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Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

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U.S. Nuclear Regulatory Commission
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Monticello Nuclear Generating Plant
Docket 50-263
License No. DPR-22

Additional Technical Specification Changes for the Monticello Nuclear Generating Plant Regarding Application of Alternative Source Term (AST) Methodology to Re-Evaluate the Fuel-Handling Accident (TAC No. MC3299)

References:
1) NMC letter to NRC, "License Amendment Request: Selective Scope Application of an Alternative Source Term Methodology for Re-evaluation of the Fuel Handling Accident," (L-MT-04-023) dated April 29, 2004.
2) NMC letter to NRC, "Supplement 1 to License Amendment Request: Selective Scope Application of an Alternative Source Term Methodology for Re-evaluation of the Fuel Handling Accident," (L-MT-04-064) dated November 23, 2004, (TAC No. MC3299).
3) NRC letter to NMC, "Monticello Nuclear Generating Plant - Request for Additional Information Related to Technical Specifications Change Request to Apply Alternative Source Term (AST) Methodology to Re-Evaluate the Fuel-Handling Accident (TAC No. MC3299)," dated January 11, 2005.
4) NMC letter to NRC, "Response to Request for Additional Information Related to Technical Specifications Change Request to Apply Alternative Source Term (AST) Methodology to Re-Evaluate the Fuel-Handling Accident, dated January 11, 2005 (TAC No. MC3299)," (L-MT-05-001) dated January 20, 2005.

Pursuant to 10 CFR 50.67 and 10 CFR 50.90, the Nuclear Management Company, LLC, (NMC) hereby request changes to the Technical Specifications (TS) for the Monticello Nuclear Generating Plant (MNGP), Operating License No. DPR-22. This License Amendment Request (LAR) proposes a change to the MNGP TS regarding Spent Fuel Pool (SFP) water level and provides supplemental changes to previously submitted TS changes associated with an alternative source term (AST) to the fuel handling accident (FHA) (Reference 1).

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On November 23, 2004, (Reference 2) NMC provided a supplemental submittal to the April 29, 2004, FHA-AST amendment request. The U.S. Nuclear Regulatory Commission (NRC) staff requested additional information (RAI) regarding NMC's submittals (Reference 1 and 2) on January 11, 2005 (Reference 3). NMC provided a response on January 20, 2005 (Reference 4), in which NMC stated that a TS change request for the spent fuel pool water level would be submitted under separate correspondence. This submittal provides that TS change and proposes additional changes to the previously submitted LAR dated April 29, 2004.

Enclosure 1 provides the introduction, summary, background, description of the proposed changes, and technical and regulatory evaluations. The no significant hazards and environmental considerations determinations submitted in Reference 1 were reviewed and were determined to be applicable and continue to bound the proposed changes discussed herein. Enclosure 2 provides a mark-up of the proposed changes to the MNGP TS and Bases. Enclosure 3 provides a retyped version of the proposed TS and Bases changes. Enclosure 4 provides a copy of calculation CA-05-072, "Effect of Reduced Pool Water Levels on Fuel Handling Accident Consequences," which provides the analysis basis for the portion of the proposed amendment that provides a TS change related to the spent fuel storage pool water level.

This letter contains no new commitments, but it does supersede a prior commitment (Item 1. below) made in Reference 1. This letter also completes the commitment made in Reference 4 (Item 2. below), to provide a TS change for SFP water level during irradiated fuel movement. Submittal of this letter fulfills these two commitments.

1. "NMC will revise refueling procedure(s) to require a minimum of 23 feet of water above stored fuel in the Spent Fuel Pool during irradiated fuel movement."

2. "NMC will propose a Technical Specification for Spent Fuel Pool water level during irradiated fuel movement under separate correspondence."

The MNGP Operations Committee has reviewed this submittal and a copy is being forwarded to our appointed state official pursuant to 10 CFR 50.91(b)(1).

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

Executed on 4/12/2005.

Thomas J. Palmisano
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosures (4)

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

## ENCLOSURE 1

# Additional Technical Specification Changes for Monticello Nuclear Generating Plant Regarding Application of Alternative Source Term (AST) Methodology to Re-Evaluate the Fuel-Handling Accident

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## ENCLOSURE 1

**Additional Technical Specification Change for Monticello Nuclear Generating Plant Regarding Application of Alternative Source Term (AST) Methodology to Re-Evaluate the Fuel-Handling Accident**

### 1.0 INTRODUCTION

Pursuant to 10 CFR 50.67 and 10 CFR 50.90, the Nuclear Management Company, LLC, (NMC) hereby requests changes to the Technical Specifications (TS) for the Monticello Nuclear Generating Plant (MNGP), Operating License No. DPR-22. This License Amendment Request (LAR) proposes a change to the MNGP TS regarding Spent Fuel Pool (SFP) water level and provides supplemental changes to previously submitted TS changes associated with an alternative source term (AST) to the fuel handling accident (FHA) (Reference 1).

On November 23, 2004, (Reference 2) NMC provided a supplemental letter discussing shutdown administrative controls for Secondary Containment, ventilation system and radiation monitor availability during refueling, and validation of the FHA radiological consequence analysis Control Room inleakage assumptions. The U.S. Nuclear Regulatory Commission (NRC) staff requested additional information (RAI) regarding NMC's submittals (Reference 1 and 2) on January 11, 2005 (Reference 3). NMC provided a response on January 20, 2005 (Reference 4), in which NMC stated that a TS change request for the spent fuel pool water level would be submitted under separate correspondence. This submittal provides that TS change and proposes additional changes to the previously submitted LAR dated April 29, 2004.

### 2.0 SUMMARY

In Reference 4, NMC committed to providing a TS change for the SFP water level during irradiated fuel movement by separate correspondence. This submittal provides the proposed TS change and additional changes to relocate action statements to the appropriate TS, correct an inadvertent omission in the TS and clarify TS changes proposed in the original submittal.

In support of the proposed TS change for the SFP water level during irradiated fuel movement, Enclosure 4 (Reference 5) provides a copy of calculation CA-05-072, "Effect of Reduced Pool Water Levels on Fuel Handling Accident Consequences." This calculation provides a generic analysis of the change in decontamination factor (DF) as a function of pool water depth. This analysis evaluates the drop of a fuel assembly in the Refuel Pool (RP) and the SFP. The results indicate that the design basis accident (DBA) FHA, which involves a drop of a fuel assembly onto the reactor core, bounds the postulated fuel handling accidents in the RP and the SFP.

## 3.0 BACKGROUND

In the April 29, 2004, LAR (Reference 1), NMC made the following statement consistent with RG 1.183 assumptions for the FHA:

> "NMC will revise refueling procedure(s) to require a minimum of 23 feet of water above stored fuel in the spent fuel pool during irradiated fuel movement."

In Question 1 in the January 11, 2005, RAI (Reference 3) the NRC asked the following:

> "One of the proposed commitments associated with this license amendment request is to change the refueling procedures to require a minimum of 23 feet of water above stored fuel in the spent fuel pool during irradiated fuel movement. *Such a commitment is usually linked with a technical specification* (TS) requirement. Why wasn't a TS surveillance requirement proposed to require 23 feet of water above stored fuel?"

NMC in the January 20, 2004, RAI response to question 1 above, stated the following:

> "NMC will propose a Technical Specification for Spent Fuel Pool water level during irradiated fuel movement under separate correspondence."

This submittal requesting a MNGP LAR provides the proposed TS change.

### 3.1 *Current Monticello Technical Specifications*

Currently MNGP Technical Specification (TS) 3.10.C, "Fuel Storage Pool Water Level," states:

> "Whenever irradiated fuel is stored in the fuel storage pool, the pool water level shall be maintained at a level of greater or equal to 33 feet."

The Bases for this specification, Specification 3.10.C, state:

> "To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 33 feet is established because it would be a significant change from the normal level (37' 9") and well above a level to assure adequate cooling."

Handling of irradiated fuel in the reactor cavity and SFP is only permitted by plant procedures when the water level in the reactor cavity is flooded up and the spent fuel pool is at a high water level. During refueling or other operations involving irradiated fuel movement, the water level is verified to be at the nominal level of the SFP skimmer plates (approximately 37 feet 9 inches from the bottom of the Spent Fuel Storage Pool). Establishment and maintenance of a specified depth of water above the top of the irradiated fuel assemblies seated in the spent fuel storage pool racks is not a TS requirement in the MNGP custom specifications, but rather an administrative practice in the refueling procedures. As described in the TS Bases, the minimum water level was provided specifically to shield and cool irradiated fuel assemblies stored in the pool.

NMC is proposing to revise TS 3/4.10.C, "Fuel Storage Pool Water Level," as discussed in this submittal.

### 3.2 Additional Changes Proposed in this Submittal

NMC is in the process of developing a submittal to implement a Full-Scope AST at MNGP. During the development of that LAR, reviews and evaluations are ongoing to update the MNGP design and licensing basis. During these NMC reviews it was determined that the previously submitted TS pages (Reference 1) require additional changes for clarity, as well as relocation of action statements to their appropriate location with respect to the TS content. These changes are considered administrative changes and do not affect the previously submitted determination of no significant hazards consideration or the environmental assessment.

## 4.0 DESCRIPTION OF THE PROPOSED CHANGES

As stated above, NMC reviewed the AST-FHA submittal to determine impact to the implementation of a Full-Scope AST TS change at MNGP. NMC's review determined that additional changes needed to be made to the proposed revised TS pages that were submitted by letter dated April 29, 2004. Sections 4.1 through 4.3 describe the revisions to the TS changes proposed in that submittal. NMC is also proposing the change discussed in Section 4.4 to provide a revised Spent Fuel Storage Pool Water Level TS consistent with the commitment in NMC response to Request for Additional Information dated January 20, 2005.

### 4.1 Specification 3.3 – Control Rod System (Relocate TS Actions)

Section 4.1.3 in NMC's original AST-FHA LAR (Reference 1), the third and fourth paragraphs described proposed changes to MNGP TS 3.7.C. That submittal stated, in part:

> *"Specification paragraph 3.7.C, "Secondary Containment," directs compliance with Specification 3.3.A via paragraphs 3.7.C.2.a and 3.7.C.2.c and provides the actions to take if compliance cannot be maintained, since individual action statement paragraphs are not provided under Specification 3.3.A.1, "Reactivity Limitations, Reactivity margin – core loading." Due to this difference in presentation between NUREG-1433 and the MNGP TS, it is necessary to separate the actions pertaining to the movement of 'recently' irradiated fuel and OPDRVs from those required for shutdown margin considerations.*
>
> *The term 'Alterations of the reactor core' will be removed from action statement paragraph 3.7.C.4 as it is now embodied in new action statement 3.7.C.5. New action statement 3.7.C.5 will continue to require, as currently required by paragraph 3.7.C.4, that alterations of the reactor core be suspended if Specification 3.3.A can not be met. Providing a separate action for this condition*

*is appropriate since it no longer applies except with respect to shutdown margin (SDM) considerations. This will maintain consistency with the format of the MNGP TS. When SDM is not met during refueling the operator must immediately suspend operations that could reduce SDM. Inserting control rods or removing fuel from the core will reduce the total reactivity and are excluded from the suspended actions. The proposed action statement will say:*

5. *"With the shutdown margin below the limit specified in specification 3.3.A, immediately suspend core alterations except for fuel assembly removal.*

   *AND*

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

*The word 'recently' will be added before 'irradiated fuel' in action statement paragraph 3.7.C.4 to clarify that Secondary Containment is not required during handling of irradiated fuel that has decayed for longer than 24-hours, consistent with results of the AST FHA analysis. For the reasons previously indicated, the wording of action statement paragraph 3.7.C.4 will be revised to require the establishment of Secondary Containment during OPDRVs."*

The proposed changes above described compliance with TS 3.3.A (entitled "Reactivity Limitations," which corresponds to Shutdown Margin) and proposed action requirements if those requirements cannot be met. However, NMC has determined upon further review that compliance with TS 3.3.A is already required by current TS 3.7.C.2.a; any additional requirements are redundant. Furthermore, inclusion of the new TS Action (proposed TS 3.7.C.5) in the location proposed in the original NMC submittal creates a potential for operator error by locating a reactivity margin TS Action Requirement within a Secondary Containment specification. Proper location of this Action Requirement is with other Shutdown Margin (SDM) Actions (MNGP TS 3.3).

As stated in the Reference 1 submittal:

> *When SDM is not met during refueling the operator must immediately suspend operations that could reduce SDM. Inserting control rods or removing fuel from the core will reduce the total reactivity and are excluded from the suspended actions.*

The proposed revisions in this LAR will delineate between the two operational applications of SDM by separating TS 3.3.G into two subparts; one for all operating modes except for Refuel and one for the Refuel mode of operation.

Therefore, NMC is proposing to relocate this Action Requirement to TS 3.3.G, which provides Required Actions for the reactivity margin requirements of MNGP TS 3.3.A. This administrative change will revise TS 3.3.G as follows:

3.3.G Required Action**[s]**

[1.] If Specifications 3.3.A **(except when the reactor mode switch is in the Refuel position)** through 3.3.D above are not met an orderly shutdown shall be initiated and the reactor placed in the cold shutdown condition within 24 hours.

[2. **If Specification 3.3.A is not met when the reactor mode switch is in the Refuel position, immediately suspend core alterations except for fuel assembly removal and immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.]**

## 4.2 Specification 3.7.B – Standby Gas Treatment System (Clarify TS Wording)

Section 4.1.2 in NMC's original AST-FHA submittal dated April 29, 2004, discussed proposed changes to the MNGP TS 3.7.B. This proposed change added TS sections 3.7.B.1.c and 3.7.B.1.d to the existing TS 3.7.B to define actions for when one or both trains of Standby Gas Treatment (SBGT) are inoperable during movement of recently irradiated fuel in the secondary containment, or during operations with the potential to drain the reactor vessel (OPDRV). Upon further review it was determined that a TS requirement was inadvertently omitted and additional clarifications should have been included in the TS.

The intent of the originally proposed changes to MNGP TS 3.7.B.1 was to clearly define the actions to be taken when one or both trains of the SBGT System are inoperable during fuel movement. The original submittal stated:

> *"Two action statement paragraphs (3.7.B.1.c and 3.7.B.1.d) are proposed to be added to the SBGT System specification to define actions for when one or both trains of the system are inoperable during movement of 'recently' irradiated fuel in the Secondary Containment, or during OPDRVs. These actions are modeled after those contained in MNGP TS 3.17.B.1, "Control Room Emergency Filtration System," and are consistent with NUREG-1433.*
>
> *To clearly define the actions to be taken when one or both trains of the SBGT System are inoperable during fuel movement, proposed action statement paragraphs 3.7.B.1.c and 3.7.B.1.d will include the word 'recently' before 'irradiated fuel' to clarify the applicability of the specification. TS operability of the SBGT System trains would no longer be required during handling of irradiated fuel that has decayed for longer than 24 hours, consistent with results of the analysis. For consistency with current industry guidance, as promulgated by the NUREG for the BWR/4 reactor design, NMC will add 'operations having the potential for draining the reactor vessel' as an applicable condition to paragraphs 3.7.B.1.c and 3.7.B.1.d. The existing seven-day allowance for one train of the SBGT System being out-of-service in action statement paragraph 3.7.B.1.a is retained and included as part of action statement paragraph 3.7.B.1.c, consistent with the current MNGP TS. Adding OPDRVs to paragraphs 3.7.B.1.c and 3.7.B.1.d is necessary for consistency with the Control Room ventilation specifications and current industry guidance. The new action statements will say:*

c. *"With one standby gas treatment system train inoperable during movement of recently irradiated fuel assemblies in the secondary containment or during operations with the potential for draining the reactor vessel, activities may continue for up to seven days. After seven days, immediately place the operable standby gas treatment system train in operation or immediately suspend movement of recently irradiated fuel assemblies in the secondary containment or immediately suspend operations with the potential for draining the reactor vessel.*

d. *With both standby gas treatment system trains inoperable during movement of recently irradiated fuel assemblies in the secondary containment or during operations with the potential for draining the reactor vessel, immediately suspend these activities."*

*Accordingly, to define when the SBGT System is applicable, applicability paragraph 3.7.B.1 will be revised to include these two additional actions (3.7.B.1.c and 3.7.B.1.d). The term 'and fuel handling' will be removed from action statement paragraph 3.7.B.1.a since new action statements 3.7.B.1.c and 3.7.B.1.d provide for more definitive actions to be taken during fuel handling operations. Also, as described in Section 4.1.3, applicability paragraph 3.7.C.2.d is being broken up into paragraphs 3.7.C.2.d and 3.7.C.2.e and a new paragraph, 3.7.C.2.f is being added. Therefore, action statement paragraphs 3.7.B.1.a and 3.7.B.1.b are being revised to list 'Specification 3.7.C.2.(a) through (f).' Additionally, as an administrative change, the term 'circuits' will be replaced with 'trains' in specification paragraphs 3.7.B.1, 3.7.B.1.a, 3.7.B.1.b, and the term will also be utilized in new action paragraphs 3.7.B.1.c and 3.7.B.1.d for consistency with Surveillance Requirement (SR) 4.7.B.1 and to avoid confusion with electrical circuits."*

Four clarifications are required to the proposed changes in Reference 1. First, the proposed sections (3.7.B.1.c and 3.7.B.1.d) could create a conflict with each previously existing section. For example, TS 3.7.B.1.a and 3.7.B.1.c both contain LCOs for one (1) inoperable SBGT train, yet they have different action requirements. Similarly, TS 3.7.B.1.b and 3.7.B.1.d contain LCOs for when two (2) trains of SBGT are inoperable. Therefore, this LAR is clarifying that TS 3.7.B.1.a and 3.7.B.1.b are applicable when reactor water temperature is greater than or equal to 212°F. This change is required because the addition of TS 3.7.B.1.c and 3.7.B.1.d are not restricted by applicability statements and therefore, could cause confusion to the operator.

The second clarification required is to restore requirements for cask movement with the SBGT inoperable. The original submittal inadvertently omitted requirements for cask movement with the SBGT system inoperable in proposed TS 3.7.B.1.c and 3.7.B.1.d that should have been retained consistent with the current requirements.

Third, administrative changes are also needed for TS 3.7.B.1.a and 3.7.B.1.b to revise the numbering for applicability requirements regarding Specification 3.7.C.2.a through b. because of related changes to TS 3.7.C.2 discussed below.

Finally, a wording correction is needed for 3.7.B.1.c, inserting "and" instead of "or," to match the wording of TSTF-51, Rev. 2. This change clarifies that if either, or if both, evolutions are being performed that they must both be suspended.

Therefore, NMC proposes to modify the previously submitted TS changes to add the following statements:

B. Standby Gas Treatment System

1. Two separate and independent standby gas treatment system trains shall be operable at all times when secondary containment integrity is required, except as specified in sections 3.7.B.1.(a) through (d).

   a. After one of the standby gas treatment system trains is made or found to be inoperable **[with reactor water temperature ≥ 212°F,]** for any reason, reactor operation is permissible only during the succeeding seven days, provided that all active components in the other standby gas treatment system are operable. Within 36 hours following the 7 days, the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.2.(a) **[and ~~(d)~~ (b)]**.

   b. If both standby gas treatment system trains are not operable, **[with reactor water temperature ≥ 212°F,]** within 36 hours the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.2.(a) ~~through~~ **[and ~~(d)~~ (b)]**.

   c. ***[With one standby gas treatment system train inoperable,***

      **1.) The following activities may continue for up to 7days:**

      **(a) Movement of recently irradiated fuel assemblies in secondary containment;**

      **(b) Movement of the fuel cask in the reactor building; and**

      **(c) Operations with the potential to drain the reactor vessel.**

      **2.) After 7 days:**

      **(a) Immediately suspend movement of the fuel cask in the reactor building; and**

      **(b) Immediately place the operable standby gas treatment system train in operation, or**

      **(1.) Immediately suspend movement of recently irradiated fuel assemblies in secondary containment; and**

      **(2.) Immediately suspend operations with the potential to drain the reactor vessel.]**

d. With both standby gas treatment trains inoperable immediately suspend: Movement of recently irradiated fuel assemblies in secondary containment; **[Movement of the fuel cask in the reactor building;]** Operations with the potential for draining the reactor vessel.

### 4.3 Specification 3.7.C – Secondary Containment (Clarify and Relocate TS)

Section 4.1.3 in NMC's original AST-FHA submittal dated April 29, 2004, proposed changes to TS 3.7.C. The results of the AST FHA analysis indicated changes were necessary to the Secondary Containment specification.

The intent of the originally proposed changes to MNGP TS 3.7.C was to clearly define the actions to be taken when Secondary Containment was not required to be operable. The original submittal stated:

*"Secondary Containment is established via Specification 3.7.C. Secondary Containment System paragraphs 3.7.C.1 and 3.7.C.2 define the applicability of this LCO. Paragraphs 3.7.C.3 and 3.7.C.4 provide actions to be taken when the LCO cannot be met. The results of the AST FHA analysis indicated changes were necessary to the Secondary Containment specification. To reflect current industry standards (i.e., NUREG-1433) an extensive rewrite of the MNGP TS would be necessary – beyond the scope of selectively implementing an AST for the FHA. To avoid a rewrite, changes consistent with the approach of the NUREG and the current MNGP TS configuration are proposed as described below.*

*Applicability paragraph 3.7.C.2.d will be broken up into two separate paragraphs 3.7.C.2.d and 3.7.C.2.e. Paragraph 3.7.C.2.d will apply to movement of a fuel cask. Paragraph 3.7.C.2.e will apply during movement of irradiated fuel. The word 'recently' will be added before 'irradiated fuel' in new applicability paragraph 3.7.C.2.e to clarify that Secondary Containment is not required during handling of irradiated fuel that has decayed longer than 24 hours, consistent with results of the AST FHA analysis. A new applicability paragraph 3.7.C.2.f will be added to require establishment of Secondary Containment during 'operations with the potential for draining the reactor vessel.' The revised applicability statements under paragraph 3.7.C.2 will say:*

*d. "The fuel cask is not being moved within the reactor building.*

*e. Recently irradiated fuel is not being moved within the reactor building.*

*f. Operations with the potential for draining the reactor vessel are not being performed."*

*Specification paragraph 3.7.C, "Secondary Containment," directs compliance with Specification 3.3.A via paragraphs 3.7.C.2.a and 3.7.C.2.c and provides the actions to take if compliance cannot be maintained, since individual action statement paragraphs are not provided under Specification 3.3.A.1, "Reactivity Limitations, Reactivity margin – core loading." Due to this difference in presentation* *between NUREG-1433 and the MNGP TS, it is necessary to separate the actions pertaining to the movement of 'recently' irradiated fuel and OPDRVs from those required for shutdown margin considerations.*

*The term 'Alterations of the reactor core' will be removed from action statement paragraph 3.7.C.4 as it is now embodied in new action statement 3.7.C.5. New action statement 3.7.C.5 will continue to require, as currently required by paragraph 3.7.C.4, that alterations of the reactor core be suspended if Specification 3.3.A can not be met. Providing a separate action for this condition is appropriate since it no longer applies except with respect to shutdown margin (SDM) considerations. This will maintain consistency with the format of the MNGP TS. When SDM is not met during refueling the operator must immediately suspend operations that could reduce SDM. Inserting control rods or removing fuel from the core will reduce the total reactivity and are excluded from the suspended actions. The proposed action statement will say:*

5. *"With the shutdown margin below the limit specified in specification 3.3.A, immediately suspend core alterations except for fuel assembly removal.*

   *AND*

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

*The word 'recently' will be added before 'irradiated fuel' in action statement paragraph 3.7.C.4 to clarify that Secondary Containment is not required during handling of irradiated fuel that has decayed for longer than 24-hours, consistent with results of the AST FHA analysis. For the reasons previously indicated, the wording of action statement paragraph 3.7.C.4 will be revised to require the establishment of Secondary Containment during OPDRVs.*

Further review of the original submittal identified that three clarifications were required for this Section. First, TS 3.7.C.2.c and TS 3.7.C.4 contain redundant requirements to TS 3.7.C.2.a. NMC is proposing to delete these redundant requirements because it is implicit in TS 3.7.C.2.a that activities would not be performed that would reduce reactivity margins below the requirements of TS 3.3.A.

Second, the addition of TS 3.7.C.5 provided TS Action Requirements that should have been located in TS 3.3.G, "Required Action," as discussed in Section 4.1 of this LAR. Dividing the Action Requirements of TS 3.7.C.4 into two separate actions provides clarity to the operators and more clearly delineates that the actions are to be performed simultaneously. Administrative changes were also identified in the numbering of the requirements located in TS 3.7.C.2.

Finally, NMC identified that a mode of applicability statement is required to eliminate potential conflicts in TS 3.7.C.4 by differentiating between proposed TS 3.7.C.4.a, which is a required action when the reactor is in Run, Startup and Hot Shutdown modes of operation and proposed TS 3.7.C.4.b, which is required at all times when any of the tasks required to be suspended by the TS are being performed.

Therefore, NMC proposes to change the previously submitted revisions to MNGP TS's 3.7.C.2 and 3.7.C.4 to provide the following statements:

3.7.C Secondary Containment

2. Secondary Containment Integrity is not required when all of the following conditions are satisfied:

   a. The reactor is subcritical and Specification 3.3.A is met.

   b. The reactor water temperature is below 212°F.

   **[c.** The fuel cask is not being moved within the reactor building.

   **d.** Recently irradiated fuel is not being moved within secondary containment.

   **e.]** Operations with the potential for draining the reactor vessel are not being performed.

4. **[a. During Run, Startup or Hot Shutdown, i]**f Specifications 3.7.C.1 through 3.7.C.3 cannot be met, initiate a normal orderly shutdown and have the reactor in the Cold Shutdown condition within 36 hours.

   **[And**

   **b. If Specifications 3.7.C.1 through 3.7.C.3 cannot be met immediately suspend:**

      **1. Operations with a potential for draining the reactor vessel.**

      **2. Handling of recently irradiated fuel in the secondary containment.**

      **3. Movement of a fuel cask in the reactor building.]**

## 4.4 Specification 3.10.C – Fuel Storage Pool Water Level (SFP Water Level)

NMC stated the following in the January 20, 2004, RAI response:

> "NMC will propose a Technical Specification for Spent Fuel Pool water level during irradiated fuel movement under separate correspondence."

This submittal requesting a MNGP LAR provides the proposed TS change revising the spent fuel pool water level.

Currently MNGP TS 3.10.C states:

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool, the pool water level shall be maintained at a level of greater or equal to 33 feet.

The Bases for this specification states:

> "To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 33 feet is established because it would be a significant change from the normal level (37 feet 9 inches) and well above a level to assure adequate cooling."

Handling of irradiated fuel in the reactor cavity and SFP is only permitted by plant procedures when the reactor cavity is flooded up and the spent fuel pool is at a high water level. During refueling or other operations involving irradiated fuel movement, the water level is verified to be at the nominal level of the SFP skimmer plates (approximately 37 feet 9 inches from the bottom of the Spent Fuel Storage Pool). Establishment and maintenance of a specified depth of water above the top of the irradiated fuel assemblies seated in the spent fuel storage pool racks is not a TS requirement in the MNGP custom technical specifications, but rather an administrative practice in the refueling procedures.

The spent fuel storage pool water level requirement is not applicable during the transport of irradiated fuel within a spent fuel cask. In this case, the cask will be lifted and transported to and from the spent fuel storage pool with the single failure proof reactor building crane in a manner controlled by the plant heavy loads program. By using a single failure proof crane system, and complying with the MNGP heavy loads program, the potential for a cask drop is extremely small and need not be postulated. Therefore, maintaining the minimum water level of 37 ft in the spent fuel storage pool during cask handling activities is not required.

NMC is proposing to revise MNGP TS 3.10.C, "Fuel Storage Pool Water Level," to read as follows:

C. **[Spent]** Fuel Storage Pool Water Level

**[During movement of irradiated fuel assemblies, the spent fuel storage pool water level shall be maintained ≥ 37 ft above the bottom of the spent fuel storage pool.**

**If the spent fuel storage pool water level is made or found not to be within limits, immediately suspend movement of irradiated fuel assemblies.]**

NMC is also proposing to revise TS Surveillance Requirement 4.10.C to read as follows:

C. **[Spent]** Fuel Storage Pool Water Level

**[Verify that the spent fuel storage pool water level is ≥ 37 ft above the bottom of the spent fuel storage pool:**

a. **Once every 24 hours, during movement of irradiated fuel assemblies, or**

b. **Once every 7 days, when irradiated fuel assemblies are stored in the spent fuel storage pool.]**

Corresponding bases changes will be made to describe the revised licensing basis for the additional proposed changes.

## 5.0 TECHNICAL EVALUATION

By letter dated April 29, 2004, NMC submitted a License Amendment Request that proposed relaxations of the operability requirements for the Secondary Containment, secondary containment related support systems and the associated ESF systems during core alterations and movement of 'recently' irradiated fuel assemblies. These proposed changes were supported by the results of the AST FHA analysis. Results of radiological consequence analyses for an AST FHA indicated that the release of fission products would result in doses that were well within the acceptable dose criteria specified in 10 CFR 50.67 for the EAB and LPZ and for the Control Room operator.

This supplemental License Amendment Request provides changes to the previously proposed revisions to clarify TS wording, include additional TS requirements that were inadvertently omitted in the original submittal, relocate proposed TS actions to the appropriate TS and provides a new TS change request for the Spent Fuel Storage Pool water level.

### 5.1 Evaluation of TS Changes for Previously Submitted Revisions to TS

In Sections 4.1 through 4.3 above, NMC is proposing additional changes to previously submitted TS revisions. The Technical Evaluation of AST FHA Related TS Changes provided in NMC letter dated April 29, 2004, is still applicable to these proposed changes.

Other administrative changes are required to conform the current TS to the proposed revisions being requested by NMC. These consist of the following:

Section 4.1 of this LAR proposed to relocate the TS Action statement originally submitted as TS 3.7.C.5 in Reference 1 to TS 3.3.G. Relocation of the proposed action requires division of TS 3.3.G into two subparts that will delineate between the two operational applications of the action statements. The first action will be applicable when the reactor mode switch is in any position other than the Refuel position. The second is applicable only when the reactor mode switch is in the Refuel position.

Section 4.2 of this LAR proposed four clarifications to the proposed changes in Reference 1:

First, the proposed sections (3.7.B.1.c and 3.7.B.1.d) could create a conflict with each previously existing section. For example, TS 3.7.B.1.a and 3.7.B.1.c, both contain LCOs for one (1) inoperable SBGT train, yet they have different action requirements. Similarly, TS 3.7.B.1.b and 3.7.B.1.d contain LCOs for when two (2) trains of SBGT are inoperable. Therefore this LAR is clarifying that TS 3.7.B.1.a and 3.7.B.1.b are applicable when reactor water temperature is greater than or equal to 212°F. This change is required because the addition of TS 3.7.B.1.c and 3.7.B.1.d are not restricted by applicability statements and therefore, could cause confusion to the operator.

The second clarification restores requirements for cask movement with the SBGT inoperable. The original submittal inadvertently omitted requirements for cask movement with the SBGT system inoperable in proposed TS 3.7.B.1.c and 3.7.B.1.d that should have been retained consistent with the current requirements.

Third, administrative changes are also needed for TS 3.7.B.1.a and 3.7.B.1.b to revise the numbering for applicability requirements regarding Specification 3.7.C.2.a through b. because of related changes to TS 3.7.C.2.

Finally, a wording correction is needed for 3.7.B.1.c, inserting "and" instead of "or," match the wording of TSTF-51, Rev. 2. This change clarifies that if either, or if both, evolutions are being performed that they must both be suspended.

Section 4.3 of this LAR proposes three clarifications for this Section. First, TS 3.7.C.2.c and TS 3.7.C.4 contain redundant requirements to TS 3.7.C.2.a. NMC is proposing to delete these redundant requirements because it is implicit in TS 3.7.C.2.a that activities would not be performed that would reduce reactivity margins below the requirements of TS 3.3.A.

Second, the addition of TS 3.7.C.5 provided TS Action Requirements that should have been located in TS 3.3.G, "Required Action," as discussed in Section 4.1 of this LAR. Dividing the Action Requirements of TS 3.7.C.4 into two separate actions provides clarity to the operators and more clearly delineates that the actions are to be performed simultaneously. Administrative changes were also identified in the numbering of the requirements located in TS 3.7.C.2.

Finally, NMC identified that a mode of applicability statement is required to eliminate potential conflicts in TS 3.7.C.4 by differentiating between proposed TS 3.7.C.4.a, which is a required Action when the reactor is in Run, Startup and Hot Shutdown modes of operation and proposed TS 3.7.C.4.b, which is required at all times when any of the task required to be suspended by the TS are being performed.

## 5.2 Evaluation of Proposed TS for Spent Fuel Pool Water Level

Calculation CA-05-072 (Reference 5) was performed to evaluate the consequences of a FHA in the RP onto the reactor vessel flange and in the SFP. As described in the MNGP USAR Section 14.7.6.1, the limiting FHA results from the accidental dropping of a fuel assembly into the reactor vessel onto the top of the core. MNGP calculation CA-

04-041 (previously submitted as a part of Reference 1) evaluated the limiting FHA event, in which a fuel assembly is dropped onto the reactor core, in support of the MNGP Selective Scope AST FHA (Reference 1). The Reference 5 calculations for the RP and SFP quantify the effects on accident consequences relative to the limiting FHA, and demonstrate that these postulated events remain bounded by the limiting design basis FHA. Additionally, this calculation generically determines decontamination factor (DF) as a function of pool depth. The results of these analyses are used as input to the selection of an appropriate value for use in the MNGP SFP water level TS proposed in this submittal.

The water depth over the core during fuel movement is approximately 46 feet. The Reference 1 AST FHA calculation conservatively assumes a DF of 200, which corresponds to a water depth of 23 feet. This assumption is consistent with the guidance in RG 1.183: "If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200..."

Water level over the reactor pressure vessel (RPV) flange and over the stored assemblies in the SFP could potentially be less than 23 feet. RG 1.183 states: "If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1)." Although the USAR identifies the FHA onto the reactor core as the limiting FHA, a calculation was performed to quantify a decontamination factor consistent with the RG 1.183 guidance for the fuel handling events in the RP and SFP.

Two FHA's were considered in Reference 5. The drop of a fuel assembly in the SFP is similar to the FHA over the reactor vessel cavity in that it involves a drop of a single fuel assembly onto other fuel assemblies. The drop of a fuel assembly in the RP onto the reactor vessel flange involves only one fuel assembly, since it is not dropped onto other fuel assemblies. Water depth affects the radiological consequences of both accidents due to a portion of the released iodine nuclides being retained by the water, thereby reducing the amount of activity released and the resulting radiological doses. In these events, the depth of water above the fuel in the SFP and the depth of water above the reactor vessel flange are both less than the depth of the water above the core, which will result in less iodine removal by the pool water. However, the drop height of the fuel assembly is shorter in both the SFP and the RP over the reactor vessel flange, than in the reactor cavity. This results in less energy in the dropped fuel assembly and, consequently, fewer failed rods.

The Reference 5 analysis demonstrates that the limiting FHA, in which a fuel assembly is dropped into the reactor vessel onto the top of the core, bounds the results of a FHA in the SPF or in the RP over the reactor vessel flange. Dose consequences for the FHA in the SFP and in the RP over the reactor vessel flange are less than those calculated for the FHA over the reactor core (Reference 1).

The Reference 5 calculation supports the proposed TS SFP water level of 37 feet from the bottom of the SFP. Maintenance of this level assures margin to the minimum water levels determined in the analysis for the stored fuel assemblies and the vessel flange.

Thus, the specified water level preserves the assumptions of the design basis fuel handling accident analysis.

The proposed change to the TS SFP water level also affects the associated surveillance requirement (SR). This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7-day frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

Utilization of a 24-hour surveillance interval during movement of irradiated fuel assemblies, as proposed in TS SR 4.10.C, serves to incorporate the most conservative reactor pool level surveillance requirements included in NUREG-1433, "Standard Technical Specification, General Electric Plants, BWR/4, Revision 3."

# 6.0 REGULATORY EVALUATION

## 6.1 No Significant Hazards Consideration and Environmental Assessment

NMC has determined that the Determination of No Significant Hazards Consideration and Environmental Assessment submitted by the original letter dated April 29, 2004, are also applicable to this submittal. Revised marked-up TS pages and TS Bases pages are included in Enclosure 2 of this submittal and the retyped pages are included in Enclosure 3. NMC has determined that the TS Bases changes can be implemented during the implementation of this License Amendment by following the guidance of MNGP TS 6.8.K, "Technical Specifications (TS) Bases Control Program." Therefore, the revised marked-up TS Bases pages included in Enclosure 2 of this submittal are provided for information only.

## 6.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.36 prescribes criteria for consideration of items for inclusion in TS. Criterion 2 includes process variables, design features, or operating restrictions that are an initial condition of design basis accidents or transient analyses that either assume the failure of or present a challenge to the integrity of a fission barrier.

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level preserves the assumptions of the fuel handling accident analysis. As such, it is the minimum required for movement of irradiated fuel assemblies within the spent fuel storage pool. The revised TS apply during movement of irradiated fuel assemblies in the spent fuel storage pool since that is when the potential for a release of fission products exists.

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 the health and safety of the public.

## 7.0 REFERENCES


1. NMC letter to NRC, "License Amendment Request: Selective Scope Application of an Alternative Source Term Methodology for Re-evaluation of the Fuel Handling Accident," (L-MT-04-023) dated April 29, 2004.

2. NMC letter to NRC, "Supplement 1 to License Amendment Request: Selective Scope Application of an Alternative Source Term Methodology for Re-evaluation of the Fuel Handling Accident," (L-MT-04-064) dated November 23, 2004, (TAC No. MC3299).

3. NRC letter to NMC, "Monticello Nuclear Generating Plant - Request for Additional Information Related to Technical Specifications Change Request to Apply Alternative Source Term (AST) Methodology to Re-Evaluate the Fuel-Handling Accident (TAC No. MC3299)," dated January 11, 2005.

4. NMC letter to NRC, "Response to Request for Additional Information Related to Technical Specifications Change Request to Apply Alternative Source Term (AST) Methodology to Re-Evaluate the Fuel-Handling Accident, dated January 11, 2005 (TAC No. MC3299)," (L-MT-05-001) dated January 20, 2005.

5. MNGP Calculation CA-05-072,"E ffect of Reduced Pool Water Levels on Fuel Handling Accident Consequences," dated March 11, 2005.

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION AND BASES CHANGES (MARK-UP)

**This enclosure consists of current Technical Specification and Technical Specification Bases pages marked up with the proposed changes. The following pages, included in this exhibit, should be added to, or replace the pages listed in Enclosure 3 to the NMC submittal dated April 29, 2004, as instructed below:**

| Insert Pages | | | Remove Pages |
|---|---|---|---|
| 83a | TS 3.3.G - | Control Rod System | - |
| 92 | Bases 3.3 - | (Insert new pages) | - |
| 166 | TS 3.7.B - | Standby Gas Treatment System | 166 |
| 167 | | (Replace current pages) | 167 |
| 167a | | (Insert new page) | - |
| 169 | TS 3.7.C - | Secondary Containment | 169 |
| 170 | | (Replace current pages) | 170 |
| 170a | | (Insert new page) | - |
| 181 | Bases 3.7 - | Standby Gas Treatment System and Secondary Containment (Replace current page) | 181 |
| 207 | TS 3.10.C - | Fuel Storage Pool Water Level | - |
| 208 | | (Insert new pages) | - |
| 209 | Bases 3.10 - | Fuel Storage Pool Water Level | - |
| 209a | | (Insert new pages) | - |

13 pages follow

F. Scram Discharge Volume

1. During reactor operation, the scram discharge volume vent and drain valves shall be operable, except as specified below.

2. If any scam discharge volume vent or drain valve is made or found inoperable, the integrity of the scram discharge volume shall be maintained by either:

   a. Verifying daily, for a period not to exceed 7 days, the operability of the redundant valve(s), or

   b. Maintaining the inoperable valve(s), or the associated redundant valve(s), in the closed position. Periodically the inoperable and the redundant valve(s) may both be in the open position to allow draining the scram discharge volume.

   If a or b above cannot be met, at least all but one operable control rods (not including rods removed per specification 3.10.E or inoperable rods allowed by 3.3.A.2) shall be fully inserted within ten hours.

G. Required Action

[1.]If Specifications 3.3.A **[(except when the reactor mode switch is in the Refuel position)]** through 3.3.D above are not met, an orderly shutdown shall be initiated and the reactor placed in the cold shutdown condition within 24 hours.

**2. If Specification 3.3.A is not met when the reactor mode switch is in the Refuel position, immediately suspend core alterations except for fuel assembly removal and immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.**

F. Scram Discharge Volume

The scram discharge volume vent and drain valves shall be cycled quarterly.

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

Deviations beyond this magnitude would not be expected and would require thorough evaluations. One per cent 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 systems.

As was noted above reactivity anomalies can be found by comparison of the actual control rod inventory to the predicted inventory at a selected base condition. For example, the predicted control rod inventory at 100% power at a specified point in time can be compared to the actual control rod inventory at 100% power and at the specified time to determine if a reactivity anomaly exists. The Monticello Plant has been designed to increase or decrease power level as the system load demand changes. For this type of plant an equilibrium condition of the variables important to making a control rod inventory prediction, specifically the reactivity effects of the xenon, is rarely achieved. The uncertainties of calculating the control rod inventory with non-equilibrium xenon conditions can result in errors which can be misconstrued as reactivity anomalies. Therefore, this specification calls for performing of rod inventory comparisons at a time when xenon will not be a source of error.

F. The safety function of the scram discharge volume vent and drain valves is to limit the loss of reactor coolant leaked past the CRD seals while the scram valves are open. To accomplish this, the vent and drain valves must either be in the closed position or close in a timely manner upon scram initiation. The closure time of 30 seconds is based on a letter dated July 25, 1980 to J G Keppler (Region III) from D E Gilberts (NSP) concerning IE Bulletin No. 80-14. Redundant isolation valves have been provided for each vent and drain line. Closure of one of the valves in each line would be sufficient to maintain the integrity of the scram discharge volume.

G. Whenever a specification (or specifications) can not be met for a particular mode of operation, the reactor would be placed in a mode for which the specification (or specifications) are not required. This requires immediate initiation of a reactor shutdown upon discovery that specifications 3.3.A **[(except when reactor mode switch is in the Refuel position)]** through 3.3.D are not met. **[If specification 3.3.A cannot be met when the reactor mode switch is in the Refuel position, core alterations, except for fuel assembly removal, will be suspended and all insertable control rods will be inserted in core cells containing fuel assemblies.**

c. Whenever primary containment oxygen concentration is equal to or exceeds 4% by volume, except as permitted by 3.7.A.5.b above, within the subsequent 24 hour period return the oxygen concentration to less than 4% by volume.

d. If the requirements of 3.7.A.5 cannot be met, reduce Thermal Power to ≤15% Rated Thermal Power, within 8 hours.

B. Standby Gas Treatment System

1. Two separate and independent standby gas treatment system **[trains]** ~~circuits~~ shall be operable at all times when secondary containment integrity is required, except as specified in sections 3.7.B.1.(a) through (d).

   a. After one of the standby gas treatment system **[trains]** ~~circuits~~ is made or found to be inoperable ***[with reactor water temperature ≥ 212°F,]*** *for* any reason, reactor operation ~~and fuel handling~~ is permissible only during the succeeding seven days, provided that all active components in the other standby gas treatment system are operable. Within 36 hours following the 7 days, the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.2.(a) ~~through (d)~~ **[and b]**.

B. Standby Gas Treatment System

1. Once per month, operate each train of the standby gas treatment system for ≥10 continuous hours with the inline heaters operating.

b. If both standby gas treatment system ~~circuits~~ **[trains]** are not operable, **[with reactor water temperature ≥212°F]** within 36 hours the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.2.(a) **[and (b)]**.

**[c. With one standby gas treatment system train inoperable,**

**1.) The following activities may continue for up to 7days:**

**(a) Movement of recently irradiated fuel assemblies in secondary containment;**

**(b) Movement of the fuel cask in the reactor building; and**

**(c) Operations with the potential to drain the reactor vessel.**

**2.) After 7 days:**

**(a) Immediately suspend movement of the fuel cask in the reactor building; and**

**(b) Immediately place the operable standby gas treatment system train in operation, or**

**(1.) Immediately suspend movement of recently irradiated fuel assemblies in secondary containment; and**

**(2.) Immediately suspend operations with the potential to drain the reactor vessel.**

**d. With both standby gas treatment trains inoperable immediately suspend: Movement of recently irradiated fuel assemblies in secondary containment; Movement of the fuel cask in the reactor building; Operations with the potential for draining the reactor vessel.]**

2. Performance Requirements

   a. Periodic Requirements

      (1) The results of the in-place DOP tests at 3500 cfm (±10%) on HEPA filters shall show ≤1% DOP penetration.

      (2) The results of in-place halogenated hydrocarbon tests at 3500 cfm (±10%) on charcoal banks shall show ≤1% penetration.

      (3) The results of laboratory carbon sample analysis shall show ≤5% methyl iodine penetration when tested in accordance with ASTM D3803-1989 at 30°C, 95% relative humidity.

2. Performance Requirement Tests

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

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

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

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

C. Secondary Containment

1. Except as specified in 3.7.C.2 and 3.7.C.3, Secondary Containment Integrity shall be maintained during all modes of plant operation.

2. Secondary Containment Integrity is not required when all of the following conditions are satisfied:

   a. The reactor is subcritical and Specification 3.3.A is met.

   b. The reactor water temperature is below 212°F.

   ~~c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.~~

   [c.]~~d.~~ The fuel cask ~~or irradiated fuel~~ is not being moved within the reactor building.

   **[d. Recently irradiated fuel is not being moved within secondary containment.**

   **e. Operations with the potential for draining the reactor vessel are not being performed.]**

3. With an inoperable secondary containment isolation damper, restore the inoperable damper to operable status or isolate the affected duct by use of a closed damper or blind flange within eight hours.



C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

   a. Secondary containment capability to maintain at least a 1/4 inch of water vacuum under calm wind (u <5 mph) conditions with a filter train flow rate of ≤4,000 scfm, shall be demonstrated at each refueling outage prior to refueling. If calm wind conditions do not exist during this testing, the test data is to be corrected to calm wind conditions.

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

      (1) Each refueling outage.

      (2) After maintenance, repair or replacement work is performed on the damper or its associated actuator, control circuit, or power circuit.

4. **[a. During Run, Startup, or Hot Shutdown i]**f Specifications 3.7.C.1 through 3.7.C.3 cannot be met, initiate a normal orderly shutdown and have the reactor in the Cold Shutdown condition within 36 hours. ~~Alterations of the reactor core, operations with a potential for reducing the shutdown margin below that specified in spefification 3.3.A, and handling of irradiated fuel or the fuel cask in the secondary containment are to be immediately suspended if secondary containment integrity is not maintained.~~

**[And**

**b. If Specifications 3.7.C.1 through 3.7.C.3 cannot be met, immediately suspend:**

**1. Operations with a potential for draining the reactor vessel.]**

**[2.** ~~h~~ Handling of **[recently]** irradiated fuel ~~or the fuel cask~~ in the secondary containment**[.]** ~~are to be immediately suspended if secondary containment integrity is not maintained.~~

**3. Movement of a fuel cask in the reactor building.]**

D. Primary Containment Isolation Valves (PCIVs)

1. During reactor power operating conditions, all Primary Containment automatic isolation valves and all primary system instrument line flow check valves shall be operable except as specified in 3.7.D.2.

D. Primary Containment Isolation Valves (PCIVs)

1. The primary containment automatic isolation valve surveillance shall be performed as follows:

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

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

c. All normally open power-operated isolation valves shall be tested in accordance with the Inservice Testing Program. Main Steam isolation valves shall be tested (one at a time) with the reactor power less than 75% of rated.

d. At least once per week the main steam-line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

generate significant amounts of hydrogen occurring during this period. The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least once a week the oxygen concentration will be determined as added assurance.

B. Standby Gas Treatment System and C. Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required except, however, ~~for initial fuel loading prior to initial power testing~~ **[as specified in specification 3.7.C.2.]**.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. One standby gas treatment system ~~circuit~~ **[train]** is designed to automatically start upon containment isolation and to maintain the reactor building pressure at the design negative pressure so that all leakage should be in-leakage. Should one ~~circuit~~ **[train]** fail to start, the redundant alternate standby gas treatment ~~circuit~~ **[train]** is designed to start automatically. Each of the two ~~circuits~~ **[trains]** has 100% capacity. Only one of the two standby gas treatment system ~~circuits~~ **[trains]** is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance. Therefore, reactor operation or refueling operation may continue while repairs are being made. If neither ~~circuit~~ **[train]** is operable, the plant is placed in a condition that does not require a standby gas treatment system.

**[During Run, Startup or Hot Shutdown, a design basis accident could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment and standby gas treatment system operability is required under these conditions.**

**During Cold Shutdown and Refuel, the probability and consequences of events are low due to pressure and temperature limitations in these conditions. Therefore, secondary containment and standby gas treatment system operability is not required during Cold Shutdown or Refuel, except for situations for which significant releases of radioactive material can be postulated, such as during operations with the potential for draining the reactor vessel, *during movement of the fuel cask in* the reactor building, or during movement of recently irradiated fuel assemblies in secondary containment.]**

## 3.0 LIMITING CONDITIONS FOR OPERATION

B. Core Monitoring

During core alterations two SRM's shall be operable, one in and one adjacent to any core quadrant where fuel or control rods are being moved. For an SRM to be considered operable, the following conditions shall be satisfied:

1. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations is permissible as long as the detector is connected into the normal SRM circuit.)
2. The SRM shall have a minimum of 3 CPS with all rods fully inserted in the core except when both of the following conditions are fulfilled:
   a. No more than two fuel assemblies are present in the core quadrant associated with the SRM,
   b. While in core, these fuel assemblies are in locations adjacent to the SRM.

C. **[Spent]** Fuel Storage Pool Water Level

~~Whenever~~ **[During movement of]** irradiated fuel **[assemblies]** ~~is stored in the fuel storage pool~~, the **[spent fuel storage]** pool water level shall be maintained ~~at a level of greater or equal to 33 feet~~ **[≥ 37 ft above the bottom of the spent fuel storage pool.]**

**[If the spent fuel storage pool water level is made or found not to be within limits, immediately suspend movement of irradiated fuel assemblies.]**

## 4.0 SURVEILLANCE REQUIREMENTS

B. Core Monitoring

Prior to making any alterations to the core while more than two fuel assemblies are present in any reactor quadrant, the SRM's shall be functionally tested and checked for neutron response. Thereafter, the SRM's will be checked daily for response.

C. **[Spent]** Fuel Storage Pool Water Level

~~Whenever irradiated fuel is stored in the fuel storage pool the pool level shall be recorded daily.~~ **[Verify that the spent fuel storage pool water level is ≥ 37 ft above the bottom of the spent fuel storage pool:**

**a. Once every 24 hrs, during movement of irradiated fuel assemblies, or**

**b. Once every 7 days, when irradiated fuel assemblies are stored in the spent fuel storage pool.]**

D. The reactor shall be shutdown for a minimum of 24 hours prior to movement of fuel within the reactor.

E. Extended Core and Control Rod Drive Maintenance
More than one control rod may be withdrawn from the reactor core during outages provided that, except for momentary switching to the Startup mode for interlock testing, the reactor mode switch shall be locked in the Refuel position. The refueling interlock signal from a control rod may by bypassed after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core.

Bases 3.10/4.10:

A. Refueling Interlocks

During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality. The core reactivity limitation of Specification 3.3 limits the core alterations to assure that the resulting core loading can be controlled with the reactivity control system and interlocks at any time during shutdown or the following operating cycle.

Addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn.

For a new core the dropping of a fuel assembly into a vacant fuel location adjacent to a withdrawn control rod does not result in an excursion or a critical configuration, thus adequate margin is provided.

B. Core Monitoring

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's, one in and one adjacent to any core quadrant where fuel or control rods are being moved, assures adequate monitoring of that quadrant during such alterations. Requiring a minimum of 3 counts per second whenever criticality is possible provides assurance that neutron flux is being monitored. Criticality is considered to be impossible if there are no more than two assemblies in each quadrant and if these are in locations adjacent to the SRM. If it is not possible to achieve criticality, the SRM or dunking type detector count rate is permitted to be less than 3 counts per second and these detectors need not be demonstrated to be operable.

~~C. Fuel Storage Pool Water Level~~ **[Insert for C on page 209a]**

~~To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 33 feet is established because it would be a significant change from the normal level (37293) and well above a level to assure adequate cooling.~~

Bases 3.10/4.10 (Continued):

C. **[Spent]** Fuel Storage Pool Water Level

**[The establishment of a spent fuel storage pool minimum water level]** assures that there is adequate water to shield and cool the irradiated fuel assemblies stored in the **[spent fuel storage]** pool.

**[The spent fuel storage pool minimum water level assures that there is sufficient water in the spent fuel storage pool to meet the assumptions of iodine decontamination factors following a fuel handling accident. The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core. The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.**

**The specified water level preserves the assumptions of the fuel handling accident analysis. As such, it is the minimum required for fuel movement within the spent fuel storage pool. This minimum water level applies during movement of irradiated fuel assemblies in the spent fuel storage pool since the potential for a release of fission products exists.**

**The spent fuel storage pool water level requirement is not applicable during the transport of irradiated fuel within a spent fuel cask. In this case, the cask will be lifted and transported to and from the spent fuel storage pool with the single failure proof reactor building crane in a manner controlled by the plant heavy loads program. By using a single failure proof crane system, and complying with the MNGP heavy loads program, the potential for a cask drop is extremely small and need not be postulated. Therefore, maintaining the minimum water level of 37 ft in the spent fuel storage pool during cask handling activities is not required.]**

D. Minimum Shutdown Period

A minimum shutdown period of 24 hours is specified prior to movement of fuel within the reactor since analysis of refueling accidents assume a 24-hour decay time following extended operation at power. Since the reactor must be shut down, depressurized, and the head removed prior to moving fuel, it is not expected that fuel could actually be moved in less than 24 hours.

## ENCLOSURE 3

## PROPOSED TECHNICAL SPECIFICATION AND BASES CHANGES (RETYPED)

This enclosure consists of revised Technical Specification and Technical Specification Bases pages that incorporate the proposed changes. The following pages, included in this exhibit, should be added to, or replace the pages listed in Enclosure 4 to the NMC submittal dated April 29, 2004, as instructed below:

| Insert Pages | | | Remove Pages |
|---|---|---|---|
| 83a<br>83b | TS 3.3.G - | Control Rod System<br>(Insert new pages) | -<br>- |
| 92 | Bases 3.3 - | (Insert new page) | - |
| 166<br>167<br>167a | TS 3.7.B - | Standby Gas Treatment System<br>(Replace current pages) | 166<br>167<br>167a |
| 169<br>170 | TS 3.7.C - | Secondary Containment<br>(Replace current pages) | 169<br>170 |
| 181 | Bases 3.7 - | Standby Gas Treatment System and Secondary Containment<br>(Replace current page) | 181 |
| 207<br>208 | TS 3.10.C - | Fuel Storage Pool Water Level<br>(Insert new pages) | -<br>- |
| 209<br>209a | Bases 3.10 - | Fuel Storage Pool Water Level<br>(Insert new pages) | -<br>- |

13 pages follow

| 3.0 LIMITING CONDITIONS FOR OPERATION | 4.0 SURVEILLANCE REQUIREMENTS |
|---|---|
| F. Scram Discharge Volume<br><br>1. During reactor operation, the scram discharge volume vent and drain valves shall be operable, except as specified below.<br><br>2. If any scam discharge volume vent or drain valve is made or found inoperable, the integrity of the scram discharge volume shall be maintained by either:<br><br>a. Verifying daily, for a period not to exceed 7 days, the operability of the redundant valve(s), or<br><br>b. Maintaining the inoperable valve(s), or the associated redundant valve(s), in the closed position. Periodically the inoperable and the redundant valve(s) may both be in the open position to allow draining the scram discharge volume.<br><br>If a or b above cannot be met, at least all but one operable control rods (not including rods removed per specification 3.10.E or inoperable rods allowed by 3.3.A.2) shall be fully inserted within ten hours.<br><br>G. Required Action<br><br>1. If Specifications 3.3.A (except when the reactor mode switch is in the Refuel position) through 3.3.D above are not met, an orderly shutdown shall be initiated and the reactor placed in the cold shutdown condition within 24 hours. | F. Scram Discharge Volume<br><br>The scram discharge volume vent and drain valves shall be cycled quarterly.<br><br>Once per operating cycle verify the scram discharge volume vent and drain valves close within 30 seconds after receipt of a reactor scram signal and open when the scram is reset. |

| 3.0 LIMITING CONDITIONS FOR OPERATION | 4.0 SURVEILLANCE REQUIREMENTS |
|---|---|
| 2. If Specification 3.3.A is not met when the reactor mode switch is in the Refuel position, immediately suspend core alterations except for fuel assembly removal and immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. | |

Bases 3.3/4.3 (Continued):

Deviations beyond this magnitude would not be expected and would require thorough evaluations. One per cent 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 systems.

As was noted above reactivity anomalies can be found by comparison of the actual control rod inventory to the predicted inventory at a selected base condition. For example, the predicted control rod inventory at 100% power at a specified point in time can be compared to the actual control rod inventory at 100% power and at the specified time to determine if a reactivity anomaly exists. The Monticello Plant has been designed to increase or decrease power level as the system load demand changes. For this type of plant an equilibrium condition of the variables important to making a control rod inventory prediction, specifically the reactivity effects of the xenon, is rarely achieved. The uncertainties of calculating the control rod inventory with non-equilibrium xenon conditions can result in errors which can be misconstrued as reactivity anomalies. Therefore, this specification calls for performing of rod inventory comparisons at a time when xenon will not be a source of error.

F. The safety function of the scram discharge volume vent and drain valves is to limit the loss of reactor coolant leaked past the CRD seals while the scram valves are open. To accomplish this, the vent and drain valves must either be in the closed position or close in a timely manner upon scram initiation. The closure time of 30 seconds is based on a letter dated July 25, 1980 to J G Keppler (Region III) from D E Gilberts (NSP) concerning IE Bulletin No. 80-14. Redundant isolation valves have been provided for each vent and drain line. Closure of one of the valves in each line would be sufficient to maintain the integrity of the scram discharge volume.

G. Whenever a specification (or specifications) can not be met for a particular mode of operation, the reactor would be placed in a mode for which the specification (or specifications) are not required. This requires immediate initiation of a reactor shutdown upon discovery that specifications 3.3.A (except when reactor mode switch is in the Refuel position) through 3.3.D are not met. If specification 3.3.A cannot be met when the reactor mode switch is in the Refuel position core alterations, except for fuel assembly removal, will be suspended and all insertable control rods will be inserted in core cells containing fuel assemblies.

| 3.0 LIMITING CONDITIONS FOR OPERATION | 4.0 SURVEILLANCE REQUIREMENTS |
|---|---|

c. Whenever primary containment oxygen concentration is equal to or exceeds 4% by volume, except as permitted by 3.7.A.5.b above, within the subsequent 24 hour period return the oxygen concentration to less than 4% by volume.

d. If the requirements of 3.7.A.5 cannot be met, reduce Thermal Power to ≤ 15% Rated Thermal Power, within 8 hours.

B. Standby Gas Treatment System

1. Two separate and independent standby gas treatment system trains shall be operable at all times when secondary containment integrity is required, except as specified in sections 3.7.B.1.(a) through (d).

a. After one of the standby gas treatment system trains is made or found to be inoperable with reactor water temperature ≥ 212°F, for any reason, reactor operation is permissible only during the succeeding seven days, provided that all active components in the other standby gas treatment system are operable. Within 36 hours following the 7 days, the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.2.(a) and (b).

B. Standby Gas Treatment System

1. Once per month, operate each train of the standby gas treatment system for ≥ 10 continuous hours with the inline heaters operating.

| 3.0 LIMITING CONDITIONS FOR OPERATION | 4.0 SURVEILLANCE REQUIREMENTS |
|---|---|
| b. If both standby gas treatment system trains are not operable, with reactor water temperature ≥ 212°F, within 36 hours the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.2.(a) and (b).<br><br>c. With one standby gas treatment system train inoperable,<br><br>1) The following activities may continue for up to 7 days:<br><br>(a) Movement of recently irradiated fuel assemblies in secondary containment;<br><br>(b) Movement of the fuel cask in the reactor building; and<br><br>(c) Operations with the potential to drain the reactor vessel.<br><br>2) After 7 days:<br><br>(a) Immediately suspend movement of the fuel cask in the reactor building; and<br><br>(b) Immediately place the operable standby gas treatment system train in operation, or | |

| 3.0 LIMITING CONDITIONS FOR OPERATION | 4.0 SURVEILLANCE REQUIREMENTS |
|---|---|

**3.0 LIMITING CONDITIONS FOR OPERATION**

(1) Immediately suspend movement of recently irradiated fuel assemblies in secondary containment; and

(2) Immediately suspend operations with the potential to drain the reactor vessel.

d. With both standby gas treatment trains inoperable immediately suspend: Movement of recently irradiated fuel assemblies in secondary containment; Movement of the fuel cask in the reactor building; Operations with the potential for draining the reactor vessel.

2. Performance Requirements

a. Periodic Requirements

(1) The results of the in-place DOP tests at 3500 cfm (±10%) on HEPA filters shall show ≤1% DOP penetration.

(2) The results of in-place halogenated hydrocarbon tests at 3500 cfm (±10%) on charcoal banks shall show ≤1% penetration.

(3) The results of laboratory carbon sample analysis shall show ≤5% methyl iodine penetration when tested in accordance with ASTM D3803-1989 at 30°C, 95% relative humidity.

**4.0 SURVEILLANCE REQUIREMENTS**

2. Performance Requirement Tests

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

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

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

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

| 3.0 LIMITING CONDITIONS FOR OPERATION | 4.0 SURVEILLANCE REQUIREMENTS |
|---|---|

**3.0 LIMITING CONDITIONS FOR OPERATION**

C. Secondary Containment

1. Except as specified in 3.7.C.2 and 3.7.C.3, Secondary Containment Integrity shall be maintained during all modes of plant operation.

2. Secondary Containment Integrity is not required when all of the following conditions are satisfied:

    a. The reactor is subcritical and Specification 3.3.A is met.

    b. The reactor water temperature is below 212°F.

    c. The fuel cask is not being moved within the reactor building.

    d. Recently irradiated fuel is not being moved within secondary containment.

    e. Operations with the potential for draining the reactor vessel are not being performed.

3. With an inoperable secondary containment isolation damper, restore the inoperable damper to operable status or isolate the affected duct by use of a closed damper or blind flange within eight hours.

**4.0 SURVEILLANCE REQUIREMENTS**

C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

    a. Secondary containment capability to maintain at least a 1/4 inch of water vacuum under calm wind ($u < 5$ mph) conditions with a filter train flow rate of ≤ 4,000 scfm, shall be demonstrated at each refueling outage prior to refueling. If calm wind conditions do not exist during this testing, the test data is to be corrected to calm wind conditions.

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

        (1) Each refueling outage.

        (2) After maintenance, repair or replacement work is performed on the damper or its associated actuator, control circuit, or power circuit.

| 3.0 LIMITING CONDITIONS FOR OPERATION | 4.0 SURVEILLANCE REQUIREMENTS |
|---|---|

**3.0 LIMITING CONDITIONS FOR OPERATION**

4. a. During Run, Startup, or Hot Shutdown if Specifications 3.7.C.1 through 3.7.C.3 cannot be met, initiate a normal orderly shutdown and have the reactor in the Cold Shutdown condition within 36 hours.

   And

   b. If Specifications 3.7.C.1 through 3.7.C.3 cannot be met, immediately suspend:

      1. Operations with a potential for draining the reactor vessel.

      2. Handling of recently irradiated fuel in the secondary containment.

      3. Movement of a fuel cask in the reactor building.

D. Primary Containment Isolation Valves (PCIVs)

   1. During reactor power operating conditions, all Primary Containment automatic isolation valves and all primary system instrument line flow check valves shall be operable except as specified in 3.7.D.2.

**4.0 SURVEILLANCE REQUIREMENTS**

D. Primary Containment Isolation Valves (PCIVs)

   1. The primary containment automatic isolation valve surveillance shall be performed as follows:

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

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

      c. All normally open power-operated isolation valves shall be tested in accordance with the Inservice Testing Program. Main Steam isolation valves shall be tested (one at a time) with the reactor power less than 75% of rated.

      d. At least once per week the main steam-line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

generate significant amounts of hydrogen occurring during this period. The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least once a week the oxygen concentration will be determined as added assurance.

B. Standby Gas Treatment System and C. Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required except, however, as specified in specification 3.7.C.2.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. One standby gas treatment system train is designed to automatically start upon containment isolation and to maintain the reactor building pressure at the design negative pressure so that all leakage should be in-leakage. Should one train fail to start, the redundant alternate standby gas treatment circuit is designed to start automatically. Each of the two trains has 100% capacity. Only one of the two standby gas treatment system trains is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance. Therefore, reactor operation or refueling operation may continue while repairs are being made. If neither train is operable, the plant is placed in a condition that does not require a standby gas treatment system.

During Run, Startup or Hot Shutdown, a design basis accident could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment and standby gas treatment system operability is required under these conditions.

During Cold Shutdown and Refuel, the probability and consequences of events are low due to pressure and temperature limitations in these conditions. Therefore, secondary containment and the standby gas treatment system operability is not required during Cold Shutdown or Refuel, except for situations for which significant releases of radioactive material can be postulated, such as during operations with the potential for draining the reactor vessel, during movement of a fuel cask in the reactor building, or during movement of recently irradiated fuel assemblies in secondary containment.

## 3.0 LIMITING CONDITIONS FOR OPERATION

B. Core Monitoring

During core alterations two SRM's shall be operable, one in and one adjacent to any core quadrant where fuel or control rods are being moved. For an SRM to be considered operable, the following conditions shall be satisfied:

1. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations is permissible as long as the detector is connected into the normal SRM circuit.)

2. The SRM shall have a minimum of 3 CPS with all rods fully inserted in the core except when both of the following conditions are fulfilled:

   a. No more than two fuel assemblies are present in the core quadrant associated with the SRM,

   b. While in core, these fuel assemblies are in locations adjacent to the SRM.

C. Spent Fuel Storage Pool Water Level

During movement of irradiated fuel assemblies, the spent fuel storage pool water level shall be maintained ≥37 feet above the bottom of the spent fuel storage pool.

If the spent fuel storage pool water level is made or found not to be within limits, immediately suspend movement of irradiated fuel assemblies.

## 4.0 SURVEILLANCE REQUIREMENTS

B. Core Monitoring

Prior to making any alterations to the core while more than two fuel assemblies are present in any reactor quadrant, the SRM's shall be functionally tested and checked for neutron response. Thereafter, the SRM's will be checked daily for response.

C. Spent Fuel Storage Pool Water Level

Verify that the spent fuel storage pool water level is ≥37 feet above the bottom of the spent fuel storage pool:

1. Once every 24 hours, during movement of irradiated fuel assemblies, or

2. Once every 7 days, when irradiated fuel assemblies are stored in the spent fuel storage pool.

| 3.0 LIMITING CONDITIONS FOR OPERATION | 4.0 SURVEILLANCE REQUIREMENTS |
|---|---|
| D. The reactor shall be shutdown for a minimum of 24 hours prior to movement of fuel within the reactor.<br><br>E. Extended Core and Control Rod Drive Maintenance<br><br>More than one control rod may be withdrawn from the reactor core during outages provided that, except for momentary switching to the Startup mode for interlock testing, the reactor mode switch shall be locked in the Refuel position. The refueling interlock signal from a control rod may by bypassed after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. | |

Bases 3.10/4.10:

A. Refueling Interlocks

During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality. The core reactivity limitation of Specification 3.3 limits the core alterations to assure that the resulting core loading can be controlled with the reactivity control system and interlocks at any time during shutdown or the following operating cycle.

Addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn.

For a new core the dropping of a fuel assembly into a vacant fuel location adjacent to a withdrawn control rod does not result in an excursion or a critical configuration, thus adequate margin is provided.

B. Core Monitoring

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's, one in and one adjacent to any core quadrant where fuel or control rods are being moved, assures adequate monitoring of that quadrant during such alterations. Requiring a minimum of 3 counts per second whenever criticality is possible provides assurance that neutron flux is being monitored. Criticality is considered to be impossible if there are no more than two assemblies in each quadrant and if these are in locations adjacent to the SRM. If it is not possible to achieve criticality, the SRM or dunking type detector count rate is permitted to be less than 3 counts per second and these detectors need not be demonstrated to be operable.

C. Spent Fuel Storage Pool Water Level

The establishment of a spent fuel storage pool minimum water level assures that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool.

The spent fuel storage pool minimum water level assures that there is sufficient water in the spent fuel storage pool to meet the assumptions of iodine decontamination factors following a fuel handling accident. The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core. The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The specified water level preserves the assumptions of the fuel handling accident analysis. As such, it is the minimum required for fuel movement within the spent fuel storage pool. This minimum water level applies during movement of irradiated fuel assemblies in the spent fuel storage pool since the potential for a release of fission products exists.

The spent fuel storage pool water level requirement is not applicable during the transport of irradiated fuel within a spent fuel cask. In this case, the cask will be lifted and transported to and from the spent fuel storage pool with the single failure proof reactor building crane in a manner controlled by the plant heavy loads program. By using a single failure proof crane system, and complying with the MNGP heavy loads program, the potential for a cask drop is extremely small and need not be postulated. Therefore, maintaining the minimum water level of 37 feet in the spent fuel storage pool during cask handling activities is not required.

D. Minimum Shutdown Period

A minimum shutdown period of 24 hours is specified prior to movement of fuel within the reactor since analysis of refueling accidents assume a 24-hour decay time following extended operation at power. Since the reactor must be shut down, depressurized, and the head removed prior to moving fuel, it is not expected that fuel could actually be moved in less than 24 hours.