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Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

September 27, 2005

L-MT-05-097
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Monticello Nuclear Generating Plant
Docket 50-263
License No. DPR-22

Additional Information to Support Technical Specification Changes for 24-Month Fuel Cycle License Amendment Request (TAC No. MC3692)

References: 1) Letter from NMC to NRC, "License Amendment Request to Support 24-Month Fuel Cycles," (L-MT-04-036) dated June 30, 2004.

On June 30, 2004, the Nuclear Management Company, LLC (NMC) submitted a license amendment request (Reference 1) to revise the Monticello Nuclear Generating Plant (MNGP) Technical Specifications (TS) to implement a 24-month fuel cycle. On September 19, 2005, during a telephone discussion with the NRC additional information was requested on whether the setpoints proposed to be changed were Limiting Safety System Settings (LSSS) associated with a variable upon which a safety limit had been placed.

NMC has reviewed the trip setpoints proposed in the 24-month fuel cycle license amendment request to determine whether the trip setpoints constitute LSSS in accordance with 10 CFR 50.36(c)(1)(ii)(A). Results of our review, as presented in Enclosure 1, indicate that none of the trip setpoints proposed to be revised constitute a LSSS in accordance with the regulation.

This letter contains no new commitments and makes no revisions to existing commitments.

A001

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I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 27, 2005

John T. Conway
Site Vice President, Monticello Nuclear Generating Plant
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Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
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ENCLOSURE 1

1.0 SUMMARY

On June 30, 2004, the Nuclear Management Company, LLC (NMC) submitted a license amendment request (Reference 1) to revise the Monticello Nuclear Generating Plant (MNGP) Technical Specifications (TS) to implement a 24-month fuel cycle. The following trip setpoints are proposed to be revised:

- High Temperature in the Main Steam Line Tunnel
- Core Spray and LPCI Pump Reactor Low Pressure Permissive Bypass Timer
- 4.16 kV Essential Bus Loss of Voltage Protection
- Low-Low Set Reactor Coolant System Pressure for Opening / Closing
- Low-Low Set Discharge Pipe Pressure Inhibit and Position Indication
- Low-Low Set Inhibit Timer

NMC has reviewed these trip setpoints to determine whether they are limiting safety system settings (LSSS) in accordance with 10 CFR 50.36(c)(1)(ii)(A). None of the trip setpoints were determined to constitute a LSSS in accordance with the regulation.

2.0 BACKGROUND

On March 31, 2005, the U.S. Nuclear Regulatory Commission (NRC) issued a letter to Nuclear Energy Institute (NEI) (Reference 2) providing their position related to resolving the ongoing instrument setpoint methodology issue. In a letter dated May 18, 2005, (Reference 3) the NEI Setpoint Methods Task Force (SMTF) proposed seven concepts to resolve the NRC staff issues. These concepts were clarified during a public meeting with the NEI SMTF on June 2, 2005.

On June 3, 2005, the NRC issued a request for additional information (RAI) (Reference 4) stating that the NRC "staff is reviewing your request and finds that it needs Nuclear Management Company (NMC) to incorporate certain commitments and technical specification changes into its license amendment request (LAR) in order for us to complete our review." On July 1, 2005, NMC provided a response (Reference 5).

On August 23, 2005, the NRC issued a letter to NEI (Reference 6) providing their current requirements for resolving this issue. The staff stated in this letter that they do "not anticipate further changes to these concepts, and intends to follow them in its current reviews of plant-specific license amendment requests." On September 19, 2005, during a telephone discussion with the NRC additional information was requested on whether the setpoints proposed to be changed were LSSS associated with a variable upon which a safety limit had been placed.

3.0 DETERMINATION OF TRANSIENT AND ACCIDENT ANALYSIS LIMITS

Events analyzed as part of the safety analysis are categorized by type and frequency of occurrence. Safety Limits and LSSS may be specified as acceptance criteria or input criteria for normal operations and anticipated operational occurrences (transients). Emergency Core Cooling System (ECCS) performance criteria and release limits are specified for the accidents related to these setpoints. Various design limits may also apply. NMC has reviewed the proposed trip setpoints and determined that the criteria associated with accidents and certain design limits, i.e., categories 2 and 3 below, apply.

1. Safety Limits and Associated LSSS Specified in the Technical Specifications
   - Fuel Cladding Integrity Safety Limit
   - Reactor Coolant System Pressure Safety Limit
   - Reactor Vessel Water Level Above the Top of Active Fuel Safety Limit
2. Accident Acceptance Criteria Specified by Regulation
   - 10 CFR 50.46 Emergency Core Cooling Performance Criteria
   - 10 CFR 100 Release Limits
3. Design Acceptance Criteria
   - Containment and Safety / Relief Valve Piping Loads

None of the trip setpoints were determined to constitute a LSSS in accordance with 10 CFR 50.36(c)(1)(ii)(A) as discussed in Section 4.

10 CFR 50.36 requires the TS to include safety limits, LSSS, and Limiting Conditions for Operation (LCO) among other items. 10 CFR 50.36(c)(1) sets forth the criteria for safety limits and LSSS. 10 CFR 50.46 specifies the evaluation models and ECCS performance criteria. 10 CFR 100 specifies the release limits in the event of an accident. The safety limits and LSSS for each reactor design are specified in Section 2 of TS.

3.1 Specification of Safety Limits

10 CFR 50.36(c)(1)(i)(A) states:

> "Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity."

The safety limits for the MNGP are defined in Section 2 of the TS. They are:

"A. Reactor Core Safety Limits

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

   Thermal power shall be ≤ 25% Rated Thermal Power.

2. With the reactor steam dome pressure greater ≥ 785 psig and core flow ≥ 10% 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.

B. Reactor Coolant System Pressure Safety Limit

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 Water Level (Shutdown Condition) - 2.1.A.3

During periods when the reactor is shutdown, consideration must also be given to water requirements due to the effect of decay heat. If reactor water level should drop below the top of active fuel (TAF) during this time, the ability to cool the core is reduced. Establishment of the safety limit above the top of the fuel provides adequate safety margin.

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 a reactor scram is initiated on high pressure by RPS to assure the RCS safety limit is not exceeded. The RPS does not require operation of any safety system auxiliaries to perform its safety function and the safety function of the SRVs is independent of the Low-Low Set actuation of the S/RVs.

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

3.2 Specification of Limiting Safety System Settings

10 CFR 50.36(c)(1)(ii)(A) states:

> "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

<u>MNGP Limiting Safety System Settings</u>

The LSSS for MNGP are listed below.[2] They are the only LSSS identified in the MNGP licensing basis.

- Neutron Flux Intermediate Range Monitor (IRM) – 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 (MSIV) Closure
- Turbine Control Valve Fast Closure
- Turbine Stop Valve Closure
- Main Steam Line Low Pressure Initiates MSIV Closure

<u>Current Industry / NRC Formal Understanding on Limiting Safety System Settings</u>

The improved STS as reflected through the associated NUREGs, reflects the most recent agreed to NRC / industry position on LSSS (prior to the development of the current staff concern). It identifies that LSSS are associated with RPS. The Background to the RPS Instrumentation section states:

> 'The Reactor Protection System (RPS) initiates a reactor scram when one or more monitored parameters exceed their specified limits, to preserve the integrity of the fuel cladding and the Reactor Coolant System (RCS) and minimize the energy that must be absorbed following a loss of coolant accident (LOCA).
>
> The protection and monitoring functions of the RPS have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by

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2 Instrumentation trip setpoints listed in the LSSS table were incorporated into Section 3 of the MNGP TS in Amendment 128.

> the RPS, as well as LCOs on other reactor system parameters and equipment performance.'

A more current staff position was presented in a letter to the Nuclear Energy Institute (NEI) on March 31, 2005 (Reference 2), which stated on page 3:

> "As you [NEI] noted in your March 18, 2005, letter the systems these instruments [LSSS related] are typically associated with are the ... reactor protection system (RPS) and emergency core cooling system (ECCS) for boiling water reactors (BWRs). The NRC staff agrees that ... RPS and ECCS are the typical systems for ... BWRs ... for which LSSS are established to protect the safety limits." [emphasis added]

Therefore, as stated in the March 31, 2005, letter from the NRC to NEI (paraphrased above for BWRs) there is general agreement between the industry and the NRC that for BWRs the RPS and ECCS are the typical systems that have instrumentation setpoints which may be LSSS related.

4.0 REVIEW OF 24-MONTH FUEL CYCLE INSTRUMENT TRIP SETPOINTS VERSUS LSSS REQUIREMENTS

The NRC staff indicated a concern in a March 18, 2005, NRC letter to NEI with explicitly limiting the scope of systems covered by a generic solution to resolve the setpoint issue since "there may be other plant-specific systems that could be included within the scope of systems covered by 10 CFR 50.36(c)(1)(ii)(A)" and the solution "may not be consistent with some plant-specific licensing bases.."

Consistent with this concern, NMC has reviewed the trip setpoints proposed in the 24-month fuel cycle license amendment request, listed below, for the MNGP to determine whether the trip setpoints constitute a LSSS in accordance with 10 CFR 50.36(c)(1)(ii)(A).

- High Temperature in the Main Steam Line Tunnel
- Core Spray and LPCI Pump Reactor Low Pressure Permissive Bypass Timer
- 4.16 kV Essential Bus Loss of Voltage Protection
- Low-Low Set Reactor Coolant System Pressure for Opening / Closing
- Low-Low Set Discharge Pipe Pressure Inhibit and Position Indication
- Low-Low Set Inhibit Timer

It was concluded that the trip setpoints are not directly associated with any safety limit. Therefore, the trip setpoints for these variables are not LSSS. A discussion of the determination for each trip setpoint is provided below.

A. High Temperature in Main Steam Line Tunnel

Background

Main steam line tunnel temperature is monitored to detect a leak in the steam tunnel and provide isolation. The Primary Containment Isolation System (PCIS) automatically initiates closure of appropriate primary containment isolation valves, in combination with other accident mitigation systems, to limit fission product release during and following postulated Design Basis Accidents (DBAs). Primary containment isolation within the times specified for the automatic isolation valves ensures the release of radioactive material to the environment is consistent with DBA analysis assumptions.

This function was not listed as a LSSS in Section 2 of the MNGP TS. It is included in Table 3.2.1 as part of the PCIS instrumentation.

Discussion

The current and proposed trip setpoints for main steam line tunnel high temperature in Table 3.2.1, "Instrumentation That Initiates Primary Containment Isolation Functions," are:

	Current Trip Setpoint	Proposed Trip Setpoint
Main Steam and Recirculation Sample Line (Group 1)		
High Temperature in Main Steam Line Tunnel (Function 1.c)	≤ 200 °F	≤ 209 °F

Functional diversity was applied in the PCIS design. The isolation signals are assumed in the safety analyses to initiate closure of the Group 1 isolation valves to limit offsite doses. Main steam line tunnel (high) temperature is monitored at several locations above each main steam line (MSL) in the steam tunnel to detect small leaks (5 to 10 gpm) to mitigate the consequences of a high energy line break (HELB). The main steam line high temperature function serves as a backup to the MSL high flow monitoring instrumentation (greater than 140% of rated), which is credited for large HELBs, and as backup to the MSL pressure monitoring instrumentation for intermediate and large HELBs (when MSL pressure is < 825 psig).

The automatic closure of the Group 1 isolation valves maintains the release of radioactivity to the environs well below 10 CFR 100 guidelines. However, credit for these instruments is not taken in any transient or accident analysis in the USAR, since bounding analyses are performed for large breaks, such as main steam line breaks.

Conclusion

The limit of concern due to a HELB in the steam tunnel is a radioactive release exceeding 10 CFR 100 guidelines. However, the 10 CFR 100 release limits are not directly related to either the fuel cladding integrity MCPR safety limit or the RCPB integrity safety limit. Since the 10 CFR 100 limits are not one of the safety limits in accordance with 10 CFR 50.36(c)(1)(ii)(A) the trip setpoint for the main steam line tunnel high temperature is not a variable on which a safety limit has been placed and does not meet the criteria for being an LSSS.

B. Reactor Low Pressure Permissive Bypass Timer

Background

The purpose of the ECCS instrumentation is to initiate appropriate responses from the ECCS and support systems to ensure that the fuel is adequately cooled in the event of a DBA or transient. The bypass timer allows the core spray and LPCI pumps to start on reactor low-low water level after the timer times out. This ensures that the low pressure ECCS subsystems are started on low-low reactor water level for small break situations, where the reactor has not depressurized, after the time delay has expired.

This function was not listed as a LSSS in Section 2 of the MNGP TS. It is included in Table 3.2.2 as part of the ECCS instrumentation.

Discussion

The proposed trip settings for the core spray and LPCI pump start reactor low pressure permissive bypass timer in Table 3.2.2, "Instrumentation That Initiates Emergency Core Cooling Systems," are:

	Current Trip Setpoint	Proposed Trip Setpoint
Core Spray and LPCI Pump Start	20 ± 1 minutes	20 ± 2 minutes
Reactor Low Pressure Permissive Bypass Timer (Function A.1.b.ii)		

Initiation of the core spray or LPCI systems occurs on a signal indicating: 1) reactor low-low water level coincident with low reactor steam dome pressure or 2) reactor low-low water level sustained for 20 minutes or 3) high drywell pressure. Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. After the bypass timer times out the core spray and LPCI pumps can start. This ensures that the low pressure ECCS subsystems are started on low-low reactor water level for small break situations, where the reactor has not depressurized, after the time delay has expired. Opening of the motor operated isolation valves occurs after reactor pressure decays below a permissive signal.

The bypass timer provides time for an operator to assess whether level control will be regained by the high pressure makeup features before the timer times out permitting low pressure makeup. The bypass timer also ensures that given no operator action, the Automatic Depressurization System will initiate automatically, removing the dependence on operator action to blowdown the vessel when required.

The actions of the ECCS, including bypass timer operation, are explicitly assumed in the accident analyses. The bypass timer is not an instrument used for, or capable of, detecting a significant abnormal degradation of the fuel cladding or of the RCPB prior to a DBA. It also is not used for, nor capable of, monitoring a process variable that is an initial condition of a DBA or transient analysis. As such it neither detects, nor monitors a process variable that is an initial condition of a DBA or transient analysis.

Conclusion

The ECCS are initiated to preserve the integrity of the fuel cladding by limiting the post-LOCA peak cladding temperature to less than the 10 CFR 50.46 limits. However, the 10 CFR 50.46 limits are not directly related to either the fuel cladding integrity MCPR safety limit or the RCPB integrity safety limit. Since the 10 CFR 50.46 limits are not one of the safety limits in accordance with 10 CFR 50.36(c)(1)(ii)(A) the trip setpoint for the bypass timer is not a variable on which a safety limit has been placed and does not meet the criteria for being an LSSS.

C. Loss of Voltage Protection

Background

ECCS operation is dependent upon the availability of adequate power sources for energizing the various components. Offsite power is the preferred source of power for the 4.16 kV essential safeguards buses. Degraded and loss of voltage protection instrumentation monitor these buses. If this instrumentation determines sufficient voltage is unavailable, the essential safeguards buses are disconnected from the offsite power sources and connected to the onsite EDG power sources.

This function was not listed as a LSSS in Section 2 of the MNGP TS. It is included in Table 3.2.6 as part of the Loss of Power instrumentation.

Discussion

The current and proposed trip settings for Loss of Voltage Protection in Table 3.2.6, "Instrumentation for Safeguards Bus Degraded Voltage and Loss of Voltage Protection," are:

	Current Trip Setpoint	Proposed Trip Setpoint
Loss of Voltage Protection (Function 2)	2625 ± 175 volts	2625 ± 280 volts

Loss of voltage on a 4.16 kV essential safeguards bus indicates that offsite power may be completely lost to the bus and it will be unable to supply sufficient power for proper operation of the applicable equipment. Therefore, the power supply to the bus is transferred from offsite power to EDG power when the voltage drops below the loss of voltage trip setpoint.

Initiation of the EDGs occurs on a signal indicating: 1) reactor low-low water level 2) high drywell pressure, 3) degraded voltage, or 4) loss of voltage. The LOCA analysis credits the EDGs starting on low-low water level.

The actions of the ECCS are explicitly assumed in the safety analyses. The EDGs provide an alternative source of AC power in the event that the multiple redundant offsite power supplies are lost. The ECCS are initiated to preserve the integrity of the fuel cladding by limiting the post-LOCA peak cladding temperature to less than the 10 CFR 50.46 limits. However, the 10 CFR 50.46 limits are not directly related to either the fuel cladding integrity MCPR safety limit or the RCPB integrity safety limit. Since the 10 CFR 50.46 limits are not one of the safety limits in accordance with 10 CFR 50.36(c)(1)(ii)(A) the trip setpoint for the loss of voltage protection function is not a variable on which a safety limit has been placed and does not meet the criteria for being a LSSS.

In a similar argument , Southern California Edison (SCE) stated the following in a LAR dated May 27, 2005 (Reference 7), concerning degraded voltage protection at the San Onofre Nuclear Generating Station:

> "SCE is aware of the ongoing generic issues with setpoint calculation methodology, particularly as described in the NRC's letter to NEI dated March 31, 2005. ... Because the degraded voltage protection setpoints do not constitute a Limiting Safety System Setting (LSSS), the NRC concerns identified in Reference 12 [3] do not apply to this proposed change."

3 Reference 12 was identified in the SCE letter dated May 27, 2005, as ISA-RP67.04-02-2000, dated January 1, 2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." This appears to be an incorrect reference.

The NRC did not refute Southern California Edison's claim that degraded voltage protection setpoints do not constitute a LSSS in their Safety Evaluation (Reference 8) approving the license amendment.

Conclusion

The loss of voltage protection instruments monitor power conditions on the essential busses. The 10 CFR 50.46 limits are not directly related to either the fuel cladding integrity MCPR safety limit or the RCPB integrity safety limit. Since the 10 CFR 50.46 limits are not one of the safety limits in accordance with 10 CFR 50.36(c)(1)(ii)(A) the trip setpoint for the loss of voltage protection function was not placed on a variable on which a safety limit has been placed and does not meet the criteria for being an LSSS.

D. Low-Low Set Instrumentation

Background

The Low-Low Set logic and instrumentation is designed to mitigate the effects of postulated thrust loads on the safety/relief valve (S/RV) discharge lines by preventing subsequent actuations of an S/RV with an elevated water leg in the S/RV discharge line. It also mitigates the effects of postulated pressure loads on the torus shell or suppression pool by preventing multiple actuations in rapid succession of the S/RVs subsequent to their initial actuation. The Low-Low Set instrumentation and logic function ensures that the containment loads remain within the primary containment design basis.

The opening of the Low-Low Set S/RV actuates tailpipe pressure switches and starts inhibit timers in the trip logic. The inhibit timers prevent the plant operators or the Low-Low Set S/RV logic from immediately reopening the valve, providing the time necessary for the water leg in the S/RV discharge line to recede.

This function was not listed as a LSSS in Section 2 of the MNGP TS. It is included in Table 3.2.7 as part of the Low-Low Set logic instrumentation.

Discussion

The current and proposed trip settings in Table 3.2.7, "Instrumentation for Safety/ Relief Valve Low-Low Set logic," are:

	Current Trip Setpoint	Proposed Trip Setpoint
Reactor Coolant System 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 sec	10 ± 2 sec

a) The Low-Low Set S/RV trip setpoints specified in Table 3.2.7 (Allowable Values) are staggered and chosen to be less than the S/RV safety lift setpoints. This choice of setpoints assures that the Low-Low Set valves are the first valves to open and the last to close, to ensure a sufficient blowdown range to make certain that containment loads remain within the primary containment design basis.

   The ASME Boiler and Pressure Vessel Code require that the pressure relief system prevent excessive overpressurization of the primary system process barrier and the pressure vessel. Consequently, Specification 3.6.E, "Safety/Relief Valves," specifies that the safety valve function (self actuation or mechanical) setpoint of the S/RVs shall be less than or equal to 1120 psig. This safety function of the S/RVs directly supports the maintenance of the RCPB integrity safety limit, and is differentiated from the Low-Low Set instrumentation.

b) The trip setpoint specified for the S/RV discharge pipe pressure inhibit and position indication function is chosen to provide detection of an S/RV opening.

c) The trip setpoint specified for the inhibit timers was chosen to ensure that subsequent S/RV openings will not occur until after the elevated water leg in the respective S/RV discharge line is lowered to minimize water clearing thrust loads on the discharge line.

Conclusion

The RCPB Safety Limit is ultimately protected by the safety function of the S/RVs. This is reverified each cycle as part of the reload overpressure protection analysis. The function of the Low-Low Set instrumentation and logic is to ensure that containment loads remain within the primary containment design basis and is not to protect the RCPB Safety Limit. Because the trip setpoints specified for the Low-Low Set instrumentation functions are not placed on variables directly associated with a safety limit (in this case the RCPB Safety Limit), they are not LSSS.

Summary

The LSSS for MNGP are listed below.

- Neutron Flux Intermediate Range Monitor (IRM) – 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 (MSIV) Closure
- Turbine Control Valve Fast Closure
- Turbine Stop Valve Closure
- Main Steam Line Low Pressure Initiates MSIV Closure

The trip setpoint values for the following functions modified as part of the proposed 24-month fuel cycle license amendment request are not associated with any safety limit and are not LSSS.

- High Temperature in Main Steam Line Tunnel
- Core Spray and LPCI Pump Reactor Low Pressure Permissive Bypass Timer
- 4.16 kV Essential Bus Loss of Voltage Protection
- Low-Low Set Reactor Coolant System Pressure for Opening / Closing
- Low-Low Set Discharge Pipe Pressure Inhibit and Position Indication
- Low-Low Set Inhibit Timer

The applicable event analysis criteria for the above setpoints, with exception of the Low-Low Set setpoints, are discussed in 10 CFR 100 and 10 CFR 50.46, neither of which is directly related to either the fuel cladding integrity MCPR safety limit or the RCPB integrity safety limit. The Low-Low Set setpoints are related to satisfying containment load design limits. Since none of these trip setpoints are safety limits in accordance with 10 CFR 50.36(c)(1)(ii)(A), and hence were not placed on a variable on which a safety limit has been placed, they do not meet the criteria for being LSSS.

REFERENCES

1. Letter from NMC to NRC, "License Amendment Request to Support 24-Month Fuel Cycles," (L-MT-04-036) dated June 30, 2004.

2. Letter from NRC to NEI, " Instrumentation, Systems, and Automation Society S67.04 Methods for Determining Trip Setpoints and Allowable Values for Safety-Related Instrumentation," dated March 31, 2005.

3. Letter from NEI to the NRC, "Instrumentation, Systems, and Automation Society S67.04 Methods for Determining Trip Setpoints and Allowable Values for Safety-Related Instrumentation," dated May 18, 2005.

4. Letter from NRC to NMC, "Monticello Nuclear Generating Plant – Third Request for Additional Information Related to Technical Specifications Change Request to Implement a 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005.

5. Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," (L-MT-05-075) dated July 1, 2005.

6. NRC letter to NEI, "Instrumentation, Systems, And Automation Society (ISA) S67.04 Methods For Determining Trip Setpoints And Allowable Values For Safety-Related Instrumentation," dated August 23, 2005.

7. Southern California Edison letter to NRC, "San Onofre Nuclear Generating Station, Units 2 and 3, Docket Nos. 50-361 and 50-362, Proposed Change Number (PCN) 561, Degraded Voltage Setpoints," dated May 27, 2005 (ADAMS Accession No. ML051530034).

8. NRC letter to Southern California Edison, "San Onofre Nuclear Generating Station, Units 2 and 3 - Issuance of Amendments on Degraded Voltage Setpoints," (TAC Nos. MC7190 and MC7191), dated July 1, 2005 (ADAMS Accession No. ML051860407).