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Constellation Energy Group
Nine Mile Point Nuclear Station

December 19, 2002
NMP1L 1705

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

SUBJECT: Nine Mile Point Unit 1
Docket No. 50-220

License Amendment Request Pursuant to 10 CFR 50.90: Update and Clarification of Shutdown Margin Requirements
TAC No. MB6940

Gentlemen:

Pursuant to 10 CFR 50.90, Nine Mile Point Nuclear Station, LLC, (NMPNS) hereby requests an amendment to Nine Mile Point Unit 1 (NMP1) Operating License DPR-63. The proposed changes to the Technical Specifications (TSs) contained herein would revise Sections 1.0, "Definitions," and 3/4.1.1, "Control Rod System." Specifically, Section 1.0 is revised to add the definition of Shutdown Margin (SDM) as Definition 1.32. Specification 3.1.1a(1) is revised to incorporate new, more restrictive, SDM limits and add the associated Limiting Condition for Operation (LCO) actions and completion times for each applicable operating condition if the SDM is not met. Specification 4.1.1a(1) is revised to add the conditional surveillance requirements for verifying the SDM. The Bases for TS 3/4.1.1 and TS 3/4.7.1, "Special Test Exception - Shutdown Margin Demonstrations," have been revised to reflect the proposed changes to the TSs. The TS Bases changes are provided for information only and do not require NRC issuance.

The proposed changes update and clarify the TS requirements for demonstrating SDM. The proposed changes incorporate new, more restrictive, SDM limits; add the required LCO actions if the SDM is not met; and also add the surveillance requirements for verifying the SDM. These LCO actions and surveillance requirements are not currently specified in the TSs. The revised SDM limits account for the uncertainty in the demonstration of adequate SDM analytically (0.38% $\Delta k/k$) or by measurement (0.28% $\Delta k/k$). The proposed changes also eliminate the unnecessary restriction requiring SDM demonstration in the cold shutdown condition. The option for SDM demonstration in the cold shutdown condition is retained consistent with the existing special test exception.

The proposed changes are, in general, consistent with the Improved Standard TSs (ISTS) for Boiling Water Reactors (NUREG-1433 and NUREG-1434, Revision 2). Certain

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deviations from the ISTS were necessary due to the non-standard content and format of the current custom TSs for NMP1. The NRC previously approved similar TS changes for the Oyster Creek Nuclear Generating Station (also a custom TS plant) in License Amendment No. 178, dated March 21, 1995 (TAC No. M89741).

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that the changes involve no significant hazards considerations.

NMPNS requests approval of this application and issuance of the TS amendment by March 30, 2003 with 60 days allowed for implementation. The amendment is needed to support plant startup following the Spring 2003 refueling outage (RFO17). This letter contains no new commitments.

Pursuant to 10CFR50.91(b)(1), NMPNS has provided a copy of this license amendment request and the associated analyses regarding no significant hazards considerations to the appropriate state representative.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 19, 2002.

Sincerely,

John T. Conway

John T. Conway
Vice President Nine Mile Point

JTC/CDM/jm

Attachments:
1. Evaluation of Proposed Technical Specification Changes
2. Proposed Technical Specification Changes (Mark-up)
3. Technical Specification Bases Changes (Mark-up For Information Only)

cc: Mr. H. J. Miller, NRC Regional Administrator, Region I
Mr. G. K. Hunegs, NRC Senior Resident Inspector
Mr. P. S. Tam, Senior Project Manager, NRR (2 copies)
Mr. John P. Spath, NYSERDA

ATTACHMENT 1

EVALUATION OF PROPOSED TECHNICAL SPECIFICATION CHANGES

Subject: License Amendment Request Pursuant to 10 CFR 50.90: Update and Clarification of Shutdown Margin Requirements

1.0 DESCRIPTION

This letter is a request to amend Operating License DPR-63 for Nine Mile Point Unit 1 (NMP1).

The proposed changes would amend the Operating License to revise Technical Specification (TS) Sections 1.0, "Definitions," and 3/4.1.1, "Control Rod System." Specifically, Section 1.0 is revised to add the definition of Shutdown Margin (SDM) as Definition 1.32. Specification 3.1.1a(1) is revised to incorporate new, more restrictive, SDM limits and add the associated Limiting Condition for Operation (LCO) actions and completion times for each applicable operating condition if the SDM is not met. Specification 4.1.1a(1) is revised to add the conditional surveillance requirements for verifying the SDM. The Bases for TS 3/4.1.1 and TS 3/4.7.1, "Special Test Exception - Shutdown Margin Demonstrations," have been revised to reflect the proposed changes to the TSs.

The proposed changes to the TSs and the associated changes to the TS Bases are indicated in the mark-up pages provided in Attachments 2 and 3, respectively. The TS Bases changes are provided for information only and do not require NRC issuance as they will be controlled by the Nine Mile Point Nuclear Station, LLC, (NMPNS) TS Bases change control process.

The next refueling outage for NMP1 (RF017) is scheduled for Spring 2003. NMPNS requests that this proposed amendment be approved and issued by March 30, 2003 to support an anticipated plant startup the first week in April, following the scheduled refueling outage. Approval of the proposed changes would eliminate the necessity for demonstrating SDM in the cold shutdown condition and represents a significant reduction in outage critical path time.

2.0 PROPOSED CHANGE

The proposed change to TS Section 1.0 adds the definition of SDM to the TSs as Definition 1.32.

The proposed changes to TS Section 3.1.1 are incorporated into Specification 3.1.1a(1), "Reactivity Limitations, Reactivity margin - core loading," as described below:

Proposed Specification 3.1.1a(1)(a) replaces the current specification and incorporates new, more restrictive, SDM limits. The revised SDM limits account for the uncertainty in the demonstration of adequate SDM analytically (0.38% $\Delta k/k$) or by measurement (0.28% $\Delta k/k$).

Proposed Specification 3.1.1a(1)(b) is added to provide the required action and associated completion time (6 hours) for determining whether the SDM limits in Specification 3.1.1a(1)(a) are met while in the power operating condition when one or more control rods are inoperable as currently defined in Specification 3.1.1a(2). If this determination cannot be made within the allowed time, Specification 3.1.1a(1)(a) is assumed to be not met.

Proposed Specification 3.1.1a(1)(c) is added to provide the required action and associated completion time (6 hours) for restoring SDM if the limits in Specification 3.1.1a(1)(a) are not met while in the power operating condition. If SDM is not restored within the allowed time, the plant must be placed in a shutdown condition within the following 10 hours.

Proposed Specification 3.1.1a(1)(d) is added to provide the required actions and associated completion times if the SDM limits in Specification 3.1.1a(1)(a) are not met while in the hot shutdown or cold shutdown condition. Immediate action is required to fully insert all insertable control rods. In addition, action must be initiated within 1 hour to (1) restore secondary containment to operable status, (2) restore one emergency ventilation system to operable status, and (3) restore isolation capability in each required secondary containment penetration flow path not isolated.

Proposed Specification 3.1.1a(1)(e) is added to provide the required actions and associated completion times if the SDM limits in Specification 3.1.1a(1)(a) are not met while in the refueling condition. Immediate action is required to suspend core alterations, except for fuel assembly removal, and to fully insert all insertable control rods in core cells containing one or more fuel assemblies.

The proposed changes to Specifications 3.1.1a(2), 3.1.1b(2), and 3.1.1f incorporate necessary cross-referencing corrections to properly reflect the changes to the referenced specifications as described above.

The proposed changes to Section 4.1.1 are incorporated into Specification 4.1.1a(1), "Reactivity Limitations, Reactivity margin - core loading," and serve to establish the conditions for which the SDM is required to be verified within limits. The two conditions are (1) prior to in vessel fuel movement during the fuel loading sequence and (2) once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement.

The Bases for TS 3.1.1 and 4.1.1 are revised to reflect the proposed changes to the TSs and provide the necessary background and basis information for the new SDM limits and associated requirements. The content of the Bases are also strengthened by providing additional clarification to assure that the LCO actions for each applicable operating condition are properly applied if the SDM limits are not met.

The Bases for TS 3.7.1 and 4.7.1 are revised to eliminate the restriction requiring SDM demonstration prior to power operation. Currently, the Bases require an SDM demonstration to be performed in the cold shutdown condition with the vessel head in place, prior to the reactor coolant system pressure and control rod scram time tests, following refueling outages when core alterations are performed. The Bases are modified to allow the optional (rather than mandatory) performance of this SDM demonstration as a special test exception under the previously specified conditions.

In summary, the proposed changes update and clarify the TS requirements for demonstrating SDM. The proposed changes incorporate new, more restrictive, SDM limits; add the required LCO actions if the SDM is not met; and also add the surveillance requirements for verifying the SDM. These surveillance requirements and LCO actions are not currently specified in the TSs. The revised SDM limits account for the uncertainty in the demonstration of adequate SDM analytically (0.38% $\Delta k/k$) or by measurement (0.28% $\Delta k/k$). The proposed changes also eliminate the unnecessary restriction requiring SDM demonstration in the cold shutdown condition. The option for SDM demonstration in the cold shutdown condition is retained consistent with the existing special test exception.

The proposed changes are, in general, consistent with the Improved Standard TSs (ISTS) for Boiling Water Reactors (NUREG-1433 and NUREG-1434, Revision 2). Certain deviations from the ISTS were necessary due to the non-standard content and format of the current custom TSs for NMP1. The NRC previously approved similar TS changes for the Oyster Creek Nuclear Generating Station (also a custom TS plant) in License Amendment No. 178, dated March 21, 1995 (TAC No. M89741).

3.0 BACKGROUND

SDM requirements are specified to ensure: (1) the reactor can be made subcritical from all applicable operating conditions, transients, and design basis events; (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. These requirements are satisfied by the control rods, as described in 10 CFR 50, Appendix A, General Design Criterion (GDC) 26, which can compensate for the reactivity effects of the fuel and water temperature changes experienced during the applicable operating conditions. The control rods, in conjunction with the use of burnable poison in the fuel and reactor coolant recirculation flow control, have the capability of controlling reactivity changes resulting from load changes, xenon burnout, and fuel burnup. The control rods and drive system are described in detail in Section IV-B.6 of the NMP1 Updated Final Safety Analysis Report (UFSAR). Reactor fuel and recirculation flow control are described in UFSAR Section IV-B.5.1 and Sections IV-B.3.1.1 and VIII-B.2.2, respectively.

Prevention or mitigation of reactivity insertion events, such as an inadvertent continuous control rod withdrawal transient or control rod drop accident (CRDA), is necessary to limit energy deposition in the fuel to prevent significant fuel damage, which could result in undue release of radioactivity. Adequate SDM provides assurance that inadvertent criticalities and potential CRDAs involving high worth control rods (i.e., the first control rod withdrawn) will not cause significant fuel damage. The control rod withdrawal error transient and CRDA are discussed in UFSAR Sections XV-B.3.4 and XV-C.4, respectively.

In the power operating condition, SDM must be provided because subcriticality with the highest worth control rod withdrawn is assumed in the CRDA analysis. In the hot shutdown and cold shutdown conditions, SDM is required to provide assurance that the reactor will be held subcritical with margin for a single withdrawn control rod. SDM is required in the refueling condition to prevent an inadvertent criticality during the withdrawal of a single control rod from a core cell containing one or more fuel assemblies.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) for inclusion in the TSs, in that it is an operating restriction that is an initial condition of a design basis accident or transient analysis. The proposed changes will provide the necessary TS LCOs, LCO actions, and surveillance requirements to assure that the SDM is maintained in accordance with 10 CFR 50.36(c)(2)(i) and 10 CFR 50.36(c)(3) for all applicable reactor operating conditions.

4.0 TECHNICAL ANALYSIS

The LCO of Specification 3.1.1a(1) is revised to incorporate new SDM limits and to specify, for each applicable operating condition, the associated required actions and completion times if the SDM is not met. In addition, the definition of SDM (Definition 1.32) is added to the TSs for clarification consistent with the ISTS and the surveillance requirements of Specification 4.1.1a(1) are revised to specify the conditions under which SDM must be verified. The Bases for TS 3/4.1.1 and TS 3/4.7.1 have been revised to reflect the proposed changes to the TSs. The proposed changes are discussed in more detail below.

The proposed SDM limits, as specified in Specification 3.1.1a(1)(a), account for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically (0.38% $\Delta k/k$) or by measurement (0.28% $\Delta k/k$). This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. In both cases, the proposed SDM limit is consistent with the ISTS and more restrictive than the limit it is replacing (0.25% $\Delta k/k$). Core reactivity will vary during the fuel cycle as a function of fuel depletion and poison burnup. As such, it is currently required that the SDM limit be increased by an adder "R" to account for changes in core reactivity during the cycle. The value of "R" must either be a positive quantity or zero (i.e., no correction is required if the beginning of the cycle is the most reactive point in the cycle). The proposed changes do not alter the current requirements regarding "R." Therefore, the proposed SDM limits are more restrictive and the margin of safety is increased relative to the SDM assumptions for the control rod withdrawal error transient and CRDA analyses.

The Bases for TS 3/4.1.1 are revised to reflect the proposed changes to the TSs and provide the necessary background and basis information for the new SDM limits.

Currently, the TSs do not specify the required actions in the event that the SDM limit is not met. The proposed SDM limits will apply to all reactor operating conditions, except the major maintenance condition when no fuel is in the reactor. Therefore, LCO actions and completion times are now proposed for each of the applicable operating conditions when the SDM is not within the specified limits. The proposed actions and associated completion times are evaluated below by the applicable operating condition(s):

Power Operating Condition

The inability to meet the SDM limits in the power operating condition would most likely be due to withdrawn control rods that cannot be inserted (i.e., stuck control rods). Proposed Specifications 3.1.1a(1)(b) and 3.1.1a(1)(c) address this condition by requiring that within 6 hours a determination be made establishing whether the SDM limits are met, and if not met, requiring restoration within 6 hours or plant shutdown within the following 10 hours.

A reduced SDM is not considered an immediate threat to nuclear safety; therefore, time is allowed for analysis to ensure the SDM limits are met, and for repair and restoration before requiring the plant to undergo a transient to achieve a shutdown condition. The proposed completion time of 6 hours for SDM analysis is more restrictive than the ISTS (which allows up to 72 hours) since the control rod operability requirements in the ISTS include additional actions and verifications that do not currently exist in the NMP1 TSs. The proposed completion time of 6 hours for restoration of the SDM is consistent with the ISTS. The total allowed completion time of 12 hours for SDM analysis, repair, and restoration is acceptable considering that the reactor can still be shutdown, assuming no additional stuck control rods, and the low probability of an event occurring during this interval. Failure to reach the cold shutdown condition is only likely if an additional control rod, adjacent to the stuck control rod, also fails to insert during a required scram. Even with the postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain hot shutdown conditions. The proposed actions and completion times for the power operating condition are consistent with those previously approved by the NRC for the Oyster Creek Nuclear Generating Station in License Amendment No. 178, dated March 21, 1995 (TAC No. M89741).

If the SDM cannot be restored within the allowed 12 hours (6 hours for SDM analysis plus 6 hours for restoration), a plant shutdown (to at least the hot shutdown condition) is required to minimize the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed completion time of 10 hours is consistent with other TS shutdown LCOs for NMP1 (including TS 3.1.1f) and considered reasonable for achieving a shutdown condition from full power in an orderly manner and without challenging plant systems.

The Bases for TS 3/4.1.1 are revised to reflect the proposed changes to the TSs and provide the necessary background and basis information to assure that the LCO actions for the power operating condition are properly applied if the SDM limits are not met.

Hot Shutdown Condition and Cold Shutdown Condition

The inability to meet the SDM limits in the hot shutdown or cold shutdown condition could be due to withdrawn control rods that cannot be inserted, discovery of errors in the SDM analysis, or discovery of errors in previous core alterations. Proposed Specification 3.1.1a(1)(d) addresses these conditions by requiring the immediate insertion of all insertable control rods, which results in the least reactive condition for the reactor core and maximizes the SDM. The proposed specification also includes 1 hour actions to initiate action to (1) restore secondary containment to operable status, (2) restore at least one emergency ventilation system to operable status, and (3) restore isolation capability in each required secondary containment penetration flow path not isolated. These additional actions are intended to provide the means for control of potential radioactive releases by maintaining secondary containment integrity. Note that the 1 hour actions only apply to the hot shutdown condition when the reactor water temperature is between 212° F and 215° F since secondary containment integrity (see TS Definition 1.12) is currently required to be fully operable above 215° F in accordance with TS 3.4.0 and the supporting specifications (TSs 3.4.1 - 3.4.5). The total allowed completion time of 1 hour for these additional actions is acceptable considering that the reactor can still be shutdown and maintained shutdown, assuming no additional stuck control rods. The proposed actions and completion times for the hot shutdown and cold shutdown conditions are consistent with the ISTS.

The Bases for TS 3/4.1.1 are revised to reflect the proposed changes to the TSs and provide the necessary background and basis information to assure that the LCO actions for the hot shutdown and cold shutdown conditions are properly applied if the SDM limits are not met.

Refueling Condition

The inability to meet the SDM limits in the refueling condition would most likely be due to fuel loading errors. Proposed Specification 3.1.1a(1)(e) addresses this condition by requiring the immediate suspension of core alterations, except for fuel assembly removal, and to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Fuel assembly removal and control rod insertion reduce total reactivity and are allowed in order to recover SDM. Note that control rod insertion is currently allowed by Definition 1.13, "Core Alteration," since control rod movement with the control rod drive hydraulic system is not considered to be a core alteration. Also note that a means for control of potential radioactive releases is provided since secondary containment integrity (see TS Definition 1.12) is currently required to be fully operable in the refueling condition in accordance with

TS 3.4.0 and the supporting specifications (TSs 3.4.1 - 3.4.5). The proposed actions and completion times for the refueling condition, in conjunction with the existing TS provisions, are consistent with the ISTS.

The Bases for TS 3/4.1.1 are revised to reflect the proposed changes to the TSs and provide the necessary background and basis information to assure that the LCO actions for the refueling condition are properly applied if the SDM limits are not met.

The proposed changes to Specifications 3.1.1a(2), 3.1.1b(2), and 3.1.1f incorporate necessary cross-referencing corrections. Accordingly, these proposed changes are considered to be editorial and, as such, are administrative.

Adequate SDM must be demonstrated to ensure the reactor can be made subcritical from any initial operating condition, except the major maintenance condition when there is no fuel in the reactor. This can be accomplished by a test, an evaluation, or a combination of the two. Adequate SDM can be demonstrated by testing before or during the first startup after fuel movement, or shuffling within the reactor pressure vessel, or control rod replacement. The SDM may be demonstrated during an in-sequence control rod withdrawal for the purpose of bringing the reactor critical, in which the highest worth control rod is analytically determined, or during local criticals, where the highest worth control rod is determined by testing. Local critical tests require the withdrawal of out of sequence control rods.

During the refueling condition, adequate SDM is also required to ensure the reactor does not reach criticality during control rod withdrawals. An evaluation of each in vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. Spiral offload and reload sequences inherently satisfy the SDM requirements and removing fuel from the core will always result in an increase in SDM.

The proposed changes to Specification 4.1.1a(1) require the SDM to be verified within limits (1) prior to in vessel fuel movement during the fuel loading sequence and (2) once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement. These surveillance requirements establish the conditions under which the SDM must be demonstrated. The proposed frequency of 4 hours after reaching criticality is based on allowing a reasonable amount of time to perform the required calculations and to have appropriate verification.

Currently, the NMP1 TSs do not contain provisions which allow the SDM to be demonstrated during plant startup following a refueling outage. This resulted from a now outdated commitment (Reference: Letter NMP1L 0241, dated April 5, 1988) related to License Amendment No. 99, dated June 9, 1988 (TAC No. 67863). TS 3/4.7.1, "Special Test Exception - Shutdown Margin Demonstrations," and the associated Bases which currently require an SDM demonstration to be performed in the cold shutdown condition following refueling outages when core alterations are performed. The proposed changes to

Specification 4.1.1a(1), in conjunction with the changes to the Bases for TS 3/4.7.1, are intended to update and clarify the requirements for SDM demonstration/verification consistent with the ISTS.

The proposed changes to Specification 4.1.1a(1) will require the SDM to be verified both during the fuel loading sequence and during the plant startup process following a refueling outage. Thus, an SDM demonstration (i.e., verification) in the cold shutdown condition (between the fuel loading and startup verifications) is redundant and unnecessary. TS 3/4.7.1 is retained to continue to provide the special testing requirements for performing an SDM demonstration in the cold shutdown condition. The Bases for TS 3/4.7.1 have been modified to allow the optional (rather than mandatory) performance of this SDM demonstration as a special test exception under the previously specified conditions.

Based on the above analysis, the proposed changes incorporate new SDM limits that are more restrictive than the current limit and provide adequate margin for uncertainties. The proposed changes also provide for the appropriate verifications to assure that the SDM is maintained within the specified limits, and for the proper and timely responses in the event the SDM limits are not met. Therefore, the proposed changes will not decrease the margin of safety, and are considered acceptable.

5.0 REGULATORY SAFETY ANALYSIS

5.1 <u>No Significant Hazards Consideration Analysis</u>

The proposed changes would update and clarify the Technical Specification (TS) requirements for demonstrating Shutdown Margin (SDM). The proposed changes incorporate new, more restrictive, SDM limits; add the required Limiting Condition for Operation (LCO) actions if the SDM is not met; and also add the surveillance requirements for verifying the SDM. These surveillance requirements and LCO actions are not currently specified in the TSs. The revised SDM limits account for the uncertainty in the demonstration of adequate SDM analytically (0.38% $\Delta k/k$) or by measurement (0.28% $\Delta k/k$). The proposed changes also eliminate the unnecessary restriction requiring SDM demonstration in the cold shutdown condition. The option for SDM demonstration in the cold shutdown condition is retained consistent with the existing special test exception.

Nine Mile Point Nuclear Station, LLC, (NMPNS) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

Adequate SDM provides assurance that inadvertent criticalities and potential control rod drop accidents (CRDAs) involving high worth control rods will not cause significant fuel damage. The SDM is not an accident initiator and, as such, will have no effect on the probability of an accident. The proposed changes incorporate more restrictive SDM limits and provide the necessary actions and verifications to assure that there will be no adverse effect on the initial conditions and assumptions of the accidents previously evaluated in the Updated Final Safety Analysis Report (UFSAR). The proposed changes do not involve physical changes to the plant or introduce any new modes of operation. Accordingly, continued assurance is provided that the process variables, structures, systems, and components are maintained such that there will be no degradation of any fission product barrier which could increase the radiological consequences of an accident. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed changes to the SDM limits and requirements will have no adverse effect on the design or assumed accident performance of any structure, system, or component, or introduce any new modes of system operation or failure modes. Moreover, the proposed changes will have no impact on conformance to 10 CFR 50, Appendix A, General Design Criterion 26 (GDC 26), in that the control rods will continue to satisfy the SDM requirements and provide assurance that the reactor can be made subcritical from all applicable operating conditions, transients, and design basis events. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   The proposed changes provide separate SDM limits for testing consistent with the Improved Standard Technical Specifications (NUREG-1433 and NUREG-1434) where the highest worth control rod is determined analytically (0.38% $\Delta k/k$) or by measurement (0.28% $\Delta k/k$). The proposed SDM limits are more restrictive than the current limit (0.25% $\Delta k/k$) and account for the uncertainty in the demonstration of SDM by testing. The SDM will continue to account for changes in core reactivity during the fuel cycle. Therefore, the margin of safety is increased relative to the SDM assumptions for the control rod withdrawal error transient and CRDA analyses. Accordingly, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, NMPNS concludes that the proposed amendment presents no significant hazards considerations under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) for inclusion in the TSs, in that it is an operating restriction that is an initial condition of a design basis accident or transient analysis. The proposed changes will provide the necessary TS LCOs, LCO actions, and surveillance requirements to assure that the SDM is maintained in accordance with 10 CFR 50.36(c)(2)(i) and 10 CFR 50.36(c)(3) for all applicable reactor operating conditions. The proposed changes will have no impact on conformance to GDC 26 since the control rods will continue to satisfy the SDM requirements to ensure: (1) the reactor can be made subcritical from all applicable operating conditions, transients, and design basis events; (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

In conclusion, based on the considerations discussed above, (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.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

The current versions of Technical Specification pages 8, 29 through 31, and 36 have been marked-up by hand to reflect the proposed changes.

1.28 (Deleted)

1.29 (Deleted)

1.30 Reactor Coolant Leakage

a. Identified Leakage

(1) Leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered and conducted to a sump or collecting tank, or

(2) Leakage into the primary containment atmosphere from sources that are both specifically located and known not to be from a through-wall crack in the piping within the reactor coolant pressure boundary.

b. Unidentified Leakage

All other leakage of reactor coolant into the primary containment area.

1.31 Core Operating Limits Report

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1f. Plant operation within these operating limits is addressed in individual specifications.

INSERT 1.32

INSERT 1.32

1.32 Shutdown Margin (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:

a. The reactor is xenon free,

b. The moderator temperature is 68° F, and

c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

LIMITING CONDITION FOR OPERATION

3.1.1 CONTROL ROD SYSTEM

Applicability:

Applies to the operational status of the control rod system.

Objective:

To assure the capability of the control rod system to control core reactivity.

Specification

a. Reactivity Limitations

(1) Reactivity margin - core loading

INSERT 3.1.1a(1)

~~The core loading shall be limited to that which can be made subcritical in the most reactive condition during the operating cycle with the strongest control rod in its full-out position and all other operable rods fully inserted.~~

SURVEILLANCE REQUIREMENT

4.1.1 CONTROL ROD SYSTEM

Applicability:

Applies to the periodic testing requirements for the control rod system.

Objective:

To specify the tests or inspections required to assure the capability of the control rod system to control core reactivity.

Specification:

The control rod system surveillance shall be performed as indicated below.

a. Reactivity Limitations

(1) Reactivity margin - core loading

INSERT 4.1.1a(1)

~~Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.25 percent Δk that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.~~

<u>INSERT 3.1.1a(1)</u>

(a) The Shutdown Margin (SDM) under all operational conditions shall be equal to or greater than:

0.38% $\Delta k/k$, with the highest worth control rod analytically determined, <u>or</u>

0.28% $\Delta k/k$, with the highest worth control rod determined by test.

(b) If one or more control rods are determined to be inoperable as defined in Specification 3.1.1a(2) while in the power operating condition, then a determination of whether Specification 3.1.1a(1)(a) is met must be made within 6 hours. If a determination cannot be made within the specified time period, then assume Specification 3.1.1a(1)(a) is not met.

(c) If Specification 3.1.1a(1)(a) is not met while in the power operating condition, restore compliance with Specification 3.1.1a(1)(a) within 6 hours or be in a shutdown condition within the following 10 hours.

(d) If Specification 3.1.1a(1)(a) is not met while in the hot shutdown condition or the cold shutdown condition, then:

Immediately initiate action to fully insert all insertable control rods, <u>and</u>

Initiate action within 1 hour to restore secondary containment to operable status, <u>and</u>

Initiate action within 1 hour to restore one emergency ventilation system to operable status, <u>and</u>

Initiate action within 1 hour to restore isolation capability in each required secondary containment penetration flow path not isolated.

(e) If Specification 3.1.1a(1)(a) is not met while in the refueling condition, then:

Immediately suspend core alterations, except for fuel assembly removal, <u>and</u>

Immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.

INSERT 4.1.1a(1)

The SDM shall be verified within limits:

(a) Prior to each in vessel fuel movement during the fuel loading sequence, and

(b) Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement.

LIMITING CONDITION FOR OPERATION

(2) Reactivity margin - stuck control rods

Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. Inoperable control rods shall be valved out of service, in such positions that Specification 3.1.1a(1)(a) is met. In no case shall the number of non-fully inserted rods valved out of service be greater than six during power operation. If this specification is not met, the reactor shall be placed in the cold shutdown condition. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing.

b. Control Rod Withdrawal

(1) The control rod shall be coupled to its drive or completely inserted and valved out of service. When removing a control rod drive for inspection, this requirement does not apply as long as the reactor is in a shutdown or refueling condition.

SURVEILLANCE REQUIREMENT

(2) Reactivity margin - stuck control rods

Each partially or fully withdrawn control rod shall be exercised at least once each week. This test shall be performed at least once per 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

b. Control Rod Withdrawal

(1) The coupling integrity shall be verified for each withdrawn control rod by either:

(a) Observing the drive does not go to the overtravel position, or

(b) A discernible response of the nuclear instrumentation.

LIMITING CONDITION FOR OPERATION

(2) The control rod drive housing support system shall be in place during power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.1.1a(1)(a) is met.

(3)(a) Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would not make the core more than 0.013 Δk supercritical.

SURVEILLANCE REQUIREMENT

(2) The control rod drive housing support system shall be inspected after reassembly.

(3)(a) To consider the rod worth minimizer operable, the following steps must be performed:

(i) The control rod withdrawal sequence for the rod worth minimizer computer shall be verified as correct.

(ii) The rod worth minimizer computer on-line diagnostic test shall be successfully completed.

(iii) Proper annunciation of the select error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

(iv) The rod block function of the rod worth minimizer shall be verified by attempting to withdraw an out-of-sequence control rod beyond the block point.

LIMITING CONDITION FOR OPERATION

f. If specification 3.1.1.~~a~~b through e, above, are not met, the reactor shall be placed in the hot shutdown condition within ten hours ~~except as noted in 3.1.1.a(2)~~.

g. Reactivity Anomalies

The difference between an observed and predicted control rod inventory shall not exceed the equivalent of one percent in reactivity. If this limit is exceeded, the reactor shall be brought to the cold shutdown condition by normal orderly shutdown procedure. Operation shall not be permitted until the cause has been evaluated and the appropriate corrective action has been completed.

SURVEILLANCE REQUIREMENT

g. Reactivity Anomalies

The observed control rod inventory shall be compared with a normalized computed prediction of the control rod inventory during startup, following refueling or major core alteration.

These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the actual control rod configuration will be compared with the expected configuration based upon appropriately corrected past data. This comparison will be made every equivalent full power month.

ATTACHMENT 3

TECHNICAL SPECIFICATION BASES CHANGES

(MARK-UP FOR INFORMATION ONLY)

The current version of Technical Specification Bases pages 37, 43,and 341 have been marked-up by hand to reflect the proposed changes. These Bases pages are provided for information only and do not require NRC issuance.

a. Reactivity Limitations

(1) Reactivity margin - core loading

INSERT B3/4.1.1a(1)

~~The core reactivity limitation is a restriction to be applied to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading or reloading and must be such that it will apply to the entire subsequent fuel cycle. It is sufficient that the core in its maximum reactivity condition be subcritical with the control rod of highest worth fully withdrawn and all other rods fully inserted. In order to implement this requirement, it will be required that the amount of shutdown margin will be at least R + 0.25 percent Δk in the cold, xenon-free condition. In this generalized expression, the value of R is the difference between the calculated value of core reactivity anytime later in the cycle where it may be greater than at the beginning. R must be a positive quantity or zero. A core which contains temporary control curtains or other burnable neutron absorbers may have a reactivity characteristic which increases the core lifetime, goes through a maximum, and then decreases thereafter.~~

~~A 0.25 percent Δk in the expression R + 0.25 percent Δk is provided as a finite, demonstrable, subcriticality margin. For the first fuel cycle, core reactivity is calculated never to be greater than the beginning-of-life value; hence, R = 0. The new value of R must be determined for each fuel cycle.~~

(2) Reactivity margin - stuck control rods

The specified limits provide sufficient scram capability to accommodate failure to scram of any one operable rod. This failure is in addition to any inoperable rods that exist in the core, provided that those inoperable rods met the core reactivity Specification 3.1.1 a(1)~~.~~ (a)

Control rods which cannot be moved with control rod drive pressure are indicative of an abnormal operating condition on the affected rods and are, therefore, considered to be inoperable. Inoperable rods are valved out of service to fix their position in the core and assure predictable behavior. If the rod is fully inserted and then valved out of service, it is in a safe position of maximum contribution to shutdown reactivity. If it is valved out of service in a non-fully inserted position, that position is required to be consistent with the shutdown reactivity limitation stated in Specification 3.1.1a(1)~~,~~ (a) which assures the core can be shut down at all times with control rods.

INSERT B3/4.1.1a(1)

The control rod drop accident analysis assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high reactivity worth and, should the core be critical during the withdrawal of the first control rod, the consequences of a control rod drop accident could exceed the fuel damage limits for the accident.

Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel to prevent significant fuel damage, which could result in undue release of radioactivity. Adequate SDM ensures inadvertent criticalities and potential control rod drop accidents involving high worth control rods (namely the first control rod withdrawn) will not cause significant fuel damage.

The SDM limits specified in Specification 3.1.1a(1)(a) account for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. When SDM is demonstrated by calculations not associated with a test (e.g., to confirm SDM during the fuel loading sequence), additional margin must be added to the specified SDM limit to account for uncertainties in the calculation. To ensure adequate SDM, a design margin is included to account for uncertainties in the design calculations (Reference (7)).

The inability to meet the SDM limits during power operating conditions would most likely be due to withdrawn control rods that cannot be inserted. A reduced SDM is not considered an immediate threat to nuclear safety; therefore, time is allowed for analysis to ensure Specification 3.1.1a(1)(a) is met, and for repair before requiring the plant to undergo a transient to achieve a shutdown condition. The allowed completion times of 6 hours for analysis and an additional 6 hours for repair, if Specification 3.1.1a(1)(a) is not met, are considered reasonable while limiting the potential for further reductions in SDM or the occurrence of a transient.

If the SDM cannot be restored within the allowed time, a plant shutdown is required to minimize the potential for, and consequences of, an accident or malfunction of equipment important to safety. The allowed completion time of 10 hours is considered reasonable to achieve the shutdown condition from full power in an orderly manner and without challenging plant systems.

The inability to meet the SDM limits in the hot shutdown condition or the cold shutdown condition could be due to withdrawn control rods that cannot be inserted, discovery of errors in the SDM analysis, or discovery of errors in previous core alterations. The immediate action to fully insert all insertable control rods will result in the least reactive condition for the core and maximizes SDM. This action must continue until all insertable control rods are fully inserted. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is operable, at least one emergency ventilation system is operable, and secondary containment isolation capability is available in each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are operable, or other acceptable administrative controls to assure isolation capability. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated). This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the surveillances needed to demonstrate the operability of the components. If, however, any required component is inoperable, then it must be restored to operable status. In this case, surveillances may need to be performed to restore the component to operable status. Actions must continue until all required components are operable.

The inability to meet the SDM limits in the refueling condition would most likely be due to fuel loading errors. The immediate action to suspend core alterations (e.g., fuel loading) prevents further reductions in SDM. Suspension of core alterations shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and is, therefore, allowed in order to recover SDM. Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. This action must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and, therefore, do not have to be inserted.

Adequate SDM must be verified to ensure that the reactor can be made subcritical from any initial reactor operating condition, except the major maintenance condition. This can be accomplished by a test, an evaluation, or a combination of the two. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, or shuffling within the reactor pressure vessel, or control rod replacement. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC is required. For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% $\Delta k/k$)

INSERT B3/4.1.1a(1) (continued)

must be added to the SDM limit of 0.28% Δk/k to account for uncertainties in the calculation.

The SDM may be demonstrated during an in-sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals, where the highest worth control rod is determined by testing. Local critical tests require the withdrawal of out of sequence control rods.

The frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During the refueling condition, adequate SDM is also required to ensure the reactor does not reach criticality during control rod withdrawals. An evaluation of each in vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload or reload sequences inherently satisfy the surveillance, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

f. Reactivity Anomalies

During each fuel cycle excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary controls is burned. The magnitude of this excess reactivity is indicated by the integrated worth of control rods inserted into the core, referred to as the control rod inventory in the core. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of actual rod inventory at any base equilibrium core state to predicted rod inventory at that state. Equilibrium xenon, samarium and power distribution are considered in establishing the steady-state base condition to minimize any source of error. During an initial period, (on the order of 1000 MWD/T core average exposure following core reloading or modification) rod inventory predictions can be normalized to actual rod patterns to eliminate calculational uncertainties. Experience with other operating BWR's indicates that the control rod inventory should be predictable to the equivalent of one percent in reactivity. Deviations beyond this magnitude would not be expected and would require thorough evaluation. One percent reactivity limit is considered safe since an insertion of this reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

(1) Paone, C. J., Stirn, R.C., and Wooley, J.A., "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, March 1972.

(2) Stirn, R. C., Paone, C. J., and Young, R. M., "Rod Drop Accident Analysis for Large BWRs," Supplement 1 - NEDO-10527, July 1972.

(3) Stirn, R. C., Paone, C. J., and Haun, J. M., "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores," Supplement 2 - NEDO-10527, January 1973.

(4) Report entitled "Technical Basis for Changes to Allowable Rod Worth Specified in Technical Specification 3.3.B.3," transmitted by letter from L. O. Mayer (NSP) to J. F. O'Leary (USAEC) dated October 4, 1973.

(5) Letter, R. R. Schneider, Niagara Mohawk Power Corporation to A. Giambusso, USAEC, dated November 15, 1973.

(6) To include the power spike effect caused by gaps between fuel pellets.

(7) Licensing Topical Report, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, latest approved revision.

BASES FOR 3.7.1 AND 4.7.1 SHUTDOWN MARGIN DEMONSTRATION

The shutdown margin demonstration ~~has to~~ may be performed prior to power operation. However, the mode switch must be placed in the startup position to allow withdrawal of more than one control rod. Specifications 3.7.1 and 4.7.1 require certain restrictions in order to ensure that an inadvertent criticality does not occur while performing the shutdown margin demonstration.

~~T~~he shutdown margin demonstration ~~will~~ to be performed in the cold shutdown condition with the vessel head in place. ~~T~~he shutdown margin demonstration will be performed prior to the reactor coolant system pressure and control rod scram time tests following refueling outages when core alterations are performed. The shutdown margin demonstration is performed using the in-sequence non-critical method.

This special test exception provides the appropriate additional controls to allow

Compliance with this special test exception is optional and applies only if