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10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Monticello Nuclear Generating Plant
Docket 50-263
Renewed Facility Operating License No. DPR-22

License Amendment Request: Revise the Modes of Applicability for Reactor Water Cleanup System Isolation on a Standby Liquid Control System Initiation

As part of the full-scope alternative source term amendment, the applicability of the Standby Liquid Control (SLC) System in Specification 3.1.7 was revised to add Mode 3 to the existing modes, Modes 1 and 2 (Enclosure 1, Reference 1). Pursuant to 10 CFR 50.90, the Northern States Power Company – Minnesota (NSPM), a wholly owned subsidiary of Xcel Energy, Inc., proposes to revise the applicability for isolation of the Reactor Water Cleanup System on a SLC System initiation, Function 5.d, in Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," to align with the modes stated in Specification 3.1.7.

Summary of Commitments

This letter proposes no new commitments or revisions to any existing commitments.

Enclosure 1 provides a description of the proposed changes and includes the technical evaluation and associated no significant hazards determination and environmental evaluation. Enclosure 2 provides a marked-up copy of the TS pages showing the proposed changes. Enclosure 3 provides a copy of the associated draft marked-up TS Bases pages for information.

The NSPM requests approval of the proposed license amendment by one year from the date of submittal, with an implementation period of 90 days.

The MNGP Plant Operations Review Committee has reviewed this application. In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Minnesota Official.

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I declare under penalty of perjury that the foregoing is true and correct.
Executed on May 27, 2009.

Timothy J. O'Connor
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosures (3)

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC
Minnesota Department of Commerce

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ENCLOSURE 1

DESCRIPTION OF CHANGE

LICENSE AMENDMENT REQUEST
REVISE MODES FOR ISOLATION OF REACTOR WATER CLEANUP SYSTEM ON A STANDBY LIQUID CONTROL SYSTEM INITIATION

1.0 SUMMARY

As part of the alternative source term (AST) amendment the Standby Liquid Control (SLC) System applicability in Specification 3.1.7, "Standby Liquid Control (SLC) System," was revised to add Mode 3 to the then existing modes, Modes 1 and 2. Amendment 148 (Reference 1) approved application of a full-scope AST at the Monticello Nuclear Generating Plant (MNGP). The applicability of Specification 3.1.7 was revised to include Mode 3 consistent with the modes for which SLC System operation is credited to maintain suppression pool pH within post-Loss of Coolant Accident (LOCA) limits, thereby, ensuring the offsite and Control Room doses remain within 10 CFR 50.67, i.e., AST limits.

Function 5.d in Table 3.3.6.1-1 of Specification 3.3.6.1, "Primary Containment Isolation Instrumentation," however, indicates that upon a SLC System initiation that Reactor Water Cleanup (RWCU) System isolation is only necessary in Modes 1 and 2. To correct this inconsistency, the Northern States Power Company – Minnesota (NSPM) proposes to add Mode 3 to the modes of applicability listed for Function 5.d to align the primary containment isolation instrumentation and SLC System specifications and, the full-scope AST analysis.

2.0 BACKGROUND

On June 29, 2005, the Nuclear Management Company, (NMC) LLC a predecessor license holder to the NSPM[1] submitted a license amendment request (LAR) for a full-scope conversion from the custom Technical Specifications (CTS) to the improved standard technical specifications (ITS) format (Reference 2). As part of the conversion CTS Section 3.4, "Standby Liquid Control System," became Specification 3.1.7, "Standby Liquid Control (SLC) System," in the ITS, with Modes 1 and 2 specified as the modes of applicability in accordance with the Boiling Water Reactor (BWR) standard technical specification (TS) NUREG applicable to the BWR-3, MNGP, design (Reference 3). This current BWR standard TS NUREG includes an instrument function in Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," requiring isolation of the RWCU System on a SLC System initiation, specified as required in Modes 1 and 2. This RWCU System isolation function was not included in the preceding MNGP CTS.

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1. NSPM is incorporated as a wholly owned subsidiary of Xcel Energy, Inc. Transfer of operating authority from the NMC to NSPM occurred on September 15, 2008.

On September 15, 2005, the NMC submitted a LAR for a full-scope conversion to an AST (Reference 4). This request was developed and submitted while the MNGP was operating under the CTS. With adoption of the full-scope AST the SLC System became credited for control of suppression pool pH to mitigate the radiological consequences of a design basis accident (DBA) loss-of-coolant accident (LOCA) in which fuel is damaged.

On June 5, 2006, Amendment 146 (Reference 5) was issued approving conversion to the ITS. A supplement to the AST full-scope LAR, dated August 18, 2006 (Reference 6), was submitted translating the proposed AST changes from the CTS format to the corresponding ITS format. The applicability of the SLC System in Specification 3.1.7 was revised in this supplement to reflect the addition of Mode 3 to the previously existing modes of applicability for the SLC System, i.e., Modes 1 and 2, which had been carried over from CTS to ITS. The supplement also included full-scope AST related changes to other ITS specifications/functions which had not existed in the previous CTS.

The applicability for isolation of the RWCU System on a SLC System initiation, Function 5.d in Table 3.3.6.1-1, should have been changed to include Mode 3, with the adoption of full-scope AST, but this then newly introduced ITS function was not recognized as being affected by adoption of an AST.

Note that the actual impact of the SLC System initiation function not being required OPERABLE in Mode 3 by the current TS is inconsequential because the channels associated with the function are actuated based on the position of the SLC System initiation switch.

3.0 SUMMARY DESCRIPTION OF THE STANDBY LIQUID CONTROL SYSTEM DESIGN AND LICENSING BASIS

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies 10 CFR 50.62 for an anticipated transient without scram (ATWS).

Additionally, following adoption of a full-scope AST (Reference 1), the SLC System is used to maintain suppression pool pH following a LOCA involving significant fission product release. This ensures that iodine will be retained within the suppression pool water and that offsite and Control Room doses from the accident will remain within the 10 CFR 50.67, i.e., AST limits. The full-scope AST analyses credit the manual initiation of the SLC System for maintaining the pH of the suppression pool. Initiation of the SLC System is a manual operator action assumed to occur within the first hour of the accident.

As described in the Bases for TS Table 3.3.6.1-1, isolation of the RWCU System on initiation of Function 5.d, "SLC System Initiation," is desirable to prevent dilution and removal of the boron solution by the RWCU System. The current Bases state:

> The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 4). SLC System initiation signals are initiated from the SLC initiation switch.
>
> Two channels of the SLC System Initiation Function are available and are required to be OPERABLE only in MODES 1 and 2, since these are the only MODES where the reactor can be critical, and these MODES are consistent with the Applicability for the SLC System (LCO 3.1.7, "Standby Liquid Control (SLC) System").
>
> There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

Extending the requirement for RWCU isolation on a SLC System initiation to also include Mode 3, is necessary to prevent dilution and removal of the boron solution by the RWCU System. The proposed change aligns the applicability of this function, i.e., 5.d for SLC Initiation in TS Table 3.3.6.1-1 with the Applicability of the SLC System in accordance with the specification for the system (Specification 3.1.7).

4.0 DESCRIPTION OF THE PROPOSED CHANGE

Based upon the SLC System design function under full-scope AST to control suppression pool pH post-LOCA, the following change is proposed:

- Revise the "Applicable Modes or Other Specified Conditions" in Table 3.3.6.1-1, Function 5.d, "SLC System Initiation," under the RWCU System Isolation heading, to add Mode 3, to the listed Modes 1 and 2.

- Revise the TS Bases for Function 5.d in Table 3.3.6.1-1 to add Mode 3 to the modes requiring RWCU isolation on a SLC System initiation. Also, expand the definition to include the RWCU isolation in conjunction with SLC System operation with respect to AST.

A mark-up of the proposed change to Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," in Specification 3.3.6.1 is provided in Enclosure 2. Enclosure 3 provides a copy of the associated draft TS Bases pages. The Bases changes will be issued in accordance with Specification 5.5.9, "Technical Specification TS Bases Control Program," following approval of this LAR.

5.0 TECHNICAL ANALYSIS

The full-scope AST analysis, approved under Amendment 148, credited the SLC System for suppression pool pH control to mitigate the consequences of a DBA-LOCA in which fuel is damaged. As described in Section 3.1.1.2, "LOCA Fission Product Transport," (page 4) of the NRC staff safety evaluation for the full-scope AST amendment (Reference 1):

> The licensee proposes to use the standby liquid control (SLC) system to inject sodium pentaborate to the RPV [reactor pressure vessel], where it will mix with ECCS flow and spill over into the suppression pool. Sodium pentaborate, being a base, will neutralize acids generated in the post-accident primary containment environment. Use of the SLC to add buffering agent assures that the assumptions on iodine release and speciation as used in the AST dose analysis are acceptable.

Under the CTS the applicable modes or other specified conditions for which SLC System operation was credited, after full-scope AST approval, were Run, Startup, and Hot Shutdown, which correlate to Modes 1, 2 and 3 under the ITS. Section 3.2, "Standby Liquid Control System," (page 19) of the NRC staff safety evaluation for the full-scope AST amendment states:

> The specific changes being made to TS Section 3.4.A.1 [the CTS SLC System Specification], to require the SLC system to be operable at all times during Run, Startup, and Hot Shutdown time, is appropriate for this action. On the basis of the above discussion, the staff finds this change to be acceptable.

Therefore, the same modes of applicability, and the equivalent ITS modes, are acceptable as stated in Section 3.4 (page 22) of the AST full-scope safety evaluation:

> This requirement is revised to specify the SLC system to be operable at all times during Modes 1, 2 and 3. The licensee states that this change reflects the use of the SLC system for maintaining suppression pool pH following a design-basis LOCA involving fuel damage, as assumed in the radiological consequence analysis. ...

Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," in Specification 3.3.6.1, states however, that the RWCU System is isolated only in Modes 1 and 2 upon a SLC System initiation (Function 5.d). As described in the TS Bases for TS Table 3.3.6.1-1, isolation of the RWCU System on SLC initiation is necessary to prevent dilution and removal of the boron solution by the RWCU System initiation; and with approval of AST, Mode 3 should have been added consistent with the applicability for the SLC System in Specification 3.1.7. Therefore, this proposed change is consistent with the present analysis basis and merely corrects and internal inconsistency within the MNGP TS. No new requirements are imposed and the existing NRC approved safety analysis supports the proposed changes.

The actual impact of Function 5.d in Table 3.3.6.1-1 not currently being required by this TS to be OPERABLE in Mode 3 was inconsequential because while administratively, i.e., as listed in the table, the function was not required OPERABLE, in actuality the function was forced to be OPERABLE because isolation of the RWCU System occurs solely based on the position of the SLC System initiation switch. Therefore, anytime the SLC System is initiated, this requirement is met, i.e., RWCU isolates. If it an event were to occur in Mode 3 (or if the plant transitioned to Mode 3 from a higher mode in which SLC had been initiated), the requirement protected by Function 5.d , to isolate RWCU, would have occurred since on SLC System initiation the switch position results in RWCU isolation.

6.0 REGULATORY ANALYSIS

6.1 No Significant Hazards Determination

In accordance with the requirements of 10 CFR 50.90, the Northern States Power Company – Minnesota (NSPM) requests an amendment to facility Renewed Operating License DPR-22, for the Monticello Nuclear Generating Plant (MNGP) to revise the applicability of Technical Specification (TS) Table 3.3.6.1-1 Function 5.d, which isolates the Reactor Water Cleanup (RWCU) System on initiation of the Standby Liquid Control (SLC) System, to include Mode 3.

The NSPM has evaluated the proposed amendment in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of the MNGP in accordance with the proposed amendment presents no significant hazards. NSPM's evaluation against each of the criteria in 10 CFR 50.92 follows.

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change to the applicability of this function does not change the actual conditions, operating configurations, or minimum amount of operating equipment assumed in the safety analysis for accident mitigation.

The proposed change does not require any physical change to any plant systems, structures, or components nor does it require any change in systems or plant operations. The proposed change does not require any change in safety analysis methods or results. The SLC System is not an accident initiator. The proposed change to align the required modes of applicability for the RWCU isolation function on SLC initiation provide

consistency with the previously NRC approved full-scope alternative source term (AST) analysis and hence do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   There are no hardware changes nor are there any changes in the method by which any plant systems perform a safety function. This request does not affect the normal method of plant operation.

   The proposed change does not introduce new equipment, which could create a new or different kind of accident. No new equipment failure modes are created. No new accident scenarios failure mechanisms, or limiting single failures are introduced as a result of this request. Therefore, the implementation of the proposed change will not create a possibility for an accident of a new or different type than those previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

   Response: No.

   There will be no change to the manner in which the SLC System or the RWCU System is operated. This change aligns the requirements in one part of the TS with requirements imposed in another portion of the TS. No new requirenments are introduced. The proposed change improves the TS by removing an internal inconsistency and as such does not reduce or involve a significant reduction in a margin of safety.

Based on the above, the NSPM has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92(c), in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

6.2 Applicable Regulatory Requirements

The MNGP was designed largely before the publishing of the 70 General Design Criteria (GDC) for Nuclear Power Plant Construction Permits proposed by the Atomic Energy Commission (AEC) for public comment in July 1967, and constructed prior to the 1971 publication of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10 CFR Part 50. As such, the MNGP was not licensed to the Appendix A, General Design Criteria (GDC).

The MNGP USAR, Section 1.2, lists the principal design criteria (PDCs) for the design, construction and operation of the plant. USAR Appendix E provides a plant comparative evaluation to the 70 proposed AEC design criteria. It was concluded that the plant conforms to the intent of the GDCs. GDCs and PDCs considered applicable are discussed below.

PDC Criterion 38 - Reliability and Testability of Engineered Safely Features

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

GDC - Criterion 21 - Protection system reliability and testability.

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

As described in the Bases for Specification 3.1.7 and Table 3.3.6.1-1 for Function 5.d, isolation of the RWCU System on SLC System initiation is necessary to prevent dilution and removal of the boron solution by the RWCU System. While the MNGP was designed before the publishing of NUREG-0800, the Standard Review Plan (SRP) (Reference 7) provides guidance therein germane to the design of these systems. SRP 5.4.8, "Reactor Water Cleanup

System (BWR)," states under Section III, "Review Procedures," that the RWCU System design (referred to as the RWCS therein) should include:

> Provisions to automatically terminate flow to the RWCS following liquid poison injection into the reactor water.

This provision was included in the SLC System design. Later this provision was captured in the applicable TS requirements. Therefore, the proposed change merely serves to clearly restate system design requirements in the applicable portions of the present TS.

The NSPM has evaluated the proposed changes against the applicable regulatory requirements and acceptance criteria. The technical analysis concludes that the proposed changes will continue to assure that the SLC System and RWCU System design requirements and assumptions of the AST analysis are met. Based on this, there is reasonable assurance that the health and safety of the public, following approval of this change, is unaffected.

7.0 ENVIRONMENTAL EVALUATION

The NSPM has determined that the proposed amendment would not change a requirement with respect to installation or use of a facility or component located within the restricted area, as defined in 10 CFR 20, nor would it change an inspection or surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, or (ii) authorize a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) result in a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for a categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, the NSPM concludes pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

REFERENCES

1. Letter NRC to NMC, "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Full-Scope Implementation of the Alternative Source Term Methodology (TAC No. MC8971)," dated December 7, 2006.

2. Letter NMC to NRC, "License Amendment Request: Conversion of Current Technical Specifications (CTS) to Improved Technical Specifications (ITS)," (L-MT-05-072) dated June 29, 2005.

3. NUREG-1433, Revision 3.0, "Standard Technical Specifications General Electric Plants, BWR/4," June 2004.

4. Letter NMC to NRC, "License Amendment Request - Full Scope Application of an Alternative Source Term," (L-MT-05-093) dated September 15, 2005.

5. Letter NRC to NMC, "Monticello Nuclear Generating Plant (MNGP) - Issuance of Amendment for the Conversion to the Improved Technical Specifications with Beyond-Scope Issues (TAC Nos. MC7505, MC7597 through MC7611, and MC8887)," dated June 5, 2006.

6. Letter NMC to NRC, "Supplemental Submittal Regarding Full Scope Application of an Alternative Source Term (AST)(TAC No. MC8971)," (L-MT-06-054) dated August 18, 2006.

7. NUREG-0800, Standard Review Plan 5.4.8, "Reactor Water Cleanup System (BWR)," Revision 2, July 1981.

ENCLOSURE 2

MONTICELLO NUCLEAR GENERATING PLANT

MARKED-UP PROPOSED TECHNICAL SPECIFICATION CHANGES

(1 page follows)

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. RWCU System Isolation					
d. SLC System Initiation	1, 2, 3	1	H	SR 3.3.6.1.6	NA
e. Reactor Vessel Water Level - Low Low	1, 2, 3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -48 inches
6. Shutdown Cooling System Isolation					
a. Reactor Steam Dome Pressure - High	1, 2, 3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 81.8 psig
b. Reactor Vessel Water Level - Low	3, 4, 5	2[(a)]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 7 inches
7. Traversing Incore Probe System Isolation					
a. Reactor Vessel Water Level - Low	1, 2, 3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 7 inches
b. Drywell Pressure - High	1, 2, 3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 2.0 psig

(a) Only one channel per trip system, with an isolation signal available to one shutdown cooling supply isolation valve, is required in MODES 4 and 5, provided RHR Shutdown Cooling System integrity is maintained.

ENCLOSURE 3

MONTICELLO NUCLEAR GENERATING PLANT

DRAFT TECHNICAL SPECIFICATION BASES PAGES

(FOR INFORMATION)

(2 pages follow)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The RWCU Room Temperature - High Allowable Value is set low enough to detect a leak equivalent to 210 gpm.

This Function isolates the Group 3 valves.

5.c. Drywell Pressure - High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded. The Drywell Pressure - High Function, associated with isolation of the primary containment, is implicitly assumed in the USAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Four channels of Drywell Pressure - High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure - High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

This Function isolates the Group 3 valves.

5.d. SLC System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 4). SLC System initiation signals are initiated from the SLC initiation switch.

in MODES 1, 2 and 3,

Two channels of the SLC System Initiation Function are available and are required to be OPERABLE ~~only in MODES 1 and 2, since these are the only MODES where the reactor can be critical, and these MODES are~~ consistent with the Applicability for the SLC System (LCO 3.1.7, "Standby Liquid Control (SLC) System") following adoption of an alternative source term, Amendment 148 (Ref. 7).

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

This Function isolates the Group 3 valves.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.4 and SR 3.3.6.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.6.1.4 is based on the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.6.1.5 is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.6.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on PCIVs in LCO 3.6.1.3 overlaps this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. USAR, Section 14.7.2.
2. USAR, Section 14.7.3.
3. USAR, Section 7.6.3.2.4.
4. USAR, Section 6.6.1.1.
5. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.
6. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.
7. Amendment No. 148, "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Full-Scope Implementation of the Alternative Source Term Methodology (TAC No. MC8971)," December 7, 2006.