection System. These MG sets are powered from 480 VAC buses and are used to supply power to the scram logic channels as well as neutron and radiation monitoring systems. These systems are designed to provide a continuous output of 120 VAC power that is free of transients and is extremely reliable. Switching transients and momentary losses of input power will not cause substantial changes in output voltage or frequency.

The normal power supply consists of two MG sets, each consisting of a three-phase induction motor driving a 120 VAC single-phase generator with flywheel. The flywheel provides energy to maintain generator output during momentary system faults or transients, which do not otherwise impair reactor operation. An alternate power source is provided to permit servicing of either MG set. Manual circuit breakers with a mechanical interlock prevent paralleling a motor generator set and the alternate source while transferring the load between them.

Electrical Protection Assemblies monitor the electric power in each of the three sources of power (RPS MG sets A and B, and the alternate source) to the RPS. Each assembly consists of two identical and redundant packages. Each package includes a circuit breaker and a monitoring module. If either module detects abnormal electric power, the respective circuit breaker will trip and disconnect the RPS from the abnormal power source. Each monitoring module will trip its associated breaker on overvoltage, undervoltage or under frequency. With the protective packages installed, abnormal output type failures (random or seismically caused) in either of the two RPS MG sets (or the alternate supply) results in a trip of either one or both of the two Class 1E protective packages.

The following SRs for the RPS power monitoring system were evaluated relative to extending their respective testing intervals. These SRs ensure the RPS will function as designed during an analyzed event. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are discussed below:

SR 4.1.C.2 At least once each Operating Cycle an instrument Calibration of each RPS power monitoring channel shall be performed to verify over-voltage, under-voltage, and under-frequency setpoints.

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that both channels of the power monitoring system for the MG set or alternate source supplying each RPS bus is operable. The drift associated with the timers was determined utilizing the Monticello-specific methodology, based upon the recomendations of EPRI TR-103335. The drift for the extended surveillance interval was appropriately considered in the development of the timer setpoints. An Instrument Functional Test and Calibration of each RPS power-monitoring channel (except for the timer function) is performed at least once every six months, which assures that the RPS power monitoring system is operable during the cycle.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 3.2.A Primary Containment Isolation Functions

The objective of Primary Containment Isolation System (PCIS) is to provide protection against the onset and consequences of accidents involving the gross release of radioactive materials from the Primary Containment. This protection is the automatic isolation of appropriate piping that penetrates the primary containment whenever certain monitored variables exceed their pre-selected operational limit. To accomplish this objective the containment isolation system was designed to prevent the release of radioactive materials in excess of the limits of 10 CFR 100 as a result of the design basis accidents, to function safely following any single component malfunction and to function independently of other plant controls and instrumentation.

The following SRs for the PCIS isolation functions were evaluated relative to extending their respective testing intervals. These SRs ensure the PCIS will function as designed during an analyzed event. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are discussed below:

<u>TS Table 3.2.1 Primary Containment Isolation Functions</u>
Function 1.a Main Steam and Recirc Sample Line –
Low-Low Reactor Water Level

The RPV low-low water level may indicate that an Anticipated Transients Without Scram (ATWS) event has occurred. Accordingly, an ATWS system trip is initiated when the level is ≥ -48 inches. To prevent the ATWS trip on low-low water level from affecting the ECCS system performance, a time delay is provided for this trip. The design basis for the Primary Containment Isolation System is to provide protection against the onset and consequences of accidents involving gross release of radioactive materials from the Primary Containment.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this transmitter SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The associated trip units are functionally tested and calibrated once every 3 months and a sensor check is performed every 12 hours.

An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval was appropriately considered in the development of the associated plant setpoints.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

In addition, this SR in TS Table 4.2.1 “MAIN STEAM LINE (GROUP 1) ISOLATION,” item 4, has been reworded to change the current TS language from “Every Operating Cycle” to “Once/Operating Cycle.” This is an administrative change required to maintain consistency with other Monticello SRs in TS Table 4.2.1.

<u>TS Table 3.2.1 Primary Containment Isolation Functions</u>
Function 1.c Main Steam and Recirc Sample Line –
High Temperature in Main Steam Line Tunnel

High temperature in the vicinity of the main steam lines is detected by 16 bimetallic temperature switches located along the main steam lines in the steam tunnel

between the drywell wall and the turbine building. The detectors are positioned so that they are sensitive to air temperature and not the radiated heat from hot equipment. A temperature sensor is located near each main steam line for remote temperature readout and alarm. The temperature sensors activate an alarm at high temperature to give the alarm condition. The main steam line space temperature detection instrumentation is designed to have the capability of detecting a minimum leak of 5 to 10 gpm. A revision to the Trip Setting is discussed below.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period.

An evaluation of the surveillance interval extension for the bimetallic temperature switches was performed, based upon the approach described in GL 91-04. The drift associated with the temperature switches was determined, applying Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval relative to Allowable Value and the associated plant Trip Setpoints was considered, as a result, the Trip Setting value will change from " ≤ 200°F" to " ≤ 209°F. " Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR Table 4.2.1 Perform FUNCTIONAL TEST

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. Extending the surveillance test interval for the Functional Test is acceptable, because the network, including the actuating logic, is designed to be single failure proof and, therefore, is highly reliable. Furthermore, as stated in Reference 1:

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the

probability of a relay or contact failure is small relative to the probability of mechanical failure, increasing the Logic System Functional Test interval represents no significant change in the overall safety system unavailability."

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

<u>TS Table 3.2.1 Primary Containment Isolation Functions</u>
Function 2.a RHR System, Head Cooling, Drywell, Sump, TIP – *Low Reactor Water Level*

The RPV low water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage can result. Therefore, the low reactor water level instrumentation is set to trip when water level is ≥ 7" on the instrument. This corresponds to a lower water level inside the shroud at 100% power due to the pressure drop across the dryer/separator. This has been accounted for in the affected safety analyses.

Reactor Pressure Vessel low water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the RPV.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months including the 25% grace period. An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

<u>TS Table 3.2.1 Primary Containment Isolation Functions</u>
Function 3.b Reactor Water Cleanup (RWCU) System – Low-Low Reactor Water Level

The RPV low-low water level may indicate that an ATWS event has occurred. Accordingly, an ATWS system trip is initiated when the level is ≥ -48 inches. To prevent the ATWS trip on low-low water level from affecting the ECCS system performance a time delay is provided for this trip. The design basis for the Primary Containment Isolation System is to provide protection against the onset and consequences of accidents involving gross release of radioactive materials from the Primary Containment.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The transmitters provide input to the associated trip units which are functionally tested and calibrated once every 3 months and have a sensor check performed every 12 hours.

An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Extending the surveillance test interval for the Channel Calibration for the transmitters is acceptable, because the Function is verified to be operating properly by the more frequent performance of Sensor Checks every 12 hours, Channel Functional Test every 3 months, and Channel Calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm the major portions of the monitoring loop are maintained within allowances.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS Table 3.2.1 Primary Containment Isolation Functions
Function 3.c Reactor Water Cleanup (RWCU) System – High RWCU Room Temperature

High temperature in the RWCU room could indicate a break in a RWCU line. The automatic closure of the RWCU line prevents the excessive loss of reactor coolant and the release of significant amounts of radioactive material from the nuclear system process barrier. The high temperature isolation setting was selected far enough above anticipated normal RWCU operational levels to avoid spurious operation but low enough to provide timely detection of RWCU line breaks.

Four temperature detectors are used to sense temperature and provide input to four channels. The outputs of two of four channels are arranged in a one-out-of-two-once logic to form a trip system. Therefore, the four channels comprise two separate trip systems. Coincident tripping of both trip systems is required for isolation initiation. Temperature in the RWCU room is detected by four resistance temperature detectors (RTDs). The RTDs are located throughout the RWCU room to detect breaks.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. Calibration of instrument channels with RTDs may consist of a qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.

The circuitry of the associated trip units are functionally tested and calibrated and the setpoints are verified once every 3 months and sensor checks are performed every 12 hours.

An evaluation of the surveillance interval extension for the RTD input circuitry of the trip units was performed, based upon the approach described in GL 91-04. These trip units are part of a recently installed trip circuit. An insufficient amount of calibration data is available to perform a statistical drift analysis based upon the recommendations of EPRI TR-103335. Evaluation of the drift data as determined from the available as-found and as-left calibration information shows that the vendor drift values are conservative and appropriate for this application. The new instrument drift-monitoring program will assure the drift value used in the setpoint determination for this application is conservative relative to actual detector performance. The drift for the extended surveillance interval was appropriately considered in the development of the associated plant setpoints.

Extending the surveillance test interval for the Channel Calibration for the RTD input circuitry of the trip units is acceptable, because the Function is verified to be operating properly by the more frequent performance of Sensor Checks every 12

hours, Channel Functional Test every 3 months, and Channel Calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm the major portions of the monitoring loop are maintained within allowances.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS Table 3.2.1 Primary Containment Isolation Functions
Function 3.d Reactor Cleanup System – High RWCU System Flow

RWCU high flow could indicate a break in a RWCU line. The automatic closure of the RWCU isolation valves prevents the excessive loss of reactor coolant and the release of significant amounts of radioactive materials from the nuclear system process barrier. The RWCU system is isolated upon detection of RWCU high flow.

Spurious RWCU system isolation during momentary system flow disturbances is avoided by a time delay provided in the isolation logic. A trip on negative differential pressure is also provided to protect against an instrument line failure disabling the high flow protection.

Four flow transmitters are used to sense flow and provide input to four channels. The outputs of two of four channels are arranged in a one-out-of-two-once logic to form a trip system. The four channels comprise two separate trip systems. Coincident tripping of both trip systems is required for isolation initiation.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. These flow transmitters provide an input to the associated trip units which are functionally tested and calibrated once every 3 months and have a sensor check performed every 12 hours.

An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. These transmitters are part of a recently installed trip circuit and a limited number of calibrations have been performed for this application. However, the same transmitter model is used in another application at MNGP. Statistical testing of the transmitter as-found and as-left data shows that grouping of the transmitter data to determine a drift value is acceptable. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-

103335. The new instrument drift-monitoring program will ensure the drift value used in the setpoint determination for this application is conservative relative to actual switch performance. The drift for the extended surveillance interval was appropriately considered in the development of the associated plant setpoints.

Extending the surveillance test interval for the Channel Calibration for the transmitters is acceptable, because the Function is verified to be operating properly by the more frequent performance of Sensor Checks every 12 hours, Channel Functional Test every 3 months, and Channel Calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm the major portions of the monitoring loop are maintained within allowances.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 3.2.B Emergency Core Cooling Subsystems Actuation

The ECCS in conjunction with the primary and secondary containment is designed to limit the release of radioactive materials to the environment following a loss of coolant accident (LOCA). The ECCS uses two independent methods (flooding and spraying) to cool the core during a LOCA. The ECCS network consists of the High Pressure Coolant Injection (HPCI) System, the Core Spray (CS) System, the low-pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System, and the Automatic Depressurization System (ADS). Each of the systems is designed to cover a specific range of accident conditions and collectively provide a redundancy of function to avoid undetected common failure mechanisms. An integrated system performance evaluation to determine the ECCS capability has been made and is discussed in the Monticello USAR.

The following SRs for the ECCS actuation functions were evaluated relative to extending their respective testing intervals. These SRs ensure the ECCS will function as designed during an analyzed event. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are discussed below:

<u>TS Table 3.2.2 Emergency Core Cooling Systems</u>

Function A.1.a Core Spray and Low Pressure Coolant Injection (LPCI) – Low-Low Reactor Water Level

The RPV low-low water level may indicate that an ATWS event has occurred. Accordingly, an ATWS system trip is initiated when the level is ≥ -48 inches. To prevent the ATWS trip on low-low water level from affecting the ECCS system performance, a time delay is provided for this trip.

The design basis for the Core Spray (CS) System is to restore and maintain the coolant in the reactor vessel in combination with other emergency core cooling systems such that the core is adequately cooled to preclude fuel damage. Low Pressure Coolant Injection operation involves restoring the water level in the reactor vessel to a sufficient height for adequate cooling after a loss-of-coolant-accident (LOCA). Core Spray provides adequate cooling along with LPCI for intermediate and large line break sizes up to and including the design basis double-ended recirculation line break, without assistance from any other Emergency Core Cooling system.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this transmitter SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The transmitters provide input to the associated trip units which are functionally tested and calibrated once every 3 months and have a sensor check performed every 12 hours.

An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval was appropriately considered in the development of the associated plant setpoints.

Extending the Channel Calibration surveillance test interval for the transmitters is acceptable because the Function is verified to be operating properly by the more frequent performance of Sensor Checks every 12 hours, Channel Functional Tests every 3 months, and Channel Calibrations of the remaining circuitry of the associated trip units every 3 months. These checks confirm the major portions of the monitoring loop are maintained within allowances.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

In addition, this SR in TS Table 4.2.1 “ECCS INSTRUMENTATION,” item 1, has been reworded to change the current TS language from “Every Operating Cycle” to “Once/Operating Cycle.” This is an administrative change required to maintain consistency with other Monticello SRs in TS Table 4.2.1.

TS Table 3.2.2 Emergency Core Cooling Systems

Function A.1.b.ii Core Spray and LPCI – Reactor Low Pressure Permissive Bypass Timer

One of the initiating signals for Core Spray and LPCI is reactor low-low water level sustained for twenty minutes. Upon receiving a low-low water level signal, the Core Spray and LPCI pumps will start if either the low reactor pressure permissive or the bypass timer is satisfied. The twenty-minute initiation delay is provided by Time Delay Relays that provide this Core Spray and LPCI bypass timer Function. The drift for the extended surveillance interval relative to the plant Trip Setting for this function was evaluated, it has been determined that a new line item should be added to TS Table 4.2.1, ECCS Instrumentation, as item 10, Reactor Low Pressure (Bypass Timer). A revision to the Trip Setting is discussed below.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this time delay relay SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period.

An evaluation of the surveillance interval extension for the time delay relays was performed, based upon the approach described in GL 91-04. The drift associated with the time delay relays was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval relative to the plant Trip Setting for this function was analyzed; as a result, the Trip Setting is changed from " 20 ± 1 minute” to “20 ± 2 minutes." Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR Table 4.2.1 Perform FUNCTIONAL TEST

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that the ECCS logic for specific trips will function as designed in response to an analyzed condition. Extending the surveillance test interval for the Functional Test is acceptable, because these timers are part a network that includes actuating logic which is designed to be single failure proof and, therefore, highly reliable. Furthermore, as stated in Reference 1:

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical failure, increasing the Logic System Functional Test interval represents no significant change in the overall safety system unavailability."

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

<u>TS Table 3.2.2 Emergency Core Cooling Systems</u>
Function A.3 Core Spray and LPCI – Loss of Auxiliary Power

The auxiliary power system is designed to provide adequate power to operate all the plant auxiliary loads necessary for plant operation. The auxiliary power system power sources, distribution equipment, power and control cabling and loads are arranged so that failure of a single component in the auxiliary power system will not reduce plant safety or impair the operation of essential plant functions. Power is supplied from the Emergency Diesel Generators (EDGs) if normal auxiliary power fails. Shedding of non-safeguard loads is accomplished by transfer relays associated with the reserve auxiliary transformer or the EDGs. A group of relays for this purpose is provided for each of the two sources and for each of the two buses. These relays directly trip the breakers supplying the loads that are not required.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months,

including the 25% grace period. An increase in the surveillance interval to accommodate a 24-month fuel cycle does not affect the contacts relative to drift.

Evaluations of the surveillance interval extension for these relays and contacts were performed, based upon the approach described in GL 91-04. These relays and contacts are mechanical devices in nature and drift does not apply to these devices. Therefore, an increase in surveillance interval does not affect the distribution equipment, relays and contacts relative to drift, and the extension of this surveillance requirement test interval is justified based upon a review of the surveillance test history.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR Table 4.2.1 Perform FUNCTIONAL TEST

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR assures that the ECCS logic for specific trips will function as designed in response to an analyzed condition. Extending the surveillance test interval for the Functional Test is acceptable because these contacts are mechanical devices in nature and drift does not apply to the devices. Additional justification for extending the surveillance test interval is that the network, including the actuating logic, is designed to be single failure proof and, therefore, is highly reliable. Furthermore, as stated in Reference 1:

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical failure, increasing the Logic System Functional Test interval represents no significant change in the overall safety system unavailability."

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

<u>TS Table 3.2.2 Emergency Core Cooling Systems</u>
Function B.2 High Pressure Coolant Injection (HPCI) System – Low-Low Reactor Water Level

The RPV Low-Low water level may indicate that an ATWS event has occurred. Accordingly, an ATWS system trip is initiated when the level is ≥ -48 inches. To prevent the ATWS trip on low-low water level from affecting the ECCS system performance, a time delay is provided for this trip. The HPCI System is designed to pump water into the reactor vessel under loss-of-coolant conditions that do not result in rapid depressurization of the pressure vessel. The HPCI System is designed to provide adequate reactor core cooling for small breaks below the capability of the unassisted Core Spray or LPCI and to depressurize the reactor primary system to aid the LPCI and Core Spray. A detailed discussion of the performance of the HPCI in conjunction with the LPCI and Core Spray is given in the Monticello USAR.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The transmitters provide input to the associated trip units which are functionally tested and calibrated once every 3 months and have a sensor check performed every 12 hours.

An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval was appropriately considered in the development of the associated plant setpoints.

Extending the surveillance test interval for the Channel Calibration for the transmitters is acceptable, because the Function is verified to be operating properly by the more frequent performance of Sensor Checks every 12 hours, Channel Functional Test every 3 months, and Channel Calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm the major portions of the monitoring loop are maintained within allowances.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

In addition, this SR in TS Table 4.2.1 “ECCS INSTRUMENTATION,” item 1, has been reworded to change the current TS language from “Every Operating Cycle” to “Once/Operating Cycle.” This is an administrative change required to maintain consistency with other Monticello SRs in TS Table 4.2.1.

TS Table 3.2.2 Emergency Core Cooling Systems
Function C.1 Automatic Depressurization – Low-Low Reactor Water Level

The RPV Low-Low water level may indicate that an ATWS event has occurred. Accordingly, an ATWS system trip is initiated when the level is ≥ -48 inches. To prevent the ATWS trip on low-low water level from affecting the ECCS system performance, a time delay is provided for this trip. The ADS is designed to depressurize the reactor to permit either LPCI or Core Spray to cool the reactor core during a small break Loss-Of-Coolant-Accident; this size break would result in a coolant loss without a significant pressure reduction, so neither low-pressure system alone could prevent clad damage.

Analysis has determined that by operating two of the relief valves, the depressurization requirement is met. However, three valves are signaled to open and remain open upon detection of "low-low reactor water level" and after up to a 2-minute time delay to provide adequate margin. The time delay is provided to prevent an unnecessary depressurization if the abnormal condition is removed during this time or if the operator determines depressurization is not desirable.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The associated trip units are functionally tested and calibrated once every 3 months and a sensor check is performed every 12 hours.

An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval was appropriately considered in the development of the associated plant setpoints.

Extending the surveillance test interval for the Channel Calibration for the transmitters is acceptable, because the Function is verified to be operating properly by the more frequent performance of Sensor Checks every 12 hours, Channel Functional Test every 3 months, and Channel Calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm the major portions of the monitoring loop are maintained within allowances.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

In addition, this SR in TS Table 4.2.1 "ECCS INSTRUMENTATION," item 1, has been reworded to change the current TS language from "Every Operating Cycle" to "Once/Operating Cycle." This is an administrative change required to maintain consistency with other Monticello SRs in TS Table 4.2.1.

TS Table 3.2.2 Emergency Core Cooling Systems
Function C.2 Automatic Depressurization – Auto Blowdown Timer

The ADS accomplishes reactor vessel depressurization by blowdown through automatic opening of the safety/relief valves, which vent steam to the suppression pool. A time delay circuit is located in series with the blowdown activation signal to provide time for the HPCI or Condensate and Feedwater System to achieve proper operation restoring reactor coolant level.

The time delay also provides surveillance time in which the operator can evaluate possible spurious activation signals. A permissive signal from the time delay circuit serves as the confirming signal to a two-out-of-two-once network to activate the relief valves unless the operator interrupts the sequence.

Excessive reactor vessel pressure is automatically relieved by a steam operated pilot valve, which in turn activates the main valve. The drift for the extended surveillance interval relative to the plant Trip Setting for the automatic blowdown function was evaluated and as a result NMC determined that a new line item should be added to TS Table 4.2.1, ECCS Instrumentation, as item 11. Auto Blowdown Timer.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this timer SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR assures that the ECCS logic for specific trips will function as designed in response to an analyzed condition. Extending the surveillance test interval for the Channel Calibration is acceptable because these timers are part a network that includes actuating logic that is designed to be single failure proof and therefore highly reliable.

An evaluation of the surveillance interval extension for the timers was performed, based upon the approach described in GL 91-04. The drift associated with the

timers was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR Table 4.2.1 Perform FUNCTIONAL TEST

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR assures that the ECCS logic for specific trips will function as designed in response to an analyzed condition. Extending the surveillance test interval for the Functional Test is acceptable because these timers are part a network that includes actuating logic that is designed to be single failure proof and therefore highly reliable. Furthermore, as stated in Reference 1:

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical failure, increasing the Logic System Functional Test interval represents no significant change in the overall safety system unavailability."

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

<u>TS Table 3.2.2 Emergency Core Cooling Systems</u>
Function D.2 Diesel Generator – Low-Low Reactor Water Level

The RPV Low-Low water level may indicate that an ATWS event has occurred. Accordingly, an ATWS system trip is initiated when the level is ≥ -48 inches. To prevent the ATWS trip on low-low water level from affecting the ECCS system performance, a time delay is provided for this trip. Two independent Emergency Diesel Generators (EDGs) provide redundant standby power sources. Each EDG is capable of providing sufficient power to safely shut down the reactor upon the loss of all outside power simultaneous with the design basis accident. Starting of

the EDGs is initiated by a degradation or loss of voltage on an essential 4160 VAC bus. Automatic starting is also initiated by low-low reactor water level or high drywell pressure.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The transmitters provide input to the associated trip units which are functionally tested and calibrated once every 3 months and have a sensor check performed every 12 hours.

An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval was appropriately considered in the development of the associated plant setpoints.

Extending the surveillance test interval for the Channel Calibration for the transmitters is acceptable, because the Function is verified to be operating properly by the more frequent performance of Sensor Checks every 12 hours, Channel Functional Test every 3 months, and Channel Calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm the major portions of the monitoring loop are maintained within allowances.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

In addition, this SR in TS Table 4.2.1 “ECCS INSTRUMENTATION,” item 1, has been reworded to change the current TS language from “Every Operating Cycle” to “Once/Operating Cycle.” This is an administrative change required to maintain consistency with other Monticello SRs in TS Table 4.2.1.

TS 3.2.D Other Instrumentation

The ECCS consists of a HPCI System, the CS System, the LPCI mode of the RHR System, and the ADS.

Monticello is also equipped with a Reactor Core Isolation Cooling (RCIC) System, which is an alternative source of make-up water for the reactor. It is designed to

provide adequate makeup to the reactor during normal plant shutdowns and transient events that lead to a loss of feedwater flow. The RCIC System is not part of the ECCS.

The following SRs were evaluated relative to extending their respective testing intervals. These SRs ensure the availability of safety functions that respond to plant transients and design basis events. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval reduction is discussed below:

TS Table 3.2.8 Other Instrumentation
Function A.1 RCIC Initiation –
Low-Low Reactor Level

The RPV Low-Low water level may indicate that an ATWS event has occurred. Accordingly, an ATWS system trip is initiated when the level is ≥ -48 inches. To prevent the ATWS trip on low-low water level from affecting the ECCS system performance, a time delay is provided for this trip. The design basis for the RCIC System is to provide adequate makeup to the reactor during normal plant shutdowns and transient events that lead to a loss of feedwater flow. The RCIC System is not part of the ECCS network.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The transmitters provide input to the associated trip units which are functionally tested and calibrated once every 3 months and have a sensor check performed every 12 hours.

An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval was appropriately considered in the development of the associated plant setpoints.

Extending the surveillance test interval for the Channel Calibration for the transmitters is acceptable, because the Function is verified to be operating properly by the more frequent performance of Sensor Checks every 12 hours, Channel Functional Test every 3 months, and Channel Calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm the major portions of the monitoring loop are maintained within allowances.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS Table 3.2.8 Other Instrumentation
Function B.1 HPCI/RCIC Turbine Shutdown – High Reactor Water Level

Automatic trip of the HPCI turbine occurs on high turbine exhaust pressure, low pump suction pressure, high reactor water level, low steam supply line pressure, HPCI auto isolation or turbine overspeed. The high reactor water level trip seals in until either a manual restart is initiated or a low-low reactor water level signal is received.

Provisions exist for automatic shutdown of the RCIC turbine upon receipt of any one of the following signals: turbine overspeed, high water level in the reactor vessel, low pump suction pressure or high turbine exhaust pressure. Provisions are also included for remote-manual startup, operation, and shutdown of the RCIC system provided Low-Low and high water level signals do not exist.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The transmitters provide input to the associated trip units, which are functionally tested and calibrated once every 3 months and have a sensor check every 12 hours.

An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval was appropriately considered in the development of the associated plant setpoints.

Extending the surveillance test interval for the Channel Calibration for the transmitters is acceptable, because the Function is verified to be operating properly by the more frequent performance of Sensor Checks every 12 hours, Channel Functional Test every 3 months, and Channel Calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm the major portions of the monitoring loop are maintained within allowances.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

In addition, this SR in TS Table 4.2.1 "ECCS INSTRUMENTATION," item 9, has been reworded to change the current TS language from "Every Operating Cycle" to "Once/Operating Cycle." This is an administrative change required to maintain consistency with other Monticello SRs in TS Table 4.2.1.

TS Table 3.2.8 Other Instrumentation
Function C.1 HPCI/RCIC Turbine Suction Transfer – Condensate Storage Tank Low Level

Two sources of water are available for the HPCI System. Normally, water is supplied for the suction of the HPCI pump from two CSTs. Low level in a condensate storage tank (CST) indicates the unavailability of an adequate supply of makeup water from this normal source. When the level in either tank falls to the Technical Specification setpoint the pump suction is automatically transferred to the other source of water, which is the suppression pool. A level switch associated with each tank provides one-out-of-two-once logic for this transfer. Valves from the suppression pool suction line are automatically opened when condensate storage water level drops to the setpoint specified in the TS. The condensate storage tank suction valve is automatically closed via an interlock with the suppression pool suction valves to prevent losing suction to the pump.

The RCIC System turbine driven pump supplies demineralized makeup water from the condensate storage tank to the reactor vessel; an alternate safety related source of water is available from the suppression pool. Valves from the suppression pool suction line are automatically opened when condensate storage water level drops to the setpoint specified in the Technical Specifications. The condensate storage tank suction valve is automatically closed via an interlock with the suppression pool suction valves to prevent losing suction to the pump.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this level switch SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. Float switches perform this Function. The float switches are mechanical devices that require mechanical setting at the proper level only. The devices cannot be significantly adjusted without a physical change in the location of the installation. Therefore, an increase in surveillance interval does not affect the accuracy of the float switches.

An evaluation of the surveillance interval extension was performed, based upon the approach described in GL 91-04. Because these float switches are

mechanical devices in nature, drift does not apply to the devices. Thus, the extension of this surveillance requirement test interval is based upon the surveillance test history.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR Table 4.2.1 Perform FUNCTIONAL TEST

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This surveillance requirement ensures the ECCS logic for the HPCI/RCIC Turbine Suction Transfer on Condensate Storage Tank Low Level will function as designed in response to an analyzed condition. Justification for extending the surveillance test interval is that the network, including the actuating logic, is designed to be single failure proof and, therefore, is highly reliable. Furthermore, as stated in Reference 1:

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical failure, increasing the Logic System Functional Test interval represents no significant change in the overall safety system unavailability."

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 3.2.E Reactor Building Isolation and Standby Gas Treatment System

Normal ventilation system operation provides outside air to all levels and equipment rooms of the Reactor Building. This supply air may be filtered or unfiltered depending on seasonal conditions. The normal ventilation system has the capacity to provide a minimum of one air change per hour of filtered air to all portions of the Reactor Building requiring ventilation.

The standby gas treatment system is provided to maintain a small negative pressure to minimize ground level escape of airborne radioactivity whenever secondary containment isolation conditions exist. Filters are provided in the system to remove radioactive particulates, and charcoal adsorbers are provided to remove radioactive halogens. All flow from the standby gas treatment system is released through the elevated off-gas vent stack and continuously monitored by the stack gas monitoring system. The system may also be used to vent the primary containment during plant operation.

The following SRs for the Reactor Building Isolation and Standby Gas Treatment System functions were evaluated relative to extending their respective testing intervals. These SRs ensure the Reactor Building Isolation and Standby Gas Treatment System will function as designed during an analyzed event. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are discussed below:

Table 3.2.4 Reactor Building Isolation and Standby Gas Treatment System
Function 1 Low-Low Reactor Water Level

Reactor Building Isolation and Standby Gas Treatment initiation signal is activated via primary containment isolation logic due to Low-Low reactor water level. The Reactor Building Ventilation Isolation and Standby Gas Treatment System initiation provide a containment system for the potential release that may occur within secondary containment. This is accomplished by a low leakage enclosure and a standby gas treatment system that has a capacity greater than the in-leakage rate. This system purifies air from the secondary containment and exhausts it to the outside by maintaining a negative pressure in the containment relative to outside and assuring that leakage flows into the secondary containment and no significant exfiltration of untreated gases exists.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The transmitters provide input to the associated trip units which are functionally tested and calibrated once every 3 months and have a sensor check performed every 12 hours.

An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended

surveillance interval was appropriately considered in the development of the associated plant setpoints.

Extending the surveillance test interval for the Channel Calibration for the transmitters is acceptable, because the Function is verified to be operating properly by the more frequent performance of Sensor Checks every 12 hours, Channel Functional Test every 3 months, and Channel Calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm the major portions of the monitoring loop are maintained within allowances.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

In addition, this SR in TS Table 4.2.1 “REACTOR BUILDING VENTILATION & STANDBY GAS TREATMENT,” item 1, has been reworded to change the current TS language from “Every Operating Cycle” to “Once/Operating Cycle.” This is an administrative change required to maintain consistency with other Monticello SRs in TS Table 4.2.1.

TS 3.2.F Recirculation Pump Trip and Alternate Rod Injection Initiation

A Recirculation Pump Trip (RPT) System and an Alternate Rod Injection (ARI) System comprise two of the systems to mitigate Anticipated Transients Without Scram (ATWS) events. The ATWS - ARI system is designed such that no single failure of the ATWS - ARI system can cause an inadvertent reactor scram.

The ATWS system consists of two separately powered trip systems, Channel A and Channel B, each made up of two sub-channels. Each sub-channel receives an input from an independent sensor monitoring each of the ATWS trip parameters. A trip occurring in both sub-channels of logic Channel A or a trip occurring in both sub-channels of logic Channel B will cause an ATWS trip which opens both recirc MG set generator field breakers and causes control rod insertion by venting the scram air header. Each field breaker is equipped with two trip coils, one connected to logic Channel A and the other to logic Channel B. Either trip coil can trip the breaker. Two solenoid valves are installed in the scram air header upstream of the hydraulic control units. Energizing either of the valves will vent the header and cause control rod insertion (if a common-mode failure has not disabled the drives) when the scram valves fail open on low air pressure.

The following SRs for the RPT and ARI System functions were evaluated relative to extending their respective testing intervals. These SRs ensure the RPT and ARI

will function as designed during an analyzed event. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are discussed below:

Table 3.2.5 Recirculation Pump Trip and Alternate Rod Injection Initiation
Function 1 High Reactor Dome Pressure

A RPT System and an ARI System comprise two of the systems to mitigate ATWS events. Reactor vessel pressure significantly higher than the high-pressure scram setting is the primary indication of an ATWS event, therefore, an ATWS trip is initiated when reactor pressure reaches #1135 psig.

Reactor dome pressure is monitored, by four pressure transmitters and their outputs are fed to four analog trip units. The trip units energize the respective sensor relays when the transmitter output reaches the trip unit setpoint.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for the calibration of these pressure transmitters, as applied to this Function, are being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The transmitters provide input to the associated trip units which are functionally tested and calibrated once every 3 months and have a sensor check performed once per day.

An evaluation of the surveillance interval extension for the transmitters was performed based upon the approach described in GL 91-04. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval was appropriately considered in the development of the associated plant setpoints.

Extending the surveillance test interval for the Channel Calibration for the transmitters is acceptable because the Function is verified to be operating properly by the more frequent performance of Sensor Checks every 12 hours, Channel Functional Test every 3 months, and Channel Calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm the major portions of the monitoring loop are maintained within allowances.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

Table 3.2.5 Recirculation Pump Trip and Alternate Rod Injection Initiation
Function 2 Low-Low Reactor Water Level

A RPT System and an ARI System comprise two of the systems to mitigate ATWS events. Low-Low water level in the reactor vessel may indicate that an ATWS event has occurred. Accordingly, an ATWS system trip is initiated when the reactor water level reaches the Low-Low level setpoint. Four level transmitters monitor reactor water level. Transmitter outputs are fed to four analog trip units that energize the respective sensor relays when the transmitter output reaches the trip unit setpoint. To prevent the ATWS trip on Low-Low water level from affecting the ECCS system performance, a time delay is provided for this trip. If the Low-Low level condition clears before the time delay relay times out, the relay will reset and the ATWS trip will not occur.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for the calibration of these level transmitters, as applied to this Function, are being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The transmitters provide input to the associated trip units which are functionally tested and calibrated once every 3 months and have a sensor check performed once every 12 hours.

An evaluation of the surveillance interval extension for the transmitters was performed based upon the approach described in GL 91-04. The drift associated with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval was appropriately considered in the development of the associated plant setpoints.

Extending the surveillance test interval for the Channel Calibration for the transmitters is acceptable because the Function is verified to be operating properly by the more frequent performance of Sensor Checks every 12 hours, Channel Functional Test every 3 months, and Channel Calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm the major portions of the monitoring loop are maintained within allowances.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 3.2.G Safeguards Bus Voltage Protection

Three transformers are provided to supply the plant with offsite power from the substation. All three sources can independently provide adequate power for the plant's safety-related loads.

Transfer of the essential buses to either of the emergency power sources, the reserve auxiliary transformer (1AR) or the Emergengy Diesel Generators (EDGs), will occur due to loss of essential bus voltage or degraded voltage conditions on the essential bus. The essential bus transfer schemes take into account the system divisions such that a malfunction of any single component will only prevent the automatic re-energization of one division's essential bus by its associated EDG or 1AR.

The 1AR transformer and the EDGs are limited in capacity and are primarily intended to supply safeguard loads. Certain other loads may be supplied if the situation permits. To prevent overloading these limited capacity sources, and to avoid excessive voltage drop during motor acceleration periods, load shedding and load application sequencing circuits are provided.

The following SRs for the Safeguards Bus Voltage Protection functions were evaluated relative to extending their respective testing intervals. These SRs ensure the Safeguards Bus Voltage Protection will function as designed during an analyzed event. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are discussed below:

Table 3.2.6 Instrumentation for Safeguards Bus Voltage Protection
Function 2 Loss of Voltage Protection

Each division of the essential 4.16 kV buses is provided with two transfer schemes. One controls automatic transfer to the 1AR Transformer and the other controls automatic transfer to the associated EDG. Transfer of the essential buses to the 1AR transformer will normally occur on loss of voltage or degraded voltage conditions. If the 1AR no-load voltage is unacceptable, or if the essential buses are being supplied from the 1AR transformer when the loss of voltage or degraded voltage condition occurs, a transfer to the EDGs will take place. A revision to the Trip Setting is discussed below.

SR 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. Voltage sensing relays are provided on the

safeguards bus to transfer the bus to an alternate source when a loss of voltage condition or a degraded voltage condition is sensed. On loss of voltage, the voltage sensing relays trip immediately and energize auxiliary relays that control the bus transfer sequence. The transfer on degraded voltage has a time delay to prevent transfer during the starting of large loads. The degraded voltage setpoint corresponds to the minimum acceptable safeguards bus voltage for a steady state LOCA load that maintains adequate voltage at the 480V essential Motor Control Centers. An allowance for relay tolerance is included.

An evaluation of the surveillance interval extension for the undervoltage relays was performed, based upon the approach described in GL 91-04. The drift associated with the undervoltage relays was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335.

The drift for the extended surveillance interval relative to the plant Trip Setting for this function was considered and as a result the Trip Setting is being changed from " 2625 ± 175 volts" to "2625 ± 280 volts." Extending this surveillance interval for this SR is acceptable for the following reasons: the design, in conjunction with TS requirements which limit the extent and duration of inoperable AC sources, provides substantial redundancy in AC sources; breaker verification and periodic breaker maintenance is based upon performance history for the breakers and is designed for maximum availability.

Additional justification for extending the surveillance test interval is that the portions of the test not directly associated with the functioning of the offsite source and breaker movement are equivalent to a system Functional Test. For these tests, Reference 1 documents the following conclusion:

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical failure, increasing the Logic System Functional Test interval represents no significant change in the overall safety system unavailability."

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 3.2.H Safety/Relief Valve Low-Low Set Logic

The function of the Safety/Relief Valve (SRV) Low-Low Set System is to minimize the possibility of an SRV reopening with an elevated water leg in its discharge line. The elevated water leg occurs after an SRV closes. The condensing steam in the SRV discharge line creates a vacuum that draws torus water up into the discharge line. Preventing subsequent manual or automatic actuation prior to the water leg receding in the discharge line prevents excessive hydrodynamic loading of the discharge piping and suppression chamber components.

The SRV Low-Low Set System controls the operation of three SRVs and is totally independent of the Automatic Depressurization System. The low-low set protective function is fully automatic and requires no operator action. The logic is also dual divisional.

The following SRs for the SRV Low-Low Set System functions were evaluated relative to extending their respective testing intervals. These SRs ensure the SRV Low-Low Set System will function as designed during an analyzed event. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are discussed below:

TS Table 3.2.7 Safety/Relief Valve Low-Low Set Logic
Function 2 Reactor Coolant System Pressure – Opening

The low-low set SRVs have lower opening setpoints than their mechanical setpoint. These lower setpoints are for electrical/pneumatic actuation of the low-low set SRVs. The low-low set protective function is fully automatic and requires no operator action. The safety/relief valves which are also part of the SRV Low-Low Set System and have low-low set actuation are prevented from subsequent manual or automatic actuation prior to the water leg receding in the discharge line to prevent excessive hydrodynamic loading of the discharge piping and suppression chamber components. A revision to the Trip Setting is discussed below.

SR 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This Channel Calibration extension is acceptable because the associated transmitters have a Functional Test once every 3 months and sensor check once per day.

An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated

with the transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval was evaluated and as a result the Trip Setting is changed from "1072 ± 3 psig; 1062 ± 3 psig; and 1052 ± 3 psig" to "1072 ± 14 psig; 1062 ± 14 psig; and 1052 ± 14 psig." Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS Table 3.2.7 Safety/Relief Valve Low-Low Set Logic
Function 3 Reactor Coolant System Pressure – Closing

Closing of a low-low set SRV occurs after an 80 psi blowdown of reactor pressure is detected. The setpoints for the low-low set valves ensure that they will be the first SRVs to open and the last to close. After a low-low set SRV has opened and closed a time delay relay prevents the plant operators or the low-low set logic from immediately re-opening the SRV to allow the water leg in the SRV discharge line to recede. A revision to the Trip Setting is discussed below.

SR 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This Channel Calibration extension is acceptable because the associated transmitters have a Functional Test once every 3 months and sensor check once per day.

An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated with the pressure transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval relative to the plant Trip Setting for this function was evaluated and as a result the Trip Settings are changed from "992 ± 3 psig; 982 ± 3 psig; and 972 ± 3 psig" to "992 ± 14 psig; 982 ± 14 psig; and 972 ± 14 psig". Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS Table 3.2.7 Safety/Relief Valve Low-Low Set Logic
Function 4 Discharge Pipe Pressure

Safety/relief valve position indication is monitored by a differential pressure transmitter signal to an analog trip unit monitoring the steam pressure in each discharge pipe. When pressure is sensed, the trip unit will indicate the valve is open by lighting a light and alarming in the Control Room. A revision to the Trip Setting is discussed below.

SR 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This Channel Calibration extension is acceptable because the associated transmitters have a Functional Test once every 3 months and sensor check once per month.

An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. The drift associated with the pressure transmitters was determined, applying a Monticello-specific methodology, based upon the recommendations of EPRI TR-103335. The drift for the extended surveillance interval relative to the plant Trip Setting for this function was evaluated and as a result the Trip Setting is changed from “30 ± 1 psid” to “30 ± 3 psid.” Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 3.2.I Instrumentation for Control Room Habitability Protection

The main control room ventilation air radiation inlet monitors are designed to automatically prevent the injection of contaminated air into the control room resulting from a steam line break or leakage that bypasses secondary containment during a LOCA. This monitoring system assures control room habitability so that control room operators will be adequately protected against the effects of accidental release of radioactivity into the environment.

The following SR for the Control Room Habitability Protection Instrumentation functions was evaluated relative to extending its testing interval. This SR ensures the Control Room Habitibility Protection Instrumentation will function as designed during an analyzed event. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are discussed below:

TS Table 3.2.9 Instrumentation for Control Room Habitability Protection
Function 1 Radiation

The radiation detectors are sufficiently sensitive to transfer the air handling system to the filtration/pressurization mode before radiation levels in the control room become excessive. The filtration units providing make-up air for establishing positive pressure in the control room are equipped with High Efficiency Particulate Absolute (HEPA) filters and charcoal adsorbers. Two detectors arranged in a one-out-of-two-once logic scheme are provided for redundancy. Due to the close proximity of the radiation detectors and their associated signal cables, the radiation monitor system has been modified so that the Control Room Ventilation-Emergency Filtration Train (CRV-EFT) system will trip into the high radiation mode if a radiation monitor failure signal is received.

SR 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The associated detectors are functionally tested monthly and a sensor check is performed daily. The radiation detector is calibrated using a calibrated source as an input signal to the detector. Exposing the sensor-converter to a known source in a constant geometry performs this calibration. Source checks of radiation monitors are subject to far more uncertainties than electronic calibration checks because of source decay uncertainties, positioning of the sources, signal strength and the sensor response curves of the particular monitoring system. Because of the uncertainties associated with the calibration methods for these devices, any drift evaluation will provide no true indication of the instruments performance over time.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 3.3.F Scram Discharge Volume

The purpose of the Scram Discharge Volumes (SDV) is to contain the water exhausted from the control rod drives during a scram. During normal plant operation, the discharge volumes are empty and vent and drain valves are open. The RPS de-energizes when a scram occurs which allows the vent and drain valves to close to isolate the SDV to contain the reactor water. This limits the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within 10 CFR 100 limits. Redundant vent and drain valves ensure single-failure-proof capability for isolating the scram discharge header.

The following SR for the SDV function was evaluated relative to extending its testing interval. This SR ensures the SDV will function as designed during an analyzed event. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are discussed below:

SR 4.3.F Once per operating cycle verify the scram discharge volume vent and drain valves close within 30 seconds after receipt of a reactor scram signal and open when the scram is reset.

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures the SDV vent and drain valves close in ≤ 30 seconds after receipt of an actual or simulated reactor scram signal and open when the actual or simulated scram signal is reset. The scram discharge volume vent and drain valves are cycled quarterly. This ensures the mechanical components and a portion of the valve logic remain operable. This test does not ensure that the logic of the SDV vent and drain valves is operable; however, logic systems are inherently more reliable than other plant components. This is acknowledged in Reference 1, the NRC Safety Evaluation (SE), dated August 2, 1993, relating to the extension of Peach Bottom Atomic Power Station, Units 2 and 3, surveillance interval extension from 18 to 24 months as follows:

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical failure, increasing the Logic System Functional Test interval represents no significant change in the overall safety system unavailability."

Because of the inherent equipment reliability of logic systems (as demonstrated by years of operating experience in the nuclear and non-nuclear industry), and more frequent stroke testing of the subject valves to prove valve function it was concluded that the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 3.4 Standby Liquid Control System

The MNGP USAR states that the design objective of the Standby Liquid Control System (SLCS) is to provide the capability of bringing the reactor from full power to a shut down condition. To meet this objective, the Standby Liquid Control System was designed to inject a quantity of boron neutron absorber solution into the reactor if the reactor cannot be shut down or kept shut down with control rods. The SLCS is required only to shut down the reactor and keep it from going critical again as it cools. The SLCS is used only in the highly improbable event that not enough control rods can be inserted into the reactor core to accomplish shutdown and cooldown in the normal manner. The SLCS is required for postulated ATWS events.

The equipment for the SLCS consists of an unpressurized tank for low temperature sodium pentaborate solution storage, a pair of full capacity positive displacement pumps, two explosive actuated shear plug valves, heaters, piping, valves and instrumentation. The boron solution is pumped into the reactor pressure vessel so it mixes with the cooling water. The boron-10 isotope absorbs thermal neutrons and thereby terminates the nuclear fission reaction in the fuel.

The following SRs were evaluated relative to extending their respective test intervals. These SRs ensure the availability of safety functions that respond to plant transients and design basis events. Potential time-based considerations, such as failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are detailed below:

SR 4.4.A.2.a At least once during each operating cycle – Manually initiate one of the two standby liquid control systems and pump demineralized water into the reactor vessel. This test checks explosion of the charge associated with tested system, proper operation of the valves and pump capacity. Both systems shall be tested and inspected, including each explosion valve in the course of two operating cycles.

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months (on a staggered test basis) to 24 months (on a staggered test basis), for a maximum interval of 30 months, including the 25% grace period. This SR ensures the SLCS is capable of injecting into the reactor pressure vessel by verifying a flow path. The SLCS is designed to bring the reactor subcritical during the most reactive point in core life. The SLCS is also designed so that all active components are single failure proof. Each SLCS pump is tested during the operating cycle on a quarterly basis to verify system capacity. Other TS SRs ensure that the SLCS solution temperature, room temperatures and the solution volume are checked once per day to prevent the precipitation of the borated solution. TS SRs also require that once per month, or any time water or boron are added or if the solution temperature drops below the limits specified in TS, the boron concentration is determined. These tests ensure that the SLCS components are operable and an open flow path exists during the operating cycle.

Based upon the inherent system and component reliability and the testing performed during the operating cycle, the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.4.A.2.b At least once during each operating cycle – Explode one of two primer assemblies in the same batch to verify proper function. Then install, as a replacement, the second primer assembly in the explosion valve of the system tested for operation.

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months (on a staggered test basis) to 24 months (on a staggered test basis), for a maximum interval of 30 months, including the 25% grace period. This SR ensures the SLCS is capable of injecting into the reactor pressure vessel by exploding an explosion valve primer assembly. The SLCS is designed to bring the reactor subcritical during the most reactive point in core life. The SLCS is also designed so that all active components are single failure proof. Each SLCS pump is tested during the operating cycle on a quarterly basis to verify system capacity. Other TS SRs ensure that the SLCS solution temperature, room

temperatures and the solution volume are checked once per day to prevent the precipitation of the borated solution. TS SRs require that, once per month, or any time water or boron are added or if the solution temperature drops below the limits specified in TS, the boron concentration is determined. The explosive valves and all other system components are designed to be highly reliable.

Based upon the inherent system and component reliability and the testing performed during the operating cycle the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.4.B.1 At least once per operating cycle – Boron enrichment shall be determined. The laboratory analysis to determine enrichment shall be obtained within 30 days of sampling or chemical addition.

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The SLCS is designed to bring the reactor subcritical during the most reactive point in core life. The SLCS is also designed so that all active components are single failure proof. Each SLCS pump is tested during the operating cycle on a quarterly basis to verify system capacity. Other TS SRs ensure that the SLCS solution, room temperatures and the solution volume are checked once per day to prevent the precipitation of the borated solution. TS SRs require that once per month, or any time water or boron are added or if the solution temperature drops below the limits specified in TS, the boron concentration is determined. These tests ensure that the SLCS components are operable and an open flow path exists during the operating cycle.

Based upon the inherent system and component reliability and the testing performed during the operating cycle the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 3.5.A ECCS Systems

The design basis for the ECCS is to adequately cool the core and to limit potential radioactive releases from the fuel in the event of a LOCA. This is accomplished by means of the ECCS, which consists of: a CS System, a HPCI System, a LPCI mode of RHR System and ADS. The CST or the suppression pool provides the water source for the ECCS.

The HPCI System is designed to pump water into the reactor vessel under loss-of-coolant conditions that do not result in rapid depressurization of the pressure vessel. The loss-of-coolant may be due to loss of reactor feedwater or to small line breaks that do not cause depressurization of the reactor vessel. Operation of the HPCI turbine continues as long as reactor pressure, as measured at the HPCI steam line, is above 150 psig.

Two independent CS loops are provided for use under LOCA conditions associated with large pipe breaks and reactor vessel depressurization. The CS, along with LPCI, is designed to maintain continuity of core cooling for a large spectrum of LOCAs. The CS provides adequate cooling along with LPCI for intermediate and large line break sizes up to and including the design basis double-ended recirculation line break, without assistance from any other ECCS. The LPCI subsystem is an integral part of the RHR System, it operates to restore and maintain the coolant inventory in the reactor vessel after a LOCA so that the core is sufficiently cooled.

The ADS accomplishes reactor vessel depressurization by blowdown through automatic opening of the safety/relief valves that vent steam to the suppression pool. For small breaks the vessel is depressurized in sufficient time to allow either CS or LPCI to provide adequate core cooling to prevent any clad melting. A time delay is provided to prevent an unnecessary depressurization if the abnormal condition is removed during this time or if the operator determines depressurization is not desirable.

The following SRs were evaluated relative to extending their respective testing intervals. These SRs ensure the availability of safety functions that respond to plant transients and design basis events. Potential time-based considerations, such as failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are discussed below:

SR 4.5.A.3.b **Demonstrate, once per operating cycle, with reactor pressure ≤ 165 psig, the HPCI pump can develop a flow rate ≥ 2700 gpm against a system head corresponding to reactor pressure. [NOTE: Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.]**

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This test ensures the capability of the HPCI pump to overcome RPV pressure and inject coolant into the core as designed, for analyzed conditions. The surveillance requirements provide adequate assurance that the HPCI system will be operable when required. All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. Other TS SRs require that HPCI be tested quarterly to ensure required flow at normal operating pressure can be achieved. The ECCS systems are designed to cover a specific range of accident conditions and collectively provide a redundancy of function to avoid undetected common failure mechanisms. Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.5.A.4 **Perform the following tests Each Operating Cycle:**

a. ADS Valve Operability [NOTE: Safety/Relief valve operability is verified by cycling the valve and observing a compensating change in turbine bypass or control valve position.]

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that ADS is capable of performing its design function. The ADS automatically controls three selected safety-relief valves. The ADS is conservatively required to be operable whenever reactor vessel pressure exceeds 150 psig. This pressure is substantially below that for which the low-pressure core cooling systems can provide adequate core cooling for events requiring ADS. Each of the ECCS systems is designed to cover a specific range of accident conditions and collectively provide a redundancy of function to avoid undetected common failure mechanisms. Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.5.A.4 Perform the following tests Each Operating Cycle:
b. ADS Inhibit Switch Operability

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. ADS Inhibit Switch is provided to allow the operater the ability to interrupt the depressurization cycle. If the operator, as directed by emergency procedures, identifies a valid reason for stopping the auto-relief cycle, the operator would manually block its initiation or completion utilizing the ADS inhibit switches in the main control room. Auto depressurization would be blocked for as long as the operator deemed necessary. Operating experience shows this component routinely passes its SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.5.A.4 Perform the following tests Each Operating Cycle:
c. Perform a simulated automatic actuation test (including HPCI transfer to the suppression pool and automatic restart on subsequent low reactor water level)

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that HPCI, CS, LPCI and ADS are capable of performing their design functions. The HPCI pump is normally lined up to the CSTs. The suction is switched to the suppression pool upon low level being sensed in either CST or high level in the suppression pool. The instrumentation associated with the automatic transfer from the CST to the suppression pool has also been verified to be safety related and capable of withstanding a seismic event. The more frequent testing ensures major portions of the HPCI system are operating within normal boundaries. Operating experience shows that these components routinely pass the SR when performed at the 18-month interval, which is based upon the refueling cycle. Furthermore, as stated in Reference 1:

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical failure, increasing the Logic System Functional Test interval represents no significant change in the overall safety system unavailability."

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

In addition, this SR has been reworded to change the current TS language from "(including HPCI transfer to the suppression pool and automatic restart on subsequent low reactor water level)," to "(including HPCI transfer to the suppression pool and automatic restart on subsequent Low-Low reactor water level)." This is an administrative change required for clarification and to maintain consistency with actual plant practice and other Monticello TS, specifically TS Table 3.2.2, Function B.2.

SR 4.5.D.1.b Demonstrate, once per operating cycle, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure. [NOTE: Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.]

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that the RCIC system is capable of performing its design function at low RPV pressure. The surveillance requirements provide adequate assurance that the RCIC system will be operable when required. All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. Additional SRs require that RCIC be tested quarterly to ensure required flow at normal operating pressure can be achieved. The increased interval for this SR is acceptable because the functions performed by RCIC can also be performed by HPCI. Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.5.D.2 RCIC - Perform a simulated automatic actuation test (including transfer to suppression pool and automatic restart on subsequent low reactor water level) each refueling outage.

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The RCIC System Functional Test ensures a system initiation signal (actual or simulated) to the automatic initiation logic of RCIC will cause the systems or subsystems to operate as designed, including actuation of the automatic valves to their required positions. The increased interval for this SR is acceptable because the functions performed by RCIC can also be performed by HPCI. The RCIC pumps and valves are tested on a more frequent basis as part of the MNGP IST program. The more frequent testing ensures major portions of the RCIC system are operating within normal boundaries. Operating experience shows that these components routinely pass the SR when performed at the 18-month interval, which is based upon the refueling cycle. Furthermore, as stated in Reference 1:

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical failure, increasing the Logic System Functional Test interval represents no significant change in the overall safety system unavailability."

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

In addition, this SR has been reworded to change the current TS language from "(including HPCI transfer to the suppression pool and automatic restart on subsequent low reactor water level)," to "(including HPCI transfer to the suppression pool and automatic restart on subsequent Low-Low reactor water

level)." This is an administrative change required for clarification and to maintain consistency with actual plant practice and other Monticello TS, specifically TS Table 3.2.8, Function A.1.

TS 3.6.D Primary System Boundary - Reactor Coolant System (RCS)

Monticello TS require means for detecting, and to the extent practical, identifying the source of reactor coolant system leakage. The reactor coolant system includes the reactor vessel; the 2-loop reactor coolant recirculation system with its pumps, pipes and valves; the main steam piping up to the main steam isolation valves; safety/relief valves; and the reactor auxiliary systems piping. The piping susceptible to Intergranular Stress Corrosion Cracking (IGSCC) in the recirculation system, the residual heat removal system and the core spray system has been replaced with material resistant to IGSCC or protected with a cladding of resistant weld metal. To further reduce susceptibility to IGSCC a Hydrogen Water Chemistry System is in operation at MNGP.

The necessary plant controls, instrumentation and alarms for safe and orderly operation are located in the main control room. These include such controls and instrumentation as the reactor coolant system leakage detection system. 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 a computer alarm are provided in the control room to alert operators when allowable leak rates are approached.

Drywell airborne particulate radioactivity is continuously monitored as well as drywell atmospheric temperature and pressure. The Process Liquid Radiation Monitoring System also monitors systems connected to the reactor coolant system boundary for leakage.

The following SRs were evaluated relative to extending their respective testing intervals. These SRs ensure the availability of safety functions that respond to plant transients and design basis events. Potential time-based considerations, such as failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are discussed below:

SR 4.6.D.2 **RCS leakage detection instrumentation shall be demonstrated OPERABLE by:**

a. Primary containment atmosphere particulate monitoring system-performance of a channel calibration at least once per cycle.

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR also requires the performance of a sensor check once per 12 hours, and a channel Functional Test at least monthly. The more frequent testing ensures that major portions of the leakage detection system are operating within normal boundaries.

A s[illegible]s not performed for radiation monitors because the ma[illegible] accuracy of the detector and the calibration sources. In t[illegible] sources multiple readings are normally required and an [illegible] to confirm operation. The decay curves and the det[illegible] from 12% to 30% accurate. This accuracy far ove[illegible] of the electronic signal conditioning circuit therefore drif[illegible] does not provide a measure of function performance ove[illegible]ns. This is substantiated by the ANSI N42.18 acc[illegible] which also recognizes ± 30% for alarm points for rele[illegible] additional 10% error is for log scale indication and no tolerance is provided for the circuit. Since drift of the detector is not a consideration the projected performance is based on the historical performance of the monitoring circuits.

Additionally, this monitor was recently replaced for greater reliability in the performance of this Function. The new instrument drift-monitoring program will assure the drift value for this application is conservative relative to actual monitor performance.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history was performed to validate the above conclusion. The results of this review demonstrate that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.6.D.2 **RCS leakage detection instrumentation shall be demonstrated OPERABLE by:**

b. Required leakage detection instrumentation-performance of a channel calibration test at least once per cycle.

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. Identified leakage is piped to the drywell equipment drain sump. All other leakage is collected in the drywell floor drain sump. This SR requires a sensor check once per 12 hours, and a channel Functional Test at least monthly (flow instruments only). The more frequent testing ensures that major portions of the leakage detection system are operating within normal boundaries. Operating experience shows that these components routinely pass the SR when performed at the 18-month interval, which is based upon the refueling cycle.

Identifying the quantity of leakage (identified or unidentified) is accomplished by the following: 1) timing the run time for the pumps that remove the water from the sumps, 2) change in the sump level indicating instruments, and 3) monitoring the flow rate of the fluid pumped. This includes monitoring the computer point to ensure that it maintains a normal reading. The comparison of these diverse methods confirms the performance of each individually. Additionally the long-term drift has little or no effect on the time that the pumps operate or the change in the measured level. In each case the measurement interval is relatively short. The concern is not for the absolute level in the sump but on the change in level over a short interval or the number of times the pump operates. Since two of the three measurements are not affected by long-term drift an independent drift analysis is not performed for these instruments. An evaluation of the surveillance interval extension for the transmitters was performed, based upon the approach described in GL 91-04. However, this evaluation only reviewed the past performance history (failure analysis), and the extension is based on that evaluation.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 3.6.E Primary System Boundary – Safety/Relief Valves

The reactor coolant system Safety/Relief Valves assure that the reactor coolant system pressure safety limit is never reached. The Safety/Relief Valves also meet the requirements for providing the minimum number of safety/relief valves available, when required, to fulfill Automatic Depressurization System, Low-Low Set System, and alternate shutdown cooling functions under all postulated accident events and transient events coincident with the failures, required to be assumed, during these events.

The safety/relief valves are mounted on 6-in.-diameter 1500-lb primary service rated flanges so that they may be removed for maintenance or bench checks and reinstalled during normal plant shutdowns. Pressure switches are provided on the low-pressure side of the machined pressure sensing bellows to detect possible leakage of the bellows on the safety/relief valves.

The following SRs were evaluated relative to extending their respective test intervals. These SRs ensure the availability of safety functions that respond to plant transients and design basis events. Potential time-based considerations, such as failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are detailed below:

SR 4.6.E.1 Safety/Relief Valves

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

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that the safety/relief valves will perform their intended safety function. The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. The American Society of Mechanical Engineers (ASME) Code requires that these bellows be monitored for failure since bellows failure would defeat the safety function of the safety/relief valve. Other TS SRs require that the safety/relief valve bellows be continuously monitored. It is recognized that it is not feasible to test the safety/relief valve setpoints while the valves are in place or during normal plant operation. No plant instrumentation is used to satisfy this SR, so instrument drift was not analyzed for this Function.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.6.E.1 Safety/Relief Valves

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

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that the safety/relief valves will

perform their intended safety function. The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. The ASME Code requires that these bellows be monitored for failure since bellows failure would defeat the safety function of the safety/relief valve. Other TS SRs require that the safety/relief valve bellows be continuously monitored. It is recognized that it is not feasible to test the safety/relief valve setpoints while the valves are in place or during normal plant operation. No plant instrumentation is used to satisfy this SR, so instrument drift was not analyzed for this Function.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.6.E.1 Safety/Relief Valves

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

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that the bellows monitoring system will perform its intended function. The function is performed by pressure switches, which are provided on the low-pressure side of the machined pressure sensing bellows to detect possible leakage of the bellows on the safety/relief valves. The ASME Code requires that these bellows be monitored for failure since bellows failure would defeat the safety function of the safety/relief valve. Other TS SRs require that the safety/relief valve bellows be continuously monitored.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 3.6.H Snubbers

All safety related snubbers are required to be operable whenever the supported system is required to be operable. The Limiting Conditions for Operation are based on ensuring that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. The snubbers installed on non-safety related

systems are excluded from this inspection program only if their failure or the failure of the system on which they are installed would not have an adverse effect on any safety-related system.

The service life of a snubber is evaluated via manufacturer input and through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

The following SRs were evaluated relative to extending their respective test intervals. These SRs ensure the availability of safety functions that respond to plant transients and design basis events. Potential time-based considerations, such as failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are detailed below:

SR 4.6.H.1 Snubber Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible or accessible) may be inspected independently according to the schedule determined by Table 4.6-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.6-1. The initial inspection interval for new types of snubbers shall be established at 18 months +25%.

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that safety related snubbers that are required to be operable, whenever the supported system is required to be operable, are operable.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. The initial inspection interval for new types of snubbers shall be established at 24 months + 25%. This TS SR visual inspection verifies the snubber has no visible indications of damage or impaired operability, attachments to the foundation or supporting structure are functional, and fastners for the attachment of the snubber anchorage are functional. Other TS SRs require a hydraulic snubber Functional Test to verify activation is achieved within the specified range of velocity or acceleration in both tension and compression, and snubber bleed, or release rate, where required, is within the specified range in

compression or tension. Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.6.H.3 Snubber Functional Test

> **Functional testing of snubbers shall be conducted at least once per 18 months +25% during cold shutdown. Ten percent of the total number of each brand of snubber shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria in Specification 4.6.H.4 below, an additional ten percent of that brand shall be functionally tested until no more failures are found or all snubbers of that brand have been tested.**
>
> **The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of the snubbers.**
>
> **In addition to the regular sample and specified re-samples, snubbers which failed the previous functional test shall be retested during the next test period if they were reinstalled as a safety-related snubber. If a spare snubber has been installed in place of a failed safety related snubber, it shall be tested during the next period.**
>
> **If any snubber selected for functional testing either fails to lockup or fails to move (i.e. frozen in place) the cause shall be evaluated and if caused by manufacturer or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.**

The surveillance test interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that safety related snubbers that are required to be operable, whenever the supported system is required to be operable, are operable.

Functional testing of the snubbers shall be conducted at least once per 24 months + 25% during cold shutdown. TS SRs require visual inspections to verify the

snubber has no visible indications of damage or impaired operability, attachments to the foundation or supporting structure are functional, and fastners for the attachment of the snubber anchorage are functional. TS SR 4.6.H.4 requires a hydraulic snubber Functional Test to verify activation is achieved within the specified range of velocity or acceleration in both tension and compression, and snubber bleed, or release rate, where required, is within the specified range in compression or tension.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.6.H.6 Snubber Installation and Maintenance Records

> **The installation and maintenance records for each safety related snubber shall be reviewed at least once every 18 months to verify that the indicated service life will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded, the snubber service life shall be re-evaluated or the snubber shall be replaced or reconditioned to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement, or reconditioning shall be indicated in the records.**

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that safety related snubbers that are required to be operable, whenever the supported system is required to be operable, are operable.

The installation and maintenance records for each safety related snubber shall be reviewed at least once every 24 months to verify that the indicated service life will not be exceeded prior to the next scheduled service life review. TS SRs require visual inspections to verify the snubber has no visible indications of damage or impaired operability, attachments to the foundation or supporting structure are functional, and fastners for the attachment of the snubber anchorage are functional. TS SR 4.6.H.4 requires a hydraulic snubber Functional Test to verify activation is achieved within the specified range of velocity or acceleration in both tension and compression, and snubber bleed, or release rate, where required, is within the specified range in compression or tension.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 4.7 Containment Systems

The Primary Containment System, consisting of a steel inverted lightbulb-shaped drywell, a steel doughnut-shaped pressure suppression chamber and interconnecting vent pipes, provides the first containment barrier surrounding the reactor vessel and reactor primary system. Any leakage from the Primary Containment System is to the Secondary Containment System, which consists of the reactor building, the plant Standby Gas Treatment System and the plant main stack. The integrated plant containment system and its associated engineered safety features are designed so that off-site doses resulting from postulated design basis accidents are well below the reference values stated in 10 CFR 100.

The primary containment system is designed, fabricated and erected to accommodate, without failure, the pressures and temperatures resulting from or subsequent to the double-ended rupture or equivalent failure of any coolant pipe within the primary containment. Provisions are made for the removal of energy from within the primary containment and/or such other measures as may be necessary to maintain the integrity of the primary containment system as long as necessary following the various postulated design basis loss-of-coolant accidents.

The primary performance objectives of the primary containment system are to provide a barrier that, in the event of loss-of-coolant accident, controls the release of fission products to the secondary containment and to rapidly reduce the pressure in the containment resulting from the loss-of-coolant accident.

One of the basic purposes of the primary containment system is to provide a minimum of one protective barrier between the reactor core and the environmental surroundings subsequent to an accident involving failure of the piping components of the reactor primary system. To fulfill its role as a barrier the primary containment is designed to remain intact before, during and after any design basis accident of the process system installed either inside or outside the primary containment. The process system and the primary containment are considered as separate systems, but where process lines penetrate the containment the penetration design achieves the same integrity as the primary containment structure itself. The process line isolation valves are designed to achieve the containment function inside the process lines when required.

Secondary containment is a controlled volume within the Reactor Building. The primary safeguards functions of the secondary containment are to minimize ground level release of airborne radioactive materials and to provide for controlled, filtered and elevated release of the secondary containment atmosphere under postulated

design basis accident conditions. Most of the Reactor Building is part of the secondary containment and the Reactor Building provides the structural integrity of the secondary containment.

A plant Standby Gas Treatment System is provided to filter the secondary containment ventilation exhaust and discharge it to the off-gas vent stack during plant secondary containment system isolation conditions.

The following SRs were evaluated relative to extending their respective test intervals. These SRs ensure the availability of safety functions that respond to plant transients and design basis events. Potential time-based considerations, such as failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are detailed below:

SR 4.7.A.1 Suppression Pool Volume and Temperature

c. A visual inspection of the suppression chamber interior including water line regions and the interior painted surfaces above the water line shall be made at each refueling outage.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The interiors of the drywell and suppression chamber are painted to prevent corrosion. This inspection of the paint during each refueling outage assures the paint is intact and is not deteriorating. Operating experience shows this component routinely passes its SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.7.A.4 Pressure Suppression Chamber-Drywell Vacuum Breakers

a. Operability and full closure of the drywell-suppression chamber vacuum breakers shall be verified by performance of the following:

(2) Once each operating cycle, drywell to suppression chamber leakage shall be demonstrated to be less than that equivalent to a one-inch diameter orifice and each vacuum breaker shall be visually inspected.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures the leakage between the drywell and

suppression chamber is within limits. This leak test is conservatively designed to demonstrate that leakage is less than about 3% of the maximum allowable. Operating experience shows these components routinely pass the SR when performed at the 18-month interval. Extension of this SR interval is further justified because of the more frequent testing required by other TS SRs, and the system is designed to be single failure proof.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.7.A.4 Pressure Suppression Chamber-Drywell Vacuum Breakers

a. Operability and full closure of the drywell-suppression chamber vacuum breakers shall be verified by performance of the following:

(3) Once each operating cycle, vacuum breaker position indication and alarm systems shall be calibrated and functionally tested.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures the availability of safety functions that respond to plant transients and design basis events. Limit switches perform this Function. Limit switches are mechanical devices that require mechanical adjustment only; drift is not applicable to these devices. The capacity of the drywell vacuum breakers is sufficient to limit the pressure differential between the suppression chamber and drywell during post-accident cooling operations to less than the design limit. Operating experience shows these components routinely pass the SR when performed at the 18-month interval. Extension of this SR interval is further justified because of the more frequent testing required by other TS SRs, and the system is designed to be single failure proof.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.7.A.4 Pressure Suppression Chamber-Drywell Vacuum Breakers

a. Operability and full closure of the drywell-suppression chamber vacuum breakers shall be verified by performance of the following:

(4) Once each operating cycle, the vacuum breakers shall be tested to determine that the force required to open each

valve from fully closed to fully open does not exceed the equivalent to 0.5 psid acting on the suppression chamber face of the valve disc.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. Demonstration of vacuum breaker opening force is necessary to ensure the vacuum breaker full open differential pressure of ≤ 0.5 psid is valid. Vacuum breaker valves between the suppression chamber and the drywell provide the capability to vent the torus air space gases back to the drywell in order to assure pressure equilibrium between the compartments. Operating experience shows these components routinely pass the SR when performed at the 18-month interval. Extension of this SR interval is further justified because of the more frequent testing required by other TS SRs, and the system is designed to be single failure proof.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.7.B.2 Standby Gas Treatment System - Performance Requirement Tests

a. At least once per 720 hours of system operation; or once per operating cycle, but not to exceed 18 months, whichever occurs first; or following painting, fire, or chemical release in any ventilation zone communicating with the system while the system is operating that could contaminate the HEPA filters or charcoal adsorbers, perform the following:

(1) In-place DOP test the HEPA filter banks.

(2) In-place test the charcoal adsorber banks with halogenated hydrocarbon tracer.

(3) Remove one carbon test sample from the charcoal adsorber in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978. Subject this sample to a laboratory analysis to verify methyl iodide removal efficiency.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that filters provided in the system to remove radioactive particulates and the charcoal adsorbers that are provided to remove radioactive halogens are functioning properly. Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Extension of this SR interval is further justified because of the more frequent testing required by other TS SRs, and the system is designed to be single failure proof.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

This SR extension also takes an exception to the requirements of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978. This exception to the Regulatory Guide states that sample removal for laboratory testing in accordance with the Regulatory Guide shall be at least once per 24 months instead of once per 18 months.

SR 4.7.B.2 Standby Gas Treatment System - Performance Requirement Tests

b. Once per operating cycle the operability of inline heater at nominal rated power shall be verified for each standby gas treatment system.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that there is sufficient flow across the combined filters and the heaters that reduce relative humidity are functioning properly. Operating experience shows these components routinely pass the SR when performed at the 18-month interval. Extension of this SR interval is further justified because of the more frequent testing required by other TS SRs, and the system is designed to be single failure proof.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

In addition, this TS SR has been renumbered to separate 4.7.B.2.b, 4.7.B.2.c and 4.7.B.2.d. This is an administrative change required to allow the performance of the 4.7.B.2.c SR on a once per 3-month basis and maintain the sequential numbering of the Monticello TS SRs.

SR 4.7.B.2 Standby Gas Treatment System - Performance Requirement Tests

d. At least once per operating cycle, automatic initiation of each standby gas treatment system circuit shall be demonstrated.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that, whenever secondary containment isolation conditions exist, a small negative pressure to minimize ground level escape of airborne radioactivity is maintained. Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Extension of this SR interval is further justified because of the more frequent testing required by other TS SRs, and the system is designed to be single failure proof.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

In addition, this TS SR has been renumbered to separate 4.7.B.2.b, 4.7.B.2.c and 4.7.B.2.d. This is an administrative change required to allow the performance of the 4.7.B.2.c SR on a once per 3-month basis and maintain the sequential numbering of the Monticello TS SRs.

SR 4.7.C.1 Secondary Containment surveillances shall be performed as indicated below:

a. Secondary containment capability to maintain at least a ¼ inch of water vacuum under calm wind (u < 5 mph) conditions with a filter train flow rate of ≤ 4,000 scfm, shall be demonstrated at each refueling interval prior to refueling. If calm wind conditions do not exist during this testing, the test data is to be corrected to calm wind conditions.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that the exfiltration rate, which is almost directly proportional to the initial in-leakage rate for a negative building pressure, is maintained within the post-accident 10 CFR 100 dose limits. Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.7.C.1 Secondary Containment surveillances shall be performed as indicated below:

b. Verification that each automatic damper actuates to its isolation position shall be performed:

(1) Each refueling outage.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that both the inlet and outlet ventilation ducts of the secondary containment can be isolated when high radiation levels are detected in the reactor building ventilation plenum or in the area of the fuel pool. Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.7.D.1 Primary Containment Isolation Valves (PCIVs) – Automatic isolation valve surveillance shall be performed as follows:

a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The SR Table 4.2.1 surveillances overlap this SR to provide complete testing of the system function. The interval was based upon the operating cycle, because it is prudent that this SR be performed only during a unit shutdown since isolation of penetrations eliminates cooling water flow and disrupts the normal operation of many critical components. These PCIVs, including the actuating logic, are designed to be single failure proof and, therefore, the PCIVs are highly reliable. Operating experience shows that these components routinely pass this SR when performed at the 18-month interval.

Based upon the testing of the valves, the reliability of the PCIVs, the redundant nature of containment isolation, and the operating experience of the testing the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.7.D.1 Primary Containment Isolation Valves (PCIVs) – Automatic isolation valve surveillance shall be performed as follows:

b. At least once per operating cycle the primary system instrument line flow check valves shall be tested for proper operation.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR requires a demonstration that each reactor instrumentation line excess flow check valve (EFCV) is operable by verifying that the valve reduces flow on an actual or simulated instrument line break condition. This SR ensures the instrumentation line EFCVs will perform as designed. The interval is based upon the need to perform this SR under shutdown conditions to reduce the potential for an unplanned transient if the SR were performed with the reactor at power. Operating experience shows that these components routinely pass this SR when performed at the 18-month interval. The operational mechanism for an EFCV is not subject to drift or other time-based changes affected by the change to a 24-month cycle.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 4.9 Auxiliary Electrical Systems

The auxiliary power system consists of power sources, distribution equipment, instrumentation and control and utilization devices.

Provisions for loss of power have been made in the design. The multiplicity of sources feeding the auxiliary buses, the redundancy of transformers and buses within the plant, and the division of critical loads between buses yields a system that has a high degree of reliability. The physical separation of buses and service components are designed to limit or localize the consequences of electrical faults

or mechanical accidents occurring at any point in the system. Cables and components of redundant circuits are physically separated by means of concrete walls, floors or space, and barriers to assure maximum independence of redundant channels. Cables of 4160 VAC circuits are principally installed in conduit. Protection for 4160 VAC circuits is provided by magnetic air circuit breakers.

Cables of 480 VAC power circuits and 125 VDC control circuits are installed in punched metal type trays and conduits. The current carrying capacity of all cables is conservatively calculated to preclude the possibility of thermal overloading.

The following SRs were evaluated relative to extending their respective test intervals. These SRs ensure the availability of safety functions that respond to plant transients and design basis events. Potential time-based considerations, such as failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are detailed below:

SR 4.9.B.3.a.2) At least once each Operating Cycle during shutdown simulate a loss of offsite power in conjunction with an ECCS actuation test signal, and:

(a) Verify de-energization of the emergency busses and load shedding from the emergency busses.

(b) Verifying diesel starts from ambient conditions on the auto-start signal and is ready to accept emergency loads within ten seconds, energizes the emergency busses with permanently connected loads, energizes the auto-connected emergency loads in proper time sequence, and operates for greater than 5 minutes while its generator is loaded with emergency loads.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This surveillance requirement demonstrates the diesel generator operation during a loss of offsite power actuation test signal in conjunction with an ECCS initiation signal. Testing that adequately shows the capability of the diesel generator system to demonstrate the ability to connect and energize loads instead of actually performing these functions is acceptable. Extending the surveillance interval for this surveillance requirement is acceptable for the following reasons:

1. During the operating cycle the diesel generators are subjected to operational testing once every month. This testing provides confidence of diesel generator operability and the capability to perform its intended function. The testing will provide prompt identification of any substantial diesel degradation or failure.

2. Diesel generators are infrequently operated, usually only to satisfy a surveillance requirement; thus, the risk of wear-related degradation is minimal.

3. Diesel generator attributes subject to degradation due to aging, such as fuel oil quality, are subject to the requirements for replenishment and testing.

4. Historical testing and surveillance testing during operation prove the ability of the diesel generators to start and operate under various load conditions.

The portions of the tests not directly associated with the functioning of the diesel and breaker movement are equivalent to a Logic System Functional Test. For these logic tests, Reference 1 documents the following conclusion:

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical failure, increasing the Logic System Functional Test interval represents no significant change in the overall safety system unavailability."

Extending the testing interval for this SR is also acceptable because the network, including the actuation logic, is designed to be single failure proof and is highly reliable.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

SR 4.9.B.4 Station Battery System

c. Every refueling outage, the station batteries shall be subjected to a rated load discharge test. Determine specific gravity and voltage of each cell after the discharge.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. The purpose of this test is to ensure the availability of necessary power to engineered safety feature systems from Class 1E battery

sources. Two divisions of batteries are required for the mitigation of an accident in the event of a loss of offsite power coincident with the worst-case single failure. Extending the surveillance interval for this surveillance requirement is acceptable for the following reasons:

1. The design, in conjunction with TS requirements, which limit the extent and duration of inoperable DC sources, provides substantial redundancy in DC sources.

2. Battery specific gravity and voltage of the pilot cell and temperature of the adjacent cells and overall battery voltage is measured every week and will provide prompt identification of any substantial battery degradation or failure.

3. Battery voltage measurements of each cell to the nearest 0.01 volt, specific gravity of each cell, and temperature of every fifth cell are monitored every three months during the operating cycle. Therefore, any substantial degradation of the subject components will be evident prior to the scheduled performance of these tests.

4. Monticello tracks all load additions and deletions and verifies that any changes to loading are within the capacity of the batteries through the normal engineering design process. Battery loading calculations are maintained based upon the as-built configuration, which consider the effects of planned design changes.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 4.13 Alternate Shutdown System

The alternate shutdown system (ASDS) provides control of the minimum necessary systems, once the transfer switches are activated, to achieve safe shutdown.

The alternate safe shutdown system is required for a 10 CFR 50, Appendix R event in the control room and/or cable spreading room. The system utilizes existing systems and equipment and a remote ASDS control panel. The ASDS control panel is located on the third floor of the emergency filtration train (EFT) building adjacent to the turbine building and the control room. This ASDS panel is safety grade (Class 1E) and provides the controls, AC circuitry and instrument

readouts for the operator to safely shutdown the plant at a centralized location assuming a fire in the control room or the cable spreading room.

The following surveillance requirements were evaluated relative to extending their respective testing intervals. These surveillance requirements ensure the availability of safety functions that respond to plant transients and design basis events. Potential time-based considerations, such as instrument drift, failures types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extensions are discussed below:

SR 4.13.A Alternate Shutdown System

1. **Switches on the alternate shutdown system panel shall be functionally tested once per operating cycle.**
2. **The alternate shutdown system panel master transfer switch shall be verified to alarm in the control room when unlocked once per operating cycle.**

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This surveillance requirement ensures that the Alternate Shutdown System (ASDS) instrumentation will function as designed during an analyzed event and that during testing the control room operators are aware that control of the required instrumentation has been transferred to the ASDS panel. This test is essentially a Logic System Functional Test for the transfer circuits associated with shifting indication and control from the main control room to the ASDS panel. The justification for extending Logic System Functional Test is valid for the extension of this surveillance requirement. As stated in Reference 1:

> "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical failure, increasing the Logic System Functional Test interval represents no significant change in the overall safety system unavailability."

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 4.14 Accident Monitoring Instrumentation

The primary purpose of the accident monitoring instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions after design basis events. The operability of the accident monitoring instrumentation ensures there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident.

The following surveillance requirements were evaluated relative to extending their respective testing intervals. These surveillance requirements ensure the availability of safety functions that monitor plant transients and design basis events. Potential time-based considerations, such as failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The results of these evaluations and an explanation of how the results justify the surveillance interval extension are discussed below:

Table 4.14.1 Accident Monitoring Instrumentation
Function 1 Reactor Vessel Fuel Zone Water Level Monitor

Reactor vessel water level is measured by differential pressure transmitters and indicated and recorded in the control room. Reactor vessel water level is measured to provide information which can be used to assure that the core is covered and that the separators are not flooded. The use of the level signals in the reactor protection system and the feedwater control system assures that the reactor is shut down automatically if the proper level is not maintained.

SR 4.14.1 Perform Channel Calibration

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. A separate drift evaluation was not performed for the accident monitoring instrumentation based upon the design accuracy requirements and equipment history of the instruments. The following discussion supports this conclusion.

The accident monitoring function is supported by a combination of process transmitters, indicators and recorders. These components differ from other TS instruments in that they are not associated with a single action point but may be required to function anywhere within their range capability. An additional difference, based upon the time of function, is the process and environmental conditions that may be present when the instruments are required. Trip devices function during the first several seconds of an accident (normally prior to any significant environment changes) to prevent or mitigate the consequences. The

design requirement for these devices considers the environmental conditions as well as the specific process conditions associated with the protective trip. The accident monitoring instrumentation devices must maintain their function after the accident has occurred and track the progress of the event and event mitigation over a long period of time. Accident monitoring instrumentation is designed to operate in a wide variety of environments (ranging from normal to high temperature, high radiation and high humidity) and to maintain functionality. Accident instrumentation may also be expected to monitor the process over a wide range of process conditions. However, these instruments are not expected to function with the same high degree of accuracy demanded of accident detection and mitigation trip devices. The accident monitoring instrumentation devices are expected to maintain sufficient accuracy to detect trends or the existence or non-existence of a condition within wider boundaries (e.g., Is there water in the RPV?). The specific expectations for accuracy of accident instruments in the various environments are defined in a Monticello Specific EQ Instrument Accuracy Requirements document.

The accident monitoring instrumentation is designed with a high degree of reliability. The indicators and recorders used for accident monitoring are compared with other channels of instruments measuring the same variable where possible to verify Operability for normal conditions. A sensor check is also required once per month. These tests verify that the indication and recording instruments are acceptable and operating within established tolerances. The primary error contributor for the transmitters during normal operations is drift. However, for accident monitoring conditions the major errors are associated with the change in process conditions and the change in environmental conditions. These process and environmental errors are in most cases orders of magnitude larger than the errors associated with drift. A drift analysis will not verify that these devices will maintain acceptable accuracy for the instruments during accident monitoring conditions. The accident analyses have adequate margin to account for instrumentation errors based on:

1. Accident monitoring instrumentation is designed to be highly reliable and accurate.

2. Instrument sensor checks are performed on a more frequent basis.

3. That for most accident monitoring instrumentation drift is not a primary error contributor during the time that operation of the equipment is required.

4. The fact that neither the TS nor the EOPs have specific accuracy requirements.

Therefore, a drift calculation for these instruments is not necessary and a review of the surveillance test history provides an acceptable method to determine if the

instrument calibration interval can be extended to a 24-month operating cycle, including the 25% grace period, for a maximum interval of 30 months.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

Table 4.14.1 Accident Monitoring Instrumentation
Function 2 Safety/Relief Valve Position (Pressure Switches)

Safety/relief valve (SRV) position indication is provided by a differential pressure transmitter and an analog trip unit that monitors the steam pressure in each discharge pipe. The trip unit will indicate the valve is open by lighting a light and alarming in the Control Room when a significant change in pressure in the SRV discharge line is sensed. These lights and alarms are the primary indication used by the operator during an accident. There are also indicating lights that indicate the state of the air actuator solenoid for each safety/relief valve in the control room. Therefore, the accident monitoring instrumentation specification addresses specifically this portion of the instrument channel.

SR 4.14.1 Perform Channel Calibration.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. A separate drift evaluation was not performed for the accident monitoring instrumentation based upon the design of the accident monitoring instruments, the specific surveillance requirements, the availability of alternate indication of SRV position and equipment history. (See the justification for Function 1 of Table 4.14.1.)

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

Table 4.14.1 Accident Monitoring Instrumentation
Function 3 Safety/Relief Valve Position (Thermocouples)

For the safety/relief valves there are thermocouples installed in each of the discharge pipes, which give an indication of increased temperature, which means that, a safety/relief valve is opening or leaking. There is a multiple channel

recorder, in the Control Room, which receives output from the thermocouples and is alarmed in the Control Room. There are also indicating lights, which indicate the state of the air actuator solenoid for each safety/relief valve, in the control room. The recorder is the primary indication used by the operator during an accident. Therefore, the accident monitoring instrumentation specification addresses specifically this portion of the instrument channel. The setpoint is set high enough to account for normal and expected deviations in tailpipe temperatures but low enough to provide early detection of a sudden step increase in temperature indicative of an SRV opening. This is also essentially a qualitative evaluation of SRV position.

SR 4.14.1 Perform Channel Calibration

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. A separate drift evaluation was not performed for the accident monitoring instrumentation based upon the design of the accident monitoring instruments, the specific surveillance requirements, the availability of alternate indication of SRV position and equipment history. (See the justification for Function 1 of Table 4.14.1.)

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

Table 4.14.1 Accident Monitoring Instrumentation
Function 4 Drywell Wide Range Pressure Monitors

The drywell wide range pressure monitoring system continuously indicates and records the drywell pressure in the control room. This variable is used to verify and provide long-term surveillance of ECCS functions. The wide-range drywell pressure monitors provide the operator with sufficient information to assess the status of accident mitigation and drywell conditions. Two wide range pressure signals are transmitted from separate pressure transmitters and are continuously provided in the control room. These recorders are the primary indication used by the operator during an accident. Therefore, the accident monitoring instrumentation specification addresses specifically this portion of the instrument channel.

SR 4.14.1 Perform Channel Calibration

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the

25% grace period. A separate drift evaluation was not performed for the accident monitoring instrumentation based upon the design of the accident monitoring instruments and equipment history. (See the justification for Function 1 of Table 4.14.1.)

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

Table 4.14.1 Accident Monitoring Instrumentation
Function 5 Suppression Pool Wide Range Level Monitors

The suppression pool wide range level monitoring system continuously indicates and records the water level of the suppression pool in the control room. This variable is used to verify and provide long-term surveillance of ECCS functions. The wide-range suppression pool water level monitors provide the operator with sufficient information to assess the status of the water supply to ECCS. Two wide range water level signals are transmitted from separate level transmitters and are continuously provided in the control room. These recorders are the primary indication used by the operator during an accident. Therefore, the accident monitoring instrumentation specification addresses specifically this portion of the instrument channel.

SR 4.14.1 Perform Channel Calibration

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. A separate drift evaluation was not performed for the accident monitoring instrumentation based upon the design of the accident monitoring instruments and equipment history. (See the justification for Function 1 of Table 4.14.1.). TS SR requires that the suppression pool volume is checked once each day this check verifies the proper operation of the wide range level monitors during normal conditions. This ensures that the suppression pool level monitoring system is functioning and capable of meeting its post accident indication requirements.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

Table 4.14.1 Accident Monitoring Instrumentation
Function 6 Suppression Pool Temperature

The Suppression Pool Temperature Monitoring System (SPOTMOS) is an integral part of the overall post-accident monitoring capability of the plant. The SPOTMOS consists of two independent and redundant divisions. The safety function of the SPOTMOS is to provide the plant operator with reliable information on the suppression pool temperature such that the plant can be operated within Technical Specification limits. Each division of the SPOTMOS is physically separated from the other, and either division is capable of providing an accurate measure of the suppression pool bulk temperature. These recorders are the primary indication used by the operator during an accident. The accident monitoring instrumentation specification addresses specifically the portion of the instrument channel measuring the suppression pool bulk temperature.

SR 4.14.1 Perform Channel Calibration

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. A separate drift evaluation was not performed for the accident monitoring instrumentation based upon the design of the accident monitoring instruments and equipment history. (See the justification for Function 1 of Table 4.14.1.). There is also a TS requirement to check the suppression pool temperature once each day. This check verifies the proper operation of the SPOTMOS during normal operation and any significant deviation from normal conditions would be detected. This ensures that the suppression pool temperature monitoring system is functioning and capable of meeting its post accident indication requirements.

These recorders were recently replaced for greater reliability in the performance of this Function. The new instrument drift-monitoring program will ensure the drift value used for this application is conservative relative to actual recorder performance.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

Table 4.14.1 Accident Monitoring Instrumentation
Function 7 Drywell High Range Radiation Monitors

The Containment High Radiation Monitoring System is comprised of sensor units that are located in the drywell. Each sensor is an ionization chamber with an internal U-234 source for operation verification. Increasing gamma radiation

increases the rate of ionization with proportional increases in the signal current outputs to the readout module. The readout modules convert the output current from the detectors to readout of radiation. There are two trip points on each unit, indicating Hi Radiation and Hi Hi Radiation. Both readout modules alarm to an annunciator and drive recorders. These recorders are the primary indication used by the operator during an accident. The accident monitoring instrumentation specification addresses specifically this portion of the instrument channel.

SR 4.14.1 Perform Channel Calibration

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. A separate drift evaluation was not performed for the accident monitoring instrumentation based upon the design of the accident monitoring instruments, equipment history and the following discussion.

The major error contributor for radiation monitors is the accuracy of the detector and the calibration sources. The decay curves and the detector sensitivity may be from 12% to 30% accurate. This accuracy far overshadows the accuracy of the electronic signal conditioning circuit. The drift of the electronic circuit does not provide a measure of function performance over time between calibrations. This is substantiated by the American National Standards Institute (ANSI) N42.18 acceptance criteria of ± 20% which also recognizes that ± 30% for alarm points satisfies the accuracy needed for Emergency Plan (E-plan) decisions and license requirements. See the justification for Function 1 of Table 4.14.1 for additional post accident discussions.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

Table 4.14.1 Accident Monitoring Instrumentation
Function 8 Offgas Stack Wide Range Radiation Monitors

Radiation monitors are provided in the plant main stack as a back-up detection of high activity release. Radiation levels, in excess of the allowable "instantaneous" release rate, alarm in the control room and isolate the hold-up line. These recorders are the primary indication used by the operator during an accident. The accident monitoring instrumentation specification addresses specifically this portion of the instrument channel.

SR 4.14.1 Perform Channel Calibration

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. A separate drift evaluation was not performed for the accident monitoring instrumentation based upon the design of the accident monitoring instruments, equipment history and the following discussion.

The major error contributor for radiation monitors is the accuracy of the detector and the calibration sources. In the case of the calibration sources, normally multiple readings are required and an average reading is used to confirm operation. The decay curves and the detector sensitivity may be from 12% to 30% accurate. This accuracy far overshadows the accuracy of the electronic signal conditioning circuit. The drift of the electronic circuit does not provide a measure of function performance over time between calibrations. This is substantiated by the ANSI N42.18 acceptance criteria of ± 20% which also recognizes that ± 30% for alarm points satisfies the accuracy needed for E-plan decisions and license requirements. See the justification for Function 1 of Table 4.14.1 for additional post accident discussions.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

Table 4.14.1 Accident Monitoring Instrumentation
Function 9 Reactor Building Wide Range Radiation Monitors

The reactor building wide range radiation monitoring system continously monitors radioactivity in the Reactor Building ventilation exhaust and provides a permanent record of the observed radiation levels. These recorders are the primary indication used by the operator during an accident. The accident monitoring instrumentation specification addresses specifically this portion of the instrument channel.

SR 4.14.1 Perform Channel Calibration

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. A separate drift evaluation was not performed for the accident monitoring instrumentation based upon the design of the accident monitoring instruments, equipment history and the following discussion.

The major error contributor for radiation monitors is the accuracy of the detector and the calibration sources. In the case of the calibration sources, normally multiple readings are required and an average reading is used to confirm

operation. The decay curves and the detector sensitivity may be from 12% to 30% accurate. This accuracy far overshadows the accuracy of the electronic signal conditioning circuit. The drift of the electronic circuit does not provide a measure of function performance over time between calibrations. This is substantiated by the ANSI N42.18 acceptance criteria of ± 20% which also recognizes that ± 30% for alarm points satisfies the accuracy needed for E-plan decisions and license requirements. See the justification for Function 1 of Table 4.14.1 for additional post accident discussions.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

4.17 Control Room Habitability

The main control room ventilation air radiation inlet monitors are designed to automatically prevent the injection of contaminated air into the control room resulting from a steam line break or leakage, which bypasses secondary containment during a LOCA.

This monitoring system assures control room habitability so that control room operators will be adequately protected against the effects of accidental release of radioactivity into the environment.

The radiation detectors are sufficiently sensitive to transfer the air handling system to the filtration/pressurization mode before radiation levels in the control room become excessive. The filtration units providing make-up air for establishing positive pressure in the control room are equipped with High Efficiency Particulate Absolute (HEPA) filters and charcoal adsorbers.

The following SRs were evaluated relative to extending their respective test intervals. These SRs ensure the availability of safety functions that respond to plant transients and design basis events. Potential time-based considerations, such as failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval extension are detailed below:

SR 4.17.B Control Room Emergency Filtration System

2. Performance Requirement Test

The in-place performance testing of HEPA filter banks and charcoal adsorber banks shall be conducted in accordance with Sections 10 and 11 of ASME N510-1989. The carbon sample test shall be conducted in accordance with ASTM D 3803-1989. Sample removal shall be in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978.

a. At least once per operating cycle, but not to exceed 18 months; or following painting, fire, or chemical release while the system is operating that could contaminate the HEPA filters or charcoal adsorbers, perform the following:

(1) In-place DOP test the HEPA filter banks.

(2) In-place test the charcoal adsorber banks with halogenated hydrocarbon tracer.

(3) Remove one carbon test sample from each charcoal adsorber bank. Subject this sample to a laboratory analysis to verify methyl iodide removal efficiency.

(4) Initiate from the control room 1000 cfm (± 10%) flow through both trains of the emergency filtration treatment system.

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that control room personnel are protected against radioactive gases so that the plant can be safely shut down under design basis accident condition.

The frequency of tests and sample analysis is necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber tray is installed which can accommodate a sufficient number of representative adsorber sample modules for estimating the amount of penetration the system adsorbs through its life. Sample modules are installed with the same batch characteristics as the system adsorbent and are withdrawn for the methyl iodide removal efficiency tests. Each module withdrawn is replaced or blocked off. In-place testing procedures utilize applicable sections of ASME N510-1989 with exceptions described in Section 6.7 of the USAR. If test results are unacceptable, all adsorbent in the train is replaced. Any HEPA filters found defective are replaced. Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

This TS SR extension takes an exception to the requirements of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978. This exception to the Regulatory Guide will state that sample removal for laboratory testing in accordance with the Regulatory Guide shall be at least once per 24 months instead of once per 18 months.

SR 4.17.B Control Room Emergency Filtration System

2. Performance Requirement Test

The in-place performance testing of HEPA filter banks and charcoal adsorber banks shall be conducted in accordance with Sections 10 and 11 of ASME N510-1989. The carbon sample test shall be conducted in accordance with ASTM D 3803-1989. Sample removal shall be in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978.

c. At least once per operating cycle, but not to exceed 18 months, the following conditions shall be demonstrated for each emergency filtration system train:

(1) Pressure drop across the combined filters of each train shall be measured at 1000 cfm (± 10%) flow rate.

(2) Operability of inlet heater at nominal rated power shall be verified.

(3) Verify that on a simulated high radiation signal, the train switches to the pressurization mode of operation and the control room is maintained at a positive pressure with respect to adjacent areas at the design flow rate of 1000 cfm (± 10 %).

The surveillance interval for this SR, as applied to this Function, is being increased from 18 months to 24 months, for a maximum interval of 30 months, including the 25% grace period. This SR ensures that control room personnel are protected against radioactive gases so that the plant can be safely shutdown under design basis accident conditions. A pressure drop across the combined HEPA filters and charcoal adsorbers of less than or equal to 8 inches of water at the system design flow rate indicates that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Operating experience shows these components routinely pass the SR when performed at the 18-month interval.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

This TS SR extension takes an exception to the requirements of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978. This exception to the Regulatory Guide will state that sample removal for laboratory testing in accordance with the Regulatory Guide shall be at least once per 24 months instead of once per 18 months.

BASES FOR CHANGE REQUEST FOR TECHNICAL SPECIFICATIONS AND SURVEILLANCE REQUIREMENTS INTERVAL REDUCTION

The purpose of this submittal is to request an amendment to Appendix A of Operating License DPR-22 for the Monticello Nuclear Generating Plant (MNGP). This license amendment request proposes revisions to the Monticello Technical Specification (TS) and Surveillance Requirements (SR) intervals, as identified. A description of, and the justification for, each proposed TS change is provided below:

TS 3.2.C Control Rod Block Actuation

The control rod block functions are provided to prevent excessive control rod withdrawal so that Minimum Critical Power Ratio (MCPR) remains above the Safety Limit. The trip logic for this function is 1 out of n; e.g., any trip on one of the six Average Power Range Monitors (APRM's), eight Intermediate Range Monitors (IRM's), or four Source Range Monitors (SRM's) will result in a rod block. The minimum instrument channel requirements for the IRM and rod block monitor (RBM) may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. The circuitry is arranged to initiate a rod block that prevents rod withdrawal regardless of the position of the mode switch. During high-power operation, the RBM provides protection for control rod withdrawal error events. During low-power operation, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch – Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent critically.

The following SR was evaluated relative to reducing its respective testing interval. This SR ensures the availability of safety functions that respond to plant transients and design basis events. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval reduction is discussed below:

TS Table 3.2.3 Control Rod Block Instrumentation
Function 5 Scram Discharge Volume – Water Level High

The circuitry is arranged to initiate a rod block that prevents rod withdrawal regardless of the position of the mode switch for scram discharge volume high

water level. This assures that no control rod is withdrawn unless enough capacity is available in either scram discharge volume to accommodate a scram. The east scram discharge volume receives the water displaced by the motion of the east control rod drive pistons and the west scram discharge volume receives the water displaced by the motion of the west control rod drive pistons during a scram. Should either scram discharge volume fill up with water to the point where not enough space remains for the water displaced during a scram, control rod movement would be hindered in the event a scram was required. The reactor is scrammed to prevent this situation when the water level in either discharge volume attains a value high enough to verify that the volume is filling up yet low enough to ensure that the remaining capacity in the volume can accommodate a scram.

SR Table 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for these level switches SR, as applied to this Function, is being decreased from 18 months to once per 3 months, for a maximum interval of 115 days, including the 25% grace period. Although the TS Frequency requirement for this test is every Refueling Outage, this test has been performed on a quarterly basis for numerous years.

Thermal switches and associated electronics perform this Function. The thermal switches consist of a thermal probe that provides input to electronic level switches. The calibration is checked quarterly during the Functional Test because it is convient to perform the calibration at this time. Operating experience shows these components routinely pass the SR when performed at the 3-month interval. The statistical data and surveillance test history do not currently support an extension of this SR. Therefore, the SR interval is being reduced to be consistent with current plant practice.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 3.2.H Safety/Relief Valve Low-Low Set Logic

The function of the Safety/Relief Valve (SRV) Low-Low Set System is to minimize the possibility of an SRV reopening with an elevated water leg in its discharge line. The elevated water leg occurs after an SRV closes. The condensing steam in the SRV discharge line creates a vacuum that draws torus water up into the discharge line. Preventing subsequent manual or automatic actuation prior to the water leg receding in the discharge line prevents excessive hydrodynamic loading of the discharge piping and suppression chamber components.

The SRV Low-Low Set System controls the operation of three SRVs and is totally independent of the Automatic Depressurization System. The low-low set protective function is fully automatic and requires no operator action. The low-low set logic is dual divisional.

The following SR was evaluated relative to reducing its respective testing interval. This SR ensures the availability of safety functions that respond to plant transients and design basis events. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval reduction is discussed below:

TS Table 3.2.7 Safety/Relief Valve Low-Low Set Logic
Function 5 Inhibit Timer

Once a low-low set SRV has opened and closed a time delay relay prevents the plant operators or the low-low set logic from immediately re-opening the SRV to allow the water leg in the SRV discharge line to recede. A revision to the Trip Setting is discussed below.

SR 4.2.1 Perform CHANNEL CALIBRATION

The surveillance test interval for this timer SR, as applied to this Function, is being decreased from 18 months to once per 3 months, for a maximum interval of 115 days, including the 25% grace period. Although the TS Frequency requirement for this test is every Refueling Outage, this test has been performed on a quarterly basis for numerous years.

An evaluation of the surveillance interval reduction for the timer was performed, based upon the recommendations of EPRI TR-103335. The timer and associated electronics perform this Function. The calibration is checked quarterly during the Functional Test and it is convient to perform the calibration at this time. The drift for the proposed reduced surveillance interval was appropriately considered in the development of the associated plant setpoints. An evaluation was performed for the reduced surveillance interval relative to the plant Trip Setting for this function that resulted in a revision of the Trip Setting from “10 ± 1 second” to “10 ± 2 seconds.” Operating experience shows these components routinely pass the SR when performed at the 3-month interval.

The statistical data and surveillance test history do not currently support an extension of this SR. Therefore, the SR interval is being reduced to be consistent with current plant practice.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

TS 4.7 Containment Systems

Any leakage from the Primary Containment System is to the Secondary Containment System that consists of the reactor building, the plant Standby Gas Treatment System and the plant main stack. The integrated plant containment system and its associated engineered safety features are designed so that off-site doses resulting from postulated design basis accidents are well below the reference values stated in 10 CFR 100.

Secondary containment is a controlled volume within the Reactor Building. The primary safeguards functions of the secondary containment are to minimize ground level release of airborne radioactive materials, and to provide for controlled, filtered, elevated release of the secondary containment atmosphere under postulated design basis accident conditions. Most of the Reactor Building is part of the secondary containment, and the Reactor Building provides the structural integrity of the secondary containment.

A plant Standby Gas Treatment System is provided to filter the secondary containment ventilation exhaust and discharge it to the off-gas vent stack during plant secondary containment system isolation conditions.

The following SR was evaluated relative to reducing its respective testing interval. This SR ensures the availability of safety functions that respond to plant transients and design basis events. Potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability, were evaluated during this effort. The evaluation results and an explanation of how the results justify the surveillance interval reduction is discussed below:

SR 4.7.B.2 Standby Gas Treatment System - Performance Requirement Tests

c. Once per operating cycle, but not to exceed 18 months, demonstrate that the pressure drop across the combined filters of each standby gas treatment system circuit shall be measured at 3500 cfm (± 10%) flow rate.

The surveillance test interval for this SR, as applied to this Function, is being decreased from, at least once per operating cycle, but not to exceed 18 months, to once per 3 months, for a maximum interval of 115 days, including the 25% grace

period. Although the TS Frequency requirement for this test is once per operating cycle, this test has been performed on a quarterly basis for numerous years.

It is convenient to perform this TS SR on a once per 3 months basis. Operating experience shows these components routinely pass the SR when performed at the 3-month interval. The statistical data and surveillance test history do not currently support an extension of this SR. Therefore, the SR interval is being reduced to be consistent with current plant practice.

Based upon the above discussion the effect, if any, of this proposed change on system availability is minimal.

A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect, if any, of this proposed change on system availability is minimal.

In addition, this TS SR has been renumbered to separate it from 4.7.B.2.b. This is an administrative change required to allow the performance of this SR on a once per 3-month basis and maintain the sequential numbering of the Monticello TS SRs.

References:

1. NRC Safety Evaluation (SE) dated August 2, 1993, relating to the extension of Peach Bottom Atomic Power Station, Units 2 and 3, surveillance interval extension from 18 to 24 months.

2. NRC Safety Evaluation (SE) dated July 15, 1987, letter from A. Thadani (NRC) to T. A. Pickens (NSP), relating to “Technical Specification Improvement Analysis for BWR Reactor Protection System,” for Monticello Nuclear Generating Plant.

MONTICELLO NUCLEAR GENERATING PLANT
LICENSE AMENDMENT REQUEST TO
SUPPORT 24-MONTH FUEL CYCLES

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT

Nuclear Management Company, LLC (NMC) is hereby requesting an amendment to Appendix A of Operating License DPR-22 for the Monticello Nuclear Generating Plant (MNGP). This license amendment request proposes revisions to the Monticello Technical Specification (TS) and Surveillance Requirement (SR) intervals to support the implementation of a 24-month fuel cycle. The NMC is proposing revisions to extend most Monticello TS SR intervals from 18 months to 24 months on the basis of the Monticello 24-Month Surveillance Interval History Study and the Monticello Drift Analysis (Instrumentation and Controls). The NMC is also proposing, for Monticello, revisions to reduce some SR intervals from 18 months to three months on the basis of Monticello 24-Month Surveillance Interval History Study, Monticello Drift Analysis (Instrumentation and Controls), and the plants preferred operational and testing procedures. In addition, NMC is proposing Monticello TS revisions to the Nominal Trip Setpoint's on the basis of Monticello 24-Month Surveillance Interval History Study and Monticello Drift Analysis (Instrumentation and Controls). Proposed administrative revisions to the TS and the TS Bases supporting the preceding areas of revision are also being provided. The TS Bases changes are provided for information only to support the NRC's review of this License Amendment Request.

A. 10 CFR 50.92 Significant Hazards Evaluation

The standards used to determine that a request for amendment does not involve a significant hazards consideration are included in 10 CFR 50.92, which states that operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

Nuclear Management Company, LLC has reviewed the proposed amendment with respect to these three factors, and determined that the

proposed changes do not involve a significant hazard based on the following:

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

   a. Surveillance Testing Interval Extensions

      The proposed Technical Specification (TS) changes involve changes in the surveillance testing to facilitate a change in the operating cycle from 18 months to 24 months. The proposed TS changes do not physically impact the normal operation of the plant, nor do they impact any design or functional requirements of the associated systems. That is, the proposed TS changes neither impact the TS SRs themselves nor the manner in which the surveillances are performed.

      In addition, the proposed TS changes do not introduce any accident initiators, since no accidents previously evaluated relate to the frequency of surveillance testing. Also, evaluations of the proposed TS changes demonstrate that the availability of equipment and systems required to prevent or mitigate the radiological consequences of an accident are not significantly affected because of other, more frequent testing that is performed, the availability of redundant systems and equipment, or the high reliability of the equipment. Since the impact on the systems is minimal NMC has concluded that the overall impact on the plant safety analysis is negligible.

      A historical review of surveillance test results and associated maintenance records indicated that there was no evidence of any failure that would invalidate the above conclusions.

      Therefore, the proposed TS changes do not significantly increase the probability or consequences of an accident previously evaluated.

   b. TS Trip Setting Changes

      Changes are proposed to the Monticello TS Trip Settings. The proposed changes are a result of application of the Monticello Instrument Setpoint Methodology using plant-specific drift values. Application of this methodology results in Trip Setpoints that more accurately reflect total instrumentation loop accuracy, as well as that of test equipment and calculated drift between surveillances. The proposed changes will not result in hardware changes. The instrumentation is not assumed to be initiators of any analyzed events, nor do they impact any design or functional requirements of

the associated systems. Existing operating margins between plant conditions and actual plant setpoints are not significantly reduced due to the proposed changes. The role of the instrumentation is in mitigating and thereby, limiting the consequences of accidents.

The Nominal Trip Setpoints were developed to ensure the design and safety analysis limits are satisfied. The methodology used for the development of the Trip Settings ensures: 1) the affected instrumentation remains capable of mitigating design basis events as described in the safety analysis; and, 2) the results and radiological consequences described in the safety analysis remain bounding. The proposed changes do not alter the plant's ability to detect and mitigate events.

Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

c. Surveillance Testing Interval Reductions

The proposed TS changes involve reductions in the surveillance testing intervals from once per operating cycle or refueling outage to once every three (3) months or once per quarter for the equipment associated with these TS SRs. The shorter intervals are based upon the plant-specific results of a review of the surveillance test history for this equipment. The implementing procedures for these SRs have been performed on a once per three (3) month or once per quarter interval for a number of years, and these changes more accurately reflect actual plant maintenance practices. The proposed, more restrictive TS changes do not physically impact the plant, nor do they impact any design or functional requirements of the associated systems. That is, the proposed TS changes neither degrade the performance of, nor increase the challenges to, any safety system assumed to function in the safety analysis. These proposed TS changes neither impact the TS SRs themselves nor the manner in which the surveillances are performed.

The proposed TS changes do not introduce any accident initiators, since no accident previously evaluated relate to the frequency of surveillance testing. The proposed TS intervals demonstrate that the equipment and systems required to prevent or mitigate the radiological consequences of an accident are continuing to meet the assumptions of the setpoint evaluation on a more frequent basis. Since the impacts on systems are minimal and the assumptions of the safety analyses are maintained, NMC has concluded that the overall impact on the plant safety analysis is negligible.

Therefore, the proposed TS changes do not significantly increase the probability or consequences of any accident previously evluated.

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

   a. Surveillance Testing Interval Extensions

      The proposed TS changes involve changes in the surveillance testing intervals to facilitate a change in the operating cycle length. The proposed TS changes do not introduce any failure mechanisms of a different type than those previously evaluated. There are no physical changes being made to the facility. No new or different equipment is being installed. No installed equipment is being operated in a different manner. As a result no new failure modes are introduced. The SRs themselves, and the manner in which surveillance tests are performed, remain unchanged.

      A historical review of surveillance test results and associated maintenance records indicated that there was no evidence of any failure that would invalidate the above conclusions.

      Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any previously evaluated.

   b. TS Trip Setting Changes

      The proposed changes to the Trip Settings are a result of applying the Monticello Instrument Setpoint Methodology using plant-specific drift values. The application of this methodology does not create the possibility of any new or different kinds of accidents from any accidents previously evaluated. This is based upon the fact that the method and manner of plant operations are unchanged.

      The use of the proposed Trip Setpoints does not impact the safe operation of the plant in that the safety analysis limits are maintained. The proposed changes in Trip Settings involve no system additions or physical modifications to plant systems. The Trip Settings are revised to ensure the affected instrumentation remains capable of mitigating accidents and transients. Plant equipment will not be operated in a manner different from previous operation. Since operational methods remain unchanged and the operating parameters were evaluated to maintain the plant within existing design basis criteria no different type of failure or accident is created.

Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any previously evaluated.

c. Surveillance Testing Interval Reductions

The proposed TS changes involve reductions in the surveillance testing intervals from once per operating cycle or refueling outage to once every three (3) months or once per quarter for the equipment associated with these TS SRs. The shorter intervals are based upon the plant-specific results of a review of the surveillance test history for this equipment. The implementing procedures for these SRs have been performed on a once per three (3) month or once per quarter interval for a number of years and these changes more accurately reflect actual plant maintenance practices. The proposed more restrictive TS changes do not physically impact the plant, nor do they impact any design or functional requirements of the associated systems. That is, the proposed TS changes neither degrade the performance of, nor increase the challenges to, any safety system assumed to function in the safety analysis. These proposed TS changes neither impact the TS SRs themselves nor the manner in which the surveillances are performed.

The proposed TS changes do not introduce any failure mechanism of a different type than those previously evaluated. The proposed changes make no physical changes to the plant. No new or different equipment is being installed. No installed equipment is being operated in a different manner.

A historical review of surveillance test results and associated maintenance records indicate that there is no evidence of any failure that would invalidate the above conclusions.

Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any previously evluated.

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

a. Surveillance Testing Interval Extensions

Although the proposed TS changes result in changes in the interval between surveillance tests, the impact, if any, on system availability is minimal based upon other, more frequent testing that is performed, the existence of redundant systems and equipment or overall system reliability. Evaluations show there is no evidence of any time-dependant failure that would impact system availability.

The proposed changes do not significantly impact the condition or performance of structures, systems and components relied upon for accident mitigation. The proposed TS changes do not physically impact the plant, nor do they impact any design or functional requirements of the associated systems. The proposed changes do not significantly impact any safety analysis assumptions or results.

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

b. TS Trip Setting Changes

The proposed changes do not involve a reduction in a margin of safety. The proposed changes were developed using a Monticello Instrument Setpoint Methodology using plant-specific drift values. This methodology ensures no safety analysis limits are exceeded. The proposed TS changes do not physically impact the plant, nor do they impact any design or functional requirements of the associated systems.

As such, these proposed changes do not involve a reduction in a margin of safety.

c. Surveillance Testing Interval Reductions

The proposed TS changes result in a shorter interval between surveillance tests to ensure the assumptions of the safety analysis are maintained. The impact, if any, on system availability is minimal, as a result of the more frequent testing that is performed. The proposed changes do not significantly impact the condition or performance of structures, systems and components relied upon for accident mitigation. The proposed TS changes do not physically impact the plant, nor do they impact any design or functional requirements of the associated systems. The proposed changes do not significantly impact any safety analysis assumptions or results.

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

B. Environmental Assessment

Nuclear Management Company (NMC) has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, “Criteria for identification of licensing and regulatory actions

requiring environmental assessments." NMC has determined that the proposed changes meet the criteria for a categorical exclusion as set forth in 10 CFR 51.52(c)(9), "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b), "Issuance of amendment." This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or that changes an inspection or surveillance requirement, and the amendment does not result in the following:

(i) A significant hazards consideration,

The proposed amendment does not involve a significant hazard. See the significant hazards consideration determination evaluation.

(ii) A significant change in the type or significant increase in the amounts of any effluent that may be released offsite, or

The proposed amendment is consistent with and does not change the design basis of the plant. The proposed amendment will not result in an increase in power level, will not increase the production of radioactive waste and byproducts and will not alter the flowpath or method of disposal of radioactive waste or byproducts. Therefore, the proposed amendment does not involve any change in the type or amount of any effluent that may be released offsite.

(iii) A significant increase in individual or cumulative occupational radiation exposure.

The proposed amendment does not result in changes in the level of control or methodology used for processing radioactive effluents or handling of solid radioactive waste. There will be no change to the normal radiation levels within the plant. Therefore, the amendment does not involve an increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.52(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment is not required.