NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

PROPRIETARY INFORMATION ENCLOSED

Pursuant to 10 CFR 2.390, withhold the proprietary compact disc for the calculations listed in the table behind the cover letter from public disclosure. Upon separation of the proprietary CD, the remainder of this letter may be decontrolled.

April 29, 2004

L-MT-04-023
10 CFR 50.90
10 CFR 50.67

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

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

License Amendment Request: Selective Scope Application of an Alternative Source Term Methodology for Re-evaluation of the Fuel Handling Accident

Pursuant to 10 CFR 50.67 and 10 CFR 50.90, the Nuclear Management Company, LLC (NMC) proposes to amend the Monticello Nuclear Generating Plant licensing basis and Technical Specifications (TS) based upon the radiological consequences of a revised fuel handling accident.

This license amendment request (LAR) proposes a selective scope application of an alternative source term (AST) for the fuel handling accident (FHA) in accordance with the provisions of 10 CFR 50.67, "Accident Source Term." NMC requests U.S. Nuclear Regulatory Commission (NRC) review and approval of an AST FHA methodology for application to the Monticello Nuclear Generating Plant (MNGP). Revisions to the TS associated with ensuring that the revised safety analyses assumptions are met for a postulated FHA in containment are proposed. The guidance of TS Task Force Traveler TSTF-51, "Revise containment requirements during handling irradiated fuel and core alterations," was utilized in the development of these proposed TS changes. Consistent with TSTF-51, NMC is implementing the guidelines of the Reviewer's Note for the assessment of systems removed from service during movement of irradiated fuel at MNGP. Draft TS Bases changes are also provided supporting the proposed TS changes.

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Enclosure 1 provides a description of the proposed changes, background, technical and regulatory safety analyses, and no significant hazards and environmental considerations. Enclosure 2 provides a comparison of the MNGP FHA analysis assumptions versus Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Enclosure 3 provides a mark-up of the proposed changes to the MNGP TS and Bases. Enclosure 4 provides a retyped version of the proposed TS and Bases changes.

A table, included behind this cover letter, lists the calculations and meteorological information used in support of the AST FHA analysis, provided in the compact disc (CD) format. Two CDs, one proprietary and the other non-proprietary, are included as enclosures to this submittal. These CDs contain copies of the calculations and meteorological data, as well as input data files for the ARCON96, PAVAN and RADTRAD computer codes to facilitate NRC review. Non-proprietary MNGP calculations/information and proprietary methodology/calculations utilized by Applied Analysis Corporation in the AST FHA analysis are identified. Enclosure 5 provides an affidavit for the proprietary AST FHA calculations performed by Applied Analysis Corporation (AAC). Pursuant to 10 CFR 2.390, it is requested that the AAC calculations contained on the proprietary CD be withheld from public disclosure. Upon separation of the proprietary CD from this letter, the remainder of this letter and the non-proprietary CD may be decontrolled.

NMC proposes the following commitments:

1. NMC will revise the guidelines for assessing MNGP systems removed from service during handling of irradiated fuel assemblies or core alterations to implement the provisions of Section 11.3.6.5 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3.

2. 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.

The MNGP Operations Committee has reviewed this application. A copy of this submittal, including the No Significant Hazards Consideration determination, is being forwarded to our appointed state official pursuant to 10 CFR 50.91(b)(1).

NMC requests approval of this proposed amendment by February 5, 2005. A sixty (60) day implementation period is requested following issuance.

If you have any questions regarding this submittal, please contact Rick Loeffler, Senior Regulatory Affairs Engineer, at 763-295-1247.

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

Executed on April 29, 2004

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

Enclosures 5 (and two enclosed CDs)

cc: Administrator, Region III, USNRC (w/o proprietary CD)
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC (w/o proprietary CD)
Minnesota Department of Commerce (w/o proprietary CD)

Table Listing Compact Discs and Proprietary Non-proprietary Calculations and Information

	MNGP Calculation No. (AAC Calculation No.)	Calculation Title / Description	
1.	CA-03-194	MNGP Calculation CA-03-194, Alternate Source Term Core Isotopic Inventory, Rev 0 (Main body without attachments.)	Non-proprietary CD
2.	CA-03-190	MNGP Calculation CA-03-190, Design Inputs for Alternate Source Term (AST) Radiological Analysis, Rev 0 (Design input sections relevant to FHA analysis, including meteorological data files.)	Non-proprietary CD
3.	CA-04-036	MNGP Calculation CA-04-036, MNGP AST - Offsite Post-Accident Atmospheric Dispersion Analysis, Rev 0 (Part 1)	Non-proprietary CD
4.	CA-04-036 (MNGP-001)	MNGP Calculation CA-04-036, MNGP AST - Offsite Post-Accident Atmospheric Dispersion Analysis, Rev 0 (Part 2) Proprietary AAC Calculation MNGP-001 identified in AAC affidavit provided in Enclosure 5.	Proprietary CD
5.	CA-04-037	MNGP Calculation CA-04-037, MNGP AST - CR/TSC Post-Accident Atmospheric Dispersion Analysis, Rev 0 (Part 1)	Non-proprietary CD
6.	CA-04-037 (MNGP-002)	MNGP Calculation CA-04-037, MNGP AST - CR/TSC Post-Accident Atmospheric Dispersion Analysis, Rev 0 (Part 2) Proprietary AAC Calculation MNGP-002 identified in AAC affidavit provided in Enclosure 5.	Proprietary CD
7.	CA-04-041	MNGP Calculation CA-04-041, MNGP AST - FHA Radiological Consequence Analysis, Rev 0 (Part 1)	Non-proprietary CD
8.	CA-04-041 (MNGP-006)	MNGP Calculation CA-04-041, MNGP AST - FHA Radiological Consequence Analysis, Rev 0 (Part 2) Proprietary AAC Calculation MNGP-006 identified in AAC affidavit provided in Enclosure 5.	Proprietary CD

MASTER INDEX

ENCLOSURE 1 – License Amendment Request

MASTER INDEX

1.0 SUMMARY

Pursuant to 10 CFR 50.67 and 10 CFR 50.90, the Nuclear Management Company, LLC (NMC) proposes to amend the Monticello Nuclear Generating Plant licensing basis and Technical Specifications (TS) based upon the radiological consequences of a revised fuel handling accident.

A selective scope application of an alternate source term (AST) is proposed for the fuel handling accident (FHA) in accordance with the provisions of 10 CFR 50.67, "Accident Source Term." NMC requests U.S. Nuclear Regulatory Commission (NRC) review and approval of an AST FHA methodology for application to the Monticello Nuclear Generating Plant (MNGP). In accordance with AST FHA analysis results, revisions to the TS and Bases to ensure that the revised safety analysis assumptions are met for a postulated FHA in the Reactor Building (Secondary Containment) are required.

The guidance of the NRC approved TS Task Force Traveler TSTF-51, "Revise containment requirements during handling irradiated fuel and core alterations," (Reference 1) was utilized, in-part, in development of the proposed TS and associated Bases changes. The proposed TS and Bases changes are consistent with this guidance. The NRC has approved similar TS changes for other licensees in conjunction with implementing AST. Draft changes to the TS Bases are provided to aid in review. These TS Bases changes will be processed in accordance with the MNGP TS Bases Control Program following issuance of this amendment.

Secondary Containment integrity and operation of associated Engineered Safety Feature (ESF) Systems is required during handling of 'recently' irradiated fuel, i.e., fuel that has resided in a critical core within the past 24 hours. Following this 24-hour post-shutdown period, Secondary Containment integrity and selected ESF System operability can be relaxed during handling of irradiated fuel, based on the results of the AST FHA analysis.

Current industry guidance, e.g., the Reviewer's Note to TSTF-51, indicates that plants may utilize shutdown safety administrative controls in lieu of TS requirements on Secondary Containment and ventilation system operability provided they meet certain considerations. Shutdown safety administrative controls during fuel handling and core alterations are acceptable provided that ventilation system and radiation monitor availability are assessed and unavailability is minimized consistent with shutdown risk considerations. Shutdown safety administrative controls are to maintain ventilation system filtration and radiation monitor availability to reduce doses even further below that provided by the natural decay and avoid unmonitored releases. NMC commits to the guidelines of the TSTF-51 Reviewer's Note for the assessment of systems removed from service during movement of irradiated fuel at MNGP.

2.0 INTRODUCTION

10 CFR 50.67, "Accident Source Term," (Reference 2) provides a mechanism for currently licensed nuclear power reactors to replace the traditional source term used in design basis accident (DBA) analyses with an alternative source term. Under this regulation, licensees who seek to revise the accident source term utilized in the radiological consequence analyses must apply for a license amendment in accordance with 10 CFR 50.90.

AST methodology was applied in a re-analysis of the DBA FHA described in Section 14.7.6, "Refueling Accident," of the MNGP Updated Safety Analysis Report (USAR). The guidance of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," (Reference 3), and the NRC Standard Review Plan (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," (Reference 4) were applied.

The proposed license amendment modifies the MNGP licensing basis to reflect a selective implementation of an AST methodology limited to the FHA. The current accident source term used in the FHA is replaced with an alternative source term as prescribed in 10 CFR 50.67 and by the establishment of total effective dose equivalent (TEDE) limits as the new radiological acceptance criteria. Analysis results demonstrate that using AST methodology, the post-accident Control Room and offsite doses associated with an FHA remain within regulatory acceptance limits.

RG 1.183 recommends that either changes to the USAR reflecting the revised analyses or calculations be submitted to the NRC staff as part of the license amendment review. Copies of the applicable calculations are included in lieu of providing the revised USAR pages. Following issuance of a license amendment, conforming USAR changes will be processed and submitted in accordance with 10 CFR 50.71(e) as part of the regular USAR revision process.

Licensing basis changes for a selective AST implementation for a FHA include:

- Utilize new offsite and Control Room atmospheric dispersion factors (χ/Q's), calculated using current site-specific meteorology data collected between January 1998 and December 2002.
- Development of a new core source term reflecting AST methodologies and assumptions, which was applied to the AST FHA analysis.
- Utilization of the ARCON96 computer code to calculate relative concentrations in plumes in the vicinity of the release point (used for ground level releases).

- Utilization of the PAVAN-PC computer code to calculate elevated releases.
- Utilization of the RADTRAD computer code to calculate radiological consequences of the design basis FHA.
- Secondary Containment integrity is not assumed during the postulated AST FHA. (Offsite and Control Room doses demonstrated to be within regulatory acceptance limits.)
- Operation of the Standby Gas Treatment (SBGT) and Control Room Emergency Filtration (EFT) Systems are not assumed during the postulated AST FHA. (Offsite and Control Room doses demonstrated to be within regulatory acceptance limits.)

Implementation of a revised FHA analysis based on the utilization of an AST allows a relaxation of requirements for the following specifications, while still meeting FHA acceptance criteria with acceptable margins. Changes to the following MNGP TS and associated TS Bases are proposed:

- Table 3.2.4 – Instrumentation That Initiates Reactor Building Ventilation Isolation And Standby Gas Treatment System Initiation
- Specification 3.7.B.1 – Standby Gas Treatment System
- Specification 3.7.C. – Secondary Containment
- Specification 3.17.A. – Control Room Ventilation System
- Specification 3.17.B.1. – Control Room Emergency Filtration System
- Define within the TS Bases the 24-hour time period for 'recently' irradiated fuel based upon AST FHA analysis results.

3.0 BACKGROUND

The power reactor siting regulation, 10 CFR 100 (Reference 5), requires that a fission product release into containment be postulated and that offsite radiological consequences be evaluated against the guideline dose values specified in that regulation. The fission product releases into containment are used for evaluating the acceptability of both the plant site and the effectiveness of engineered safety feature components and systems. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," (Reference 6) references TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," (Reference 7). The U.S. Atomic Energy Commission published the current source term, referred to as TID-14844 in March 1962. Until development of the alternate source term, in the 1990's, TID-14844 was used as the basis for determining DBA analysis source terms for power reactors.

The current radiological consequence analysis for the FHA is based on the TID-14844 accident source term. Since the 1960's, there have been significant advances in the understanding of timing, magnitude, and chemical forms of fission product releases from severe reactor accidents. In 1995, the NRC staff published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," (Reference 8). NUREG-1465 provided estimates of accident source terms that were more physically based and that could be applied to the design of future light-water power reactors. NUREG-1465 presented a representative accident source term for a boiling-water reactor (BWR) and a pressurized-water reactor (PWR). The composition and magnitude of the radioactive material, the chemical and physical properties of the material, and timing of the release to the containment characterize the more recent BWR and PWR source terms.

The NRC staff considered the applicability of the revised source terms in NUREG-1465 to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety. Operating reactors licensed under the TID-14844 source term are not required to re-analyze accidents using the revised source term. The NRC staff also determined that some licensees might wish to use AST analyses to support cost-beneficial licensing actions.

On December 23, 1999, the NRC published a new regulation, 10 CFR 50.67, "Accident Source Term," in the Federal Register. The regulation allows a holder of an operating license issued prior to January 10, 1997 (or a renewed license under 10 CFR 54), to voluntarily revise the current accident source term on a full or selective basis. 10 CFR 50.67(b) states that licensees who seek to revise their current accident source term in design basis radiological consequence analyses must apply for a license amendment in accordance with 10 CFR 50.90. Regulatory guidance for implementation of ASTs is provided in RG 1.183 and SRP Section 15.0.1. As part of the implementation of an AST, the total effective dose equivalent acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR 50, Appendix A, General Design Criterion 19 (Reference 9).

4.0 PROPOSED CHANGES

The MNGP TS impose requirements related to Secondary Containment integrity and associated engineered safety feature system operability during activities when irradiated fuel assemblies are handled or during core alterations. Certain specifications restrict activities during operations with the potential for draining the reactor vessel (OPDRV). These requirements are imposed, in part, to ensure that the radiological consequences of an FHA do not exceed those predicted by the DBA FHA analysis. The proposed TS changes will relax Secondary Containment and ESF system operability requirements (based on the AST FHA results) to require ESF system and Secondary Containment operability only during the handling of 'recently' irradiated fuel. The proposed changes are generally changes to reflect the format and content of the MNGP custom TS. These changes are consistent with NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," (Reference 10) which includes TSTF-51.

Currently, the Secondary Containment, the SBGT System, and the Control Room EFT System and associated actuating instrumentation are assumed operable to mitigate the potential effects of an FHA, i.e., they are assumed to be part of the primary success path for mitigating the FHA.

Using the proposed AST, the FHA analysis results demonstrate that after sufficient radioactive decay of the fuel has occurred the radiological dose consequences (the off-site and Control Room operator doses) due to a release from an FHA are within 10 CFR 50.67 limits. The AST FHA analysis does not assume Secondary Containment. The AST FHA analysis also does not assume operation of the SBGT and Control Room EFT Systems.

The results of the AST FHA analysis indicate that the operability requirements for Secondary Containment and selected ESF systems during core alterations can be eliminated during shutdown conditions based on the results of the analyses. No changes are proposed to requirements associated with operations with the potential for draining the reactor vessel. The associated TS Bases for each affected section will be revised to reflect this change in licensing basis.

4.1 General Discussion of Fuel Handling Accident Considerations

The FHA (or refueling accident) is discussed in Section 14.7 of the MNGP USAR. The design basis scenario is for a fuel assembly to fall onto the top of the reactor core. The results of the AST FHA analysis indicate that the TS requirements pertaining to core alterations and handling of irradiated fuel in the Secondary Containment could be modified based on the recognition that after reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. Fuel that has been 'recently' irradiated contains sufficient fission products to require operability of accident mitigation systems (ESF systems) and requires Secondary Containment integrity to meet AST FHA analysis assumptions. 'Recently' irradiated fuel is classified as fuel that has decayed for less than 24 hours from the time of reactor shut down, i.e., the reactor is subcritical (defined as all rods fully inserted). The 24-hour irradiated fuel decay period is a fundamental assumption of the AST FHA analysis. Following this period, based on the AST FHA analysis results, the TS requirements for operability of certain plant systems, structures, and components (SSCs) may be removed during fuel handling because sufficient radioactive decay has occurred such that these SSCs are no longer necessary to ensure offsite and Control Room dose limits are not exceeded. Implementation of a revised FHA analysis based on the utilization of an AST allows a relaxation of requirements while still meeting FHA acceptance criteria with acceptable margins.

MNGP Specifications 3.17.A, "Control Room Ventilation System," and 3.17.B.1, "Control Room Emergency Filtration System," provide action statements for occurrences happening during operations and for operations involving the movement of irradiated fuel assemblies and during activities with the potential for draining the reactor vessel. Action statements are being added to the SBGT System and Secondary Containment specification paragraphs for operations involving the movement of irradiated fuel assemblies and activities (or operations) with the potential for draining the reactor vessel for consistency with Specifications 3.17.A and 3.17.B.1 and general industry practice.

For the TS that are being revised, the paragraphs that discuss applicability are being modified to be appropriate for operations involving the movement of 'recently' irradiated fuel. 'Recently' irradiated fuel contains sufficient fission products to require the operability of accident mitigation systems (ESF systems) for meeting analysis assumptions. The new terminology is being used to differentiate between conditions where fuel handling activities can involve situations for which significant radioactive releases are not postulated and those for which significant radioactive releases can be postulated. Thus, the new terminology is being used to modify the operability requirements for the identified ESF systems. The AST analyses demonstrate that a 24-hour decay period is sufficient to ensure Secondary Containment isolation, SBGT System and Control Room EFT System automatic initiation/isolation features are not required to mitigate an FHA. The Applicability Statements related to operations (or activities) with a potential for draining the reactor vessel are unaffected by the proposed changes.

Enclosure 3 provides a mark-up of proposed TS and associated Bases changes related to the relaxation of Secondary Containment integrity and operability requirements for certain systems during handling of 'recently' irradiated fuel. Enclosure 4 provides a retyped version of these proposed changes to the MNGP TS and associated TS Bases.

4.1.1 TS Table 3.2.4 – Instrumentation That Initiates Reactor Building Ventilation Isolation And Standby Gas Treatment System Initiation

Revisions to TS Table 3.2.4, "Instrumentation That Initiates Reactor Building Ventilation Isolation And Standby Gas Treatment System Initiation," are proposed to allow the applicable modes or operating conditions for each instrument function to be individually specified. TS Table 3.2.4 is sorted by four sets of instruments that currently initiate the Reactor Building Ventilation and SBGT Systems. The results of the AST FHA analysis indicate that initiation of the SBGT is only required during 'operations with a potential for draining the reactor vessel' and during movement of 'recently' irradiated fuel assemblies in the secondary containment.' The AST FHA analysis assumed a 24-hour decay period prior to beginning irradiated fuel movement. Following the 24-hour decay period AST FHA analysis results indicate that operation of the SBGT System, Control Room EFT System, and Secondary Containment Integrity are not required. To implement these system conditions, the following modifications to Table 3.2.4 are proposed.

(a) Revise Table 3.2.4 by adding a column entitled *"Applicable Modes or Other Specified Conditions for Which the Function Must be Operable or Operating. #"* This new column is inserted between the present *"Trip Settings"* and *"Total No. of Instrument Channels Per Trip System"* columns. This change is to allow the applicable modes or operating conditions for each instrument function to be individually specified. The purpose for addition of the "#" is described in section (b) below.

The addition of the new column simply reflects the applicability of each trip system and states when each function is required to be operable. This change is acceptable because it is being made to make this table consistent with other MNGP TS instrumentation tables and to clarify the system applicability requirements.

(b) At the end of Table 3.2.4, the "#" sign condition is listed under the notes section with notes (a) and (b) immediately following. The proposed notes say:

"Other specified conditions for which the function must be operable or operating:"

(a) "During operations with the potential for draining the reactor vessel."

(b) "During movement of recently irradiated fuel assemblies in secondary containment."

These other specified conditions are consistent with the applicability paragraphs and action statement paragraphs being added to the SBGT System specifications as discussed in later sections.

(c) For the 'Low Low Reactor Water Level' function the 'Applicable Modes or Other Specified Conditions...' under this new column are listed as 'Hot Shutdown, Startup and Run.' This nomenclature is similar to, and consistent with that used in a similar application in Monticello TS Table 3.2.3, "Instrumentation That Initiates Rod Block." Specifying reactor modes/ conditions in this fashion clarifies the applicability of the each instrument function. This allows the specified conditions for when a particular function is required to be listed. Note (a) is added (referred to via the # sign), indicating that the 'Low Low Reactor Water Level' function must be operable 'During operations with the potential for draining the reactor vessel.'

The Low Low Reactor Water Level function is required to be operable in the mode switch positions of Run and Startup (except as specified in existing note 3 to the table) and in the Hot Shutdown condition because these are areas of operation where considerable energy exists in the Reactor Coolant System (RCS); thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This function is not required in Cold Shutdown and Refuel conditions, because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these conditions. In addition, the function is also required to be operable during OPDRVs because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs.

(d) For the 'High Drywell Pressure' function the 'Applicable Modes or Other Specified Conditions...' under this new column are listed as 'Hot Shutdown, Startup and Run.' This nomenclature is similar to, and consistent with that used in a similar application in Monticello TS Table 3.2.3. Specifying reactor modes/conditions in this fashion clarifies the applicability of each instrument function. No additional notes are required.

The High Drywell Pressure function is required to be operable in the mode switch positions of Run and Startup (except as specified in existing note 3 to the table) and in the Hot Shutdown condition because these are areas of operation where considerable energy exists in the RCS; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This function is not required in Cold Shutdown and Refuel conditions, because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these conditions.

(e) For the 'Reactor Building Plenum Radiation Monitors' function the 'Applicable Modes or Other Specified Conditions...' under this new column are listed as 'Hot Shutdown, Startup and Run.' This nomenclature is similar to, and consistent with that used in a similar application in Monticello TS Table 3.2.3. Specifying reactor modes/conditions in this fashion clarifies the applicability of the each instrument function. This allows the specified conditions for when a particular function is required to be listed. Notes (a) and (b) will be added (referred to via the # sign), indicating that the 'Reactor Building Plenum Radiation Monitors' function must be operable 'During operations with the potential for draining the reactor vessel' and 'During movement of recently irradiated fuel assemblies in secondary containment.'

The Reactor Building Plenum Radiation Monitors function is required to be operable in the mode switch positions of Startup and Run and in the Hot Shutdown condition because these are areas of operation where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This function is not required in Cold Shutdown and Refuel conditions, because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these conditions; thus, this function is not required. In addition, this function is also required to be operable during OPDRVs and movement of 'recently' irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. Due to radioactive decay, this function is only required to isolate secondary containment during fuel handling accidents involving handling 'recently' irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

(f) For the 'Refueling Floor Radiation Monitors' function the 'Applicable Modes or Other Specified Conditions...' under this new column are listed as 'Hot Shutdown, Startup and Run.' This nomenclature is similar to, and consistent with that used in a similar application in Monticello TS Table 3.2.3. Specifying reactor modes/conditions in this fashion clarifies the applicability of the each instrument function. This allows the specified conditions for when a particular function is required to be listed. Notes (a) and (b) will be added (referred to via the # sign), indicating that the 'Refueling Floor Radiation Monitors' function must be operable 'During operations with the potential for draining the reactor vessel' and 'During movement of recently irradiated fuel assemblies in secondary containment.'

The Refueling Floor Radiation Monitors function is required to be operable in the mode switch positions of Startup and Run and in the Hot Shutdown condition because these are areas of operation where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This function is not required in Cold Shutdown and Refuel conditions, because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these conditions; thus, this function is not required. In addition, this function is also required to be operable during OPDRVs and movement of 'recently' irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. Due to radioactive decay, this function is only required to isolate secondary containment during fuel handling accidents involving handling 'recently' irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

Changes to TS Bases are provided to define the above-proposed conditions. Note that additional changes to TS Table 3.2.4 were provided under a LAR dated December 23, 2003. The proposed changes in this LAR do not conflict with the previously requested changes.

Safety Assessment of Table 3.2.4 Instrumentation Changes

Revising Table 3.2.4 by adding a new column entitled "Applicable Modes or Other Specified Conditions for Which the Function Must be Operable or Operating" establishes a place to list the applicable reactor modes or other specified conditions for each instrument function. This provides additional clarification as the applicable modes or conditions for each instrument function supporting SBGT System operation or isolation of the Reactor Building Ventilation System will be clearly identified. The "#" sign after the new column title clarifies the use of the column in the table by directing the operator to the applicable notes at the end of the instrumentation table. Clearly stipulating within the table the reactor modes and other specified conditions when each instrument function must be operable instead of referring to the governing system specification clarifies the specification. This is appropriate since operation of the specific instrument functions in support of the SBGT System is only meaningful or required in accordance with the applicable reactor modes or specified conditions. These other specified conditions are consistent with the applicability paragraphs and action statement paragraphs within the governing specifications, as discussed in later sections. Also, specifying the 'applicable reactor modes' and 'other specified conditions' under this new column for the following functions increases the ease of use by the operators.

The other specified conditions, i.e., conditions (a) and (b) (referred to via the # sign), are applicable based upon the results of the AST FHA analysis and are consistent with NUREG-1433 (implements guidance of TSTF-51). Corresponding changes to TS Bases are proposed consistent to fully define these proposed changes. Therefore, based on the preceding discussion these changes are acceptable.

4.1.2 Specification 3.7.B – Standby Gas Treatment System

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.

<u>Safety Assessment of SBGT System Changes</u>

The two new action statement paragraphs (3.7.B.1.c and 3.7.B.1.d) provide specific descriptive actions to be taken when one or both trains of the SBGT System are inoperable, enhancing the direction provided to the operators. Clearly defining the actions to be taken when one or both trains of the SBGT System are inoperable during fuel movement improves the usability of the specification. Retaining and replicating the existing seven-day allowance for one train of the SBGT System being out-of-service (current paragraph 3.7.B.1.a) within new action statement paragraph 3.7.B.1.c is consistent with the current MNGP TS. Defining the post-shutdown period when operability in accordance with this specification is required by introducing the term 'recently' before 'irradiated fuel' maintains the assumptions of the AST FHA analysis. Modifying system applicability paragraph 3.7.B.1 to include these two new actions has no effect on safety. 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 SBGT System is required to be operable during OPDRVs to provide filtering and holdup of activity to reduce offsite and Control Room doses if an event were to occur.

Removing the term 'and fuel handling' from action statement paragraph 3.7.B.1 is acceptable because new action statements 3.7.B.1.c and 3.7.B.1.d provide more definitive actions to be taken during fuel handling operations. Replacing the term 'circuits' with 'trains' in the specification paragraphs avoids potential confusion with electrical circuits and is consistent with SR 4.7.B.1. Therefore, these changes are acceptable.

Also, as discussed in Section 4.2 of this Enclosure, "Supplemental Risk Discussion – Shutdown Controls," the unavailability of the SBGT System (and associated radiation monitoring instrumentation) will be assessed and minimized as part of the outage planning process, in accordance with the guidance of NUMARC 93-01 (Reference 11), to further reduce doses in case of an event, below that provided by natural decay, and decontamination by water in the reactor cavity or Spent Fuel Pool.

4.1.3 Specification 3.7.C. – Secondary Containment

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.

Safety Assessment of Secondary Containment Changes

Breaking up applicability paragraph 3.7.C.2.d into two separate paragraphs (3.7.C.2.d and 3.7.C.2.e) to clearly define the applicability of Secondary Containment during movement of a fuel cask or during movement of 'recently' irradiated fuel is consistent with the results of the AST FHA analysis. Adding an additional applicability paragraph, 3.7.C.2.f, will clearly define the applicability of Secondary Containment during operations with the potential for draining the reactor vessel.

Due to the presentation in the MNGP custom TS, it was necessary to relocate the concept of the term 'Alterations of the reactor core' to a new separate action (3.7.C.5) requiring suspension of core alterations if the shutdown margin (SDM) requirements of Specification 3.3.A are not met. Action statement paragraph 3.7.C.5 is consistent with the definitions of a core alteration in the MNGP TS and provides additional guidance, directing that all insertable control rods in core cells containing one or more fuel assemblies be fully inserted. When SDM is not met during refueling the operator must immediately suspend operations that could reduce SDM. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and are excluded from the suspended actions. Relocating and modifying the action related to core alterations will allow SDM considerations to be maintained consistent with current industry practice and the MNGP TS format.

Adding the word 'recently' before 'irradiated fuel' in action statement paragraph 3.7.C.4 clarifies that Secondary Containment is not required during handling of irradiated fuel that has decayed for longer than 24 hours, consistent with AST FHA analysis results. Adding OPDRVs to the conditions under which establishing Secondary Containment is necessary in action statement paragraph 3.7.C.4 for consistency with the Control Room ventilation specifications and current industry guidance. This function is also required to be operable during OPDRVs because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs. Therefore, these changes are acceptable.

4.1.4 Specification 3.17.A. – Control Room Ventilation System

It is proposed to modify the Control Room Ventilation (CRV) System specification applicability paragraph 3.17.A.1 and action statement paragraphs 3.17.A.2.c and 3.17.A.3.c to remove the term 'core alterations.' Also, consistent with the results of the AST FHA analysis it is proposed to revise 3.17.A.3.c to require that this action, 3.17.A.3.c, be immediately entered with both Control Room ventilation trains inoperable.

Safety Assessment of CRV System Changes

Modifying the CRV System specification applicability paragraph 3.17.A.1 and action statement paragraphs 3.17.A.2.c and 3.17.A.3.c to remove the term 'core alterations' is acceptable since the only accident that could result in the release of significant quantities of fission products to the Secondary Containment during this mode of operation is the accidental dropping of a fuel bundle. Therefore, the results of an event during a core alteration are bounded by the results of the AST FHA analysis. Removing the reference to action statement paragraph 3.17.A.3.a in action statement paragraph 3.17.A.3.c provides additional conservative direction consistent with current industry guidance when both trains of the CRV System are inoperable during movement of irradiated fuel. Requiring suspension of fuel movement in action paragraph 3.17.A.3.c when both CRV System trains are inoperable, exits the condition requiring the operation of one CRV train, preserving the AST FHA analysis assumption. Therefore, these changes are acceptable.

4.1.5 Specification 3.17.B.1. – Control Room Emergency Filtration System

The Control Room EFT System specification applicability paragraph (3.17.B.1) and action statement paragraphs 3.17.B.1.c and 3.17.B.1.d are being modified to remove the term 'core alterations.' To define when this system is required to be operable based on the AST FHA analysis, the specification applicability (3.17.B.1) and action statement paragraphs (3.17.B.1.c and 3.17.B.1.d) are modified by adding the word 'recently' before 'irradiated fuel assemblies' to clarify that this specification does not apply during handling of irradiated fuel that has decayed for longer than 24 hours.

Safety Assessment of CRV EFT System Changes

Modifying the Control Room EFT System specification applicability paragraph (3.17.B.1) and action statement paragraphs 3.17.B.1.c and 3.17.B.1.d to remove the term 'core alterations' is acceptable since the only accident that could result in the release of significant quantities of fission products to the Containment during this mode of operation is the accidental dropping of a fuel bundle. Therefore, the results of an event during a core alteration are bounded by the results of the AST FHA analysis. Defining in the specification applicability paragraph (3.17.B.1) and action statement paragraphs (3.17.B.1.c and 3.17.B.1.d) when the Control Room EFT System (or train) is required to be operable and the actions to be taken when inoperable, by adding the word 'recently' before 'irradiated fuel assemblies' to clarify that this specification does not apply during handling of irradiated fuel that has decayed for longer than 24 hours clarifies the specification. Therefore, these changes are acceptable.

4.1.6 Revise Technical Specification Bases

Draft revised MNGP TS Bases consistent with the results of the AST FHA analysis are provided. These proposed Bases changes will be processed in accordance with the MNGP TS Bases Control Program in conjunction with the processing and NRC approval of this amendment. Enclosure 2 provides a mark-up and Enclosure 3 provides a retyped version of the proposed changes to the MNGP TS and associated TS Bases. The proposed TS Bases changes include:

1. Discussions of the applicable modes or other specified conditions for which the SBGT instrumentation listed in Table 3.2.4; the SGBT System, the Control Room EFT System, and/or Secondary Containment must be operable or operating.

2. Discussion of actions for Specification 3.7.C.5 related to shutdown margin.

3. Discussion of 'recently' irradiated fuel, i.e., fuel that has decayed for less than 24 hours is discussed in the applicable TS Bases specifications paragraphs.

4.2 Supplemental Risk Discussion – Shutdown Controls

The following discussion of shutdown risk is provided to supplement the analysis and justification of the changes to relax the operational constraints during shutdown. It is applicable primarily to those TS affected by the proposed changes regarding the terminology 'recently irradiated fuel assemblies.'

The containment and associated ESF systems are only required by the TS to respond to the specific events, which are postulated to result in a significant release of radioactivity (e.g., an FHA or a reactor cavity or fuel pool drain down). As a result, the requirements of the TS are based on the plant being in certain specified conditions and are not based on providing requirements associated with shutdown risk considerations. Shutdown risk issues are instead addressed by utility outage management programs that follow the guidance of NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," (Reference 12). NUMARC 91-06, Section 4.5, discusses the need to assure that secondary containment closure can be achieved to prevent fission product release during severe accidents. NUMARC 91-06 also identifies that the time to effect closure should be consistent with plant conditions (e.g., reactor coolant system inventory and decay heat load).

As described in the Reviewer's Note to TSTF-51, the addition of the term 'recently' associated with handling irradiated fuel in all of the containment function TS requirements is only applicable to those licensees who have demonstrated by analysis that after sufficient radioactive decay has occurred, off-site doses resulting from a fuel handling accident remain below the SRP limits (well within 10 CFR 100).

Additionally, licensees adding the term 'recently' must make the following commitment, which is consistent with NUMARC 93-01. In NUMARC 93-01, Revision 3 (Reference 11), Section 11.3.6.5, the following guidance is provided:

"The following guidelines are included in the assessment of systems removed from service during movement of irradiated fuel:

- During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.

- A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt measures need not completely block the penetration or be capable of resisting pressure.

The purpose of the "prompt methods" mentioned above is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."

Accordingly, as required by the TSTF-51 Reviewer's Note, NMC will adopt the NUMARC 93-01 guidance discussed above.

5.0 TECHNICAL EVALUATION

The 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 are 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 will result in doses that are well within the acceptable dose criteria specified in 10 CFR 50.67 for the EAB and LPZ and for the Control Room operator.

5.1 Evaluation of AST FHA Related TS Changes

AST FHA analysis results indicate that TS operability requirements for the Secondary Containment and certain ESF features (i.e., SBGT and the Control Room EFT Systems) may be removed after sufficient radioactive decay has occurred to ensure offsite doses remain below the 10 CFR 50.67. Following a reactor shutdown, the decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed TS changes are based on the results of performing the design basis AST FHA analysis, demonstrating that with the assumed fuel decay period, the reduced radionuclide inventory available for release would result in radiological dose consequences within the regulatory acceptance limits.

Following sufficient fuel decay, the primary success path for mitigating an FHA no longer includes the functioning of the active containment systems. Additionally, ESF atmospheric treatment systems are no longer necessary to protect the Control Room operator dependent on the AST assumptions and methodology. Therefore, TS operability requirements are proposed to be modified to reflect that water level and decay time comprise the primary success path for mitigating a FHA.

Handling of irradiated fuel in the reactor vessel is only allowed by plant procedures when the water level in the reactor cavity is at a high water level. The proposed TS changes only affect containment requirements during periods of relatively low shutdown risk during refueling outages. Therefore, the proposed changes do not significantly increase the shutdown risk.

The accidents postulated to occur during core alterations, in addition to FHAs, are: inadvertent criticality (due to a control rod withdrawl error or continuous control rod withdrawal error during refueling) and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Since the only accident postulated to occur during core alterations that results in a significant radioactive release is the FHA, the proposed change to omit core alterations as a condition requiring secondary containment or ESF mitigation features is justified.

5.2 Radiological Consequences of a Design Basis FHA

The radiological consequences of a design basis FHA were analyzed using MNGP specific design inputs and assumptions. No specific ESF functions were credited in the analysis. The calculations assumed that the Control Room EFT System was not operating and that the Control Room Ventilation System remained operating in its normal (non-emergency) mode. Similarly, the SBGT System was assumed not to be operating. Plant-specific design inputs were validated to ensure that they are representative of 'as-built' plant design conditions. The primary assumptions used in this analysis are:

- An alternate source term in accordance with the guidance of RG 1.183 was used. The guidance of RG 1.183 Appendix B was followed.
- A fission product inventory for a fuel burnup of less than 60 GWd/MT was assumed. The reactor is assumed to have operated at 1880 MWt (consistent with the initial analyses performed in accordance with the power rerate plus an additional 2 percent power to account for uncertainty).

- A radial peaking factor of 1.7 was assumed.
- 125 fuel rods are assumed damaged by a drop of a fuel bundle onto the reactor core releasing the entire fission product contents in the fuel gap.
- The fraction of the fission product inventory in the gap as provided in RG 1.183 was assumed.
- An irradiated fuel fission product decay period of 24 hours was assumed (time period from the reactor shutdown (all rods in) to the first fuel movement).
- An overall effective decontamination factor of 200 for the iodine isotopes in the reactor cavity and spent fuel pool was assumed in accordance with the minimum water depth of 23 feet assumed in RG 1.183 (see discussion in Section 5.3).
- No credit is taken for Primary and Secondary Containment – assumed not isolated.
- No credit is taken for operation or fission product removal by the Standby Gas Treatment System (an ESF system).
- No credit is taken for atmospheric dilution or mixing in the Secondary Containment.
- No credit is taken for operation or fission product removal by the Control Room EFT System (an ESF system).

5.3 Fuel Handling Accident Description

MNGP USAR Section 14.7.6, "Refueling Accident Analysis," provides a description of the present FHA analysis. The AST FHA analysis considers changes in various parameters in accordance with the AST methodology, determines requirements for operation of associated ESF systems (and other ventilation systems), and determines requirements for isolation of Secondary Containment (the Reactor Building) and Control Room. The FHA analysis postulates that a fuel assembly is dropped on to the top of the reactor core during refueling operations.

The MNGP reactor core consists of 484 fuel assemblies. The number of fuel rods within an assembly has increased from initially 7x7 array designs, to 8x8, 9x9 and 10x10 arrays (minus the water rods/channels associated with a particular design). Fuel assembly designs now also include partial length fuel rods. The total number of fuel rods in a reactor core may vary each cycle due to the number and type of fuel assemblies specified each cycle by the core nuclear design. The actual number of rods postulated to fail in an FHA is dependent on the fuel type and also dependent upon the fuel handling equipment used at the site. (The dropped fuel assembly together with the fuel grapple head and refueling mast are assumed to impact the core for additional conservatism in the accident.) USAR Section 14.7.6.3.1, "Assumptions," states that the radiological analysis conservatively assumes failure of 125 rods of GE 8x8 array fuel. As described therein, the relative amount of activity released for a 9x9 and a 10x10 array fuel is 0.91 and 0.95 times, respectively, the activity released for an 8x8 fuel array. Therefore, the present FHA analysis assumption of 125 'equivalent' damaged rods bounds the available fuel designs. Based on these considerations and consistent with the existing scenario, 125 fuel rods out of a core total of 29040 rods were assumed damaged.

The AST FHA analysis was performed for a drop of a fuel bundle in the reactor cavity, over the reactor core, since it is more limiting, i.e., damages more fuel rods, than any drops that could occur over the spent fuel pool (SFP). The depth of water over a fuel bundle greatly exceeds 23 feet in the reactor cavity. The SFP low water level alarm point corresponds to a depth of approximately 22 feet above the stored fuel, less than the 23 feet of water assumed in RG 1.183 for allowing a decontamination factor of 200 to be automatically used. A drop over the reactor core is more limiting (i.e., damages more fuel rods) than any drop that could occur over the SFP, releasing more activity, off-setting the approximately 12-inch reduction in the depth over the fuel during movement in the SFP. However, for conservatism NMC makes the following commitment.

> 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.

TS 3.10.D and refueling procedures require that the reactor be shut down for a minimum of 24 hours prior to fuel movement within the reactor. Therefore, a post-shutdown 24-hour decay period was assumed in determining the release activity inventory. A radial peaking factor of 1.7 was assumed. All the activity in the gap between the fuel pellets and the cladding of the damaged fuel rods is assumed to be released instantaneously into the pool. A decontamination factor of 200 was used (per RG 1.183), which is higher than the value of 100 used in the existing licensing basis analysis. NMC has assumed no decontamination for noble gases released in the pool (reactor cavity or SFP) and full retention of all aerosol and particulate fission products by the pool water. Any activity leaving the fuel pool enters the Reactor Building. All of the AST FHA activity is assumed to be released within 2 hours from the Reactor Building vent as a ground release, with no credit for holdup or dilution by the Reactor Building, and no credit for operation of the SBGT System. Not crediting any dilution, holdup, filtration (cleanup), and elevated release by the SBGT System of the activity released from the pool represents a more conservative basis than that used in the existing licensing basis FHA analysis.

The AST FHA was analyzed assuming no credit for Control Room isolation or operation of the Control Room EFT System. The activity is assumed to enter the Control Room via the normal Control Room Ventilation System at a bounding rate of 7440 scfm with an assumed unfiltered Control Room inleakage rate of up to 1000 scfm.

No Control Room isolation is assumed during the course of the accident, which is more conservative than the existing licensing basis FHA analysis. This results in conservative dose assumptions for Control Room personnel.

5.4 Fuel Handling Accident Results

The radiological consequences of the design basis FHA were analyzed using the RADTRAD computer code. NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation," (Reference 13) describes the application of the code. Analysis results indicate that the offsite (EAB and LPZ) calculated and Control Room operator doses are within regulatory limits after AST implementation.

<table>
<tr><th colspan="4">Fuel Handling Accident Radiological Consequence Analysis
(rem TEDE)</th></tr>
<tr><th rowspan="2">Proposed Analysis</th><th colspan="2">Offsite Dose</th><th rowspan="2">Control Room Dose</th></tr>
<tr><th>EAB</th><th>LPZ</th></tr>
<tr><td>Refueling Accident Inside Containment</td><td>1.80</td><td>0.34</td><td>4.71</td></tr>
<tr><td>AST Regulatory Acceptance Limit (10 CFR 50.67)</td><td>6.3</td><td>6.3</td><td>5</td></tr>
</table>

<table>
<tr><th rowspan="2">Current Analysis</th><th>Thyroid</th><th>Whole Body</th><th>Thyroid</th><th>Whole Body</th><th rowspan="2"></th></tr>
<tr><th colspan="4">(rem)</th></tr>
<tr><td>Refueling Accident Current Analysis</td><td>2.04</td><td>0.192</td><td>0.71</td><td>0.067</td><td>Not calculated</td></tr>
<tr><td>Present Regulatory Acceptance Limit (10 CFR 100)</td><td>300</td><td>25</td><td>300</td><td>25</td><td>Not calculated</td></tr>
</table>

1. MNGP does not have a separate Fuel Handling Building.

The results of the AST FHA analysis indicate that the dose at the EAB would be 1.80 rem TEDE and at the LPZ would be 0.34 rem TEDE, both well below the regulatory acceptance criterion of 6.3 rem TEDE set forth in RG 1.183, Table 6, and therefore acceptable. The dose to the Control Room operators is 4.71 rem TEDE, less than the regulatory acceptance criterion of 5 rem TEDE contained in 10 CFR 50.67. On this basis, this proposed license amendment application to change the TS and the licensing and design bases regarding the DBA FHA does not represent a significant offsite or onsite radiological impact and is acceptable.

For the FHA, the AST analysis demonstrates that a 24-hour decay period is sufficient to ensure Secondary Containment and Control Room automatic isolation are not required during core alterations or fuel handling.

5.5 Discussion of Control Room Ventilation Systems Operation

The current MNGP licensing basis assumes Control Room isolation and initiation of the Control Room EFT System following a FHA. The AST FHA analysis demonstrates that a 24-hour decay period is sufficient to ensure that Control Room isolation/Control Room EFT initiation are not required during core alterations or fuel handling activities.

During normal operation, the CRV System operates to recirculate and condition (heat or cool) air within the Control Room habitability envelope, which includes the Control Room and the first and second floors of the EFT Building. CRV System design includes a common outside air intake with the Control Room EFT system; however, blanking plates were placed downstream of the air intake, blanking off the CRV System air intake duct because of inleakage concerns, leaving the CRV as a recirculation-only system. The AST FHA analysis and calculations bound CRV System operation with or without the CRV air intake blanking plates installed.

5.6 Offsite Atmospheric Dispersion Factors

NMC calculated new offsite atmospheric dispersion factors (χ/Q's) for the MNGP Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) using site-specific inputs and the PAVAN-PC computer program. The PAVAN program, documented in NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," (Reference 14) uses the methodology described in RG 1.145 (Reference 15). PAVAN calculates the diffusion from a source and relative concentration at a receiver due to an accidental release of radioactivity into the environment in accordance with the guidance in RG 1.145. Current site-specific meteorological data was used for the five-year period of 1998-2002 to calculate the revised atmospheric dispersion factors.

Meteorological data files suitable for PAVAN input were generated by the site MIDAS computer program, providing data in the form of a joint frequency distribution of wind direction, wind speed, and atmospheric stability. MIDAS is an acronym for "Meteorological Information and Dose Assessment System," developed by Pickard, Lowe and Garrick, Inc. to collect meteorological and radiological effluent data, perform off-site dose projections, and perform routine effluent release calculations.

MNGP Calculation CA-04-036, "MNGP AST - Offsite Post-Accident Atmospheric Dispersion Analysis," (AAC Calculation #MNGP-001) describes how these files were transformed for PAVAN input and provides a card-by-card description of the inputs. NMC does not plan on submitting further updates of this calculation for review as part of AST unless requested. The data files are listed within the calculation and are provided on the enclosed calculation compact disk (CD).

5.7 Control Room Atmospheric Dispersion Factors

NMC used the ARCON96 computer code methodology, described in NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes," (Reference 16) and the PAVAN-PC computer program to calculate the Control Room χ/Q values in accordance with RG 1.194 (Reference 17). The ARCON96 code calculates relative concentrations in plumes from nuclear power plants at control room air intakes in the vicinity of the release point. As recommended by RG 1.194, ARCON96 was used to calculate ground level releases and PAVAN-PC was used to calculate elevated releases. The Reactor Building Exhaust Vent was determined to be the limiting and representative release point for the AST FHA. Values for elevated releases under fumigation conditions were calculated using PAVAN-PC, as recommended by RG 1.194.

Meteorological data files suitable for ARCON96 input were generated containing data collected by MIDAS and reformatted for ARCON96 input as specified in NUREG/CR-6331:

Location identifier
Julian Day
Hour of the Day
Lower-level wind direction
Lower-level wind speed
Stability Class
Upper-level wind direction
Upper-level wind speed

Wind direction is in degrees. Wind speed is meters/sec (nearest tenth without decimal). Stability class is in accordance with RG 1.23, "Meteorological Programs in Support of Nuclear Power Plants," (Reference 18).

Joint frequency meteorological data files suitable for PAVAN input were provided as discussed previously.

MNGP Calculation CA-04-037, "MNGP AST - CR/TSC Post-Accident Atmospheric Dispersion Analysis," (AAC Calculation #MNGP-002) describes how these files were transformed for ARCON96 and PAVAN-PC input and provides a description of the ARCON96 inputs and a card-by-card description of the PAVAN inputs. NMC does not plan on submitting further updates of this calculation for review as part of AST unless requested. All data files used are listed in the calculation and can be found on the calculation CD.

6.0 REGULATORY SAFETY ANALYSIS

6.1 No Significant Hazards Consideration

Pursuant to 10 CFR 50.67 and 10 CFR 50.90, the Nuclear Management Company, LLC (NMC) proposes to revise the Monticello Nuclear Generating Plant (MNGP) licensing basis to reflect a selective application of an Alternative Source Term (AST) methodology. The fuel handling accident (FHA) was re-analyzed applying the guidance of Regulatory Guide (RG) 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000, and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms." The current accident source term is replaced with a source term as prescribed in 10 CFR 50.67 for the fuel handling accident only.

The fuel handling AST analyses were performed without credit for the functioning of the systems listed below or isolation of the Secondary Containment during or after the FHA. As such, it is proposed to relax the applicable operability requirements during fuel handling consistent with Revision 2 of the Technical Specification Task Force (TSTF), Improved Standard Technical Specifications Change Traveler TSTF-51 entitled, "Revise containment requirements during handling irradiated fuel and core alterations," which was approved by the NRC on November 1, 1999, for the following systems, structures and components.

- Instrumentation that initiates Reactor Building ventilation isolation and initiates the Standby Gas Treatment System,
- Secondary Containment,
- Standby Gas Treatment System,
- Control Room Ventilation System, and
- Control Room Emergency Filtration (EFT) System.

NMC has evaluated whether a significant hazards consideration is involved with the proposed change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   The proposed amendment involves implementation of the AST for the fuel handling accident at MNGP. There are no physical design modifications to the plant associated with the proposed amendment. The revised calculations do not impact the initiators of an FHA in any way.

   The changes also do not impact the initiators for any other design basis accident (DBA) or events. Therefore, because DBA initiators are not being altered by adoption of the AST analyses, the probability of an accident previously evaluated is not affected.

   With respect to consequences, the only previously evaluated accident that could be affected is the FHA. The AST is an input to calculations used to evaluate the consequences of the accident, and does not, in and of itself, affect the plant response or the actual pathways to the environment utilized by the radiation/activity released by the fuel. It does however, better represent the physical characteristics of the release, so that appropriate mitigation techniques may be applied. For the FHA, the AST analyses demonstrate acceptable doses that are within regulatory limits after 24 hours of radiological decay, without credit for Secondary Containment integrity, selected ESF filtration system operation (i.e., SBGT System or Control Room EFT System) or Control Room isolation. Therefore, the consequences of an accident previously evaluated are not significantly increased.

   Based on the above conclusions, this proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed amendment does not involve a physical alteration of the plant. No new or different types of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed changes. Also, no changes are proposed to the methods governing plant/system operation during handling of irradiated fuel, so no new initiators or precursors of a new or different kind of accident are created. New equipment or personnel failure modes that might initiate a new type of accident are not created as a result of the proposed amendment.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

Response: No.

The proposed amendment is associated with the implementation of a new licensing basis for the MNGP FHA. Approval of this change from the original source term to an alternative source term derived in accordance with the guidance of RG 1.183 is being requested. The results of the FHA accident analysis, revised in support of the proposed license amendment, are subject to revised acceptance criteria. The AST FHA analysis has been performed using conservative methodologies, as specified in RG 1.183. Safety margins have been evaluated and analytical conservatism has been utilized to ensure that the analyses adequately bound the postulated limiting event scenario. The dose consequences of the limiting FHA remain within the acceptance criteria presented in 10 CFR 50.67 and RG 1.183.

The proposed changes continue to ensure that the doses at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) boundaries, as well as the Control Room, are within the corresponding regulatory limits. For the FHA, RG 1.183 conservatively sets the EAB and LPZ limits below the 10 CFR 50.67 limit, and sets the Control Room limit consistent with 10 CFR 50.67.

Since the proposed amendment continues to ensure the doses at the EAB, LPZ and Control Room are within corresponding regulatory limits, the proposed license amendment does not involve a significant reduction in a margin of safety.

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

6.2 Applicable Regulatory Requirements/Criteria

This section describes how the proposed changes and NMC's technical analyses satisfy applicable regulatory requirements and acceptance criteria.

- 10 CFR 50, Appendix A, GDC 61, "Fuel storage and handling and radioactivity control"

  The GDC in place today became effective after the MNGP construction permit was issued. A memorandum, dated September 18, 1992, to the NRC Executive Director for Operations from the Secretary of the NRC summarized the results of a Commissioners vote in which the Commissioners instructed the NRC staff not to apply the GDC to plants with construction permits issued prior to May 21, 1971. The Northern State Power Company (predecessor to NMC) construction permit was issued on June 19, 1967.

  NMC's design and licensing basis for fuel storage and handling and radiological controls is detailed in the USAR and other plant-specific licensing basis documents. Appendix E of the MNGP FSAR evaluated the MNGP design against the GDC presented in 10 CFR 50, Appendix A, effective May 21, 1971.

  "The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions."

- 10 CFR 50, Appendix A, GDC 19, "Control Room"

  "A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses under part 52 of this chapter who do not reference a standard design certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident."

- 10 CFR 50.67, "Accident Source Term"

  10 CFR 50.67 permits licensees to voluntarily revise the accident source term used in design-basis radiological consequence analyses. This document is part of a 10 CFR 50.90 license amendment application and evaluates the consequences of a design basis FHA previously reported in the safety analysis report.

- 10 CFR 100, "Paragraph 11 - Determination of Exclusion Area, Low Population Zone and Population Center Distance"

  This paragraph provides criteria for evaluating the radiological aspects of reactor sites. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based on a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products. A similar footnote appears in 10 CFR 50.67.

  In accordance with the provisions of 10 CFR 50.67(a), the radiation dose reference values in 10 CFR 50.67(b)(2) were used in these analyses in lieu of those prescribed in 10 CFR 100.

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000"

  This guide outlines acceptable applications of AST's: the scope, nature and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. It also establishes acceptable ASTs and identifies the attributes of ASTs acceptable to the NRC staff. This guide also identified acceptable radiological analysis assumptions for use in conjunction with the AST.

NMC used this regulatory guide extensively in the preparation of this 'selective implementation' evaluation, the supported application and the supporting analyses. This application and the supporting analyses comply with this guidance to the extent practical.

- NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants"

  NUREG-1465 provides more realistic estimates of 'source term' releases into containment in terms of timing; nuclide types, quantities, and chemical form, given a severe core melt, than TID-14844. NUREG-1465 provides much of the technical basis for the regulatory positions in RG 1.183.

- NUREG-0800, Standard Review Plan, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents"

  This SRP section covers the review of the radiological effects of a postulated FHA. Revision 1 does not reflect the guidance in RG 1.183 or the promulgation of 10 CFR 50.67.

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

7.0 ENVIRONMENTAL CONSIDERATION

Nuclear Management Company (NMC) has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for identification of licensing and regulatory actions requiring environmental assessments." NMC has determined that the proposed changes meet the criteria for a categorical exclusion as set forth in 10 CFR 51.52(c)(9), "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b), "Issuance of amendment." This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or that changes an inspection or surveillance requirement, and the amendment does not result in the following:

(i) A significant hazards consideration,

The proposed amendment does not involve a significant hazard. See the no *significant hazards consideration determination evaluation.*

(ii) A significant change in the type or significant increase in the amounts of any effluent that may be released offsite, or

The proposed amendment is consistent with and does not change the design basis of the plant. The proposed amendment will not result in an increase in power level, will not increase the production of radioactive waste and byproducts, and will not alter the flowpath or method of disposal of radioactive waste or byproducts. Therefore, the proposed amendment does not involve any change in the type or amount of any effluent that may be released offsite.

(iii) A significant increase in individual or cumulative occupational radiation exposure.

The proposed amendment does not result in changes in the level of control or methodology used for processing radioactive effluents or handling of solid radioactive waste. There will be no change to the normal radiation levels within the plant. Therefore, the amendment does not involve an increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.52(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment is not required.

8.0 REFERENCES

1. Technical Specification Task Force (TSTF), Improved Standard Technical Specifications Change Traveler, TSTF-51, "Revise containment requirements during handling irradiated fuel and core alterations," Revision 2, NRC approved on November 1, 1999.

2. 10 CFR 50.67, "Accident Source Term."

3. U.S. Nuclear Regulatory Commission [NRC], Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000.

4. U.S. NRC, NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition, Section 15.0.1, 'Radiological Consequence Analyses Using Alternative Source Terms,' Revision 0, July 2000.

5. 10 CFR 100, "Reactor Site Criteria."

6. 10 CFR 100.11, "Reactor Site Criteria - Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

7. U.S. Atomic Energy Commission, Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962.

8. U.S. NRC, NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995.

9. 10 CFR 50, Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," GDC-19, 'Control room.'

10. U.S. NRC, NUREG-1433, Revision 3, "Standard Technical Specifications, General Electric Plants, BWR/4," dated March 31, 2004.

11. Nuclear Energy Institute, NUMARC 93-01, Revision 3, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated July 2000.

12. Nuclear Management and Resources Council, Inc. (NUMARC) 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," dated December 1991.

13. U.S. NRC, NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation," dated December 1997, including Supplement 1, dated June 8, 1999, and Supplement 2, dated October 2002.

14. U.S. NRC, NUREG/CR-2858 (PNL-4413), "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," dated November 1982.

15. U.S. NRC, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, dated November 1982.

16. U.S. NRC, NUREG/CR-6331 (PNNL-10521), Revision 1, "Atmospheric Relative Concentrations in Building Wakes," dated May 1997.

17. U.S. NRC, Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." dated June 2003.

18. U.S. NRC, Regulatory Guide 1.23, "Meteorological Programs in Support of Nuclear Power Plants," dated February 17, 1972.

Regulatory Guide 1.183 Versus MNGP Analysis Comparison Matrix

Regulatory Guidance	Basis of Compliance
3. ACCIDENT SOURCE TERM	
3.1 Fission Product Inventory	In compliance.
The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty[8]. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values[9]. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID-14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels. [8]The uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02. [9]Note that for some radionuclides, such as Cs-137, equilibrium will not be reached prior to fuel offload. Thus, the maximum inventory at the end of life should be used.	Generic BWR Source Term program (NYSIS) used to determine bounding core isotopic inventory. SAS2H code utilized by NYSIS contains same method of solution as ORIGEN-ARP. NYSIS reports a maximum activity level for each isotope in the inventory, for the set of sixty isotopes required as RADTRAD inputs. (CA-03-194, Methodology section) Current licensed thermal power is 1775 MWth. Analyses use thermal power of 1880 MWth and Appendix K uncertainty factor of 1.02 (1918 MWth). (CA-03-194, Input 6)
For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the	For FHA, fission product inventory was determined by dividing total core inventory by number of fuel rods in core. (CA-04-041, Design Input 4.4)

Regulatory Guidance	Basis of Compliance
damaged rods.	Peaking factor of 1.7 was used. MNGP does not specify radial peaking factors in Tech Specs or the COLR. Value of 1.7 was chosen as conservative and bounding based on anticipated core designs from MNGP's Nuclear Analysis Department and review of previous calculation assumptions. (CA-04-041, Assumption 3.11)
No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	For FHA, decay of 24 hours from time of shutdown is assumed. (CA-04-041, Assumption 3.10 and Design Input 4.9) (T.S. 3.10.D)
3.2 Release Fractions[10, 11] For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. [10, 11]The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.	In compliance. For FHA, release fractions are as given in Table 3. Radial peaking factor is 1.7. (CA-04-041, Design Input 4.5, Assumption 3.11) Peak fuel burnup is assumed less than 60 GWD/MTU. (CA-03-194, Table 1) (LAR submittal Enclosure 1, Section 5.2 Assumptions invoke Table 3 for release fractions and associated footnote) MNGP core contains uranium dioxide fuel. (USAR Rev 20, Section 3.4.2.2)
3.3 Timing of Release Phases	In compliance.

Regulatory Guidance	Basis of Compliance
For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.	For FHA, instantaneous release assumed. (CA-04-041, Assumption 3.6)
3.4 Radionuclide Composition Table 5 lists the elements in each radionuclide group that should be considered in DBAs.	In compliance. See Appendix B Section 1.2 below. This item provides specific instructions for nuclides to be considered for FHA (Xe, Kr, I, Br, Cs, Rb). The other nuclide groups in Table 5 are not expected as part of a gap release so are not included in FHA analysis. (RG 1.183, Section 3.2 Tables 1 and 3, and Appendix B Section 1.2)
3.5 Chemical Form Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	In compliance. For FHA, 95% CsI, 4.85% elemental iodine, 0.15% organic iodide assumed. (CA-04-041, Design Input 4.6)
3.6 Fuel Damage in Non-LOCA DBAs The amount of fuel damage caused by a FHA is addressed in Appendix B of this guide.	In compliance. See Appendix B below.

Regulatory Guidance	Basis of Compliance
4. DOSE CALCULATIONAL METHODOLOGY	
4.1 Offsite Dose Consequences The following assumptions should be used in determining the TEDE for persons located at or beyond the boundary of the exclusion area (EAB):	
4.1.1 The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences and the released radioactivity.[13] [13]The prior practice of basing inhalation exposure on only radioiodine and not including radioiodine in external exposure calculations is not consistent with the definition of TEDE and the characteristics of the revised source term.	In compliance. For FHA, TEDE doses were determined. RADTRAD Version 3.03 was utilized to model the accident. RADTRAD contains a default table of 60 isotopes based on NUREG-1465 data. (CA-04-041, Section "Computer Data", Assumption 3.3, Design Input 4.16, and Section 5.0, "Methodology") (NUREG/CR-6604, RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation, Section 1.4.3.2)
4.1.2 The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	In compliance. For FHA, RADTRAD Version 3.03 utilized to model the accident. RADTRAD uses dose conversion factors from FGR 11 and FGR 12. (CA-04-041, Section "Computer Data", Design Input 4.15, and Section 5.0, "Methodology") (NUREG/CR-6604, RADTRAD: A Simplified Model for Radionuclide Transport and Removal

Regulatory Guidance	Basis of Compliance
	And Dose Estimation, Sections 1.4.3.3 and 2.3.1)
4.1.3 For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	In compliance. 8 hours – 3.5×10^{-4} m^3/sec 8 to 24 hours – 1.8×10^{-4} m^3/sec 1-30 days – 2.3×10^{-4} m^3/sec (CA-04-041, Design Input 4.13)
4.1.4 The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	In compliance. For FHA, RADTRAD Version 3.03 utilized to model the accident. RADTRAD uses dose conversion factors from FGR 11 and FGR 12. (CA-04-041, Section "Computer Data", Design Input 4.15, and Section 5.0, "Methodology") (NUREG/CR-6604, RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation, Sections 1.4.3.3 and 2.3.1)
4.1.5 The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67.[14] The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6). [14]With regard to the EAB TEDE, the maximum two-hour value is the basis for screening	In compliance. For the FHA, the release time is 2 hours; therefore the calculated TEDE is the maximum. The maximum (limiting) EAB χ/Q was determined and used to calculate TEDE. (CA-04-036, Assumption 3.1, Section 6.2.2) (CA-04-041, Assumption 3.7, Design Input 4.10, Section 7.0)

Regulatory Guidance	Basis of Compliance
and evaluation under 10 CFR 50.59. Changes to doses outside of the two-hour window are only considered in the context of their impact on the maximum two-hour EAB TEDE.	
4.1.6 TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	In compliance. For the FHA, the maximum LPZ χ/Q was determined and used to calculate TEDE. (CA-04-036, Assumption 3.1, Section 6.2.2) (CA-04-041, Design Input 4.10, Section 7.0)
4.1.7 No correction should be made for depletion of the effluent plume by deposition on the ground.	In compliance. No credit taken for ground deposition, verified with calculation performer AAC. PAVAN-PC and ARCON96 have no provision for plume depletion by deposition. No correction made for depletion in FHA calculation per assumptions and inputs. (NUREG/CR-2858, NUREG/CR-6331) (CA-04-041, Sections 3 and 4)
4.2 Control Room Dose Consequences The following guidance should be used in determining the TEDE for persons located in the control room:	
4.2.1 The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope,	In compliance. For the FHA, TEDE doses from applicable sources were calculated using NRC-approved RADTRAD code (i.e., air immersion dose and inhalation dose). Shine doses for sources external to the Control Room were not calculated

Regulatory Guidance	Basis of Compliance
• Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	since they would be negligible for the limited fuel damage and release of the FHA accident. (CA-04-041, Sections 5.0 and 6.1) (NUREG/CR-6604, Section 2.3.2)
4.2.2 The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in nonconservative results for the control room.	In compliance. For the FHA, the airborne isotopic activity released to the environment was evaluated. These results were then input into RADTRAD computer code to calculate doses for CR, EAB, and LPZ. (CA-04-041, Section 6.1)
4.2.3 The models used to transport radioactive material into and through the control room,[15] and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel. [15]The iodine protection factor (IPF) methodology of Reference 22 may not be adequately conservative for all DBAs and control room arrangements since it models a steady-state control room condition. Since many analysis parameters change over the duration of the event, the IPF methodology should only be used with caution. The NRC computer codes HABIT (Ref. 23) and RADTRAD (Ref. 24) incorporate suitable methodologies.	In compliance. For FHA, RADTRAD Version 3.03 utilized to model the accident. (CA-04-041, Section "Computer Data" and Section 5.0, "Methodology")
4.2.4 Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance. The control room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by engineered	In compliance. For FHA, no credit taken for ESF filtration or control room isolation or pressurization. (CA-04-041, Assumptions 3.4, 3.7, 3.8, 3.9)

Regulatory Guidance	Basis of Compliance
safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.	
4.2.5 Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	In compliance. No credit taken for personal protective equipment or prophylactic drugs. (CA-04-041, not included in assumptions and design inputs)
4.2.6 The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days.[16] For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second. [16]This occupancy is modeled in the χ/Q values determined in Reference 22 and should not be credited twice. The ARCON96 Code (Ref. 26) does not incorporate these occupancy assumptions, making it necessary to apply this correction in the dose calculations.	In compliance. Occupancy Factor: 0 to 24 hours – 1.0 1 to 4 days – 0.6 4 to 30 days – 0.4 Breathing Rate: 3.5×10^{-4} m^3/sec χ/Qs generated by ARCON96 flagged as not corrected for occupancy in FHA calculation, and occupancy factors input in appropriate RADTRAD runs (RB vent to CR). (CA-04-041, Design Inputs 4.11, 4.13 and 4.14)
4.2.7 Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors.	In compliance. For FHA, RADTRAD Version 3.03 utilized to model the accident.

Regulatory Guidance	Basis of Compliance
Equation 1 may be used to correct the semi-infinite cloud dose, $DDE_{infinity}$, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22).	RADTRAD uses dose conversion factors from FGR 11 and FGR 12 as specified in Section 4.1 above, and the correction to finite cloud dose given in Equation 1. (CA-04-041, Section "Computer Data", Design Input 4.15, and Section 5.0, "Methodology") (NUREG/CR-6604, RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation, Sections 2.3.1 and 2.3.2)
4.3 Other Dose Consequences The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	In compliance. Existing design/license basis has been established for NUREG-0737 items based on assumptions for TID 14844 and the LOCA DBA. These assumptions are bounding for the FHA. (USAR Rev 20, Section 12.03) Equipment qualification doses were not recalculated, IAW guidance in RG 1.183 Section 6.
4.4 Acceptance Criteria The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6 of RG 1.183.	In compliance. For FHA, the values in Section 4.4, Table 6 of RG 1.183 were used: Control Room – 5 rem TEDE EAB, LPZ – 6.3 rem TEDE (CA-04-041, Design Input 4.16)

Regulatory Guidance	Basis of Compliance
5. ANALYSIS ASSUMPTIONS AND METHODOLOGY	
5.1 General Considerations	
5.1.1 Analysis Quality The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	In compliance. Appendix B quality assurance provided by Sargent & Lundy. Calculations accepted by NMC IAW applicable sections of 4 AWI-05.01.25, "Calculation/Analysis Control", for safety-related vendor calculations.
5.1.2 Credit for Engineered Safeguard Features Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	In compliance. For the FHA, no credit is taken for Engineered Safeguard Features.
5.1.3 Assignment of Numeric Input Values The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis. For example, assuming minimum containment system spray flow is usually conservative for estimating iodine scrubbing, but in many cases may be nonconservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may	In compliance with limited exceptions as noted. For the FHA, numeric values chosen are conservative: Radial peaking factor of 1.7 was used. MNGP does not specify radial peaking factors in Tech Specs or the COLR. Value of 1.7 was chosen as

Regulatory Guidance	Basis of Compliance
be used in the evaluation of that portion. A single value may not be applicable for a parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by technical specifications, the value used in the analysis should be that specified in the technical specifications.[18] If a range of values or a tolerance band is specified, the value that would result in a conservative postulated dose should be used. If the parameter is based on the results of less frequent surveillance testing, e.g., steam generator nondestructive testing (NDT), consideration should be given to the degradation that may occur between periodic tests in establishing the analysis value. [18]Note that for some parameters, the technical specification value may be adjusted for analysis purposes by factors provided in other regulatory guidance. For example, ESF filter efficiencies are based on the guidance in Regulatory Guide 1.52 (Ref. 25) and in Generic Letter 99-02 (Ref. 27) rather than the surveillance test criteria in the technical specifications. Generally, these adjustments address potential changes in the parameter between scheduled surveillance tests.	conservative and bounding based on anticipated core designs from MNGP's Nuclear Analysis Department and review of previous calculation assumptions. (CA-04-041, Assumption 3.11) Decay of 24 hours from time of shutdown is assumed, consistent with TS. (CA-04-041, Assumption 3.10 and Design Input 4.9) (T.S 3.10.D) Several cases of varying Control Room ventilation flow rates and inleakage rates were performed to determine the most conservative rates and bound all possible operating configurations. (CA-04-041, Assumption 3.8 and Section 7.0) Analysis is performed at 1.02% of 1880 MWth. Current licensed power is 1775 MWth. (CA-04-041, Design Input 4.1) Most limiting χ/Qs chosen in accordance with applicable regulatory guidance. (CA-04-041, Design Inputs 4.10 and 4.11) (CA-04-036, Section 1.0, Assumption 3.1, Sections 6.1.2 and 6.2.2) Water level over damaged fuel is

Regulatory Guidance	Basis of Compliance
	determined from the level maintained by refueling procedures rather than the fuel pool water level TS (3.10.C) since the actual level during refueling is significantly higher than the TS limit, which is applicable at all times irradiated fuel is in the pool. Refueling water level is controlled by procedure and checked each shift during refueling. See discussion under Appendix B Section 2 below. (CA-04-041, Design Input 4.8) (TS 3.10.C and bases) (Procedures 9007 and 9007-B)
5.1.4 Applicability of Prior Licensing Basis The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, "Backfitting." However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	In compliance. For the FHA, significant changes to facility design basis include use of the AST and no credit taken for ESF systems. The AST FHA calculation assumptions, design inputs, and methods are compatible with the AST and TEDE criteria and comply with RG 1.183 Appendix B requirements.
5.2 Accident-Specific Assumptions The appendices to this regulatory guide provide accident-specific assumptions that are acceptable to the staff for performing analyses that are required by 10 CFR 50.67. The	In compliance. For the FHA, see Appendix B assumptions below.

Regulatory Guidance	Basis of Compliance
DBAs addressed in these attachments were selected from accidents that may involve damage to irradiated fuel. This guide does not address DBAs with radiological consequences based on technical specification reactor or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in this guide should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST. The NRC staff has determined that the analysis assumptions in the appendices to this guide provide an integrated approach to performing the individual analyses and generally expects licensees to address each assumption or propose acceptable alternatives. Such alternatives may be justifiable on the basis of plant-specific considerations, updated technical analyses, or, in some cases, a previously approved licensing basis consideration. The assumptions in the appendices are deemed consistent with the AST identified in Regulatory Position 3 and internally consistent with each other. Although licensees are free to propose alternatives to these assumptions for consideration by the NRC staff, licensees should avoid use of previously approved staff positions that would adversely affect this consistency. The NRC is committed to using probabilistic risk analysis (PRA) insights in its regulatory activities and will consider licensee proposals for changes in analysis assumptions based upon risk insights. The staff will not approve proposals that would reduce the defense in depth deemed necessary to provide adequate protection for public health and safety. In some cases, this defense in depth compensates for uncertainties in the PRA analyses and addresses accident considerations not adequately addressed by the core damage frequency (CDF) and large early release frequency (LERF) surrogate indicators of overall risk.	
5.3 Meteorology Assumptions Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide.	In compliance. For the FHA, new χ/Q values for the EAB, LPZ, and Control Room were determined using meteorological data

Regulatory Guidance	Basis of Compliance
Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 22, and 28). References 22 and 28 should be used if the FSAR X/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Ref. 29) implements Regulatory Guide 1.145 (Ref. 28) and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96 (Ref. 26) is generally acceptable to the NRC staff for use in determining control room χ/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident χ/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 30). All changes in χ/Q analysis methodology should be reviewed by the NRC staff.	from 1998 – 2002. Meteorological data was collected in accordance with USAR Section 2.3, Meteorology, and RG 1.23. χ/Qs determined using PAVAN-PC (EAB, LPZ, Control Room) and ARCON96 (Control Room). Accident modeled as ground release from RB vent (bounding over elevated stack release). For EAB, fumigation χ/Q is bounded by RB vent χ/Q so is not used. (CA-04-041, Design Inputs 4.10 and 4.11) (CA-04-036, Section 1.0, Assumption 3.1, Sections 6.1.2 and 6.2.2) (CA-04-037, Section 1.0, 5.0, 6.1.1.2, 6.2.1) (USAR Section 2.3; Chem Proc I.6.12, Meteorological/Radiological Data Review) (CA-03-190, Attachment C)

Appendix B

Regulatory Guidance	Basis of Compliance
1. Source Term Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	In compliance. See Section 3 above.
1.1 The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	In compliance. Most limiting case is event over reactor vessel since greater drop height will result in more fuel damage. (CA-04-041, Assumption 3.2 and Design Input 4.3. See also CA-03-190, Section C.5)
1.2 The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	In compliance. Note that alkali metals (Cs and Rb) not included since none would be expected – particulates with infinite DF (see Position 3 below), no airborne parents. (CA-04-041, Assumptions 3.1 and 3.6, and Design Input 4.6)
1.3 The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	In compliance. 95% CsI, 4.85% elemental iodine and 0.15% organic iodide assumed. CsI assumed to instantaneously dissociate with iodine re-evolving as elemental iodine. (CA-04-041, Design Input 4.7)

2. Water Depth 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 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method.	In compliance. DF of 200 used. Minimum fuel depth over reactor core is >46 feet. (CA-04-041, Design Input 4.8) Minimum water depth over fuel pool also >23 feet. See LAR submittal letter, NMC commitment to maintain fuel pool water level IAW procedures.
3. Noble Gases The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	In compliance. DF of 1 for noble gases, infinite DF for particulates assumed. (CA-04-041, Assumptions 3.1 and 3.5, and Design Input 4.8)
4. Fuel Handling Accidents Within The Fuel Building MNGP does not have a separate fuel building.	Not applicable.
5. Fuel Handling Accidents Within Containment For fuel handling accidents postulated to occur within the containment, the following assumptions are acceptable to the NRC staff.	
5.1 If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	Not applicable. No credit taken for containment isolation. (CA-04-041, Assumption 3.7)
5.2 If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	In compliance. No credit taken for containment isolation. (CA-04-041, Assumption 3.7)

5.3 If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.	In compliance. 2-hour period assumed. (CA-04-041, Assumption 3.7)
5.4 A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	In compliance. No credit taken for ESF filter systems. (CA-04-041, Assumptions 3.4, 3.8, 3.9)
5.5 Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	In compliance. No credit taken for dilution or mixing. (CA-04-041, Assumption 3.9)

ENCLOSURE 3

PROPOSED TECHNICAL SPECIFICATION AND BASES CHANGE(S) (MARK-UP)

This enclosure consists of current Technical Specification and Bases page(s) marked up with the proposed change(s). The page(s) included in this enclosure are listed below:

Table 3.2.4
Instrumentation That Initiates Reactor Building Ventilation Isolation
And Standby Gas Treatment System Initiation

Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System	Required Conditions*
1. Low Low Reactor Water Level	≥ -48"	2	2 (Notes 1, 3, 5, 6)	A. or B.
2. High Drywell Pressure	≤ 2 psig	2	2 (Notes 1, 3, 5, 6)	A. or B.
3. Reactor Building Plenum Radiation Monitors	≤ 100 mR/hr	1	1 (Notes 1, 2, 4)	A. or B.
4. Refueling Floor Radiation Monitors	≤ 100 mR/hr	1	1 (Notes 1, 2, 4)	A. or B.

Insert A

Notes:

(1) There shall be two operable or tripped trip systems for each function with two instrument channels per trip system and there shall be one operable or tripped trip system for each function with one instrument channel per trip system.

(2) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied action shall be initiated to:

(a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

(b) Place the plant under the specified required conditions using normal operating procedures.

(3) Need not be operable when primary containment integrity is not required.

(4) One of the two monitors may be bypassed for maintenance and/or testing.

ENCLOSURE 3

Table 3.2.4, Instrumentation That Initiates Reactor Building Ventilation Isolation And Standby Gas Treatment System Initiation

INSERT A:

Function		Applicable Modes or Other Specified Conditions for Which the Function Must be Operable or Operating #
1. Low Low Reactor Water Level		Hot Shutdown, Startup, Run, (a)
2. High Drywell Pressure		Hot Shutdown, Startup, Run
3. Reactor Building Plenum Radiation Monitors		Hot Shutdown, Startup, Run, (a), (b)
4. Refueling Floor Radiation Monitors		Hot Shutdown, Startup, Run, (a), (b)

Shown for information.

Column to be inserted in Table 3.2.4.

Notes: (cont'd)

(5) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied action shall be initiated as follows:

(a) With one required instrument channel inoperable per trip function, place the inoperable channel or trip system in the tripped condition within 12 hours, or

(b) With more than one instrument channel per trip system inoperable, immediately satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

(c) Place the plant under the specified required conditions using normal operating procedures.

(6) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided that at least one other OPERABLE channel in the same trip system is monitoring that parameter.

* Required Conditions when minimum conditions for operation are not satisfied.

A. The reactor building ventilation system isolated and the standby gas treatment system operating.

B. Establish conditions where secondary containment is not required.

Insert B

ENCLOSURE 3

Table 3.2.4, Instrumentation That Initiates Reactor Building Ventilation Isolation And Standby Gas Treatment System Initiation

INSERT B:

Other specified conditions for which the function must be operable or operating:

(a) During operations with the potential for draining the reactor vessel.

(b) During movement of recently irradiated fuel assemblies in secondary containment.

Bases 3.2:

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminate a single operator error before it results in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, and other safety related functions. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is $>7''$ on the instrument. This corresponds to a lower water level inside the shroud at 100% power due to the pressure drop across the dryer/separator. This has been accounted for in the affected transient analysis. This trip initiates closure of Group 2 primary containment isolation valves. Reference Section 7.7.2.2 FSAR. The trip setting provides assurance that the valves will be closed before perforation of the clad occurs even for the maximum break in that line and therefore the setting is adequate. The head cooling valves no longer function to provide head cooling, but continue to provide containment isolation for penetration X-17.

The low low reactor water level instrumentation is set to trip when reactor water level is $\geq -48''$. This trip initiates closure of the Group 1 and Group 3 Primary containment isolation valves, Reference Section 7.7.2.2 FSAR, and also activates the ECC systems and starts the emergency diesel generators.

Included for Information
NO CHANGES

Bases 3. 2 (Continued):

This trip setting level was chosen to be low enough to prevent spurious operation but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference Section 6.2.7 and 14.6.3 FSAR. The instrumentation also covers the full range or spectrum of breaks and meets the above criteria. Reference Section 6.2.7 FSAR.

Insert A

The high drywell pressure instrumentation is a back-up to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2 and Group 3 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 and Group 3 isolation valves include the drywell vent, purge, sump isolation, RWCU, and recirc sample valves.

Insert B

Two pressure switches are provided on the discharge of each of the two core spray pumps and each of the four RHR pumps. Two trip systems are provided in the control logic such that either trip system can permit automatic depressurization. Each trip system consists of two trip logic channels such that both trip logic channels are required to permit a system trip.

Division I core spray and RHR pump discharge pressure permissives will interlock one trip system and Division II permissives will interlock the other trip system. One pressure switch on each pump will interlock one of the trip channels and the other pressure switch will interlock the other trip channel within their respective trip system.

The pump pressure permissive control logic is designed such that no single failure (short or open circuit) will prevent auto-blowdown or allow auto-blowdown when not required. The trip setting for the low pressure ECCS pump permissive for ADS is set such that it is less than the pump discharge pressure when a pump is operating in a full flow condition and also high enough to avoid any condition that results in a discharge pressure permissive when the pumps are not operating.

Venturis are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow,

ENCLOSURE 3

TS Bases for Table 3.2.4, Instrumentation That Initiates Reactor Building Ventilation Isolation And Standby Gas Treatment System Initiation

TS BASES TABLE 3.2.4 - INSERT A:

The low low reactor water level instrumentation is required to be operable in Run, Startup and Hot Shutdown where considerable energy exists in the Reactor Coolant System (RCS); thus there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. During Cold Shutdown and Refueling, the probability and consequences of events are low due to the RCS pressure and temperature limitations under these conditions; thus, this function is not required. In addition, this function is also required to be operable during operations with a potential for draining the reactor vessel because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs.

TS BASES TABLE 3.2.4 - INSERT B:

The high drywell pressure function is required to be operable in Run, Startup and Hot Shutdown where considerable energy exists in the RCS; thus there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This function is required to be operable when primary containment integrity is required. This function is not required during Cold Shutdown and Refueling because the probability and consequences of events are low due to the RCS pressure and temperature limitations under these conditions.

The RBM bypass time delay is set low enough to assure minimum rod movement while upscale trips are bypassed.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the Safety Limit (T.S.2.1.A).

A downscale indication of an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented. The downscale rod blocks assure that there will be proper overlap between the neutron monitoring systems and thus, that adequate coverage is provided for all ranges of reactor operation. The downscale trips are set at 3/125 of full scale.

For effective emergency core cooling for the small pipe break the HPCI or Automatic Pressure Relief system must function since for these breaks, reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria is met. Reference Section 6.2.4 and 6.2.6 FSAR. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Four radiation monitors (two reactor building vent plenum and two refueling floor) are provided which initiate isolation of the reactor building and operation of the standby gas treatment system following a refueling accident. The monitors measure radioactivity in the reactor building ventilation exhaust and on the refueling floor. One upscale trip signal or two downscale/inoperable trip signals, from a pair of monitors performing the same function, will cause the desired action. Trip settings of 100 mR/hr for the reactor building vent plenum monitors and the refueling floor monitors are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation stack but that all the activity is processed by the standby gas treatment system.

The recirculation pump trip is provided to minimize reactor pressure in the highly unlikely event of a plant transient coincident with the failure of all control rods to scram. The rapid flow reduction increases core voiding, a negative reactivity feedback. High pressure sensors initiate the pump trip in the event of an isolation transient. Low level sensors initiate the trip on loss of feedwater (and the resulting MSIV closure). The recirculation pump trip is only required at high reactor power levels, where the safety/relief valves have insufficient capacity to relieve the steam which continues to be generated after reactor isolation in this unlikely postulated event, requiring the trip to be operable only when in the RUN mode is therefore conservative.

$\boxed{\text{Insert C}}$

ENCLOSURE 3

TS Bases for Table 3.2.4, Instrumentation That Initiates Reactor Building Ventilation Isolation And Standby Gas Treatment System Initiation

TS BASES TABLE 3.2.4 - INSERT C:

The reactor building plenum and refueling floor radiation monitors are required to be operable in Run, Startup and Hot Shutdown where considerable energy exists in the Reactor Coolant System (RCS); thus there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. During Cold Shutdown and Refueling, the probability and consequences of events are low due to the RCS pressure and temperature limitations under these conditions; thus, these radiation monitors are not required. In addition, the reactor building plenum and refueling floor radiation monitors are also required to be operable during operations with a potential for draining the reactor vessel and during movement of recently irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. Due to radioactive decay, these radiation monitors are only required to isolate secondary containment during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

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 circuits trains shall be operable at all times when secondary containment integrity is required, except as specified in sections 3.7.B.1.(a) and (b) through (d).

a. After one of the standby gas treatment system circuits trains is made or found to be inoperable 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). (f).

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

b. If both standby gas treatment system circuits trains are not operable, 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 d) e (f).

Insert A

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 iodide 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 adsorbers, 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 iodide removal efficiency.

ENCLOSURE 3

Specification 3.7.B, Standby Gas Treatment System

INSERT A:

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.

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

   d. The fuel cask or irradiated fuel is not being moved within the reactor building.

Insert A

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. 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. Alterations of the

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.

ENCLOSURE 3

Specification 3.7.C, Secondary Containment

INSERT A:

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.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
reactor core operations with a potential for reducing the shutdown margin below that specified in specification 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.	operations with a potential for draining the reactor vessel recently
D. Primary Containment Isolation Valves (PCIVs)	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. Insert B	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.

ENCLOSURE 3

Specification 3.7.C, Secondary Containment

INSERT B:

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.

Bases 3.7 (Continued):

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

as specified in Specification 3.7.C.2.

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~~.

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.

Insert A

ENCLOSURE 3

TS Bases for Specifications 3.7.B, Standby Gas Treatment System and 3.7.C, Secondary Containment

TS BASES SPECIFICATIONS 3.7.B AND 3.7.C - INSERT A:

During Run, Startup and 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 Refueling, the probability and consequences of events are low due to the pressure and temperature limitations in these conditions. Therefore, maintaining the secondary containment and the standby gas treatment system operable are not required during Cold Shutdown or Refueling, except for situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel or during movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, the secondary containment and the standby gas treatment system are only required to be operable during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

Specification 3.7.C.5 provides additional actions when the shutdown margin requirements of specification 3.3.A are not met during Refueling. With the shutdown margin not within limits as demonstrated by analysis during Refueling, the operator must immediately suspend core alterations that could reduce shutdown margin (e.g., insertion of fuel in the core or the withdrawal of control rods). Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and are therefore excluded from the suspended actions.

Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and therefore do not have to be inserted.

3.17 CONTROL ROOM HABITABILITY

Applicability:

Applies to the control room ventilation system equipment necessary to maintain habitability.

Objectives:

To assure the control room is habitable both under normal and accident conditions.

Specification:

A. Control Room Ventilation System

1. Except as specified in 3.17.A.2 and 3.17.A.3 below, both trains of the control room ventilation system shall be operable, whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, or during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel.

2.a With one control room ventilation train inoperable, restore the inoperable train to operable status within 30 days.

2.b If 2.a is not met, then be in hot shutdown within the next 12 hours following the 30 days and in cold shutdown within 24 hours following the 12 hours.

2.c If 2.a is not met during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel then immediately place the operable control room ventilation train in operation or immediately suspend these activities.

4.17 CONTROL ROOM HABITABILITY

Applicability:

Applies to the periodic testing requirements of systems required to maintain control room habitability.

Objectives:

To verify the operability of equipment related to control room habitability.

Specification:

A. Control Room Ventilation System

1. Once per 12 hours check control room temperature.

3.0 LIMITING CONDITIONS FOR OPERATION

3.a With both control room ventilation trains inoperable, restore at least one train to operable status within 24 hours.

3.b If 3.a is not met, then be in hot shutdown within the next 12 hours and in cold shutdown within 24 hours following the 12 hours.

3.c If 3.a is not met during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel then immediately suspend these activities.

B. Control Room Emergency Filtration System

1. Except as specified in 3.17.B.1.a through d below, two control room emergency filtration system filter trains shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, or during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel.

4.0 SURVEILLANCE REQUIREMENTS

With both control room ventilation trains inoperable,

B. Control Room Emergency Filtration System

1. At least once per month, initiate from the control room 1000 cfm ($\pm$10%) flow through both trains of the emergency filtration treatment system. The system shall operate for at least 10 hours with the heaters operable.

recently

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
a. When one control room emergency filtration system filter train is made or found to be inoperable for any reason, restore the inoperable train to operable status within seven days or be in hot shutdown within the next 12 hours following the seven days and either reduce the reactor coolant temperature to below 212°F or initiate and maintain the operable emergency filtration system filter train in the pressurization mode within the following 24 hours.	
b. When both filter trains of the control room emergency filtration system are inoperable, restore at least one train to operable status within 24 hours or be in hot shutdown within the next 12 hours following the 24 hours and reduce the reactor coolant water temperature to below 212°F within the following 24 hours.	
c. With one control room emergency filtration system filter train inoperable during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel, restore the inoperable train to operable status within 7 days or immediately after the 7 days initiate and maintain the operable emergency filtration system filter train in the pressurization mode or immediately suspend these activities.	recently
d. With both control room emergency filtration system filter trains inoperable during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel, immediately suspend these activities.	

Bases 3.17:

A. Control Room Ventilation System

The Control Room Ventilation System provides air conditioning and heating as required to maintain a suitable environment in the main control room and portions of the first and second floors of the Emergency Filtration Train (EFT) building. The system is designed to maintain a nominal temperature of 78°F dry bulb in the main control room in the summer and a nominal temperature of 72°F in the winter. During normal operation, the CRV system recirculates the air in the control room envelope as needed. During a high radiation event, the Control Room Ventilation System continues to operate, and the Control Room Emergency Filtration Train system will start automatically to pressurize the control room protective envelope. The Emergency Filtration Train system can also be started manually.

All toxic substances which are stored onsite or stored/shipped within a 5 mile radius of the plant have been analyzed for their effect on the control room operators. It has been concluded that the operators will have at least two minutes to don breathing apparatus before incapacitation. Protection for toxic chemicals is provided through operator training, self-contained breathing apparatus (SBCAs) and the Control Room Breathing Air Supply.

B. Control Room Emergency Filtration System

(discussed below).

The Control Room Emergency Filtration System assures that the control room operators will be adequately protected against the effects of radioactive leakage which may by-pass secondary containment following a loss of coolant accident, steam line break accident or fuel handling accident. The system is designed to slightly pressurize the control room on a radiation signal in the ventilation air. Two completely redundant trains are provided.

Each train has a filter unit consisting of a prefilter, HEPA filters, and charcoal adsorbers. The HEPA filters remove particulates from the Control Room pressurizing air and prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to remove any radioiodines from the pressurizing air. The verification of performance parameters combined with the qualification testing conducted on new filters and adsorbers provide a high level of assurance that the Emergency Filtration System will perform as predicted in reducing doses to plant personnel below those levels stated in Criterion 19 of Appendix A to 10 CFR 50. The allowable penetration for the laboratory test is based on a conservative adsorber efficiency of 99% and a safety factor of ≥ 2.

Dose calculations have been performed for the Control Room Emergency Filtration System which show that, assuming 85% standby gas treatment system overall removal efficiency and 98% control room emergency filtration system overall removal efficiency and radioiodine plateout, whole body and organ doses remain within NRC guidelines.

Insert A

ENCLOSURE 3

TS Bases for Specifications 3.17.B, Control Room Emergency Filtration System

TS BASES SPECIFICATIONS 3.17.B - INSERT A:

During Run, Startup and Hot Shutdown, the control room emergency filtration system must be operable to control operator exposure during and following a design basis accident (DBA), since the DBA could lead to a fission product release.

During Cold Shutdown and Refueling, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these conditions. Therefore, maintaining the control room emergency filtration system operable is not required during Cold Shutdown or Refueling, except for situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel or during movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, the control room emergency filtration system is only required to be operable during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ENCLOSURE 4

PROPOSED TECHNICAL SPECIFICATION AND BASES CHANGE(S) (RETYPED)

This enclosure consists of the revised Technical Specification and Bases page(s) that incorporate the proposed change(s). The page(s) included in this enclosure are listed below:

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Table 3.2.4 Instrumentation That Initiates Reactor Building Ventilation Isolation And Standby Gas Treatment System Initiation					
Function	Trip Settings	Applicable Modes or Other Specified Conditions for Which the Function Must be Operable or Operating #	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System	Required Conditions*
1. Low Low Reactor Water Level	≥-48"	Hot Shutdown, Startup, Run, (a)	2	2 (Notes 1, 3, 5, 6)	A. or B.
2. High Drywell Pressure	≤2 psig	Hot Shutdown, Startup, Run	2	2 (Notes 1, 3, 5, 6)	A. or B.
3. Reactor Building Plenum Radiation Monitors	≤100 mR/hr	Hot Shutdown, Startup, Run, (a), (b)	1	1 (Notes 1, 2, 4)	A. or B.
4. Refueling Floor Radiation Monitors	≤100 mR/hr	Hot Shutdown, Startup, Run, (a), (b)	1	1 (Notes 1, 2, 4)	A. or B.

<u>Notes</u>:

(1) There shall be two operable or tripped trip systems for each function with two instrument channels per trip system and there shall be one operable or tripped trip system for each function with one instrument channel per trip system.

(2) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied action shall be initiated to:

(a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

(b) Place the plant under the specified required conditions using normal operating procedures.

(3) Need not be operable when primary containment integrity is not required.

Notes: (cont'd)

(4) One of the two monitors may be bypassed for maintenance and/or testing.

(5) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied action shall be initiated as follows:

(a) With one required instrument channel inoperable per trip function, place the inoperable channel or trip system in the tripped condition within 12 hours, or

(b) With more than one instrument channel per trip system inoperable, immediately satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

(c) Place the plant under the specified required conditions using normal operating procedures.

(6) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided that at least one other OPERABLE channel in the same trip system is monitoring that parameter.

* Required Conditions when minimum conditions for operation are not satisfied.

A. The reactor building ventilation system isolated and the standby gas treatment system operating.

B. Establish conditions where secondary containment is not required.

\# Other specified conditions for which the function must be operable or operating:

(a) During operations with the potential for draining the reactor vessel.

(b) During movement of recently irradiated fuel assemblies in secondary containment.

Bases 3. 2 (Continued):

This trip setting level was chosen to be low enough to prevent spurious operation but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference Section 6.2.7 and 14.6.3 FSAR. The instrumentation also covers the full range or spectrum of breaks and meets the above criteria. Reference Section 6.2.7 FSAR.

The low low reactor water level instrumentation is required to be operable in Run, Startup and Hot Shutdown where considerable energy exists in the Reactor Coolant System (RCS); thus there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. During Cold Shutdown and Refueling, the probability and consequences of events are low due to the RCS pressure and temperature limitations under these conditions; thus, this function is not required. In addition, this function is also required to be operable during operations with a potential for draining the reactor vessel because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs.

The high drywell pressure instrumentation is a back-up to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2 and Group 3 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 and Group 3 isolation valves include the drywell vent, purge, sump isolation, RWCU, and recirc sample valves.

The high drywell pressure function is required to be operable in Run, Startup and Hot Shutdown where considerable energy exists in the RCS; thus there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This function is only required to be operable when primary containment integrity is required. This function is not required during Cold Shutdown and Refueling because the probability and consequences of events are low due to the RCS pressure and temperature limitations under these conditions.

Two pressure switches are provided on the discharge of each of the two core spray pumps and each of the four RHR pumps. Two trip systems are provided in the control logic such that either trip system can permit automatic depressurization. Each trip system consists of two trip logic channels such that both trip logic channels are required to permit a system trip.

Division I core spray and RHR pump discharge pressure permissives will interlock one trip system and Division II permissives will interlock the other trip system. One pressure switch on each pump will interlock one of the trip channels and the other pressure switch will interlock the other trip channel within their respective trip system.

Bases 3.2 (Continued):

The pump pressure permissive control logic is designed such that no single failure (short or open circuit) will prevent auto-blowdown or allow auto-blowdown when not required. The trip setting for the low pressure ECCS pump permissive for ADS is set such that it is less than the pump discharge pressure when a pump is operating in a full flow condition and also high enough to avoid any condition that results in a discharge pressure permissive when the pumps are not operating.

Venturis are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 valves are closed. For the worst case accident, main steamline break outside the drywell, this trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steamline valve closure, limit the mass inventory loss such that fuel is not uncovered, fuel clad temperatures remain less than 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Sections 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a back-up to high steam flow instrumentation discussed above, and for small breaks with the resultant small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (USAR Section 7.6.3.2.4-4). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.A is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing power to $<25\%$ RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detection low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

Bases 3.2 (Continued):

The Main Steam Line Pressure - Low Function is only required to be OPERABLE during power operation since this is when the assumed transient can occur (USAR Section 7.6.3.2.4-4).

This Function isolates the Group 1 valves.

The RWCU high flow and temperature instrumentation is provided to detect a break in the RWCU piping. Tripping of this instrumentation results in actuation of the RWCU isolation valves, i.e., Group 3 valves. The trip settings have been established so that the radiological consequences of a high energy line break in this system are bounded by a break in the main steam system. The recirc sample isolation valves, which receive a Group 1 isolation signal, also receive a redundant Group 3 isolation signal.

The HPCI and RCIC high flow and temperature instrumentation is provided to detect a break in the HPCI or RCIC piping. The trip settings of 200°F and approximately 300% of HPCI and RCIC design steam flow are such that the core will not be uncovered and the radiological consequences are bounded by the main steam line break accident. The HPCI and RCIC low steam line pressure instrumentation protects the HPCI and RCIC turbines when system operation is no longer useful or possible. Tripping of the high flow, high temperature, or low steam line pressure instrumentation results in actuation of the HPCI or RCIC steam supply valves; i.e., a Group 4 or Group 5 isolation.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR remains above the Safety Limit (T.S.2.1.A). The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, eight IRM's, or four SRM's will result in a rod block. The minimum instrument channel requirements for the IRM and RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. See Section 7.3 FSAR.

The APRM rod block trip is referenced to flow and prevents operation significantly above the licensing basis power level especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The operator will set the APRM rod block trip settings no greater than that stated in Table 3.2.3. However, the actual setpoint can be as much as 3% greater than that stated in Table 3.2.3 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 39.

Bases 3.2 (Continued):

The RBM provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is referenced to power. This power signal is provided by the APRMs. A statistical analysis of many single control rod withdrawal errors has been performed and at the 95/95 level the results show that with the specified trip settings, rod withdrawal is blocked at MCPRs greater than the Safety Limit, thus allowing adequate margin. This analysis assumes a steady state MCPR prior to the postulated rod withdrawal error. The RBM functions are required when core thermal power is greater than 30% and a Limiting Control Rod Pattern exists. When both RBM channels are operating either channel will assure required withdrawal blocks occur even assuming a single failure of one channel. With one RBM channel inoperable for no more than 24 hours, testing of the RBM prior to withdrawal of control rods assures that improper control rod withdrawal will be blocked. Requiring at least half of the normal LPRM inputs to be operable assures that the RBM response will be adequate to protect against rod withdrawal errors, as shown by a statistical failure analysis.

The RBM bypass time delay is set low enough to assure minimum rod movement while upscale trips are bypassed.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the Safety Limit (T.S.2.1.A).

A downscale indication of an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented. The downscale rod blocks assure that there will be proper overlap between the neutron monitoring systems and thus, that adequate coverage is provided for all ranges of reactor operation. The downscale trips are set at 3/125 of full scale.

For effective emergency core cooling for the small pipe break the HPCI or Automatic Pressure Relief system must function since for these breaks, reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria is met. Reference Section 6.2.4 and 6.2.6 FSAR. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Four radiation monitors (two reactor building vent plenum and two refueling floor) are provided which initiate isolation of the reactor building and operation of the standby gas treatment system following a refueling accident. The monitors measure radioactivity in the reactor building ventilation exhaust and on the refueling floor. One upscale trip signal or two downscale/inoperable trip signals, from a pair of monitors performing the same function, will cause the desired action. Trip settings of 100 mR/hr for the reactor building vent plenum monitors and the refueling floor monitors are based upon initiating normal ventilation isolation and standby gas treatment system operation.

Bases 3.2 (Continued):

The reactor building plenum and refueling floor radiation monitors are required to be operable in Run, Startup and Hot Shutdown where considerable energy exists in the Reactor Coolant System (RCS); thus there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. During Cold Shutdown and Refueling, the probability and consequences of events are low due to the RCS pressure and temperature limitations under these conditions; thus, these radiation monitors are not required. In addition, the reactor building plenum and refueling floor radiation monitors are also required to be operable during operations with a potential for draining the reactor vessel and during movement of recently irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. Due to radioactive decay, these radiation monitors are only required to isolate secondary containment during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24-hours).

The recirculation pump trip is provided to minimize reactor pressure in the highly unlikely event of a plant transient coincident with the failure of all control rods to scram. The rapid flow reduction increases core voiding, a negative reactivity feedback. High pressure sensors initiate the pump trip in the event of an isolation transient. Low level sensors initiate the trip on loss of feedwater (and the resulting MSIV closure). The recirculation pump trip is only required at high reactor power levels, where the safety/relief valves have insufficient capacity to relieve the steam which continues to be generated after reactor isolation in this unlikely postulated event, requiring the trip to be operable only when in the RUN mode is therefore conservative.

The ATWS high reactor pressure and low-low water level logic also initiates the Alternate Rod Injection System. Two solenoid valves are installed in the scram air header upstream of the hydraulic control units. Each of the two trip systems energizes a valve to vent the header and causes rod insertion. This greatly reduces the long term consequences of an ATWS event.

Voltage sensing relays are provided on the safeguards bus to transfer the bus to an alternate source when a loss of voltage condition or a degraded voltage condition is sensed. On loss of voltage, the voltage sensing relays trip immediately and energize auxiliary relays that control the bus transfer sequence. The transfer on degraded voltage has a time delay to prevent transfer during the starting of large loads. The degraded voltage setpoint corresponds to the minimum acceptable safeguards bus voltage for a steady state LOCA load that maintains adequate voltage at the 480V essential MCCS. An allowance for relay tolerance is included.

Bases 3.2 (Continued):

Safety/relief valve low-low set logic is provided to prevent any safety/relief valve from opening when there is an elevated water leg in the respective discharge line. A high water leg is formed immediately following valve closure due to the vacuum formed when steam condenses in the line. If the valve reopens before the discharge line vacuum breakers act to return water level to normal, water clearing thrust loads on the discharge line may exceed their design limit. The logic reduces the opening setpoint and increases the blowdown range of three non-APRS valves following a scram. A 15-second interval between subsequent valve actuations is provided assuming one valve fails to open and instrumentation drift has caused the nominal 80-psi blowdown range to be reduced to 60 psi. Maximum water leg clearing time has been calculated to be less than 6 seconds for the Monticello design. Inhibit timers are provided for each valve to prevent the valve from being manually opened less than 10 seconds following valve closure. Valve opening is sensed by pressure switches in the valve discharge line. Each valve is provided with two trip, or actuation, systems. Each system is provided with two channels of instrumentation for each of the above described functions. A two-out-of-two-once logic scheme ensures that no single failure will defeat the low-low set function and no single failure will cause spurious operation of a safety/relief valve. Allowable deviations are provided for each specified instrument setpoint. Valve operation within the specified allowable deviations provide assurance that subsequent safety/relief valve actuations are sufficiently spaced to allow for discharge line water leg clearing.

Control room habitability protection instrumentation assures that the control room operators will be adequately protected against the effects of accidental releases of radioactive leakage which may bypass secondary containment following a loss of coolant accident or radioactive releases from a steam line break accident, thus assuring that the Monticello Nuclear Generating Plant can be operated or shutdown safely.

Although the operator will set the setpoints within the trip settings specified in Tables 3.2.1 through 3.2.9, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. The deviations could be caused by inherent instrument error, operator setting error, drift of the set point, etc. Therefore, these deviations have been accounted for in the various transient analyses.

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 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) through (f).

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, 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 (f). 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.	

3.0 LIMITING CONDITIONS FOR OPERATION

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

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

   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.

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 $\leq 4{,}000$ 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

4. 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. Operations with a potential for reducing the shutdown margin below that specified in specification 3.3.A, operations with a potential for draining the reactor vessel, and 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.

5. With the shutdown margin below the limit specified in specification 3.3.A, immediately suspend core alternations 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.

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 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 train 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 and 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 Refueling, the probability and consequences of events are low due to the pressure and temperature limitations in these conditions. Therefore, maintaining the secondary containment and the standby gas treatment system operable are not required during Cold Shutdown or Refueling, except for situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel or during movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, the secondary containment and the standby gas treatment system are only required to be operable during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

Specification 3.7.C.5 provides additional actions when the shutdown margin requirements of specification 3.3.A are not met during Refueling. With the shutdown margin not within limits as demonstrated by analysis during Refueling, the operator must immediately suspend core alterations that could reduce SDM (e.g., insertion of fuel in the core or the withdrawal of control rods). Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and are therefore excluded from the suspended actions.

Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and therefore do not have to be inserted.

While only a small amount of particulates are released from the primary containment as a result of the loss of coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates. Laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. The allowable penetration for the laboratory test is based on 90% adsorber efficiency assumed in the off-site dose analysis and a safety factor of ≥ 2. Operation of the standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

D. Primary Containment Isolation Valves

The function of the Primary Containment Isolation Valves (PCIVs), in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents to within limits. The PCIVs help ensure that an adequate primary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. Therefore, Technical Specifications requirements provide assurance that primary containment function assumed in the safety analysis will be maintained. These valves are either passive or active (automatic). Manual valves, deactivated automatic valves (including remote manual valves) secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices.

Closure of one of the valves in each line would be sufficient to maintain the integrity of the Primary Containment. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the Primary Containment isolation valves are discussed in Section 5.2 of the USAR. A listing of all Primary Containment automatic isolation valves including maximum operating time is given in USAR Table 5.2-3b.

The Technical Specifications are modified by a footnote allowing penetration flow path(s) to be unisolated intermittently under Operations Committee approved administrative controls. These controls consist of stationing a dedicated operator at the controls of the valve who is in constant communication with the control room. In this way, the penetration can be rapidly isolated when a need for the primary containment isolation is indicated.

With one or more penetration flow paths with one PCIV inoperable, the affected penetration must be returned to operable status or isolated within 4 hours (8 hours for MSIVs and 72 hours for Excess Flow Check Valves (EFCVs)). The 4 hour completion time is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment. The 8 hour completion time for MSIVs allows a period of time to restore the MSIVs to operable status given the fact that MSIV closure will result in a potential for plant shutdown. The 72 hour completion time for EFCVs is reasonable considering the instrument and the small diameter of the penetration piping combined with the ability of the penetration to act as an isolation boundary. With one or more penetrations with two PCIVs inoperable, either the inoperable PCIVs must be returned to operable status or the affected penetration flow path must be isolated within 1 hour.

Specification 3.7.D.3 requires the containment to be purged and vented through the standby gas treatment system except during inerting and deinerting operations. This provides for iodine and particulate removal from the containment atmosphere. Use of the 2-inch flow path prevents damage to the standby gas treatment system in the event of a loss of coolant accident during purging or venting. Use of the reactor building plenum and vent flow path for inerting and deinerting operations permits the control room operators to monitor the activity level of the resulting effluent by use of the Reactor Building Vent Wide Range Gas Monitors.

E. Combustible Gas Control System

The function of the Combustible Gas Control System (CGCS) is to maintain oxygen concentrations in the post-accident containment atmosphere below combustible concentrations. Oxygen may be generated in the hours following a loss of coolant accident from radiolysis of reactor coolant.

Bases 3.7 (Continued):

The Technical Specifications limit oxygen concentrations during operation to less than four percent by volume during operation. The maintenance of an inert atmosphere during operation precludes the build-up of a combustible mixture due to a fuel metal-water reaction. The other potential mechanism for generation of combustible mixtures is radiolysis of coolant which has been found to be small.

A special report is required to be submitted to the Commission to outline CGCS equipment failures and corrective actions to be taken if inoperability of one train exceeds thirty days. In addition, if both trains are inoperable for more than 30 days, the plant is required to shutdown until repairs can be made.

3.17 CONTROL ROOM HABITABILITY

Applicability:

Applies to the control room ventilation system equipment necessary to maintain habitability.

Objectives:

To assure the control room is habitable both under normal and accident conditions.

Specification:

A. Control Room Ventilation System

1. Except as specified in 3.17.A.2 and 3.17.A.3 below, both trains of the control room ventilation system shall be operable, whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, or during movement of irradiated fuel assemblies in the secondary containment or activities having the potential for draining the reactor vessel.

2.a With one control room ventilation train inoperable, restore the inoperable train to operable status within 30 days.

2.b If 2.a is not met, then be in hot shutdown within the next 12 hours following the 30 days and in cold shutdown within 24 hours following the 12 hours.

4.17 CONTROL ROOM HABITABILITY

Applicability:

Applies to the periodic testing requirements of systems required to maintain control room habitability.

Objectives:

To verify the operability of equipment related to control room habitability.

Specification:

A. Control Room Ventilation System

1. Once per 12 hours check control room temperature.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2.c If 2.a is not met during movement of irradiated fuel assemblies in the secondary containment or activities having the potential for draining the reactor vessel then immediately place the operable control room ventilation train in operation or immediately suspend these activities. 3.a With both control room ventilation trains inoperable, restore at least one train to operable status within 24 hours. 3.b If 3.a is not met, then be in hot shutdown within the next 12 hours and in cold shutdown within 24 hours following the 12 hours. 3.c With both control room ventilation trains inoperable, during movement of irradiated fuel assemblies in the secondary containment or activities having the potential for draining the reactor vessel then immediately suspend these activities. B. Control Room Emergency Filtration System 1. Except as specified in 3.17.B.1.a through d below, two control room emergency filtration system filter trains shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, or during movement of recently irradiated fuel assemblies in the secondary containment, or activities having the potential for draining the reactor vessel.	B. Control Room Emergency Filtration System 1. At least once per month, initiate from the control room 1000 cfm (±10%) flow through both trains of the emergency filtration treatment system. The system shall operate for at least 10 hours with the heaters operable.

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3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
a. When one control room emergency filtration system filter train is made or found to be inoperable for any reason, restore the inoperable train to operable status within seven days or be in hot shutdown within the next 12 hours following the seven days and either reduce the reactor coolant temperature to below 212°F or initiate and maintain the operable emergency filtration system filter train in the pressurization mode within the following 24 hours. b. When both filter trains of the control room emergency filtration system are inoperable, restore at least one train to operable status within 24 hours or be in hot shutdown within the next 12 hours following the 24 hours and reduce the reactor coolant water temperature to below 212°F within the following 24 hours. c. With one control room emergency filtration system filter train inoperable during movement of recently irradiated fuel assemblies in the secondary containment, or activities having the potential for draining the reactor vessel, restore the inoperable train to operable status within 7 days or immediately after the 7 days initiate and maintain the operable emergency filtration system filter train in the pressurization mode or immediately suspend these activities. d. With both control room emergency filtration system filter trains inoperable during movement of recently irradiated fuel assemblies in the secondary containment, or activities having the potential for draining the reactor vessel, immediately suspend these activities.	

Bases 3.17:

A. Control Room Ventilation System

The Control Room Ventilation System provides air conditioning and heating as required to maintain a suitable environment in the main control room and portions of the first and second floors of the Emergency Filtration Train (EFT) building. The system is designed to maintain a nominal temperature of 78°F dry bulb in the main control room in the summer and a nominal temperature of 72°F in the winter. During normal operation, the CRV system recirculates the air in the control room envelope as needed. During a high radiation event, the Control Room Ventilation System continues to operate, and the Control Room Emergency Filtration Train system will start automatically to pressurize the control room protective envelope. The Emergency Filtration Train system can also be started manually.

All toxic substances which are stored onsite or stored/shipped within a 5 mile radius of the plant have been analyzed for their effect on the control room operators. It has been concluded that the operators will have at least two minutes to don breathing apparatus before incapacitation. Protection for toxic chemicals is provided through operator training, self-contained breathing apparatus (SBCAs) and the Control Room Breathing Air Supply.

B. Control Room Emergency Filtration System

The Control Room Emergency Filtration System assures that the control room operators will be adequately protected against the effects of radioactive leakage which may by-pass secondary containment following a loss of coolant accident, steam line break accident or fuel handling accident (discussed below). The system is designed to slightly pressurize the control room on a radiation signal in the ventilation air. Two completely redundant trains are provided.

During Run, Startup and Hot Shutdown, the control room emergency filtration system must be operable to control operator exposure during and following a design basis accident (DBA), since the DBA could lead to a fission product release.

During Cold Shutdown and Refueling, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these conditions. Therefore, maintaining the control room emergency filtration system operable is not required during Cold Shutdown or Refueling, except for situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel or during movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, the control room emergency filtration system is only required to be operable during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

Bases 3.17 (Continued):

Each train has a filter unit consisting of a prefilter, HEPA filters, and charcoal adsorbers. The HEPA filters remove particulates from the Control Room pressurizing air and prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to remove any radioiodines from the pressurizing air. The verification of performance parameters combined with the qualification testing conducted on new filters and adsorbers provide a high level of assurance that the Emergency Filtration System will perform as predicted in reducing doses to plant personnel below those levels stated in Criterion 19 of Appendix A to 10 CFR 50. The allowable penetration for the laboratory test is based on a conservative adsorber efficiency of 99% and a safety factor of ≥ 2.

Dose calculations have been performed for the Control Room Emergency Filtration System which show that, assuming 85% standby gas treatment system overall removal efficiency and 98% control room emergency filtration system overall removal efficiency and radioiodine plateout, whole body and organ doses remain within NRC guidelines.

ENCLOSURE 5

AFFIDAVIT FOR PROPRIETARY CALCULATIONS CONCERNING

Monticello Nuclear Generating Plant Alternate Source Term Implementation Project Selective Implementation for the FHA Event

by Applied Analysis Corporation

APPLIED ANALYSIS CORP.

AFFIDAVIT

I, Juan M. Cajigas, being duly sworn, depose and state as follows:

1) I am the President of Applied Analysis Corp. ("AAC") and have reviewed the information described in paragraph (2) and sought to be withheld.

2) The information sought to be withheld is contained in the AAC proprietary calculations listed in Attachment A.

3) In making this application for withholding of proprietary information, AAC relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9. 17(a)(4) and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission. 975F2d871 (DC Cir. 1992), and Public Citizen Health Research *Group* v. FDA, 704F2d1280 (DC Cir. 1983).

4) Some examples of categories of information which fit into the definition of proprietary information are:

   a) Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by AAC competitors without license from AAC constitutes a competitive economic advantage over other companies;

   b) Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, preparation, assurance of quality, or licensing of a similar service;

   c) Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of AAC, its customers, or its suppliers;

   d) Information which reveals aspects of past, present, or future AAC customer-funded development plans and programs, of potential commercial value to AAC;

   e) Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

   The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a and (4)b, above.

5) The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by AAC, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within AAC is limited on a "need to know" basis.

7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside AAC are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed methods and processes, which AAC has developed for the preparation of detailed safety analyses in support of the design and licensing of nuclear facilities.

   The development of these methods and processes was achieved at a significant cost to AAC and derived from company experience that constitutes a major AAC asset.

9) Disclosure of the information sought to be withheld is likely to cause substantial harm to AAC's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of AAC's nuclear safety analysis and technology base, and its commercial value includes development of the expertise to determine and apply the appropriate evaluation processes.

   The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

   AAC's competitive advantage will be lost if its competitors are able to use the results of the AAC experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar results and conclusions.

   The value of this information to AAC would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive AAC of the opportunity to exercise its competitive advantage to seek an adequate return on its investment in developing these analytical processes.

STATE OF PENNSYLVANIA)
) ss:
COUNTY OF BERKS)

Juan M. Cajigas, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Reading, Pennsylvania, this 26th day of APRIL 2004.

Juan M. Cajigas
Applied Analysis Corp.

Subscribed and sworn before me this 26th day of April 2004.

Notary Public, State of Pennsylvania

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Michael J. Hyneman, Notary Public
Cumru Twp., Berks County
My Commission Expires Oct. 6, 2007
Member, Pennsylvania Association Of Notaries

ATTACHMENT A

1. AAC Calculation No. MNGP-001, Rev. 1, "MNGP AST - Offsite Post-accident Atmospheric Dispersion Analysis" and attachments.
2. AAC Calculation No. MNGP-002, Rev. 2, "MNGP AST - CR/TSC Post-accident Atmospheric Dispersion Analysis" and attachments.
3. AAC Calculation No. MNGP-006, Rev. 0, "MNGP AST - FHA Radiological Consequence Analysis" and attachments.