September 30, 2005

Mr. John T. Conway
Site Vice President
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT RE: IMPLEMENTATION OF 24-MONTH FUEL CYCLES (TAC NO. MC3692)

Dear Mr. Conway:

The Commission has issued the enclosed Amendment No. 143 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant (Monticello). The amendment consists of changes to the technical specifications (TSs) in response to your application dated June 30, 2004, as supplemented by letters dated September 16, 2004, November 5, 2004, March 3, 2005, July 1, 2005 and September 27, 2005.

The amendment would change the TSs to support an increase in the length of the fuel cycle from 18 to 24 months at Monticello. In addition the proposed amendment requested changes in surveillance test intervals for various instruments. These changes will be evaluated in a separate license amendment.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

John F. Stang, Sr. Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 143 to DPR-22
2. Safety Evaluation

cc w/encls: See next page

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JStang	GHill(2)	THarris	MKotzalas	SJones	AHowe
DLPM DPR	RCorreia	BBurgess, RIII			

Package Accession No. ML052700255
Amendment Accession No. ML052700252
TS Accession No. ML052780367

OFFICE	PD3-1/PM	PD3-1/LA	IROB	EEIB/A	SPSB/C	OGC	PD3-1/SC
NAME	JStang	THarris	CSchulten for TBoyce	AHowe	MKotzalas	AHodgdon	LRaghavan
DATE	9/30/05	9/30/05	9/30/05	9/30/05	9/30/05	9/30/05	9/30/ 05

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Monticello Nuclear Generating Plant

cc:

Jonathan Rogoff, Esquire
Vice President, Counsel & Secretary
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
2807 W. County Road 75
Monticello, MN 55362

Manager, Regulatory Affairs
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

Robert Nelson, President
Minnesota Environmental Control
Citizens Association (MECCA)
1051 South McKnight Road
St. Paul, MN 55119

Commissioner
Minnesota Pollution Control Agency
520 Lafayette Road
St. Paul, MN 55155-4194

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Commissioner
Minnesota Department of Health
717 Delaware Street, S. E.
Minneapolis, MN 55440

Douglas M. Gruber, Auditor/Treasurer
Wright County Government Center
10 NW Second Street
Buffalo, MN 55313

Commissioner
Minnesota Department of Commerce
85 7th Place East, Suite 500
St. Paul, MN 55101-2198

Manager - Environmental Protection Division
Minnesota Attorney General's Office
445 Minnesota St., Suite 900
St. Paul, MN 55101-2127

John Paul Cowan
Executive Vice President & Chief Nuclear
Officer
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

Nuclear Asset Manager
Xcel Energy, Inc.
414 Nicollet Mall, R.S. 8
Minneapolis, MN 55401

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 143
License No. DPR-22

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated June 30, 2004, as supplemented by letters dated September 16, 2004, November 5, 2004, March 3, 2005, July 1, 2005 and September 27, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (I) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Facility Operating License No. DPR-22 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 143, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 30, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 143

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
4	4
25a	25a
33	33
49	49
52	52
60a	60a
60b	60b
61	61
62	62
63	63
102	102
105	105
127	127
129	129
131	131
132	132
156	156
167	167
168	168
169	169
203	203
229w	229w
229x	229x

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 143 FACILITY OPERATING LICENSE NO. DPR-22

NUCLEAR MANAGEMENT COMPANY, LLC

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated June 30, 2004 (Agencywide Documents Access and Management System (ADAMS Accession No. ML042040159), as supplemented by letters dated September 16, 2004 (ADAMS Accession No. ML042600576), November 5, 2004 (ADAMS Accession No. ML043150428), March 3, 2005 (ADAMS Accession No. ML050670432), July 1, 2005 (ADAMS Accession No. ML051890051) and September 27, 2005, the Nuclear Management Company, LLC (the licensee), requested changes to the technical specifications (TSs) for the Monticello Nuclear Generating Plant (Monticello). The proposed amendment would change the TSs to support an increase in the length of the fuel cycle from 18 to 24 months at Monticello. The licensee used the guidance in Generic Letter (GL) 91-04, “Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle,” dated April 2, 1991, in the development of the proposed changes to the TSs. In addition the proposed amendment requested changes in the surveillance test intervals for various instruments. These changes will be evaluated in a separate amendment.

The supplements dated September 16, 2004, November 5, 2004,, March 3, 2005, July 1, 2005 and September 27, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 18, 2005 (70 FR 2892).

2.0 REGULATORY EVALUATION

NRC issued GL 91-04 to give licensees generic guidance on preparing license amendment requests that change the TS surveillance intervals to accommodate a 24-month fuel cycle. In accordance with GL 91-04, the licensee must provide the following information to justify increasing the calibration intervals for instruments used to perform safety functions:

(1) Confirm that instrument drift as determined by as-found and as-left calibration data from surveillance and maintenance records have not, except on rare occasions, exceeded acceptable limits for a calibration interval.

(2) Confirm that the values of drift for each instrument type (make, model, and range) and application have been determined with a high probability, and a high degree of confidence. Provide a summary of the methodology and assumptions used to determine the rate of instrument drift with time based upon historical plant calibration data.

(3) Confirm that the magnitude of instrument drift has been determined with a high probability and a high degree of confidence for a bounding calibration interval of 30 months for each instrument type (make, model number, and range) and application that performs a safety function. Provide a list of the channels by TS section that identifies these instrument applications.

(4) Confirm that a comparison of the projected instrument drift errors has been made with the values of drift used in the setpoint analysis. If this results in revised setpoints to accommodate large drift errors, provide proposed TS changes to update trip setpoints. If the drift errors result in a revised safety analysis to support existing setpoints, provide a summary of the updated analysis conclusions to confirm that safety limits (SL) and safety analysis assumptions are not exceeded.

(5) Confirm that the projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation.

(6) Confirm that all conditions and assumptions of the setpoint and safety analyses have been checked and are appropriately reflected in the acceptance criteria of plant surveillance procedures for channel checks, channel functional tests, and channel calibrations.

(7) Provide a summary description of the program for monitoring and assessing the effects of increased calibration surveillance intervals on instrument drift and their effects on safety.

The NRC staff used the guidance in the GL 91-04 for evaluating the acceptability of the proposed changes to the surveillance test intervals.

3.0 TECHNICAL EVALUATION

The licensee performed a safety assessment for the proposed changes to the surveillance test intervals in accordance with the GL 91-04 guidance stated above. The proposed 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. The proposed TS changes do not introduce any accident initiators. The licensee's evaluation of the proposed TS changes has demonstrated 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. The licensee has concluded that extending the surveillance interval from 18 months to 24 months has minimal impact on the systems and the overall impact on the plant safety

analysis is negligible. The licensee also performed a historical review of surveillance test results and associated maintenance records and concluded that there was no evidence of any failure that would invalidate the above conclusions.

In Enclosure 1 of the June 30, 2004, application, the licensee discussed the methodology used to evaluate the acceptability of the change to a 24-month fuel cycle and compliance with GL 91-04. The proposed 24-month calibration frequency is supported by plant-specific analyses. To do the assessment, the licensee reviewed the historical maintenance and surveillance test data at the bounding surveillance test interval limit, performed an evaluation to ensure that a 24-month surveillance test interval would not invalidate any assumption in the plant licensing bases, and determined that the safety impact of the surveillance interval extension was acceptable.

The licensee has performed a drift evaluation, based on a Monticello plant-specific drift analysis using spreadsheets based on Electric Power Research Institute (EPRI) topical report TR-103335, “Guidelines for Instrument Calibration Extension/Reduction Programs.” The guidance described in TR-103335 provides more detail than GL 91-04 about the application of standard statistical methods to historical calibration data to determine if calibration intervals may be extended to 24 months, with a maximum of 30 months (24 months + 25 percent grace period), to support 24-month fuel cycle upgrades. The NRC staff had not formally endorsed TR-103335. However, in a letter dated December 1, 1997, from the NRC to James EPRI, “Status Report on the Staff Review of EPRI Technical Report TR-103335, Guidelines for Instrument Calibration Extension/Reduction Programs, dated March 1994,” the NRC staff stated that the TR-103335 offered acceptable guidance for GL 91-04 calibration interval extension programs except in some areas that NRC staff had comments that needed further clarification. In Enclosure 2 of the June 30, 2004, application, the licensee addressed all the concerns that the NRC staff identified in its December 1, 1997, letter.

To address the requirements of the GL 91-04, the licensee referenced the NRC safety evaluation (SE) dated August 2, 1993, relating to the extension of the Peach Bottom, Units 2 and 3, surveillance intervals from 18 months to 24 months. In this SE, the NRC staff stated the following:

> Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P), show that the overall safety systems’ reliability is not dominated by the reliability of the logic system, but by 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 component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability.

The licensee reviewed the surveillance test history at Monticello and validated this conclusion. The licensee’s review demonstrates that there are no failures that would invalidate the conclusion that changing to a 24-month operating cycle has minimal impact on safety system availability.

The November 5, 2004 and March 3, 2005, supplements responded to NRC staff requests for information on the licensee's sample data, outlier determination, time dependency, and miscellaneous other items. Based on its review of the licensee's application and supplements, the NRC staff finds the surveillance test interval extensions acceptable. A detailed explanation of each specific TS change follows.

On January 31, 2005, the NRC staff sent the licensee a request for additional information (RAI) on the Monticello setpoint methodology. By letter dated March 3, 2005, the licensee responded that the Monticello setpoint methodology uses the methodology in NEDC-31336P-A, "General Electric (GE) Instrument Setpoint Methodology," to determine the allowable value (AV). The method used to calculate the AV is similar to "Method 2" in the Instrumentation, Systems, and Automation (ISA) Society standard ANSI/ISA-S67.04-2000.

The licensee recently replaced some existing instruments and installed some new trip circuits. A new instrument drift-monitoring program will be implemented. Furthermore, as part of this license amendment request, the licensee made the following new commitment:

> Monticello will implement a trending program to address setpoints for TS calibration intervals extended to 24 months. Setpoints found to exceed the expected drift for the instruments would require an additional evaluation to ensure the instrument's performance is still enveloped by the assumptions in the drift or setpoint analysis. The trending program will also plot setpoint or transmitter As-Found/As-Left (AFAL) values to verify that the performance of the instruments is within expected boundaries and that adverse trends (repeated directional changes in AFAL even of smaller magnitudes) are detected and evaluated.

On June 3, 2005, the NRC staff sent a third RAI related to TS limiting safety system setting (LSSS) setpoint methodology. The NRC staff is concerned with instrumentation setpoint changes to the TS with respect to assessing the operability of the instruments associated with systems for which LSSS values are established. To address the NRC staff's concern with respect to operability and satisfy 10 CFR 50.36(c)(1)(ii)(A), operability needs to be assessed, in part, based on the ability of the system to initiate automatic protective actions as required to protect the safety limit (i.e., satisfy its safety function). The NRC staff's position is that in order to meet this requirement, periodic testing or calibration must demonstrate that the performance of the instrument is within the expected range, accounting for uncertainties associated with the test or calibration. Incorporating requirements into TSs that assess operability based on the previous as-left setting and the credible uncertainties when testing or calibrating the instrumentation will address the NRC staff concern.

By letters dated July 1, 2005 and September 27, 2005, the licensee responded to the NRC staff's concerns. In the July 1, 2005, supplement, the licensee stated that current plant procedures require resetting all LSSS setpoints within the specified tolerances (as-left criteria). The licensee reviewed each instrument for which it had proposed a setpoint change against the Monticello licensing basis to determine if any of the instruments directly protect SLs or are considered LSSS in the licensing basis. The SLs for Monticello are defined in Section 2 of the TS. The SLs are:

Reactor Core Safety Limits (2.1.A)

1. With the reactor steam dome pressure < 785 psig or core flow < 10 percent rated core flow:

   Thermal power shall be # 25 percent Rated Thermal Power.

2. With the reactor steam dome pressure greater $785 psig and core flow $ 10 percent rated core flow:

   MCPR [minimum critical power ratio] shall be $ 1.10 for two recirculation loop operation or $ 1.12 for single recirculation loop operation.

3. Reactor vessel water level shall be greater than the top of active irradiated fuel.

Reactor Coolant System Pressure Safety Limit (2.1.B)

Reactor steam dome pressure shall be # 1332 psig.

Fuel Cladding Integrity Safety Limit 2.1.A.1 and 2.1.A.2

The reactor core safety limits determined by reactor steam dome pressure and core flow limitations (i.e., 2.1.A.1 and 2.1.A.2 above) are protected by trip settings associated with the Reactor Protection System (RPS). A reactor scram is initiated by certain instrumentation associated with RPS to assure that fuel limits are not exceeded. Setpoints associated with RPS performing this function were originally listed in the LSSS Table[1] in the TS. The RPS does not require operation of any safety system auxiliaries to perform its safety function; thus, it is independent of the standby AC system power system (i.e., Emergency Diesel Generators (EDGs).

Reactor Vessel Water Level Above the Top of Active Fuel Safety Limit - 2.1.A.3

The safety limit directing the maintenance of the reactor vessel water level being above the top of active fuel is protected by both the RPS low level scram function, as well as the low level initiation of the Emergency Core Cooling System (ECCS). The ECCS has initiation signals to start on low low reactor pressure vessel (RPV) water level in order to maintain adequate core cooling to protect the safety limit.

Reactor Coolant System Pressure Safety Limit - 2.1.B

The reactor coolant system (RCS) pressure safety limit (2.1.B) is protected by both the RPS high pressure scram function as well as the safety relief function of the safety relief valves (S/RVs). As part of the overpressure protection analyses reactor scram is initiated on high pressure by RPS to assure the RCS safety limit is not exceeded. As previously stated the RPS

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[1] Instrumentation trip setpoints listed in the LSSS table were incorporated into Section 3 of the Monticello TS in Amendment 128.

does not require operation of any safety system auxiliaries to perform its safety function and the safety function of the S/RVs is independent of the Low-Low Set (LLS) actuation of the S/RVs.

The licensee reviewed the Monticello licensing basis and found the only place that LSSS[1] have been specified was in Section 2 of the TS associated with the automatic protective devices listed below.

Neutron Flux Intermediate Range Monitor - High-High
Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High
Flow Referenced Neutron Flux APRM - High Flow Clamp
Reactor Low Water Level Scram
Reactor Low Water Level ECCS Initiation
Main Steam Isolation Valve Closure
Turbine Control Valve Fast Closure
Turbine Stop Valve Closure
Main Steam Line Low Pressure Initiates Mainsteam Isolation Valve (MSIV) Closure

Other parameters required by the regulations, e.g., 10 CFR 50.46 ECCS acceptance criteria and 10 CFR Part 100, release limits used as acceptance criteria for accident mitigation that have significant safety functions are not cited as SLs or LSSS. The NRC staff has reviewed the information provided by the licensee in the July 1, 2005 and September 27, 2005 letters and understand that the licensee's current licensing basis is as described above. The licensee has also made the following commitments in its July 1, 2005, supplement.

- Continue resetting Limiting Safety System Settings (LSSS) setpoints within the specified tolerances (as-left criteria) until the Technical Specification Task Force's TS change pertinent to instrument setpoints has been approved by the NRC and assessed for applicability to Monticello.

- Assess applicability of the Technical Specification Task Force's TS change pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to Monticello's licensing basis are necessary.

The NRC staff concludes based on the review of the information provided by the licensee that the NRC staff's concerns with respect to operability and compliance with 10 CFR 50.36 are addressed for the scope of this amendment request. The NRC staff is continuing to review the generic issue of setpoint methodologies and finds the licensee's commitment to evaluate the Technical Specification Task Force's TS change discussed above is acceptable.

3.0 Evaluation of Proposed Technical Specification Changes

The licensee proposes to revise the following setpoints:

TS Table	Instrument Function	Current TS Value	Proposed TS Value
3.2.1 Instrumentation that Initiates Primary Containment Isolation Functions	High temp. in Main Steamline Tunnel	# 200 °F	# 209 °F
3.2.2 Instrumentation that Initiates Emergency Core Cooling Systems	Reactor Low Pressure Permissive Bypass Timer	20 ± 1min	20 ± 2min
3.2.6 Instrumentation for Safeguards Bus Degraded Voltage and Loss of Voltage Protection	Loss of Voltage Protection	2625 ± 175 volts	2625 ± 280 volts
3.2.7 Instrumentation for Safety/Relief Valve (S/RV) Low-Low Set Logic	Reactor Coolant System (RCS) Pressure for Opening/Closing	1072±3 / 992±3 psig 1062±3 / 982±3 psig 1052±3 / 972±3 psig	1072±14 / 992±14 psig 1062±14 / 982±14 psig 1052±14 / 972±14 psig
	Discharge Pipe Pressure Inhibit and Position Indication	30 ± 1 psid	30 ± 3 psid
	Inhibit Timers	10 ± 1 secs	10 ± 2 secs

The licensee stated in its November 5, 2004, submittal there was sufficient margin within the existing safety analysis to accommodate the revision in the nominal trip setpoints without revising the safety analysis in each of these cases and in no case was it necessary to change the existing analytical limit (AL) or safety analysis to accommodate a larger instrument drift error.

In the November 5, 2004, supplement the licensee submitted its calculation for the Main Steamline High Temperature Group 1 Isolation in the steam tunnel. In the supplement, the licensee shows its allowable AL is 212 °F. The NRC staff finds that the proposed value of 209 °F is still bounded by the existing safety analyses. In the March 3, 2005, supplement, the licensee submitted its current ALs for all of the setpoints it proposes to revise. The NRC staff

verified that all the proposed setpoints are within ALs. The NRC staff finds that the revised setpoints are still bounded by existing safety analyses.

The licensee stated that the proposed changes to the trip settings are a result of applying the Monticello instrument setpoint methodology using plant-specific drift values. The use of the proposed trip setpoints does not impact the safe operation of the plant because 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. Therefore, the NRC staff finds the above setpoint changes acceptable.

- TS 1.0.U - Definition - Refueling Operation and Refueling Outage

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 EF and movement of fuel or core components is in progress. Refueling Interval is a designated frequency for performing surveillance 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 proposed change follows the guidance in GL 91-04. The NRC staff considers that these changes are editorial and are, therefore, acceptable.

- TS 3.1/4.1 - Reactor Protection System (RPS)

The licensee evaluated extending the test intervals of the following surveillance requirements (SRs) for the RPS functions.

Table 3.1.1, Reactor Protection System, Function 1, Reactor Mode Switch in Shutdown
(SR Table 4.1.1, Perform Functional Test)

This function was not specifically credited in the safety analysis, but is retained for overall redundancy and diversity. The surveillance test interval for this function is being increased from 18 months to 24 months. Because no instrumentation is associated with this function, drift has no effect when the surveillance interval is increased. 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 a weekly functional test of manual scram. A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect of the proposed change has minimal impact on system availability. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

Table 3.1.1, Reactor Protection System, Function 7, Reactor Vessel Water Level - Low
(SR Table 4.1.2, Perform Channel Calibration)

The reactor vessel low-water-level signal provides automatic action to assure that the capability to cool the fuel may be maintained. The licensee evaluated the surveillance interval extension for the transmitters, using the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying the Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoint analysis. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

Table 3.1.1, Reactor Protection System, Function 10, Main Steamline Isolation Valve - Closure
(SR Table 4.1.2, Perform Channel Calibration)

A MSIV closure results in loss of the main turbine and loss of 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. Limit switches perform this function. Limit switches are mechanical devices that require mechanical setting only; drift is not applicable to these devices. The proposed surveillance interval change from 18 months to 24 months has minimal effect on system availability. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

Table 3.1.1, Reactor Protection System, Function 12, Turbine Stop Valve - Closure
(SR Table 4.1.2, Perform Channel Calibration)

Closure of the turbine stop valves results in the loss of a heat sink, producing reactor pressure, neutron flux, and heat flux transients that must be limited. A reactor scram is initiated when the turbine stop valves start to close in anticipation of the transients. The reactor scram reduces the amount of energy to be absorbed and ensures that the MCPR SL is not exceeded. Local limit switches perform this function. Limit switches are mechanical devices that require mechanical setting only; drift is not applicable to these devices. The proposed surveillance interval change from 18 months to 24 months has minimal effect on system availability. A review of the surveillance test history found no failures that invalidate the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS 3.1/4.1 - RPS Power Monitoring System
(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)

Two motor generator (MG) sets provide AC power for operation of the RPS. These MG sets are powered from 480 VAC buses and are used to supply power to the scram logic channels and the neutron-monitoring and radiation-monitoring systems. An alternate power source is provided to permit servicing of either MG set. Electrical protection assemblies monitor the electric power in each of the three sources of power 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 circuit breaker trips and disconnects the RPS from the abnormal power source. Each monitoring module trips the associated breaker on over-voltage, under-voltage or under-frequency. With the protective package installed, abnormal output type failures in either of the two RPS MG sets result in a trip of one or both of the two Class 1E protective packages.

The licensee has evaluated the effect of extending the testing interval on the RPS power-monitoring system. Potential time-based considerations, such as instrument drift, failure type, and failure frequency, affect system availability. Based on the recommendation of EPRI TR-103335, the licensee used the Monticello plant-specific methodology to determine the time drift. The proposed surveillance interval change from 18 months to 24 months has minimal effect on the system availability. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

- TS 3.2.A, Primary Containment Isolation System (PCIS)

The licensee evaluated extending the test intervals of the following SRs for the RPS functions.

TS Table 3.2.1, PCIS, Function 1.a, Main Steam and Recirculation Sample Line - Low-Low Reactor Water Level
(SR Table 4.2.1, Perform Channel Calibration)

The design basis for the PCIS is to provide protection against the onset and consequences of accidents involving gross releases of radioactive materials from the primary containment. The licensee evaluated the surveillance interval extension for the transmitters, using the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoint analysis. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.1, PCIS Function 1.c, Main Steam and Recirculation Sample Line - High Temperature in Main Steamline Tunnel
(SR Table 4.2.1, Perform channel calibration)
(SR Table 4.2.1, Perform functional test)

High temperature in the vicinity of the main steamlines is detected by 16 bimetallic temperature switches located along the main steamlines in the steam tunnel between the drywell wall and the turbine building. The surveillance test interval for this function is being increased from 18 months to 24 months. An evaluation of the surveillance interval extension for the bimetallic temperature switches was performed, based on the approach described in GL 91-04. Based on EPRI TR-103335, the drift associated with the temperature switches was determined by applying Monticello plant-specific methodology. The drift for the extended surveillance interval relative to AV and the plant trip setpoint was considered. As a result, the trip setpoint will change from #200 EF to #209 EF. 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 is highly reliable. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.1, PCIS Function 2.a, RHR [residual heat removal] System, Head Cooling, Drywell, Sump, TIP - Low Reactor Water Level
(SR Table 4.2.1, Perform channel calibration)

The reactor vessel low-water-level signal provides automatic action to assure that the capability to cool the fuel may be maintained. Should the reactor water level decrease too far, fuel damage can result. The licensee evaluated the surveillance interval extension for the transmitters, using the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoint analysis. The channel functional test is performed every 3 months, and channel calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm that the major portions of the monitoring loop are tested for proper operation at more frequent interval. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.1, PCIS Function 3.b, Reactor Water Cleanup System - Low-Low Reactor Water Level
(SR Table 4.2.1, Perform Channel Calibration)

The design basis for the PCIS is to provide protection against the onset and consequences of accidents involving gross releases of radioactive materials from the primary containment. The

automatic closure of the reactor water cleanup (RWCU) line prevents the excessive loss of reactor coolant. The licensee evaluated the surveillance interval extension for the transmitters, using the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoint analysis. The channel functional test is performed every 3 months, and channel calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm that the major portions of the monitoring loop are tested for proper operation in more frequent interval. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.1, PCIS Function 3.c, Reactor Water Cleanup (RWCU) System - High RWCU room Temperature
(SR Table 4.2.1, Perform Channel Calibration)

High temperature in the RWCU room could indicate a leak in a RWCU line. The automatic closure of the RWCU line prevents the excessive loss of reactor coolant and the release of radioactive material from the nuclear system process barrier. The surveillance test interval for this function is being increased from 18 to 24 months. An evaluation of the surveillance interval extension for the resistance temperature detector (RTD) input circuitry of the trip unit was performed based upon the approach described in GL 91-04. These trip units are part of a recently installed trip circuit. Because an insufficient amount of calibration data is available to perform a statistical drift analysis, vendor drift values were used for this application. A new instrument drift-monitoring program will ensure that the drift value used in the setpoint determination for this application is conservative relative to actual detector performance.

Extending the surveillance test interval for the channel calibration for the RTD input circuitry of the trip unit 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 that the major portions of the monitoring loop are tested for proper operation. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. Based on review of the information in the submittal, the NRC staff finds the proposed changes acceptable.

TS Table 3.2.1, PCIS Function 3.d, Reactor Water Cleanup System - High RWCU System Flow
(SR Table 4.2.1, Perform Channel Calibration)

RWCU high flow could indicate a leak in a RWCU line. The automatic closure of the RWCU line prevents the excessive loss of reactor coolant and the release of radioactive material from the nuclear system process barrier. The surveillance test interval for this function is being increased from 18 to 24 months. 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. The same transmitter model is used in another application at Monticello plant. 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 licensee evaluated the surveillance interval extension for the transmitters, using the approach described in GL 91-04. The evaluation was based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoints. The channel functional test is performed every 3 months, and channel calibration of the remaining circuitry of the associated trip units every 3 months. These checks confirm that the major portions of the monitoring loop are tested for proper operation. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

- TS 3.2.B - Emergency Core Cooling Subsystem (ECCS) 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).

TS Table 3.2.2, ECCS Function A.1.a, Core Spray and Low Pressure Coolant Injection (LPCI) - Low-Low Reactor Water Level
(SR Table 4.2.1, Perform Channel Calibration)

The design basis function of the core spray (CS) and LPCI systems is to restore and maintain the coolant in the reactor vessel so that the core is adequately cooled to preclude fuel damage. The surveillance test interval for the CS and LPCI system's transmitters, as applied to this function, is being increased from 18 months to 24 months. The licensee evaluated the surveillance interval extension for the transmitters, using the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoints. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.2, ECCS Function A.1.b.ii, Core Spray and LPCI - Reactor Low Pressure Permissive Bypass Timer
(SR Table 4.2.1, Perform Channel Calibration)
(SR Table 4.2.1, Perform Functional Test)

One of the initiating signals for CS and LPCI is reactor low-low water level sustained for 20 minutes. On receiving a low-low water level signal, the CS and LPCI pumps will start if either the low reactor pressure permissive or the bypass timer is satisfied. The 20-minute

initiation delay is provided by time delay relays for this CS and LPCI bypass timer function. The licensee evaluated drift for the extended surveillance interval relative to the plant trip setting for this function. The licensee determined that a new line item "item 10, Reactor Low Pressure (Bypass Timer)," should be added to TS Table 4.2.1, ECCS Instrumentation.

The licensee evaluated the surveillance interval extension for the time delay relays, using the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, applying Monticello plant-specific methodology. The drift for the extended surveillance interval relative to the plant trip setting for this function is changed from "20 ± 1 minute" to "20 ± 2 minutes." Extending the surveillance test interval for the functional test is acceptable because these timers are part of a network that includes actuating logic that is designed to be single-failure proof and is, therefore, highly reliable. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.2, ECCS Function A.3, Core Spray and LPCI - Loss of Auxiliary Power
(SR Table 4.2.1, Perform Channel Calibration)
(SR Table 4.2.1, Perform Functional Test)

The auxiliary power system is designed to provide adequate power to operate all the plant auxiliary loads necessary for plant operation. A failure of a single component in the auxiliary power system will not reduce plant safety or impair the operation of essential plant functions. The licensee evaluated the surveillance interval extension for these relays and contacts based on the approach described in GL 91-04. These relays and contacts are mechanical devices and not subject to drift. Extending the surveillance test interval for the functional test is therefore acceptable. The network, including the actuating logic, is designed to be single-failure proof and is, therefore, highly reliable. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.2, ECCS Function B.2, High Pressure Coolant Injection (HPCI) System - Low-Low Reactor Water Level
(SR Table 4.2.1, Perform Channel Calibration)

The HPCI system is designed to pump water into the reactor vessel under LOCA conditions that do not result in rapid depressurization of the pressure vessel. Dependent on the flow from a small line break, it is possible that the reactor water level could drop to a point where the core is not adequately cooled. Yet the reactor would be at or near rated pressure. Consequently, the low pressure core cooling systems (CS and RHR) would not be capable of injecting coolant into the vessel. The HPCI system supplies makeup coolant into the reactor vessel from fully pressurized to a preset depressurized condition. The flow rate of the system will maintain the reactor core adequately cooled until the reactor pressure drops sufficiently to permit the low

pressure core cooling systems to automatically inject coolant into the vessel. The licensee evaluated the surveillance interval extension for the transmitters, using the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoint analysis. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.2, ECCS Function C.1, Automatic Depressurization - Low-Low Reactor Water Level
(SR Table 4.2.1, Perform Channel Calibration)

The automatic depressurization system (ADS) is designed to depressurize the reactor to permit either LPCI or CS to cool the reactor core during a small-break LOCA. The licensee evaluated the surveillance interval extension for the ADS transmitters, using the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoint analysis. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.2, ECCS Function C.2, Automatic Depressurization - Auto Blowdown Timer
(SR Table 4.2.1, Perform Channel Calibration)
(SR Table 4.2.1, Perform Functional Test)

The reactor vessel is depressurized by blowdown through automatic opening of the S/RVs, which vent steam to the suppression pool. A time delay circuit is connected 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 licensee evaluated the surveillance interval extension for the ADS timers, using the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the timer drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoint analysis. Extending the surveillance test interval for the functional test is acceptable because these timers are part of a network that includes the actuating logic. It is designed to be single-failure proof and is, therefore, highly reliable. A review of the surveillance test history demonstrates that no identified failure invalidates the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.2, ECCS Function D.2, Diesel Generator - Low-Low Reactor Water Level
(SR Table 4.2.1, Perform Channel Calibration)

Two independent EDGs provide redundant standby power sources. The licensee evaluated the surveillance interval extension for the low-low water level transmitters based on the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoint analysis. Extending the channel calibration surveillance test interval for the transmitters is acceptable because the function is verified by the more frequent functional test of the trip units. The EDG automatic starting is also initiated by high drywell pressure as a diverse means for this protective function. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

- Other Instrumentation

TS Table 3.2.8, Other Instrumentation Function A.1, Reactor Core Isolation Cooling (RCIC) Initiation - Low-Low Reactor Level
(SR Table 4.2.1, Perform Channel Calibration)

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. The licensee evaluated the surveillance interval extension for the low-low reactor water level transmitters based on the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoint analysis. Extending the channel calibration surveillance test interval for the transmitters is acceptable because the function is verified by the more frequent functional test of the trip units that perform major role on protective function. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.8, Other Instrumentation Function B.1, HPCI/RCIC Turbine Shutdown - High Reactor Water Level
(SR Table 4.2.1, Perform Channel Calibration)

The HPCI turbine automatically trips 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 is activated until either a manual restart is initiated or a low-low reactor water level signal is received. The RCIC turbine auto shutdown

occurs on turbine overspeed, high water level in reactor vessel, low pump suction pressure, or high turbine exhaust pressure.

The licensee evaluated the surveillance interval extension for the high reactor water level transmitters based on the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoint analysis. Extending the channel calibration surveillance test interval for the transmitters is acceptable because the function is verified by the more frequent functional test of the trip units that perform a major role on protective function. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.8, Other Instrumentation Function C.1, HPCI/RCIC Turbine Suction Transfer - Condensate Storage Tank Low Level
(SR Table 4.2.1, Perform Channel Calibration)
(SR Table 4.2.1, Perform functional test)

Two sources of water, the condensate storage tanks (CSTs) and the suppression pool, are available for the HPCI system. When the level in either one of the two CSTs falls below the low level setpoint, the pump suction is automatically transferred to the suppression pool. The RCIC system turbine-driven pump supplies demineralized makeup water from the CST to the reactor vessel; the suppression pool is an alternate source of water. The surveillance test interval for the level switches for this function is being increased from 18 months to 24 months. The switches are float switches, which are mechanical devices and cannot be significantly adjusted without physically changing the location of the installation. Therefore, increasing the surveillance interval does not affect the accuracy of the float switches. Extending the surveillance test interval for the functional test is acceptable because these switches are part of a network that includes the actuating logic. The network is designed to be single-failure proof and is, therefore, highly reliable. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

- Reactor Building Isolation and Standby Gas Treatment (SBGT) System

TS 3.2.E, Reactor Building Isolation and SBGT System - Low-Low Reactor Water Level
(SR Table 4.2.1, Perform Channel Calibration)

Reactor building isolation and SBGT are activated by primary containment isolation logic on low-low reactor water level. The surveillance test interval for the low-low reactor water level transmitters for this function is being increased from 18 months to 24 months. The transmitters provide input to the associated trip units, which are functionally tested and calibrated once

every 3 months. The licensee evaluated the surveillance interval extension for the transmitters based upon the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoints. Extending the channel calibration surveillance test interval for the transmitters is acceptable, in part, because the trip function is verified by the more frequent functional test of the trip units that perform a major role on protective function. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

- Recirculation Pump Trip and Alternate Rod Injection Initiation

TS 3.2.F, Recirculation Pump Trip and Alternate Rod Injection Initiation, Function 1 - High Reactor Dome Pressure
(SR Table 4.2.1, Perform Channel Calibration)

A recirculation pump trip (RPT) system and an alternate rod injection (ARI) system mitigate anticipated transients without scram (ATWS) events. Reactor vessel pressure significantly higher than the high-pressure scram setting is the primary indication of an ATWS event. High reactor dome pressure inputs are used to initiate these ATWS mitigation systems. The licensee evaluated the surveillance test interval of the pressure transmitters as it applies to these functions. The transmitters provide input to the associated trip units, which are functionally tested and calibrated once every 3 months. The licensee evaluated the surveillance interval extension for the transmitters, using the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoint analysis. Extending the channel calibration surveillance test interval for the transmitters is acceptable because the function is verified by the more frequent functional test of the trip units that perform major role on protective function. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS 3.2.F, Recirculation Pump Trip and Alternate Rod Injection Initiation, Function 2 - Low-Low Reactor Water Level
(SR Table 4.2.1, Perform Channel Calibration)

A RPT system and an ARI system mitigate ATWS events. Low-low water level in the reactor vessel may indicate that an ATWS event has occurred. Low-low reactor water level signals are used to initiate these ATWS mitigation systems. The licensee evaluated the surveillance test interval of the low-low reactor water level transmitters as it applies to these functions. The transmitters provide input to the associated trip units, which are functionally tested and calibrated once every 3 months. The licensee evaluated the surveillance interval extension for

the transmitters, using the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. The drift for the extended surveillance interval was considered in the instrument setpoint analysis. Extending the channel calibration surveillance test interval for the transmitters is acceptable because the function is verified by the more frequent functional test of the trip units that perform a major protective function role. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

- Safeguards Bus Voltage Protection

TS Table 3.2.6, Instrumentation for Safeguards Bus Voltage Protection
(SR Table 4.2.1, Perform Channel Calibration)

The essential buses are transferred to either of the emergency power sources, the reserve auxiliary transformer (1AR) or the EDGs, on loss of essential bus voltage or degraded voltage conditions on the essential bus. The licensee evaluated the surveillance test interval for this function. The licensee evaluated the undervoltage relays using the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the undervoltage relays drift, applying Monticello plant-specific methodology. 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 because 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 on performance history for the breakers and is designed for maximum availability. 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. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.7, Safety/Relief Valve Low-Low Set Logic, Function 2, Reactor Coolant System Pressure - Opening
(SR Table 4.2.1, Perform Channel Calibration)

The opening setpoints for the low-low set S/RVs are lower than the valve’s mechanical setpoints. The lower setpoints are for electrical/pneumatic actuation of the low-low set S/RVs. The S/RVs, which are also part of the S/RV low-low set system and have low-low set actuation, are prevented from subsequent manual or automatic actuation before the water leg recedes in the discharge line to prevent excessive hydrodynamic loading of discharge piping and suppression chamber components. The licensee evaluated the surveillance test interval for this function. An evaluation of the surveillance interval extension for the transmitters was

performed, based upon the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. 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 "1072 ± 3psig; 1062 ± 3psig; and 1052 ± 3psig" to "1072 ± 14psig; 1062 ± 14psig; and 1052 ± 14psig." A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

- Safety/Relief Valve Low-Low Set Logic

TS Table 3.2.7, Safety/Relief Valve Low-Low Set Logic, Function 3, Reactor Coolant System Pressure - Closing
(SR Table 4.2.1, Perform Channel Calibration)

A low-low set S/RV closes after an 80 psi blowdown of reactor pressure is detected. The setpoints for the low-low set valves ensure that these valves will be the first S/RVs to open and the last to close. After a low-low set S/RV has opened and closed, a time delay relay prevents the plant operator or the low-low set logic from immediately reopening the S/RV to allow the water leg in the S/RV discharge line to recede. The licensee evaluated the surveillance test interval for this function. The licensee evaluated the surveillance interval extension for the reactor coolant pressure transmitters based on the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. 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 "992 ± 3psig; 982 ± 3psig; and 972 ± 3psig" to "992 ± 14psig; 982 ± 14psig; and 972 ± 14psig." Extending the channel calibration surveillance test interval for the transmitters is acceptable because the function is verified by the more frequent functional test of the trip units. A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 3.2.7, Safety/Relief Valve Low-Low Set Logic, Function 4, Discharge Pipe Pressure
(SR Table 4.2.1, Perform Channel Calibration)

S/RV position is indicated by a differential pressure transmitter signal to an analog trip unit that monitors the steam pressure in each discharge pipe. When pressure is sensed, the trip unit will indicate the valve is open. The licensee evaluated the surveillance test interval for this function. The licensee evaluated the surveillance interval extension for the steam pressure transmitters based on the approach described in GL 91-04. Based on the recommendation of EPRI TR-103335, the licensee determined the transmitter drift, applying Monticello plant-specific methodology. 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 "30 ± 1psid"

to “30 ± 3psid.” A review of the surveillance test history found no failures that invalidated the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

- Instrumentation for Control Room Habitability Protection

TS 3.2.1, Instrumentation for Control Room Habitability Protection Function 1, Radiation (SR Table 4.2.1, Perform Channel Calibration)

The main control room ventilation air radiation inlet monitors are designed to automatically prevent the flow of contaminated air into the control room. This system protects the control room operators from the effects of accidental release of radioactivity into the environment. The filtration units, which provide makeup air for establishing positive pressure in the control room, are equipped with high-efficiency particulate air (HEPA) filters and charcoal absorbers. Two radiation detectors arranged in a one-out-of-two-once logic. The control room ventilation emergency filtration system trips into the high-radiation mode if a radiation monitor failure signal is received. The licensee evaluated the surveillance test interval for this function. The associated detectors are functionally tested monthly and a sensor check is performed daily. The radiation detector is calibrated by exposing the sensor-converter to a calibrated source in a constant geometry. A drift evaluation will not provide a true indication of the instrument’s performance. The proposed surveillance interval change has minimal effect on system availability. The licensee reviewed the surveillance test history and found no failures that invalidate the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

- Scram Discharge Volume (SDV)

The safety function of the SDV vent and drain valves is to limit the loss of reactor coolant leaked past the control rod drive seals while the scram valves are open. To accomplish this, the vent and drain valves must either be in the closed position or close in a timely manner upon scram initiation.

SR 4.3.F requires the licensee to verify, once per operating cycle, that 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. This test ensures the mechanical components and a portion of the valve logic remain operable.

SR 4.3.F also specifies that the SDV vent and drain valves are cycled quarterly to ensure the mechanical components and a portion of the valve logic remain operable. Logic systems are considered inherently more reliable than other plant components as acknowledged in the NRC 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.

Based on the above statements and the licensee's review of its surveillance test history, which did not find any failures that indicate that the extended surveillance interval would cause a more than minimal change in system availability, the NRC staff finds the increase in surveillance interval for this SR from 18 months to 24 months for a maximum interval of 30 months acceptable.

- Standby Liquid Control (SLC) System

The design objective of the SLC system is to provide the capability of bringing the reactor from full power to a shut down condition during the event that the reactor cannot be shut down or be kept shut down with the control rods. To meet this objective the SLC system was designed to inject a quantity of boron neutron absorber solution into the reactor. The SLC system is required to mitigate postulated ATWS events. The licensee evaluated the following SRs relative to extending their respective test intervals.

One of the two SLC systems is manually initiated once each operating cycle during performance of SR 4.4.A.2.a required surveillance testing. SR 4.4.A.2.b requires the licensee to explode one of the primer assemblies then replace it. The same surveillance test satisfies the requirements of both SR 4.4.A.2.a and SR 4.4.A.2.b.

The explosive valves are double squib actuated shear plug valves that are the explosive type to provide a high assurance of opening when actuated. Each of the two primer assemblies will be in service for no more than 4 years as one of the two assemblies is replaced each operating cycle. This is within the usable lifetime of 5 years considered for these explosive squibs. The system is designed so that all active components are single failure proof. In its March 3, 2005, letter to the NRC, the licensee submitted the results of its last eight surveillance tests. During this time the licensee has seen no failures of this system.

Based on the above statements, the NRC staff finds the increase in surveillance interval for these SRs from 18 months to 24 months, for a maximum interval of 30 months, to be acceptable.

The Boron-10 (B-10) isotope absorbs thermal neutrons and thereby terminates the nuclear fission reaction in the fuel. Naturally occurring boron contains a small percentage of B-10; therefore, the boron used in the SLC system is enriched with B-10. SR 4.4.B.1 requires the licensee to determine the B-10 enrichment once per cycle and the results of the analysis must have a B-10 enrichment greater than 55-atom percent.

Two mechanisms have the potential to affect the B-10 enrichment in the SLC tank. These are (1) absorption of thermal neutrons by the B-10 in the tank solution and (2) alteration of the average B-10 enrichment resulting from addition of Boron to the SLC tank.

The SLC tank at Monticello is not subjected to a thermal neutron flux. Thus there would be no deterioration of the B-10 due to thermal neutron absorption. The sodium pentaborate material received includes documentation of isotopic and impurity analyses for a sample. Approved vendors, who have been audited/surveyed and qualified to perform applicable analyses, perform confirmatory chemical analyses whenever boron is added to the SLC tank.

The extension of the surveillance interval will not introduce any new mechanisms by which the B-10 enrichment could be adversely affected. Therefore the B-10 enrichment in the SLC system will not be adversely affected. Based on this the NRC staff finds the increase in surveillance interval for this SR from 18 months to 24 months, for a maximum interval of 30 months, to be acceptable.

- ECCS Systems - High Pressure Core Injection (HPCI), Automatic Depressurization System (ADS), and Reactor Core Isolation Cooling (RCIC)

The HPCI system is designed to pump water into the reactor vessel under LOCA conditions that do not result in rapid depressurization of the pressure vessel.

SR 4.5.A.3.b ensures the capability of the HPCI pump to overcome RPV pressure and inject coolant into the core as designed for analyzed conditions. This is achieved by demonstrating that with reactor pressure # 165 psig, the HPCI pump can develop a flow rate $2700 gallons per minutes (gpm) against a system head corresponding to reactor pressure.

All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. There are other SRs that require the licensee to test HPCI quarterly at normal operating pressure to ensure required flow can be achieved. With the HPCI inoperable, adequate core cooling is assumed by the operability of the ADS, and both the CS and the LPCI systems. In addition, RCIC, can serve as a back up to HPCI.

Based on the above statements and the licensee’s review of its surveillance test history which did not find any failures that indicate that the extended surveillance interval would cause a more than minimal change in system availability, the NRC staff finds the increase in surveillance interval for this SR from 18 months to 24 months for a maximum interval of 30 months acceptable.

The ADS system accomplishes reactor vessel depressurization by blowdown through automatic opening of the S/RVs that vent steam to the suppression pool. For small breaks, the vessel is depressurized in sufficient time to allow a lower pressure system to provide adequate core cooling. The operator has the capability to manually block auto depressurization initiation or completion by utilizing the ADS inhibit switches in the main control room.

SR 4.5.A.4.a requires the licensee to perform a test to determine ADS valve operability each operating cycle. Three S/RVs are used in the automatic circuitry. This provides redundancy since only two are needed to perform its design function. Each ADS valve is powered independently from alternate power sources which provides common failure protection. The licensee also performs additional component testing on the ADS valves that provide assurance of the mechanical integrity of the valve and its functionality beyond SR 4.5.A.4. In its March 3, 2005 submittal, the licensee provided results of the last eight surveillance tests which showed no failures that indicate that the extended surveillance interval would cause a more than minimal change in system availability.

Based on the above statements, the NRC staff finds the increase in surveillance interval for this SR from 18 months to 24 months, for a maximum interval of 30 months, to be acceptable.

SR 4.5.A.4.b requires the licensee to perform an operability test for the ADS inhibit switch. This test evaluates the circuitry logic of the switch and its ability to inhibit ADS actuation.

Logic systems are considered inherently more reliable than other plant components. Also, in its March 3, 2005, submittal, the licensee provided results of the last eight surveillance tests, which showed no failures that indicate that the extended surveillance interval would cause a more than minimal change in system availability.

Based on the above statements, the NRC staff finds the increase in surveillance interval for this SR from 18 months to 24 months, for a maximum interval of 30 months, to be acceptable.

The Automatic Actuation Test SR ensures that HPCI, CS, LPCI, and ADS are capable of performing their design functions by performing a simulated automatic actuation test. Logic systems are considered inherently more reliable than other plant components.

Based on the above and the licensee's review of its surveillance test history which did not find any failures that indicate that the extended surveillance interval would cause a more than minimal change in system availability, the NRC staff finds the increase in surveillance interval for this SR from 18 months to 24 months, for a maximum interval of 30 months, to be acceptable.

The licensee is also updating the language used in this TS 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." The licensee has stated that this is an administrative change only due to the convention that "low" is not a designated water level and this surveillance procedure is performed at "Low Low" water level. The licensee submitted additional information on March 3, 2005, demonstrating that the change is administrative and that this change is consistent with the convention used throughout the TS and TS Bases. The NRC staff finds this change acceptable due to its administrative nature.

The RCIC system can provide makeup water to the reactor vessel for core cooling under certain circumstances. RCIC is not credited in any safety analyses.

SR 4.5.D.1.b requires the licensee, once per cycle, with reactor pressure # 165 psig, to demonstrate that the RCIC pump can develop a flow rate $ 400 gpm against a system head corresponding to reactor pressure. This SR ensures that the RCIC system is capable of performing its design function at low RPV pressure.

All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. Additional SRs that require the licensee to perform additional tests quarterly ensure required RCIC flow at normal operating pressure.

Based on the above statements and the licensee's review of the surveillance test history which did not find any failures that indicate that the extended surveillance interval would cause a more than minimal change in system availability, the NRC staff finds the increase in surveillance interval for this SR from 18 months to 24 months for a maximum interval of 30 months acceptable.

SR 4.5.D.2 requires the licensee to perform a simulated automatic actuation test each refueling outage. The RCIC System Functional Test ensures a system initiation signal to the automatic initiation logic of RCIC will cause the systems or subsystems to operate as designed. Based on the inherent equipment reliability of logic systems, the fact that this system is not credited in any safety analyses, and the licensee's review of the surveillance test history which did not find any failures that indicates that the extended surveillance interval would cause a more than minimal change in system availability, the NRC staff finds the increase in surveillance interval for this SR from 18 months to 24 months, for a maximum interval of 30 months, to be acceptable.

The licensee is also updating the language used in this TS from "including transfer to suppression pool and automatic restart on subsequent low reactor water level" to "including transfer to suppression pool and automatic restart on subsequent Low-Low reactor water level." The licensee has stated that this is an administrative change only due to the convention that "low" is not a designated water level and this surveillance procedure is performed at "Low Low" water level. The licensee submitted additional information in their March 3, 2005 demonstrating that the change is administrative and that this change is consistent with the convention used throughout the TS and TS Bases. The NRC staff finds this change acceptable due to its administrative nature.

- Primary System Boundary - Reactor Coolant System (RCS)

Monticello TS require means for detecting, and to the extent practical, identifying the source of RCS leakage. The following surveillance requirements are related to detecting RCS leakage and have been evaluated relative to extending their respective testing intervals.

SR 4.6.D.2.a requires the licensee to demonstrate that the RCS leakage detection instrumentation by performing a channel calibration of the primary containment atmosphere particulate monitoring system once per cycle. This SR also requires the performance of a sensor check once per 12 hours, and a channel Functional Test at least monthly.

The licensee states that the error in these detectors is due to accuracy of the detector and the calibration sources, which is far greater than what may be experienced because of an error due to drift of the electronic signal conditioning circuit, therefore drift is not a consideration in maintaining the performance of this instrument.

Based on the above statements and the licensee's review of its surveillance test history, which did not find any failures that indicate that the extended surveillance interval would cause a more than minimal change in system availability, the NRC staff finds the increase in surveillance interval for this SR from 18 months to 24 months for a maximum interval of 30 months acceptable.

SR 4.6.D.2.b requires the licensee to demonstrate operability of the RCS leakage detection instrumentation is operable by performing a channel calibration test. This SR also requires that the licensee perform a sensor check once per 12 hours, and a channel Functional Test for flow instruments at least monthly.

The licensee states that identifying the quantity of leakage is accomplished by (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. The comparison of these diverse methods confirms the performance of each individually. Additionally the long-term drift has little effect on the time that the pumps operate or the change in the measured level.

Based on the diverse methods used to identify RCS leakage and the licensee's review of its surveillance test history which did not find any failures that indicate that the extended surveillance interval would cause a more than minimal change in system availability, the NRC staff finds the increase in surveillance interval for this SR from 18 months to 24 months, for a maximum interval of 30 months, to be acceptable.

- Primary System Boundary Safety/Relief Valves

The RCS S/RVs assure that the RCS pressure safety limit is never reached. The S/RVs also meet requirements of the ADS, Low-Low Set System, and alternate shutdown cooling functions under all postulated accident events and transient events. The following surveillance requirements are related to testing the S/RVs and have been evaluated relative to extending their respective testing intervals.

SR 4.6.E.1.a requires the licensee to test or replace each S/RV each refueling outage in accordance with the Inservice Testing Program. SR 4.6.E.1.b requires that the licensee disassemble and inspect at least two of the S/RVs each refueling outage. SR 4.6.E.1.d requires that the licensee demonstrate operability of the bellows monitoring system each operating cycle.

These SRs ensure that the S/RVs will perform their intended safety function. There are additional SRs that require continuous monitoring of the S/RV components.

Based on the above statements and the licensee's review of its surveillance test history which did not find any failures that indicate that the extended surveillance intervals would cause a more than minimal change in system availability, the NRC staff finds the increase in surveillance interval for this SR from 18 months to 24 months for a maximum interval of 30 months acceptable.

- TS 3.6.H Snubbers

The operability of snubbers is required to provide assurance that the structural integrity of the RCS and all other safety-related systems is maintained during and following seismic or other event-initiating dynamic loads. The operability is verified by an inservice inspection and testing program specified in the TS. In order to provide assurance that the hydraulic and mechanical snubbers function reliably, a representative sample of Monticello's installed snubbers will be functionally tested during plant shutdowns. 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.

SR 4.6.H.1 Snubber Visual Inspections requires that snubbers both accessible and non-accessible be visually inspected. This SR requires a visual inspection to verify the snubber has no visible indications of damage or impaired operability, attachments to the foundation or supporting structure are functional, and fasteners for the attachment of the snubber anchorage are functional. 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 is currently established at the surveillance test interval of 18 months.

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 percent grace period. Operating experience shows these components routinely pass the SR when performed at the 18-month interval. A review of the surveillance history determined that snubber failures had occurred; however, none of the failures have resulted in the inoperability of the attached piping. Based on the historical review, the change from 18 to 24-month cycles would have a small, if any, impact on the availability of the associated piping system.

SR 4.6.H.3 requires that the functional testing of snubbers shall be conducted at least once per 18 months +25 percent 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 TS 4.6.H.4, an additional 10 percent of that brand is required to be tested until no more failures are found or all snubbers of that brand have been tested. 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 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 percent grace period. A review of the surveillance history determine that snubber failures had occurred; however, the none of the failures have resulted in the inoperability of the attached piping. Based on the historical review, the change from 18 to 24-month cycles would have a small, if any, impact on the availability of the associated piping system.

SR 4.6.H.6 requires that 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. 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 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 percent grace

period. Failure to perform this surveillance would have a small, if any, impact on the availability of the associated piping system.

The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to TSs 4.6.H.1, 4.6.H.3, and 4.6.H.6 are acceptable.

- TS 4.7 - Containment Systems

The function of the primary containment, which includes the drywell and suppression chamber, is to isolate and contain fission products released from the RCS following a design-basis LOCA and to confine the postulated release of radioactive material. As such, the primary containment must provide essentially a leak tight barrier against an uncontrolled release of radioactive material to the environment.

There are eight internal vacuum breakers, located on the vent header of the vent system between the drywell and the suppression chamber, which allow air and steam flow from the suppression chamber to the drywell when drywell pressure is negative with respect to the suppression chamber. In this way, the suppression chamber-to-drywell vacuum breakers prevent excessive negative differential pressures across the suppression chamber-drywell boundary. Each suppression chamber-to-drywell vacuum breaker is a self-actuating valve, similar to a check valve, which can be remotely operated for testing purposes. The function of the suppression chamber-to-drywell vacuum breakers is to relieve vacuum in the drywell.

Design-basis accident (DBA) analyses assume the vacuum breakers to be closed initially and to remain closed and leak tight until the suppression chamber is at a positive pressure relative to the drywell. The TS Bases state that the safety analyses assume that the internal vacuum breakers are closed initially and are fully open at a differential pressure of 0.5 psid. Additionally, two of the eight vacuum breakers are assumed to fail in a closed position. The results of the analyses show that the design pressure is not exceeded even under the worst case accident scenario. The vacuum breaker opening differential pressure setpoint and the requirement that seven of eight vacuum breakers be OPERABLE are a result of the requirement placed on the vacuum breakers to limit the vent system waterleg height.

Suppression Pool Volume and Temperature

Existing TS SR 4.7.A.1.c requires a visual inspection of the suppression chamber interior be conducted, including water line regions and interior painted surfaces above the water line. This SR supports primary containment integrity verification and assures that any potential containment degradation is identified. The licensee states the drywell and suppression chamber interiors are painted to minimize corrosion, and that inspection of the paint during each refueling outage assures the paint is intact and not deteriorating.

The licensee proposed to increase the surveillance interval for this SR from 18 months to 24 months, for a maximum interval of 30 months including the 25 percent grace period. As confirmed by historical plant maintenance and surveillance data, the licensee states this

surveillance has failed to identify significant discrepancies when performed at the current 18-month interval.

The NRC staff reviewed the information presented by the licensee and concludes that, based on a review of surveillance testing history, the proposed change does not have a significant effect on safety and follows the guidance of GL 91-04. Therefore, the proposed change to extend the surveillance interval for TS 4.7.A.1.c is acceptable.

Pressure Suppression Chamber-Drywell Vacuum Breakers

Existing TS SR 4.7.A.4.a.(2) requires that drywell to suppression chamber leakage shall be demonstrated to be less than that equivalent to a one-inch diameter orifice and that each vacuum breaker shall be visually inspected. This SR ensures that a bypass leak path does not exist between the drywell and suppression chamber and that the pressure suppression capability of the suppression pool is not bypassed by excessive leakage from the drywell to suppression chamber air space.

The licensee proposed to increase the surveillance interval for this SR from 18 months to 24 months, for a maximum interval of 30 months including the 25 percent grace period. The licensee states that these components have routinely passed this surveillance when performed at the current 18-month interval, that the system is designed to be single failure proof, and that there is no identified failure that would invalidate this conclusion.

The NRC staff reviewed the information presented by the licensee and concludes that, based on surveillance history, the proposed change does not have a significant effect on safety and follows the guidance of GL 91-04. Therefore, the proposed change to extend the surveillance interval for TS 4.7.A.4.a.(2) is acceptable.

Existing TS SR 4.7.A.4.a.(3) requires that pressure suppression chamber-to-drywell vacuum breaker position indication and alarm systems be calibrated and functionally tested. This SR ensures the vacuum breaker position indication and alarm capability are assured to verify equipment functionality during normal operations and during response to plant transients and design-basis events.

The licensee proposed to increase the surveillance interval for this SR from 18 months to 24 months, for a maximum interval of 30 months including the 25 percent grace period. The licensee states that the limit switches are mechanical devices, requiring mechanical adjustment only, and that drift is not applicable. Additionally, these components have routinely passed this surveillance when performed at the current 18-month interval, the system is designed to be single failure proof, and there is no identified failure that would invalidate this conclusion.

The NRC staff reviewed the information presented by the licensee and concludes that, based on surveillance history, the proposed change does not have a significant effect on safety and follows the guidance of GL 91-04. Therefore, the proposed change to extend the surveillance interval for TS 4.7.A.4.a.(3) is acceptable.

Existing TS SR 4.7.A.4.a.(4) requires that pressure suppression chamber-to-drywell vacuum breakers be tested to determine that the force required to open from fully-closed to fully-open

does not exceed the equivalent of 0.5 psid acting on the suppression chamber face of the valve disc. This SR validates that the vacuum breaker will fully open when subjected to a differential pressure of less than or equal to 0.5 psid, thus ensuring containment integrity by preventing an excessive negative differential pressure across the drywell boundary.

The licensee proposed to increase the surveillance interval for this SR from 18 months to 24 months, for a maximum interval of 30 months including the 25 percent grace period. The licensee states that the vacuum breakers have routinely passed this surveillance when performed at the current 18-month interval, that the system is designed to be single failure proof, and that there is no identified failure that would invalidate this conclusion.

The NRC staff reviewed the information presented by the licensee and concludes that, based on surveillance history, the proposed change does not have a significant effect on safety and follows the guidance of GL 91-04. Therefore, the proposed change to extend the surveillance interval for TS 4.7.A.4.a.(4) is acceptable.

Standby Gas Treatment System

The reactor building secondary containment feature is designed, in conjunction with other engineered safeguards, to limit the ground level release of airborne radioactive materials and to provide means for controlled elevated release of the building atmosphere so that off-site doses from postulated DBAs are below the guidelines of 10 CFR Part 100. The SBGT system consists of two parallel filter trains connected to three full capacity exhaust fans. The SBGT system is designed to limit the ground level release from the reactor building and to release primary and secondary containment air at an elevated release point via the main stack.

TS SR 4.7.B.2.a ensures that the SBGT filters that remove radioactive particulates and the charcoal absorbers that remove the radioactive halogens are functioning properly. Existing SR 4.7.B.2.a requires the licensee to perform the following: 1) in-place DOP test the HEPA filter banks; 2) in-place test the charcoal absorber banks with halogenated hydrocarbon tracer; 3) remove one carbon test sample from the charcoal absorber in accordance with regulatory Position C.6.b of Regulatory Guide (RG) 1.52, Revision 2, March 1978 and subject this sample to a laboratory analysis to verify methyl iodide removal efficiency. Redundant trains are available in the event of the failure of one of the system components. The licensee proposed to increase the surveillance interval to perform the above actions from a maximum of 18 months to 24 months, for a maximum interval of 30 months (25 percent grace period).

The licensee stated that operating experience shows that these components routinely pass the TS SR when performed at the 18-month interval. Based on the redundant trains of SBGT available and the more frequent testing required by other TS SRs, the licensee concluded that the impact on system availability as a result of extending the surveillance interval is minimal. In addition, the licensee stated that a review of the surveillance data demonstrated no evidence of any failures that would invalidate that conclusion.

The NRC staff reviewed the information presented by the licensee and concluded that the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to SR 4.7.B.2, are acceptable.

TS SR 4.7.B.2.b ensures that the there is sufficient flow across the combined filters and the heaters that reduce relative humidity are functioning properly. The licensee proposed to increase the surveillance interval from 18 months to 24 months, for a maximum interval of 30 months (25 percent grace period). Based on the redundant trains of SBGT available and the more frequent testing required by other TS SRs, the licensee concluded that the impact on system availability as a result of extending the surveillance interval is minimal. In addition, the licensee stated that a review of the surveillance data demonstrated no evidence of any failures that would invalidate that conclusion.

This TS SR has been renumbered to separate SR 4.7.B.2.b(1), SR 4.7.B.2.b(2) into SR 4.7.B.2.b and SR 4.7.B.2.c. This change will allow the performance of the 4.7.B.2.b SR on a once per 3-month basis and maintain the sequential numbering of the Monticello TS SRs. This is considered an administrative change; therefore, the proposed change is acceptable.

The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of ST history and on the redundant testing associated with TS 4.7.B.2.b, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to TS 4.7.B.2.b are acceptable.

TS SR 4.7.B.2.c ensures that whenever secondary containment isolation conditions exist, a small negative pressure is maintained, to minimize ground level escape of airborne radioactivity. The licensee proposed to increase the surveillance interval from 18 months to 24 months, for a maximum interval of 30 months (25 % grace period). Based on the redundant trains of SBGT available and the more frequent testing required by other TS SRs, the licensee concluded that the impact on system availability as a result of extending the surveillance interval is minimal. In addition, the licensee stated that a review of the surveillance data demonstrated no evidence of any failures that would invalidate that conclusion.

TS SR 4.7.B.2.c has been renumbered as SR 4.7.B.2.d. This change maintains the sequential numbering of the Monticello TS SRs. This is considered an administrative change; therefore, the proposed change is acceptable.

The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of ST history and on the redundant testing associated with TS SR 4.7.B.2.c, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to TS 4.7.B.2.c are acceptable.

Secondary Containment Systems

The function of secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a DBA. The two principal accident scenarios taking credit for secondary containment operability are a loss of coolant accident and a fuel handling accident involving handling of recently irradiated fuel.

The secondary containment performs no active function in response to each of these events; however, leak tightness is required to ensure that the release of radioactive materials is restricted to leakage rates assumed in the accident analysis. To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional

structure, support systems are required to maintain the control volume at less than external pressure.

Secondary containment relies upon the SBGT system and closure of certain valves whose lines penetrate secondary containment to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be operable, or that take place outside primary containment.

Existing TS SR 4.7.C.1.a. requires that secondary containment maintain at least a 1/4-inch of water vacuum under calm wind conditions with a filter train flow rate of $\leq$ 4,000 scfm. This ensures that under accident conditions fission product release to the environment is minimized and restricted to leakage rates assumed in the accident analysis. The exfiltration rate, which is almost directly proportional to the initial inleakage rate for a negative building pressure, will be maintained within the post-accident 10 CFR Part 100 dose limits.

The licensee proposed to increase the surveillance interval for this SR from 18 months to 24 months, for a maximum interval of 30 months including the 25 percent grace period. The licensee states that these components have routinely passed this surveillance when performed at the current 18-month interval and that there is no identified failure that would invalidate this conclusion.

The NRC staff reviewed the information presented by the licensee and concludes that, based on surveillance history, the proposed change does not have a significant effect on safety and follows the guidance of GL 91-04. Therefore, the proposed change to extend the surveillance interval for TS 4.7.C.1.a. is acceptable

Existing TS SR 4.7.C.1.b. verifies that each automatic damper actuates to its isolation position. This surveillance 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 vicinity of the spent fuel pool.

The licensee proposed to increase the surveillance interval for this SR from 18 months to 24 months, for a maximum interval of 30 months including the 25 percent grace period. The licensee states that these components have routinely passed this surveillance when performed at the current 18-month interval and that there is no identified failure that would invalidate this conclusion.

The NRC staff reviewed the information presented by the licensee and concludes that, based on surveillance history, the proposed change does not have a significant effect on safety and follows the guidance of GL 91-04. Therefore, the proposed change to extend the surveillance interval for TS 4.7.C.1.b. is acceptable.

Primary Containment Isolation Valves (PCIV)

TS SR 4.7.D.1.a ensures that each automatic PCIV will actuate to its isolation position on primary containment isolation signal. The TS SR Table 4.2.1 surveillances overlap this TS SR to provide complete testing of the system function. The licensee proposed to increase the

surveillance interval to perform the above actions from a maximum of 18 months to 24 months, for a maximum interval of 30 months (25 % grace period). The licensee requested this increase in order to perform this TS SR under shutdown conditions since isolation of penetrations eliminates cooling water flow and disrupts the normal operation of many critical components.

The licensee stated that operating experience shows that these components routinely pass the TS 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 licensee concluded that the impact on system availability as a result of extending the surveillance interval is minimal. In addition, the licensee stated that a review of the surveillance data demonstrated no evidence of any failures that would invalidate that conclusion.

The NRC staff reviewed the information presented by the licensee and concluded that the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to SR 4.7.D.1.a, are acceptable.

TS SR 4.7.D.1.b ensures that the instrumentation line excess flow check valve (EFCV) will perform as designed by verifying that the EFCV reduces flow on an actual or simulated instrument line break condition. The licensee proposed to increase the surveillance interval to from 18 months to 24 months, for a maximum interval of 30 months (25 % grace period). The licensee requested this increase in order to perform this TS SR under shutdown conditions to reduce the potential fo an unplanned transient if the TS SR were performed with the reactor at power.

The licensee stated that operating experience shows that these components routinely pass this TS SR when performed at the 18-month interval. The licensee stated that 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 licensee concluded that the impact on system availability as a result of extending the surveillance interval is minimal. In addition, the licensee stated that a review of the surveillance data demonstrated no evidence of any failures that would invalidate that conclusion.

The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of ST history and on the redundant testing associated with TS 4.7.D.1.b, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to TS 4.7.D.1.b are acceptable.

- TS 4.9 Auxiliary Electrical Systems

Standby Diesel Generators

SR 4.9.B.3.a.2 requires verifying once each operating cycle during shutdown, with a simulated loss of offsite power in conjunction with an ECCS actuation test signal, verification of the de-energization of the emergency busses, load shedding from the emergency busses, starting of the EDG from ambient conditions on the auto-start signal, and readiness of the EDGs to accept emergency loads within ten seconds. This surveillance requirement demonstrates

diesel generator operation during a loss of offsite power actuation test signal in conjunction with an ECCS initiation signal.

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 percent grace period. 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 their 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, the "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."

The licensee's proposed revisions to the above TS Sections follow the guidelines in GL 91-04, therefore based on the above the NRC staff finds the proposed change acceptable.

SR 4.9.B.4.c Station Battery System

SR 4.9.B.4.c requires that every refueling outage, the station batteries are subject to a rated load discharge test, and the specific gravity and voltage of each cell is determined after the test has been performed. 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. 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 percent grace period.

Extending the surveillance interval for this surveillance requirement is acceptable for the following reasons:

1. The design, in conjunction with TS requirements, which limits 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. The licensee 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.

The licensee's proposed revisions to the above TS Sections follow the guidelines contained in GL 91-04, therefore based on the above the NRC staff finds the proposed change acceptable.

- TS 4.13 - Alternate Shutdown System

The function of the Alternate Shutdown System (ASDS) provides control of the minimum necessary systems, once transfer switches are activated, to achieve safe shutdown.

The ASDS is required for a 10 CFR Part 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 safety grade (Class 1E) and provides controls, AC circuitry and instrument readouts to allow operators to safely shut down the plant at a centralized location assuming a fire prohibited normal shutdown operations from the control room.

SR 4.13.A.1 and SR 4.13.A.2 - Alternate Shutdown System

Existing TS SR 4.13.A.1 requires that switches on the ASDS panel be functionally tested, and TS SR 4.13.A.2 requires the ASDS panel master transfer switch be verified to alarm in the control room when unlocked. These surveillances ensure that ASDS instrumentation will function as designed during an analyzed event, and that operators are aware that control of specific instrumentation has been transferred to the ASDS panel.

The licensee describes these surveillances as essentially Logic System Functional Tests (LSFT) for the transfer circuits associated with shifting indication and control from the main control room to the ASDS panel. The licensee states that potential time-based considerations, such as instrument drift, failure types and frequencies, as well as other qualitative measures of system availability, were evaluated.

The licensee provides further justification based upon industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P). The study discusses that overall safety system reliability is not dominated by the logic system, but by 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 LSFT interval represents no significant change in the overall safety system unavailability.

The licensee proposed to increase the surveillance interval for these SRs from 18 months to 24 months, for a maximum interval of 30 months including the 25 percent grace period. The NRC staff reviewed the information presented by the licensee and concludes that based on the information provided, including surveillance history, the proposed change does not have a significant effect on safety and follows the guidance of GL 91-04. Therefore, the proposed change to extend the surveillance intervals for TS 4.13.A.1 and TS 4.13.A.2 is acceptable.

- TS Accident Monitoring

TS Table 4.14.1, Accident Monitoring Instrumentation, Function 1, Reactor Vessel Fuel Zone Water Level Monitor
(SR Table 4.14.1, Perform Channel Calibration)

Reactor vessel water level is measured to assure that the core is covered and that the separators are not flooded. 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. The accident-monitoring devices must maintain their function after the accident has occurred and track the progress of the event and event mitigation over a long period. During normal operations, the primary cause of transmitter error is drift. However, for accident-monitoring conditions the major errors are associated with changes in process conditions and changes in environmental conditions. These errors are in most cases orders of magnitude larger than errors resulting from drift. Extending the surveillance interval has minimal effect on system availability. The licensee reviewed the surveillance test history and found no failures that invalidate the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 4.14.1, Accident Monitoring Instrumentation, Function 2, Safety/Relief Valve Position (Pressure Switches)
(SR Table 4.14.1, Perform Channel Calibration)

The S/RV 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 indicates the valve is open by a light and an alarm in the control room. The accident-monitoring devices must maintain their function after the accident has occurred and track the progress of the event and event mitigation over a long period. During normal operations, the primary cause of transmitter

error is drift. However, for accident-monitoring conditions the major errors are associated with changes in process conditions and changes in environmental conditions. These errors are in most cases orders of magnitude larger than errors resulting from drift. Extending the surveillance interval has minimal effect on system availability. The licensee reviewed the surveillance test history and found no failures that invalidate the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 4.14.1, Accident Monitoring Instrumentation, Function 3, Safety/Relief Valve Position (Thermocouples)
(SR Table 4.14.1, Perform Channel Calibration)

The S/RVs have thermocouples installed in each of the discharge pipes. A temperature increase means an S/RV is opening or leaking. There is a multiple-channel recorder and alarm in the control room for this function. The accident monitoring devices must maintain their function after the accident has occurred and track the progress of the event and event mitigation over a long period. During normal operations, the primary cause of transmitter error is drift. However, for accident-monitoring conditions the major errors are associated with changes in process conditions and changes in environmental conditions. These errors are in most cases orders of magnitude larger than errors resulting from drift. Extending the surveillance interval has minimal effect on system availability. The licensee reviewed the surveillance test history and found no failures that invalidate the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 4.14.1, Accident Monitoring Instrumentation, Function 4, Drywell Wide Range Pressure Monitors
(SR Table 4.14.1, Perform Channel Calibration)

The drywell wide-range pressure-monitoring continuously indicates the drywell pressure in the control room. This variable is used to verify and provide long-term surveillance of ECCS functions. Two wide-range pressure transmitters provide signals to the recorders in the control room. The accident monitoring devices must maintain their function after the accident has occurred and track the progress of the event and event mitigation over a long period. During normal operations, the primary cause of transmitter error is drift. However, for accident-monitoring conditions the major errors are associated with changes in process conditions and changes in environmental conditions. These errors are in most cases orders of magnitude larger than errors resulting from drift. Extending the surveillance interval has minimal effect on system availability. The licensee reviewed the surveillance test history and found no failures that invalidate the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 4.14.1, Accident Monitoring Instrumentation, Function 5, Suppression Pool Wide Range Level Monitors
(SR Table 4.14.1, Perform Channel Calibration)

The suppression pool wide-range level signal is used to assess the status of the water supply to ECCS and provide long-term surveillance of ECCS functions. Two wide-range level transmitters provide signals to the recorders in the control room. The accident monitoring devices must maintain their function after the accident has occurred and track the progress of the event and event mitigation over a long period. During normal operations, the primary cause of transmitter error is drift. However, for accident-monitoring conditions the major errors are associated with changes in process conditions and changes in environmental conditions. These errors are in most cases orders of magnitude larger than errors resulting from drift. Extending the surveillance interval has minimal effect on system availability. The licensee reviewed the surveillance test history and found no failures that invalidate the conclusion that the effect of the proposed change on system availability is minimal. Based on review of the information in the submittal, the NRC staff finds the proposed change acceptable.

TS Table 4.14.1, Accident Monitoring Instrumentation, Function 6, Suppression Pool Temperature
(SR Table 4.14.1, Perform Channel Calibration)

The suppression pool temperature-monitoring system gives the plant operator reliable information so that the plant can be operated within TS limits. Two independent and redundant divisions are provided for this function. The accident monitoring devices must maintain their function after the accident has occurred and track the progress of the event and event mitigation over a long period. During normal operations, the primary cause of sensor error is drift. However, for accident-monitoring conditions the major errors are associated with changes in process conditions and changes in environmental conditions. These errors are in most cases orders of magnitude larger than errors resulting from drift. Extending the surveillance interval has minimal effect on system availability. The licensee reviewed the surveillance test history and found no failures that invalidate the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 4.14.1, Accident Monitoring Instrumentation, Function 7, Drywell High Range Radiation Monitors
(SR Table 4.14.1, Perform Channel Calibration)

The containment high-radiation monitoring system consists of sensor units 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 module converts the output current to radiation readout for alarm and recording. The major cause of radiation monitor error is the inaccuracy of the detector and the calibration sources. The drift of the electronic circuit does not provide a measure of functional performance between calibrations. During normal operations, the primary cause of sensor error is drift. However, for accident-monitoring

conditions the major errors are associated with changes in process conditions and changes in environmental conditions. These errors are in most cases orders of magnitude larger than errors resulting from drift. Extending the surveillance interval has minimal effect on system availability. The licensee reviewed the surveillance test history and found no failures that invalidate the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 4.14.1, Accident Monitoring Instrumentation, Function 8, Offgas Stack Wide Range Radiation Monitors
(SR Table 4.14.1, Perform Channel Calibration)

Radiation monitors are provided in the plant main stack for backup detection of high-activity releases. Radiation levels in excess of the allowable “instantaneous” release rate alarm in the control room and isolate the holdup line. Operators use these recorders during an accident. The major cause of radiation monitor errors is the inaccuracy of the detector and the calibration sources. The drift of the electronic circuit does not provide a measure of functional performance between calibrations. During normal operations, the primary cause of sensor error is drift. However, for accident-monitoring conditions the major errors are associated with changes in process conditions and changes in environmental conditions. These errors are in most cases orders of magnitude larger than errors resulting from drift. Extending the surveillance interval has minimal effect on system availability. The licensee reviewed the surveillance test history and found no failures that invalidate the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

TS Table 4.14.1, Accident Monitoring Instrumentation, Function 9, Reactor Building Wide Range Radiation Monitors
(SR Table 4.14.1, Perform Channel Calibration)

The reactor building wide-range radiation monitoring system continuously monitors radioactivity in the reactor building ventilation exhaust and provides a permanent record of the observed radiation levels. Operators use these recorders primarily during an accident. During normal operations, the primary cause of sensor error is drift. However, for accident-monitoring conditions the major errors are associated with changes in process conditions and changes in environmental conditions. These errors are in most cases orders of magnitude larger than errors resulting from drift. Extending the surveillance interval has minimal effect on system availability. The licensee reviewed the surveillance test history and found no failures that invalidate the conclusion that the effect of the proposed change on system availability is minimal. The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of history of the surveillance testing, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to the TSs are acceptable.

- TS 4.17 Control Room Habitability

Control Room Emergency Filtration System

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 steamline 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 HEPA filters and charcoal adsorbers.

TS SR 4.17.B.2.a ensures that control room personnel are protected against radioactive gases so that the plant can be safely shut down under DBA conditions. Redundant trains are available in the event of the failure of one of the system components. The licensee proposed to increase the surveillance interval to perform the above actions from a maximum of 18 months to 24 months, for a maximum interval of 30 months (25 percent grace period).

The licensee stated that the charcoal adsorber 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 and each module withdrawn is replaced or blocked off. If test results are unacceptable, all adsorbent in the train is replaced and any HEPA filters found defective are replaced. The licensee stated that operating experience shows that these components routinely pass the TS SR when performed at the 18-month interval. Based on the above, the licensee concluded that the impact on system availability as a result of extending the surveillance interval is minimal. In addition, the licensee stated that a review of the surveillance data demonstrated no evidence of any failures that would invalidate that conclusion.

The NRC staff reviewed the information presented by the licensee and concluded that the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to SR 4.17.B.2.a, are acceptable.

TS SR 4.17.B.2.c ensures that control room personnel are protected against radioactive gases so that the plant can be safely shut down under DBA conditions. The licensee stated that 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. The licensee proposed to increase the surveillance interval from 18 months to 24 months, for a maximum interval of 30 months (25 percent grace period). Based on the above, the licensee concluded that the impact on system availability as a result of extending the surveillance interval is minimal. In addition, the

licensee stated that a review of the surveillance data demonstrated no evidence of any failures that would invalidate that conclusion.

The NRC staff reviewed the information presented by the licensee and concluded that, based on the review of surveillance test (ST) history and on the redundant testing associated with TS 4.17.B.2.c, the proposed changes do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to TS 4.17.B.2.c are acceptable.

4.0 TECHNICAL CONCLUSION

Based on the review of the Monticello 24-month fuel cycle implementation license amendment request dated June 30, 2004, as supplemented by letters dated November 5, 2004, and March 3, July 1, and September 27, 2005, the NRC staff finds that the proposed changes to extend the calibration surveillance frequency (24 months, for a maximum interval of 30 months, including the 25 percent grace period) is in conformance with guidance in the GL 91-04. The NRC staff has determined that the licensee has addressed the issues identified in GL 91-04 and has provided an acceptable basis for increasing the calibration interval for instruments used to perform safety functions. Based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the changes will not be inimical to the common defense and security or to the health and safety of the public.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (70 FR 2892). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: HLi
JStang
TBeltz
AMuniz
VKlein

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