December 7, 2006

Mr. John T. Conway
Site Vice President
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT RE: FULL-SCOPE IMPLEMENTATION OF THE ALTERNATIVE SOURCE TERM METHODOLOGY (TAC NO. MC8971)

Dear Mr. Conway:

The Commission has issued the enclosed Amendment No. 148 to Renewed Facility Operating License No. DPR-22 for Monticello Nuclear Generating Plant (MNGP), in response to your application dated September 15, 2005, as supplemented on April 13, August 21, and August 22, 2006.

The amendment revised the MNGP licensing basis by implementing the full-scope alternative source term methodology, resulting in revision of portions of the Technical Specifications to reflect this licensing basis change.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Peter S. Tam, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 148 to DPR-22
2. Safety Evaluation

cc w/encls: See next page

Mr. John T. Conway December 7, 2006
Site Vice President
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT RE: FULL-SCOPE IMPLEMENTATION OF THE ALTERNATIVE SOURCE TERM METHODOLOGY (TAC NO. MC8971)

Dear Mr. Conway:

The Commission has issued the enclosed Amendment No. 148 to Renewed Facility Operating License No. DPR-22 for Monticello Nuclear Generating Plant (MNGP), in response to your application dated September 15, 2005, as supplemented on April 13, August 21, and August 22, 2006.

The amendment revised the MNGP licensing basis by implementing the full-scope alternative source term methodology, resulting in revision of portions of the Technical Specifications to reflect this licensing basis change.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Peter S. Tam, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 148 to DPR-22
2. Safety Evaluation
(Note: According to Toni Harris Amendment Date is 12/7/06)
cc w/encls: See next page

DISTRIBUTION:	RidsNrrDraACVB	RidsNrrDraAadb	
PUBLIC	LPLIII-1 R/F	RidsNrrPMPTam	RidsNrrLATHarris
RidsNrrDorlLpl3-1	RidsOgcRp	RidsAcrsAcnwMailCenter	MHart
GHill(2)	EForrest	RidsRgn3MailCenter	RidsNrrDirsItsb
LBrown	BLee	RLobel	KParczewski
STammara			

Package Accession Number: **ML062850049**
Amendment Accession Number: **ML062790015**
Tech. Spec. pages Accession Number: **ML063420517**

OFFICE	LPL3-1/PM	LPL3-1/ LA	AADB/BC	ACVB/BC	CSGB/BC	ITSB/BC	OGC	LPL3-1/BC
NAME	PTam	THarris	MKotzalas*	RDennig*	AHiser*	TTjader forTKobetz	MZobler	LRaghavan
DATE	10/13/06	10/12/06	9/15/06*	08/11/06*	8/3/06*	11/17/06	11/13/06	11/28/06

OFFICIAL RECORD COPY

*Safety evaluation input transmitted by memo on the date indicated.

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 148
License No. DPR-22

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated September 15, 2005, as supplemented on April 13, August 21, and August 22, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Facility Operating License No. DPR-22 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 148 are hereby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented before startup from the 2007 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 7, 2006

ATTACHMENT TO OPERATING LICENSE AMENDMENT NO. 148

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following page of Renewed Facility Operating License DPR-22 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

3	3

Replace the following pages of Appendix A (Technical Specifications) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
1.1-2	1.1-2
3.1.7-1	3.1.7-1
3.3.7.1-1	3.3.7.1-1
3.3.7.1-2	3.3.7.1-2
--	3.3.7.1-3
--	3.3.7.2-1
--	3.3.7.2-2
3.4.6-1	3.4.6-1
3.4.6-2	3.4.6-2
3.6.1.3-1	3.6.1.3-1
3.6.1.3-3	3.6.1.3-3
3.6.1.3-5	3.6.1.3-5
3.6.1.3-6	3.6.1.3-6
3.6.1.3-8	3.6.1.3-8
3.7.5-1	3.7.5-1
3.7.5-2	3.7.5-2
3.8.2-1	3.8.2-1
3.8.2-2	3.8.2-2
3.8.5-1	3.8.5-1
3.8.8-1	3.8.8-1
5.5-10	5.5-10
5.5-11	5.5-11

2. Pursuant to the Act and 10 CFR Part 70, NMC to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operations, as described in the Final Safety Analysis Report, as supplemented and amended, and the licensee's filings dated August 16, 1974 (those portions dealing with handling of reactor fuel) and August 17, 1977 (those portions dealing with fuel assembly storage capacity);

3. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NMC to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

4. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NMC to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

5. Pursuant to the Act and 10 CFR Parts 30 and 70, NMC to possess, but not separate, such byproduct and special nuclear material as may be produced by operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission, now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. Maximum Power Level

   NMC is authorized to operate the facility at steady state reactor core power levels not in excess of 1775 megawatts (thermal).

2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 148, are hereby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications.

3. Physical Protection

   NMC shall implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR AMENDMENT NO. 148 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-22

NUCLEAR MANAGEMENT COMPANY, LLC

MONTICELLO NUCLEAR GENERATING PLANT (MNGP)

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated September 15, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML052640366), as supplemented by letters dated April 13 (Accession No. ML061310408), August 21 (Accession No. ML062350046), and August 22 (Accession No. ML062410341), 2006, Nuclear Management Company, LLC (the licensee) submitted an application for amendment in accordance with Title 10 of the *Code of Federal Regulations,* Part 50.67 (10 CFR 50.67), "Accident Source Term." The licensee proposed to change the MNGP licensing basis by implementing full-scope the alternative source term (AST) methodology, leading to revision of portions of the Technical Specifications (TSs) to reflect this change in the licensing basis. The licensee's September 15, 2005, letter, also requested a specific exemption from 10 CFR 50.54(o) and the requirements of Sections III.A and III.B of 10 CFR Part 50, Appendix J, Option B, for MNGP; the specific exemption is addressed by separate correspondence dated the same day as this safety evaluation.

On March 6, 2006, the Nuclear Regulatory Commission (NRC) staff met with the licensee's personnel to discuss the subject application for amendment. The summary of this meeting is dated March 21, 2006 (Accession No. ML060670035).

The licensee's supplements cited above provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 14, 2006 (71 FR 7808).

2.0 REGULATORY EVALUATION

This safety evaluation addresses the impact of AST methodology to the MNGP licensing basis, in particular, the radiological consequences of various postulated design-basis accidents. The regulatory requirements and guidance on which the NRC staff based its acceptance are set for as follows:

(1) Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident source term," and the associated guidance in:

(a) Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” dated July 2000;

(b) NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 15.0.1, “Radiological Consequence Analysis Using Alternative Source Terms,” dated July 2000.

(2) Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 of Appendix A, General Design Criterion 19 (GDC-19)[1], "Control Room," and the associated guidance in:

(a) RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995;

(b) Section 6.4 of the SRP, "Control Room Habitability System."

(3) Technical Specification Task Force Traveler (TSTF) 51, regarding containment requirements during handling of irradiated fuel and core alterations, Revision 2, approved by the U.S. Nuclear Regulatory Commission (NRC) on October 13, 1999.

(4) RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.”

(5) RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants.”

(6) SRP Section 2.3.4, “Short-Term Diffusion Estimates for Accidental Atmospheric Releases.”

The NRC staff also considered relevant licensing basis information in the MNGP Updated Safety Analysis Report (USAR).

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences of Design-Basis Accidents Using AST Methodology

The NRC staff reviewed the licensee’s regulatory and technical analyses as related to the radiological consequences of design-basis accidents (DBAs). The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impacts of the proposed license amendment. The NRC staff performed independent calculations to confirm the conservatism of the licensee’s analyses.

Revised DBA radiological consequences analyses were performed by the licensee in accordance with the guidance in RG 1.183. The licensee performed AST analyses for the DBAs that could potentially result in significant control room and offsite doses. These include

[1]MNGP's construction permit predates the implementation of the GDCs. The citing of GDC 19 is not an effort to impose GDC 19 on the licensee. The NRC staff is using GDC 19 solely as a convenient summary of acceptable review standard for control room habitability. In addition, the MNGP USAR references GDC 19 in Section 14.7 for control room dose standard.

the loss-of-coolant accident, the main steamline break accident, the fuel handling accident, and the control rod drop accident. The licensee calculated revised atmospheric dispersion factors for use in the DBA radiological consequence analyses. The licensee used the most limiting source to receptor control room atmospheric dispersion value in its control room dose analyses for each DBA. The NRC staff's review of the licensee's revised atmospheric dispersion factors is discussed in Section 3.1.5 below.

3.1.1 Loss-of-Coolant Accident (LOCA)

The objective of analyzing the radiological consequences of a LOCA is to evaluate the performance of various plant safety systems intended to mitigate the postulated release of radioactive materials from the plant to the environment. The licensee assumes an abrupt failure of a large reactor coolant pipe that results in a blowdown of the reactor coolant system (RCS) into the drywell. The emergency core cooling systems (ECCS) makes up lost water inventory and cools the core. DBA thermo-hydraulic analyses show that the ECCS can adequately cool the core and prevent significant fuel damage. Nonetheless, the licensee conservatively assumes that the ECCS is not successful and that substantial core damage occurs. The following release pathways are considered in the licensee's analysis of the LOCA; primary containment leakage, main steam isolation valve (MSIV) leakage, secondary containment bypass, and leakage from the ECCS. Secondary containment bypass is discussed below both in conjunction with the primary containment leakage and the MSIV leakage.

3.1.1.1 LOCA Source Term

MNGP is of a boiling-water reactor 3 (BWR/3) design with a Mark I containment. The licensed power is 1775 megawatts thermal (MWt). Consistent with prior analyses, the licensee assumed a power of 1880 MWt for determining radiological consequences. This value is further increased by 2 percent to 1918 MWt, consistent with the guidance of RG 1.183, to account for power measurement uncertainties. The core inventory used for the source term for the LOCA analysis was developed using the ORIGEN 2.1 isotope generation and depletion computer code. The calculation assumed operation at 1918 MWt and operation at the total average burnup expected for a 24-month fuel cycle. The licensee's determination of the core fission product inventory, including use of the ORIGEN code, is in accordance with RG 1.183.

Fission products from the damaged fuel are assumed to be released into the RCS and then into the primary containment. Due to the postulated loss of core cooling, the fuel over-heats, resulting in the release of fission products. The gap inventory release phase begins two minutes after the event starts and is assumed to continue for 30 minutes. As the core continues to degrade, the gap inventory release phase ends and the in-vessel release phase begins. This phase continues for 1.5 hours. Tables 1, 4, and 5 of RG 1.183 define the source term used for these two phases. The inventory in each release phase is released at a constant rate starting at the onset of the phase and continuing over the duration of the phase.

3.1.1.2 LOCA Fission Product Transport

The LOCA pipe break results in a blowdown of the reactor pressure vessel (RPV) liquid and steam to the drywell. The resulting pressure buildup drives the mixture of steam, water, and other gases through the suppression pool water and into the primary containment. The suppression pool water condenses the steam and reduces the pressure. After the initial RPV

blowdown, ECCS water injected into the RPV will spill into the drywell, transporting fission products to the suppression pool and then into the primary containment. In lieu of modeling this transport mechanistically, the licensee conservatively assumed that the fission product release from the fuel is homogeneously and instantaneously dispersed within the drywell free volume. Rapid mixing between the drywell and wetwell (torus) airspace is assumed after the first 2 hours for the assumed duration of 30 days to model fission products being homogeneously distributed between the drywell and torus airspace. The NRC staff finds that the licensee's assumptions regarding drywell and wetwell mixing are consistent with assumptions previously found acceptable for full implementation of an AST for other Mark I containments. The licensee did not credit any reduction by suppression pool scrubbing for fission products transferred to the primary containment through the suppression pool.

The licensee assumes that a portion of the fission products released from the RPV will plateout in the drywell due to natural deposition processes. The licensee models this deposition using the 10th percentile values in the model described in the NRC-accepted NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (i.e., the "Powers Model"). The licensee did not assume natural deposition of elemental or organic forms of iodine in the drywell or wetwell. The licensee's assumptions on drywell/wetwell mixing and natural deposition processes are consistent with the guidance in RG 1.183.

The AST methodology assumes that the iodine released to the containment consists of 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic forms. The assumption of this iodine speciation is predicated on maintaining the containment sump water at pH 7.0 or higher. The licensee proposes to use the standby liquid control (SLC) system to inject sodium pentaborate to the RPV, where it will mix with ECCS flow and spill over into the suppression pool. Sodium pentaborate, being a base, will neutralize acids generated in the post-accident primary containment environment. Use of the SLC to add buffering agent assures that the assumptions on iodine release and speciation as used in the AST dose analysis are acceptable. See Sections 3.2 and 3.3 below for use of the SLC system to maintain pH higher than 7.0.

3.1.1.3 LOCA - Containment Leakage Pathway

The primary containment is presumed to leak at its TS design leakage rate (L_a) of 1.2 percent of its contents by weight per day for the first 24 hours and is reduced to 0.61 L_a between 24 and 72 hours, then is further reduced to 0.5 L_a for the remainder of the 30-day accident duration after 72 hours. The licensee estimated the reduction to 61 percent of the primary containment design leakage rate at 24 hours and to 50 percent at 72 hours based on plant-specific analysis of the post-LOCA primary containment pressure/temperature profile.

The licensee assumed that the majority of the leakage from the primary containment collects in the free volume of the secondary containment, and is subsequently released to the environment via ventilation system exhaust. No holdup in the secondary containment is assumed. Secondary containment bypass of 35.2 standard cubic feet per hour (scfh) was assumed for the first 24 hours, which is reduced to 61 percent of that value at 24 hours, then further reduced at 72 hours to 50 percent of the initial 35.2 scfh. The reduction in the secondary containment bypass is in correlation with the reduction in the primary containment leakage reduction. The secondary containment bypass occurs through lines that drain to the main condenser, and took

credit for fission product aerosol deposition in those lines. The NRC staff's evaluation of the piping deposition credit is discussed below in Section 3.1.1.4.

Following a LOCA, the standby gas treatment (SBGT) fans start and draw down the secondary containment to create a negative pressure in relation to the environment. This pressure differential ensures that leakage from the primary containment is collected and processed by the SBGT. SBGT exhaust is processed through filter media prior to release to the environment via the SBGT vent. The licensee assumes that the requisite negative pressure will not be achieved for 5 minutes after the start of the event, assuming that only one train of the SBGT system is operable. During the draw down period, the primary containment leakage is assumed to be released unfiltered to the environment. After the secondary containment is drawn down, the release is filtered by the SBGT system and released through the offgas stack. The remainder of the leakage from the primary containment (the secondary containment bypass) is assumed to be released unfiltered to the environment for the duration of the LOCA through the condenser as a ground level release from the turbine building. The staff finds that the licensee's assumptions for the containment leakage and containment bypass release pathways are in accordance with the guidance in RG 1.183, and are, therefore, acceptable.

3.1.1.4 LOCA - Main Steam Isolation Valve Leakage

The four main steam lines which penetrate the primary containment, are automatically isolated by the MSIVs in the event of a LOCA. There are two MSIVs on each steam line, one inside the drywell (i.e., inboard) and one outside the primary containment (i.e., outboard). The MSIVs are functionally part of the primary containment boundary and design leakage through these valves provides a leakage path for fission products to bypass the secondary containment and enter the environment as a ground level release.

For the MSIV leakage pathway, the licensee made the same assumptions with regard to the release and transport within the primary containment (drywell and wetwell) as for the containment leakage pathway. All MSIV leakage is assumed to transport through two of the four main steam lines at 100 scfh each. These leak rates are consistent with the proposed surveillance requirement MSIV leak rate and main steam line pathway leakage criteria. This modeling of the MSIV leakage pathway also supports the licensee's requested exemption from the regulatory requirements for containment leakage testing. The MSIV leak rates decrease to 61 percent of the above (i.e., 61 scfh each) after 24 hours, then are further reduced to 50 percent of the above (i.e., 50 scfh each) after 72 hours, based on the decrease in containment pressure, as discussed above for the containment leakage pathway.

The licensee assumed that the inboard MSIV has failed open on one of the two lines. This assumed failure limits the piping surface area credited for natural deposition in the main steam piping. Natural deposition of radioactive aerosols is credited for the piping between the inboard and outboard MSIV on one steam line (the one of shortest distance) and in the drain lines from two of the main steam lines to the main condenser (in the two shortest drain line paths). Since a single failure of an inboard MSIV in one steam line is assumed, natural deposition is not credited between the MSIVs in this line.

The licensee's iodine removal modeling assumes well-mixed control volumes. Only the volumes associated with horizontal runs of seismically qualified main steam line piping are included in the modeling of iodine aerosol deposition. The licensee conservatively modeled the

temperatures in the piping post-LOCA by choosing boundary conditions to maximize the heat addition to the pathway and also choosing heat transfer boundary conditions and coefficients to minimize the heat transfer away from the piping. The licensee's main steam line modeling is conservative because it minimizes aerosol deposition credit.

The licensee's modeling of aerosol settling is based on the methodology used by the NRC staff in its review of the implementation of AST at the Perry Nuclear Power Plant. The aerosol settling model is described in an NRC report, AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term." AEB-98-03 gives a distribution of aerosol settling velocities that are estimated to apply in the main steam line piping. The model used in the Perry assessment assumed aerosol settling may occur in the main steam lines upstream of the outboard MSIV, at the median settling velocity given by the Monte Carlo analysis described in the AEB-98-03 report. In the Perry assessment, aerosol settling is assumed to occur in one settling volume between the inboard MSIV and the outboard MSIV for the main steam line which has been assumed broken inside the drywell. For the remaining main steam lines, aerosol settling is assumed to occur in 2 settling volumes: one between the reactor vessel and the inboard MSIV, the other volume between the two closed MSIVs.

The licensee's modeling of aerosol settling in the MSIV leakage pathway for MNGP, is similar to that for Perry with regard to the main steam line modeling, however, the licensee modeled deposition in drain lines determined to be seismically rugged and for the secondary containment bypass. The NRC staff posed questions on the licensee's model with regard to the conservatism in its use of the AEB 98-03 methodology, asking about the impact on the piping temperature profile by the aerosols deposited within the piping on the deposition efficiency. The licensee's August 22, 2006, letter, stated that, although the piping thermal profile does not include decay heat from deposited aerosols, the thermal profile is conservatively calculated to maximize the heat in the piping. The conservatism in heating profile more than compensates for the additional heating from deposited aerosols, which was estimated to be about 147 British thermal units per hour. The NRC staff had previously found a similar discussion acceptable as a precedent in an AST review submitted by the Clinton Nuclear Station, and therefore finds the licensee's pipe thermal modeling acceptable.

Another staff concern was that the removal through aerosol settling was overestimated by using settling velocities which may not take into account the change in the particle size distribution that may be expected over time and as some aerosol is settled out in the drywell before leakage through the MSIVs. The licensee's response, dated August 22, 2006, indicated that the model based on AEB 98-03 is intended to be conservative in estimating total deposition based on a deposition velocity statistically determined from expected drywell distributions of aerosol density, diameter and shape factors. Because the MNGP main steam line aerosol deposition model is a lumped-single volume, there is no aerosol size distribution behavior impact on the model itself as the deposition volume's outlet flow empties directly into the condenser and not into other pipe deposition volumes with deposition rates based on the first volume's exit conditions. Simulation of the physical changes to the aerosol constituents would require a multi-volume model to account for time-dependent changes to control volume characteristics (i.e., loss of mass, redistribution of remaining particles, etc.). However, the licensee expects that a mechanistic multi-compartment pipe deposition model would yield higher deposition rates because the downstream volumes further remove additional aerosol

particles not deposited in upstream volumes. The NRC staff finds this to be a reasonable expectation, but does not know how much additional deposition downstream volumes may add over what the licensee is currently estimating, especially in light of the fact that the licensee also takes credit for deposition in the condenser downstream, which may under-estimate iodine release to the environment. Nevertheless, considering that the licensee has conservatively modeled the amount of piping credited for deposition, and considering that the licensee has generally followed the precedent established in the review of the Perry AST in the use of the AEB 98-03 report, the NRC staff finds that the licensee's modeling of the aerosol deposition in the seismically rugged sections of the steam lines and drain lines is reasonable and acceptable.

As stated above, the licensee also takes credit for particulate and elemental mixing, holdup and deposition in the condenser using the methodology discussed in the NRC staff's safety evaluation of General Electric (GE) Topical Report, NEDC-31858P (Proprietary GE Report), Revision 2, "[Boiling-Water Reactor Owners' Group] Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," September 1993, issued March 3, 1999. This safety evaluation is referred to as an acceptable methodology in RG 1.183, and may be applied if the leakage pathway and condenser are shown to be seismically rugged. The NRC staff previously found the leakage pathways credited and the condenser to be seismically rugged in the safety evaluation supporting MNGP Amendment No. 102. The NRC staff reviewed the licensee's calculations using this methodology and has found the estimated effective removal efficiency for the condenser to be acceptable.

The NRC staff acknowledges that aerosol settling is expected to occur in the main steam line piping but, because of recent concerns with aerosol sampling used in AEB-98-03 and lack of further information, does not know how much deposition (i.e., settling velocity value) is appropriate. The licensee has used a model based on the methodology of AEB-98-03, and has applied conservatism which addresses the NRC staff's questions on the applicability of the AEB-98-03 methodology to MNGP. Given this information, the NRC staff finds the MNGP main steam line aerosol settling model to be reasonable and appropriate.

The licensee also assumed deposition of elemental iodine in the main steam line piping. The licensee used the model described in a letter report dated March 26, 1991, by J.E. Cline, "MSIV Leakage Iodine Transport Analysis," which the NRC staff has found acceptable as discussed in RG 1.183. The Cline report provides elemental iodine deposition velocities, resuspension rates and fixation rates. The deposition velocities were used in the well-mixed model formulation described above for use with AEB-98-03. Because elemental deposition is not gravity-dependent, the licensee assumed elemental iodine deposition occurs on the entire surface area of the horizontal and vertical piping. The licensee evaluated the effects of resuspension, as described in the Cline report, and found the impact to be small and more than compensated for by the conservative piping temperature profile used.

Based on the above discussion, the NRC staff finds the licensee's modeling of the release by leakage through the MSIVs conservative and appropriate.

3.1.1.5 LOCA - Leakage from Emergency Core Cooling Systems

During the progression of a LOCA, some fission products released from the fuel will be carried to the suppression pool by spilling from the RCS and by natural processes such as deposition and plateout. Post-LOCA, the suppression pool is a source of water for the ECCS. Because

portions of these systems are located outside of the primary containment, leakage from these systems is evaluated as a potential radiation exposure pathway. For the purposes of assessing the consequences of leakage from the ECCS, the licensee assumes that all of the radioiodine species released from the fuel are instantaneously and homogeneously mixed in the minimum suppression pool liquid volume. Noble gases are assumed to remain in the containment atmosphere. Since aerosols and particulate radionuclides will not become airborne upon release from the ECCS, they are not included in the ECCS source term. This source term assumption is conservative for the LOCA analysis in that all of the radioiodine released from the fuel is assumed to be the source term in both the primary containment atmosphere leakage and the ECCS leakage. In a mechanistic treatment, the radioiodine in the primary containment atmosphere would relocate to the suppression pool over time.

The licensee's analysis considers the equivalent of 2.62 gallons per minute (gpm) unfiltered ECCS leakage starting at the onset of the LOCA, which is twice the design leakage value for MNGP. The licensee assumes 10 percent of the iodine in the ECCS leakage becomes airborne and is available for release as 97 percent elemental and 3 percent organic iodine. No credit was assumed for hold-up and dilution in the secondary containment. As was assumed for the primary containment leakage pathway, the leakage enters the environment via the SBGT as a filtered elevated release, after the secondary containment draw down period. The release continues for 30 days. The NRC staff finds the licensee's assumptions for the ECCS leakage release in accordance with the guidance in RG 1.183, and are, therefore, acceptable.

3.1.1.6 LOCA Doses and Control Room Modeling

The licensee evaluated the maximum 2-hour total effective dose equivalent (TEDE) to an individual located at the exclusion area boundary (EAB) and the 30-day TEDE to an individual at the outer boundary of the low population zone (LPZ). The LOCA doses calculated for each of the pathways discussed above, were added together for the total LOCA dose estimate. The licensee's resulting offsite doses are less than the 10 CFR 50.67 dose guidelines.

The licensee evaluated the dose to the operators in the control room both from the releases from the above pathways, and through post-LOCA shine from contaminated plant components. The radiological dose to the control room operators during the postulated design-basis LOCA is minimized by the integrity of the control room envelope (CRE) and operation of the control room ventilation emergency filtration train (CRV-EFT) system. The inhalation and immersion doses calculated in the licensee's AST evaluation are based on a combination of unfiltered in-leakage and filtered intake flow, determined through studies. The unfiltered inleakage assumption is verified as conservative by CRE inleakage tests. The CRV-EFT system is in full emergency mode operation prior to radiological release at 2 minutes post-LOCA because the system initiates on the LOCA signal. The licensee chose the minimum emergency mode filtered air intake flow of 900 cubic feet per minute (cfm) to provide conservative dose results. Control room intake filter efficiencies are based on the TSs surveillance requirements in accordance with RG 1.183. The control room unfiltered inleakage for the CRV-EFT system emergency mode was assumed to be 500 cfm, which conservatively bounds the results of the control room tracer gas testing performed by the licensee in June 2004.

The licensee performed a separate calculation assessing the external shine dose to control room (CR) operators from sources confined in the reactor building airspace; contained in the external airborne cloud (plume outside the control room); deposited on the SBGT system,

CRV-EFT system, and technical support center emergency ventilation system filters; and, contained in ECCS piping recirculating reactor water inside the reactor building. This calculation was provided to the NRC staff for review by letter dated August 22, 2006. The NRC staff finds that the licensee used analysis source term and transport assumptions that are in agreement with the assumptions used in analyses described above for offsite and control room doses. The NRC staff also finds that the licensee's control room receptor and radiation shine sources were modeled to reflect the physical layout of the plant.

The shine dose results were added into the LOCA release results to give a total LOCA control room TEDE which meets the dose criteria of 10 CFR 50.67 and GDC-19.

3.1.1.7 LOCA Summary

The NRC staff found that the licensee's LOCA analysis assumptions and methodology are consistent with the guidance of RG 1.183. The analysis assumptions found acceptable to the staff are presented in Table 1. The EAB, LPZ, and CR doses estimated by the licensee for the LOCA were found to be acceptable, and are listed in Table 2. The NRC staff compared the radiation doses estimated by the licensee to the applicable acceptance criteria in SRP 15.0.1 and to the results estimated by the NRC staff in its confirmatory calculations. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses for the LOCA will continue to comply with the dose guidelines in 10 CFR 50.67, GDC-19, and the accident-specific dose acceptance criteria in SRP 15.0.1 and RG 1.183.

3.1.2 Main Steamline Break (MSLB)

The postulated MSLB accident assumes a double-ended break of one main steam line outside the primary containment with displacement of the pipe ends that permits maximum blowdown rates. There is no fuel damage as a consequence of this accident. Two activity release cases were analyzed, corresponding to the pre-accident elevated iodine concentration (spike) and maximum equilibrium concentration allowed by proposed technical specifications of 2.0 µCi/gm and 0.2 µCi/gm dose equivalent iodine-131, respectively.

The mass of coolant released is the amount in the steam line and connecting lines at the time of the break, plus the amount passing through the MSIVs prior to closure at 10.5 seconds. The assumed MSIV closure time of 10.5 seconds is consistent with the current licensing basis and allows for a maximum valve closure of 9.9 seconds, as tested, and instrument response time. A total of 91,834 pounds of steam and liquid is assumed to be released during blowdown. The mass of the blowdown is calculated based upon the thermal-hydraulic transient analysis of the MSLB as described in the MNGP USAR, and is not affected by the application of the AST methodology to the calculation of the radiological consequences of the event. The licensee increased the mass release over the calculated value for additional conservatism.

The analysis assumes an instantaneous ground level release. No holdup or dilution by mixing in the turbine building air volume is credited.

The activity in the release volume is assumed to be equal to the activity present in the 91,834 lbm of liquid and steam released. Noble gas releases correspond to a 300,000 µCi/sec offgas release rate after a 30-minute delay. The design-basis emission rate of 300,000 uCi/sec after 30 minutes decay is conservative with respect to the MNGP TS limit of 260,000 uCi/sec

after 30 minutes decay. The licensee assumed that the iodine species released from the main steam line are 95 percent cesium iodide as an aerosol, 4.85 percent elemental, and 0.15 percent organic as stated in RG 1.183, Appendix D, Regulatory Position 4.4. The licensee evaluated the effect of cesium activity on the dose consequences of the MSLBA, and found that the dose contribution was negligible (〈0.5 mRem). The NRC staff finds that the licensee's analysis source term assumptions are consistent with the guidance in RG 1.183.

The licensee took no credit for the operation of the control room emergency filtration systems or control room isolation. Instead, the licensee used the normal ventilation intake rate of 7,440 cfm as a bounding unfiltered intake assumption and additionally assumed 1,000 cfm of unfiltered inleakage. The administrative building air intake represents the limiting control room receptor location for analyzing dispersion. The analysis assumes a ground level release from the turbine building for the MSLB.

The licensee calculated revised offsite and control room atmospheric dispersion factors (χ/Qs) for use in the DBA dose analyses. The NRC staff's review of the χ/Qs is discussed below in Section 3.1.5 of this safety evaluation.

The licensee's analysis used inhalation committed effective dose equivalent dose conversion factors from Federal Guidance Report 11 and external effective dose equivalent dose conversion factors from Federal Guidance Report 12. Use of these factors is acceptable to the staff, as discussed in RG 1.183.

3.1.2.1 MSLB Summary

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The assumptions found acceptable to the NRC staff are presented in Table 1. The licensee's calculated dose results are presented in Table 2. Based upon the information provided by the licensee, the NRC staff finds that the licensee used analysis methods and assumptions consistent with the guidance of RG 1.183 in its analysis of the MSLB. The NRC staff compared the radiation doses estimated by the licensee to the applicable acceptance criteria in SRP 15.0.1 and to the results estimated by the NRC staff in its confirmatory calculations. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses for the MSLB will continue to comply with the dose guidelines in 10 CFR 50.67, GDC-19, and the accident-specific dose acceptance criteria in SRP 15.0.1 and RG 1.183.

3.1.3 Control Rod Drop Accident (CRDA)

The design-basis CRDA assumes the rapid removal of the highest worth control rod resulting in a reactivity excursion that encompasses the consequences of any other postulated CRDA. For the dose consequence analysis, the licensee assumed that 850 of the 29,040 fuel rods in the core were damaged, with melting occurring in 1.06 percent of the damaged rods. A core average radial peaking factor of 1.7 was used in the analysis. For releases from the breached fuel, 10 percent of the core inventory of noble gases and iodines are assumed to be in the fuel gap. For releases attributed to fuel melting, 100 percent of the noble gases and 50 percent of the radioiodine species are assumed to be released to the reactor coolant. These assumptions are consistent with the guidance in RG 1.183 for the CRDA.

Although not specified in Appendix C of RG 1.183, alkali metals (Cs and Rb) are assumed to be released consistent with Table 3 of RG.1.183. For the damaged fuel, a release fraction of 0.12 is assumed and, for melted fuel, a higher release fraction of 0.25 is assumed (consistent with the BWR LOCA table, Table 1, of RG 1.183). The iodine species released to the reactor coolant are assumed to be 95 percent aerosol, 4.85 percent elemental, and 0.15 percent organic. The iodine species released from the main condenser to the environment are assumed to be 97 percent elemental iodine and 3 percent organic iodine.

Instantaneous mixing of the activity released from the fuel in the reactor coolant is assumed, with 100 percent of the noble gases, 10 percent of the radioiodine, and 1 percent of the remaining radionuclides that are released into the reactor coolant, to reach the turbine and condenser. Of this activity that reaches the turbine and condenser, 100 percent of the noble gases, 10 percent of the radioiodine species and 1 percent of the particulate radionuclides are available for release to the environment. These assumptions are consistent with the guidance in RG 1.183 for the CRDA. Two separate licensing basis cases were required to model the main condenser release and its impact on control room operator and offsite doses. The first (Case 1) provides the bounding (highest) doses and models the release from the main condenser through operating steam jet air ejectors (SJAEs) to the offgas stack. Case 1 covers those operating power levels when the SJAEs are operating prior to the accident. The SJAEs are assumed to operate for the entire duration of the accident (24 hours). No other release paths are assumed.

The second licensing basis case (Case 2) was developed to evaluate the impact of mechanical vacuum pump (MVP) operation on dose during early stages of the accident. MVP operation occurs at low reactor power conditions. Main steam line radiation monitors detect high radiation causing MVP trip and isolation. The MVP release point is through the offgas stack. After the MVP is isolated, the main condenser is assumed to leak at 1 percent per day as a ground level release. Because MVP isolation is assumed for this CRDA analysis case, the licensee proposed changes to the TS operability requirements for the MVP isolation instrumentation.

The MNGP control room is modeled as a closed volume of 27,000 cubic feet. The control room was modeled without taking credit for automatic emergency filtration system actuation or isolation. Instead, the licensee used the normal ventilation intake rate of 7,440 cfm as a bounding unfiltered intake assumption and additionally assumed 1,000 cfm of unfiltered inleakage. The licensee performed sensitivity studies at other lesser flow combinations which showed that the above combination of unfiltered intake and unfiltered inleakage provided bounding (worst-case) results.

The licensee used revised offsite and control room atmospheric dispersion factors in its analysis. The staff's review of the atmospheric dispersion factors is discussed below in Section 3.1.5 of this safety evaluation. The licensee used ground level release χ/Q values for the most conservative release point to receptor locations for Case 2 after MVP isolation. For elevated releases from the offgas stack, the MVP and SJAEs are assumed to conservatively operate at maximum flows. Elevated release atmospheric dispersion values, including fumigation, are assumed for release from the offgas stack for Case 1 (SJAE operation) and Case 2 before the MVP isolation occurs.

The licensee determined that the control room elevated release χ/Q values submitted in the September 15, 2005, application for amendment were calculated with an error in the input. By

letter dated August 22, 2006, the licensee provided corrected control room elevated release χ/Q values and determined the impact on the CRDA dose analysis results due to this correction. Because the fumigation control room χ/Q values were not affected by the error, and fumigation was assumed for the duration of the pre-isolation elevated release period for Case 2 (MVP isolation), the control room dose calculated for Case 2 was not impacted by the correction. The increase in the licensee's calculated dose in the control room for the CRDA with SJAEs in operation (Case 1) was less than 0.04 rem TEDE, assuming a conservative 2.1 percent increase due to the corrected control room elevated χ/Q values. The corrected total CRDA Case 1 control room dose is 1.736 rem TEDE. The dose in the control room continues to meet the requirements of 10 CFR 50.67 and GDC-19.

3.1.3.1 CRDA Summary

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The assumptions found acceptable to the NRC staff are presented in Table 1. The licensee's calculated dose results are presented in Table 2. Based upon the information provided by the licensee, the NRC staff finds that the licensee used analysis methods and assumptions consistent with the guidance of RG 1.183 in its analysis of the CRDA. The NRC staff compared the radiation doses estimated by the licensee to the applicable acceptance criteria in SRP 15.0.1, and to the results estimated by the NRC staff in its confirmatory calculations. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses for the CRDA will continue to comply with the dose guidelines in 10 CFR 50.67, GDC-19, and the accident-specific dose acceptance criteria in SRP 15.0.1 and RG 1.183.

3.1.4 Fuel Handling Accident (FHA)

The FHA analysis postulates that a spent fuel assembly is dropped during refueling operations. For the licensing basis case, the postulated FHA involves the drop of a fuel assembly in the reactor vessel cavity over the reactor core during refueling operations. The licensee also considered the consequences of dropping a fuel assembly during fuel movement at other locations.

The drop of a fuel assembly in the reactor vessel cavity over the reactor core was found to be the limiting design basis case. A fuel assembly is postulated to drop from the maximum height allowed by the refueling platform and to fall onto the fuel in the reactor. The drywell head and reactor vessel head are assumed to be removed. At this location, the maximum drop (free fall distance) is approximately 27 feet for the fuel assembly, and fuel pin damage is postulated to occur to both the dropped assembly and to a portion of those assemblies impacted in the reactor core. The extent of damage is calculated based on the free fall distance and the resulting kinetic energy of the dropped assembly. In accordance with the current licensing basis, this drop is conservatively assumed to damage 125 fuel pins.

The gap activity from the damaged pins is the radioactive source term for this event. A radial peaking factor of 1.7 is assumed in the analysis. A post-shutdown 24-hour decay period was used to determine the release activity inventory. An effective total decontamination factor (DF) of 200 for the released radioiodine species is assumed based on a minimum water depth of 23 feet. The nominal water depth (i.e., the distance from the top of the water in the vessel to the

point of impact for the dropped assembly) for the postulated drop onto the reactor core would be approximately 46 feet (well in excess of the 23 feet needed to justify the DF of 200.).

The licensee's analysis assumed a ground level release via the normal reactor building vent over a 2-hour period. No credit is taken for radiation monitors, isolation of secondary containment, filtering, or elevated release from the SBGT system, or operation of the control room emergency filtration system. The licensee used revised offsite and control room atmospheric dispersion factors in its analysis. The NRC staff's review of the atmospheric dispersion factors is discussed below in Section 3.1.5 of this safety evaluation. The licensee's analysis assumptions with regard to fuel damage, source term, and radiation transport are all in accordance with the guidance on FHA in RG 1.183.

The MNGP control room is modeled as a closed volume of 27,000 cubic feet. The control room was modeled without taking credit for automatic emergency filtration system actuation or isolation. Instead, the licensee used the normal ventilation intake rate of 7,440 cfm as a bounding unfiltered intake assumption and additionally assumed 1,000 cfm of unfiltered inleakage. The control room χ/Q for the worst-case release path from the reactor building vent to the control room air intake is assumed and modeled as a ground level release.

For a drop in the fuel transfer area, between the reactor vessel and the spent fuel pool, or over the spent fuel pool, the resulting maximum credible drop height would be less than that assumed in the postulated FHA in the reactor vessel cavity. For a drop in the spent fuel pool or fuel transfer area, the postulated activity released would be lower. In both these areas, the minimum water depth covering the damaged fuel would be less than that available over the reactor core, resulting in less iodine removal by the water. However, in both these areas, the drop height and subsequent calculated number of impacted fuel rods is less, thus resulting in a lower source term. The licensee performed analyses to show that the reduction in source term sufficiently compensates for the reduction in water DF such that the total radiological release and calculated dose for the postulated fuel assembly drop over the reactor core is bounding for FHAs in the other areas. The NRC staff previously found acceptable the licensee's evaluation determining that the fuel drop over the reactor core is bounding over other areas in Amendment No. 145 (Accession No. ML060760284) for the selective implementation of an AST for the FHA.

The NRC staff determines that the licensee's modeling of the control room ventilation systems and the FHA support the licensee's proposed changes to the TS applicability for the control room emergency filtration system, control room ventilation system, and electrical power sources during shutdown, in accordance with TSTF-51.

3.1.4.2 FHA Summary

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The assumptions found acceptable to the NRC staff are presented in Table 1. The licensee's calculated dose results are presented in Table 2. Based upon the information provided by the licensee, the NRC staff finds that the licensee used analysis methods and assumptions consistent with the guidance of RG 1.183 in its analysis of the FHA. The NRC staff compared the radiation doses estimated by the licensee to the applicable acceptance criteria in SRP 15.0.1, and to the results estimated by the NRC staff in its confirmatory calculations. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses for the FHA will continue to

comply with the dose guidelines in 10 CFR 50.67, GDC-19, and the accident-specific dose acceptance criteria in SRP 15.0.1 and RG 1.183.

3.1.5 Atmospheric Dispersion Estimates

The licensee used two previously generated χ/Q values in the dose analysis to support this license amendment request. The χ/Q values are those calculated for the 0-2 hour time period for effluent releases to the LPZ, and from the reactor building vent to the control room intake. The χ/Q values are addressed in the safety evaluation associated with Amendment No. 145, approving use of the alternative source term in the radiological dose assessment for the postulated FHA.

The licensee also calculated new control room, EAB, and LPZ χ/Q values. The licensee used the ARCON96 (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wake") and PAVAN (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations") atmospheric dispersion computer models to calculate χ/Q values for the LOCA, MSLB, CRDA, and FHA. The resulting set of control room, EAB, and LPZ χ/Q values evaluated below represent a change from those currently presented in the MNGP USAR.

3.1.5.1 Meteorological Data

The licensee generated χ/Q values for this amendment application using meteorological data collected at the MNGP site during the period 1998–2002. The licensee previously provided, and NRC staff performed a quality review of these data as part of the review that led to Amendment No. 145. The data included wind speed and wind direction measurements in the form of hourly meteorological data files for input into the ARCON96 atmospheric dispersion computer code from measurements made at a height of 10 meters and 43 meters above the ground. Stability class was calculated using the temperature difference between the 43-meter and 10-meter levels. Data from the 10-meter, 43-meter and 100-m levels were also provided in the form of joint wind speed, wind direction, and atmospheric stability frequency distributions (joint frequency distributions) for input into the PAVAN atmospheric dispersion computer code. Stability class for use with the 10-meter and 43-meter wind data was calculated using the temperature difference between the 43-meter and 10-meter levels. Stability class for use with the 100-meter wind data was calculated using the temperature difference between the 100-meter and 10-meter levels. The NRC staff's assessment of these data is presented in the safety evaluation associated with Amendment No. 145.

As part of review of the current application, the NRC staff requested wind speed and wind direction data in the form of hourly meteorological data files for input into the ARCON96 atmospheric dispersion computer code from measurements made at a height of 10 meters and 100 meters above the ground with stability class based upon the temperature difference between the 100-meter and 10-meter levels. During a review of the data, the NRC staff observed some cases when the wind speed and/or wind direction were reported as zeroes. It was not clear that these data were valid. In its August 22, 2006, letter, the licensee stated that the presence of either a zero value for the wind direction or concurrent values of zero for the wind direction and wind speed indicated invalid data. However, a non-zero value of wind direction coupled with a value of zero for the wind speed indicated a valid measurement for a calm condition.

The NRC staff generated revised data files to include the above criteria for zero values of wind direction and wind speed, for both the 10-meter and 43-meter hourly measurements previously provided by the licensee and the 10-meter and 100-meter hourly data provided for the current amendment application. The NRC staff performed a quality review of these corrected data files using the methodology described in NUREG-0917, “Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data,” and also computer spreadsheets. With respect to atmospheric stability measurements, the time of occurrence and duration of stable and unstable conditions were found to be consistent with expected meteorological conditions. Stable and neutral conditions were reported to occur at night and unstable and neutral conditions during the day, with neutral or near-neutral conditions predominating during each year. During 1998–2002, wind direction frequency occurrences were similar among the three measurement heights and from year to year at each height. Winds were predominately from the north northwest and south at all levels. Wind speed at each height was also consistent for each of the 5 years. A comparison of the joint frequency distributions derived by the NRC staff from these ARCON96-formatted hourly data with joint frequency distributions developed by the licensee showed some slight differences. These differences appear to be the result of use of two slightly different data sets in generation of the joint frequency distributions. The licensee calculated joint frequency distributions based upon a meteorological data set of measurements from the single measurement train having the highest data recovery. The ARCON96 files used a meteorological data set based upon measurements from a second train, when data were not available from the first train. The NRC staff made an overall comparison of wind rose calculations from joint frequency distributions determined from the above two data sets and found the two sets of joint frequency distributions to be similar.

3.1.5.2 Control Room Atmospheric Dispersion Factors

The licensee used guidance provided in RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” to generate the new control room atmospheric dispersion factors for assumed releases from the turbine building vent, the reactor building vent, and the reactor building wall nearest the two assumed intake locations, i.e., the control room and administrative building intakes. All releases were assumed to be point source releases. The licensee calculated these new control room χ/Q values using the ARCON96 computer code. Because the heights of these postulated release locations are less than 2½ times the height of adjacent buildings, they were modeled using the ARCON96 ground level release option in accordance with RG 1.194. The licensee executed ARCON96 using the 1998–2002 hourly data from the MNGP onsite meteorological tower from wind measurements made at a height of 10 meters and 43 meters and atmospheric stability measurements between the 43-meter and 10-meter levels. The licensee used the taut string methodology described in RG 1.194 to estimate the shortest distance around or over intervening buildings which was input as the source-to-receptor distance. RG 1.194 states that ARCON96 is an acceptable methodology for assessing control room χ/Q values for use in DBA radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of the ARCON96 model for the MNGP site.

The licensee also generated new χ/Q values for releases from the off-gas stack using PAVAN for the actual distance to the intake locations. Fumigation was assumed to occur for one-half hour during the worst 2-hour control room exposure period. Guidance in RG 1.194 states that

comparative calculations should be made using both the PAVAN and ARCON96 computer codes. PAVAN estimates should be made for several distances in each wind direction sector with the objective of identifying the maximum χ/Q value, but fumigation need not be assumed. The NRC staff made comparison estimates using the RG 1.194 methodology. The licensee's χ/Q values for the 0-4 day time periods were larger than the NRC's χ/Q values. The NRC-generated 4-30 day χ/Q value was somewhat higher than the 4-30 day χ/Q value generated by the licensee using PAVAN only. The only DBA dose analysis assuming elevated releases after 4 days is the LOCA, for the containment release pathway. Because the majority of the containment release from the LOCA occurs within the first 4 days, and since the NRC staff found that the licensee's χ/Q for the periods up to 4 days are conservative compared to the NRC staff's calculated values, the impact of the higher χ/Q value for the 4-30 day time period on the total control room dose for the LOCA is minimal.

For each release point, dispersion factors were calculated for both the control room intake and the administrative building intake. The limiting dispersion factor of these two was chosen by the licensee to model all pathways to the control room. Unfiltered inleakage conservatively assumed the limiting X/Q value, due to the proximity of the local intake to the control room. The licensee modeled unfiltered inleakage as though it entered the control room at the same locations as the more limiting of the two χ/Q values.

The NRC staff qualitatively reviewed selected inputs to the licensee's computer runs and found them generally consistent with site configuration drawings and NRC staff practice, made confirmatory calculations using the corrected data files discussed above. Therefore, the NRC staff finds the licensee's estimates to be acceptable, with the stipulation on use of the 4-30-day offgas stack X/Q values as discussed above. These control room χ/Q values are listed in Tables 3 through 5.

3.1.5.3 Offsite Atmospheric Dispersion Factors

The licensee used RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the PAVAN computer code to calculate χ/Q values for the EAB and LPZ.

For postulated releases from the reactor building wall, reactor building vent and turbine building vent, the licensee used the PAVAN ground level release mode, inputting a reactor building height of 43.6 meters and reactor building cross-sectional area of 1829 square meters. The licensee's meteorological input to PAVAN consisted of a joint frequency distribution of wind speed, wind direction, and atmospheric stability data for the 1998–2002 period. Wind speed and direction data from the meteorological tower's 10-meter level were used. Stability class was based on the temperature difference data between the 43-meter and 10-meter levels on the onsite meteorological tower.

The licensee used the elevated release mode option of the PAVAN computer code to model the release from the 100-meter free-standing off-gas stack at MNGP. Consistent with RG 1.183, Section 5.3, fumigation was assumed to occur for one-half hour during the worst 2-hourEAB exposure period. The licensee modeled the reduction in effective stack height due to topography by inputting the highest terrain height of 9 meters in all directions.

The NRC staff qualitatively reviewed selected inputs to the licensee's computer runs, and found them generally consistent with site configuration drawings and NRC staff practice other than the wind speed categories upon which the joint frequency distributions were based. In general, joint frequency distributions used as input to the PAVAN computer code should have a large number of wind speed categories at the lower wind speeds in order to produce the best results (e.g., Section 4.6 of NUREG/CR-2858 suggests wind speed categories of calm, 0.5, 0.75,1.0, 1.25, 1.5, 2.0, 3.0, 4.0, 5.0, 6.0, 8.0, and 10.0 meters per second). The licensee used low wind speed categories as defined in RG 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," which resulted in some clustering of the data.

The NRC staff generated joint frequency distributions from the corrected ARCON96 formatted hourly meteorological data using the finer wind speed categories. The NRC staff then performed an independent evaluation of the resulting atmospheric dispersion estimates calculated by PAVAN for the postulated ground level and elevated releases and found the licensee's χ/Q values to be acceptable. However, since the maximum EAB dose should be determined for any 2 hour period of a DBA, the 0-2 hour χ/Q value should be applied to the entire 30 day period to determine the limiting 2 hour period. Thus, EAB χ/Q values calculated by the licensee for intervals longer than 0-2 hours are not required for this proposed amendment and are not listed in Table 6.

3.1.5.4 Secondary Containment Drawdown - Meteorology

RG 1.183 states that the effect of high winds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5 percent of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5 percent or 95 percent of the total hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, those exceeded only 5 percent of the time should be used). In its August 22, 2006, letter, the licensee stated that it checked both the high and low temperature limits, and considered the lowest temperatures to be limiting as per NRC Information Notice No.88-76 (September 19,1998). The site ambient 5 percent minimum air temperature of 50.9 °F in summer and -3.5 °F in winter, respectively, are considered in the application. In addition, the licensee used wind speed of 24 miles per hour, which was exceeded only 1.4 percent of the time (i.e., less than 5 percent). The NRC staff found these estimates reasonable when compared with estimates that the NRC staff generated from the 1998–2002 onsite wind data measured at the 10-meter level and climatic temperature data (Weather Data Viewer, Version 3, 2005 American Society of Heating, Refrigerating and Air-Conditioning Engineers, Inc., Atlanta, GA) for St. Cloud, Minnesota, which is approximately 20 miles northwest of the MNGP site and Minneapolis, MN, which is about 30 miles southeast of MNGP.

3.2 Standby Liquid Control System

The NRC staff reviewed the SLC system with respect to its role in delivery of sodium pentaborate to the suppression pool for pH control. The control of pH in the suppression pool is required to mitigate the consequences of a DBA in which fuel is damaged. As such, the new role being assigned to the SLC is a safety-related role. The licensee stated that the SLC is categorized in the USAR as an engineered safeguard system.

The NRC staff reviewed the licensee's submittals on the use of the SLC system for the safety-related function. The licensee stated that the SLC system is categorized in the USAR as an engineered safeguard system. As such, the SLC system, as designed and as installed, is a high quality system that provides reasonable assurance that the sodium pentaborate will be injected into the core upon activation. Specifically:

(1) The system components required for reactivity control and new suppression pool pH control functions are seismically qualified (see USAR Section 12.2).

(2) The system is provided with emergency power with the capability to supply power from the emergency diesel generators (see USAR Section 6.6.3).

(3) The system is subject to American Society of Mechanical Engineers Section XI, "Inservice Inspection Requirements," as required by 10 CFR 50.55a, "Codes and standards."

(4) The system is within the scope of the 10 CFR 50.65, regarding monitoring the effectiveness of maintenance at nuclear power plants.

(5) Most components (pumps, squib valves, etc.) are redundant in parallel trains powered from different electrical busses. The exceptions are the containment isolation check valves and the initiation switch. These are discussed below under single-failure review.

(6) Manual initiation of the SLC system is directed in the severe accident management guidelines (SAMGs), which are invoked when adequate core cooling cannot be maintained. Procedures will be updated to specifically direct boron injection without dilution until the required amount of boron is injected for pH control.

(7) Training will be provided on the new SLC injection function as part of operator re-qualification training and emergency operating procedure and SAMG training.

The NRC staff considered MNGP components that could be subject to single failure. The licensee identified two components, the containment isolation check valves on the injection line and the SLC initiation switch, as not being redundant. The containment isolation valves are 1.5 Rockwell Edwards Stainless Steel Piston Check Valves Part No. 3674F316, mounted in the injection line. In the periodic inspections and testing of these valves, the licensee has not experienced any failures of these valves to open on demand. The licensee performed a review of the industry databases EPIX and NPRDS, and identified no incidents of check valves of this type and manufacture failing to open. Although acknowledging that a single failure to open of one of the two check valves could prevent SLC injection, the NRC staff has determined that the potential for failure is very low, based on the quality as established by its procurement, periodic testing and inspection, and historical performance of the component.

The licensee stated that there have been no failures of the initiation switch. Both the switch and the valves were purchased under the requirements for nuclear safety-grade equipment that predate the requirements for Appendix B to 10 CFR Part 50. The equipment is maintained as safety-related in the MNGP 10 CFR Part 50, Appendix B program. The NRC staff finds that the use of a single penetration of the containment with the identified check valves as described by the licensee, and the single initiation switch, to be acceptable.

The NRC staff considered the transport of the sodium pentaborate from the reactor vessel to the suppression pool. The SLC system injects the sodium pentaborate to the reactor vessel through a sparger that consists of a vertical pipe in the core region. SLC system injection occurs approximately 1 hour after the onset of the LOCA. At this time, low-pressure coolant injection (LPCI) pumps are transferred from the injection mode to recirculating the suppression pool water for mixing. Core spray from at least one core spray pump (approximately 2000 gallons per minute) is injected at the top of the core for cooling and liquid level control. The core spray flows down, mixes with SLC injection and transports it to the break and, ultimately, to the suppression pool. Although transport of SLC contents by core spray alone would be slow, sufficient sodium pentaborate is present in the SLC, which, when transferred to the suppression pool, will ensure long-term pH control.

The licensee will use the LPCI in the recirculation mode for suppression pool cooling. Such recirculation would cause the sodium pentaborate to be evenly distributed in the water in the suppression pool and reactor vessel. Even distribution of sodium pentaborate in the water ensures sufficient pH control to prevent iodine re-evolution.

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

3.3 Control of Post-LOCA Suppression Pool Water pH

The NRC staff reviewed the licensee's analysis regarding maintaining suppression pool pH$\geq$7 for 30 days following a LOCA. According to RG 1.183, maintaining pH basic will prevent re-evolution of iodine from the suppression pool water.

After a LOCA, a variety of different chemical species are released from the damaged core One of them is radioactive iodine, which, when released to the outside environment, will significantly contribute to radiation doses. It is, therefore, essential to keep the iodine confined within the plant's containment. According to NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plant," the iodine is release from the core in three different chemical forms; at least 95 percent is released in ionic form as cesium iodide (CsI) and the remaining 5 percent as elemental iodine (I) and as hydriodic acid (HI), with not less than 1 percent of each as I and HI. CsI and HI are ionized in water environment and are, therefore, soluble. However, elemental iodine is scarcely soluble. It is of interest, therefore, to maintain as much as possible of the released iodine in ionic form. However, in the post-LOCA radiation environment in the containment, some of the ionic iodine dissolved in water is converted into elemental form. The degree of conversion varies significantly with the pH of water. At higher pH, conversion to elemental form is low and at pH> 7, conversion is negligibly small. The relationship between the degree of conversion and pH is specified in Figure 3.1 of NUREG/CR-5950, "Iodine Evolution and pH Control."

In MNGP, most of the iodine released from the core is assumed to end up in the suppression pool. Therefore, in order to keep it dissolved, the suppression pool water should be kept at pH$\geq$7 throughout the 30 day post-LOCA period. The licensee has demonstrated that because

of strong acid formation in the suppression pool, this is not achievable without adding buffering chemicals to control the water pH.

After a LOCA, the pH of the suppression water would be continuously decreasing due to formation of hydrochloric and nitric acids in the containment. Hydrochloric acid is formed from decomposition of the Hypalon cable insulation by radiation. About 3.08E-5 mols/liter of hydrochloric acid would be formed during the 30 day period. Nitric acid is formed by irradiation of air and water and about 2.28E-5 mols/liter of nitric acid would be formed during the same period. Both acids are strong acids and will significantly contribute to lowering suppression pool pH. In order to neutralize their effect, the licensee chooses to buffer suppression pool water by using sodium pentaborate from the SLC system. The main purpose of the SLC system is to control reactivity in the case of control rod failure. Since sodium pentaborate is derived from a strong base and a weak acid, it can also act as a buffer. Such buffering action could maintain pH in the basic range in the suppression pool despite the presence of strong acids. The licensee has calculated that in order to maintain pH in the suppression water basic, 1404 gallons of 10 percent solution of sodium pentaborate should be added into the suppression pool's water. The licensee stated that the addition is accomplished within 1 hour after a LOCA.

In order to evaluate beneficial effect of sodium pentaborate, the licensee calculated suppression pool pH for unbuffered and buffered cases. Without addition of sodium pentabarate but taking only credit for the presence of Cs(OH), the value of pH during the 30-day period was below 7, reaching a minimum value of 4.26. With the addition of sodium pentaborate, but without taking credit for Cs(OH), the pH will increase rapidly above 7 and can reach a value as high as 8.6, depending on the core thermal power and the volume of water in the suppression pool.

The NRC staff reviewed the licensee's analysis and concludes that, based on the analysis, the suppression pool pH will stay basic for a period of 30 days after a LOCA.

3.4 Changes to the Technical Specifications

The initial application, dated September 15, 2005, arrived when MNGP was still operating under the former custom TS. On June 5, 2006, the NRC staff issued Amendment No. 146, fully converting the former custom TS to the Improved Technical Specifications (ITS) format. The most obvious impact of this conversion is that the locations of various requirements are no longer the same, and that many requirements have been changed. The licensee's August 21, 2006, letter, provided reprinted ITS pages reflecting implementation of the AST methodology. Accordingly, the following subsections of this safety evaluation identify the changes both in the former custom format and in the current ITS format.

ITS 1.1, "Definitions"
(**Custom TS 1.0, "Definitions"**)

The definition of Dose Equivalent I-131 is changed in two ways: (1) "thyroid dose" is changed to "dose, and (2) the source of dose conversion factors is changed from TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated 1962, and RG 1.109, Rev. 1 to Federal Guidance Report (FGR)-11, "Limiting Values of Radionuclide Intake and Air Concentration Factors for Inhalation, Submersion, and Ingestion," September 1988, and

FGR-12, "External Exposure to Radionuclides in Air, Water and Soil," September 1993. These changes are consistent with the change to AST methodology evaluated above in Section 3.1.

ITS 3.3.7.1, "Control Room Emergency Filtration (CREF) System Instrumentation" (Custom TS 3.2.I, “Instrumentation for Control Room Habitability Protection”)

The Limiting Condition for Operation (LCO) is being revised to refer to the functions and applicability listed in the new ITS Table 3.3.7.1-1. New Required Actions A.1 and A.2 are being proposed with associated completion times. New Condition B is being proposed. Three new surveillance requirements (SRs) are proposed for the instrumentation in the new Table 3.3.7.1-1. What remains to be done is the addition of a new ITS Table 3.3.7.1-1.

The CREF system is vital to maintaining control room personnel doses within acceptable limits according to the AST dose evaluation. The NRC staff reviewed the proposed changes and found that the revised LCO and Required Actions will assure the CREF system to function within the assumptions employed in the AST dose evaluation model in Section 3.1 above.

New ITS Table 3.3.7.1-1, "Control Room Emergency Filtration System Instrumentation" (Custom TS Table 3.2.9, “Instrumentation for Control Room Habitability Protection”)

The licensee proposed to revise Custom TS Table 3.2.9 to delete radiation instrumentation requirements. These requirements will be replaced with requirements for Low Low Reactor Water Level and High Drywell Pressure instrumentation. The new Table 3.3.7.1-1 also adds two additional CREF system instrumentation initiation signals, one from Reactor Building Ventilation Exhaust - High, and the other from Refueling Floor Radiation - High,. that will initiate the CREF system.

The control room radiation instrumentation requirements will be relocated to the licensee-controlled Technical Requirements Manual.

The NRC staff reviewed the proposed ITS Table 3.3.7.1-1 and concluded that control room habitability protection is adequately provided during a LOCA by use of Low Low Reactor Water Level and High Drywell Pressure instrumentation. Since operation of control room habitability protection is not credited in other DBAs, there is no need for TS requirements for controls on the radiation monitoring instrumentation to actuate the control room filter train system. On the basis that the new requirements in Table 3.3.7.1-1 would assure that the CREF system would be actuated as is depicted in the AST dose analysis model, the NRC staff finds the proposed requirements in Table 3.3.7.1-1 to be acceptable.

New ITS 3.3.7.2, "Mechanical Vacuum Pump Isolation Instrumentation" (This would have been added as new Custom TS 3.2.J, “Mechanical Vacuum Pump Isolation Instrumentation” if the Custom TS had not been replaced by ITS.)

The licensee proposed new requirements in the form of a new ITS 3.3.7.2 for MVP isolation instrumentation. Isolation of the MVP is assumed in the AST design-basis CRDA radiological consequence analysis (see Section 3.1.3 above). Operability of the MVP isolation instrumentation is only required during Run or Startup operating conditions with the MVP in service. During shutdown or refueling with control rods inserted, a CRDA is not expected to result in any fuel damage or fission product release. The NRC staff reviewed the proposed new

ITS 3.3.7.2 and finds that it appropriately reflects the assumptions used in the AST CRDA analysis.

ITS 3.1.7, "Standby Liquid Control (SLC) System"
(**Custom TS 3.4.A.1, "Standby Liquid Control")**

This requirement is revised to specify the SLC system to be operable at all times during Modes 1, 2 and 3. The licensee states that this change reflects the use of the SLC system for maintaining suppression pool pH following a design-basis LOCA involving fuel damage, as assumed in the radiological consequence analysis. Existing TS performance requirements (e.g., minimum required flow and boron concentration) of the SLC system were assumed for its pH control. The SLC system is fully capable, without modification, of performing the pH control function.

New Required Action D.2 is added with associated completion times.

As a result of the pH control function, the SLC system is now required to be operable during plant conditions when a LOCA may result. This change is consistent with the NRC staff's evaluation of the SLC system in Section 3.1.1 above, and is thus acceptable.

ITS 3.4.6, "RCS Specific Activity"
(Custom TS 3.6.C and TS 4.6.C, "Coolant Chemistry")

The licensee proposed, consistent with assumptions contained in the AST design-basis main steamline break accident (DBA MSLBA) radiological consequence analysis on radioiodine concentration in reactor coolant, and the guidance of Section 2.0 and Footnote 1 of Appendix D of RG 1.183, changes to existing TS limits on reactor coolant dose equivalent I-131 activity concentrations. The proposed changes will require more restrictive (lower) coolant activity limits on radioiodine concentration in plant operating modes where the AST DBA MSLB accident has been postulated to occur (see evaluation details in Section 3.1.2 above). These changes are, therefore, acceptable.

ITS 3.7.5, "Control Room Ventilation System"
(Custom TS 3.17, "Control Room Habitability")

The licensee proposed to add the word "recently" to various requirements such that two control room ventilation subsystems shall be operable during movement of recently irradiated fuel assemblies in the secondary containment. These changes are consistent with the assumptions of the configuration of these subsystems used in the AST radiological analysis evaluated above.

ITS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"
(Custom TS SR 4.7.D.1.e, "Primary Containment Isolation Valves (PCIVs)"

Condition E is added to include revised leakage requirements for the main steam isolation valves (MSIVs) and the main steam pathway. The main steam pathway includes the combined leakage through the eight MSIVs and through two PCIVs, which isolate the inboard MSIV drain lines. Leakage through the main steam pathway is quantified by summing the leakage from Type C tests for these valves.

This leakage is separate, and in addition to, the primary containment leakage limit of 1.2 percent per day (L_a) at a pressure (P_a) of 42 pounds per square inch gauge (psig). [Note that on the same date of this safety evaluation, the NRC staff has issued an exemption to the licensee on the subject of leakage pathway.] The new Condition E adds specific leakage requirements for this pathway consistent with assumptions included in the AST DBA LOCA radiological consequence analysis regarding radiological releases (see Section 3.1.1 above), and is acceptable.

ITS 3.8.2, "AC [Alternating Current] Sources - Shutdown"
(No corresponding Custom TS)

The licensee proposed to revise various requirements concerning offsite circuit and emergency diesel generators such that under certain conditions of inoperability movement of recently irradiated fuel assemblies in the secondary containment will be suspended. These changes are consistent with the assumptions of the configuration of these subsystems used in the AST radiological analysis evaluated above.

ITS 3.8.5, "DC [Direct Current] Sources - Shutdown"
(No corresponding Custom TS)

The licensee proposed to revise a requirement concerning the DC electrical power subsystem such that when it is inoperable, movement of recently irradiated fuel assemblies in the secondary containment will be suspended. This change is consistent with the assumptions of the configuration of these subsystems used in the AST radiological analysis evaluated above.

ITS 3.8.8, "Distribution Systems - Shutdown"
(No corresponding Custom TS)

The licensee proposed to revise a requirement concerning the AC or DC electrical power distribution subsystems such that when it is inoperable, movement of recently irradiated fuel assemblies in the secondary containment will be suspended. This change is consistent with the assumptions of the configuration of these subsystems used in the AST radiological analysis evaluated above.

ITS 5.5.11, "Primary Containment Leakage Rate Testing Program"
(Custom TS SR 4.7.D.1.e, "Primary Containment Isolation Valves (PCIVs)"

Two new sections are added regarding steam line pathway leakage contribution. The guidelines of RG 1.163 endorses, with certain exceptions, Nuclear Energy Institute (NEI) 94-01 and American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-1994, "Containment System Leakage Testing Requirements." With the granting of the exemption (see Section 3.1.1.4 and **ITS 3.6.1.3** above), certain exceptions are also needed from these associated guidelines. For Type A tests, exceptions are needed from Section 3.2 of ANSI/ANS-56.8-1994 and Sections 8.0 and 9.0 of NEI 94-01, Revision 0. For Type B and Type C tests, exceptions are needed from Section 6.4.4 of ANSI/ANS-56.8-1994 and Section 10.2 of NEI 94-01, Revision 0. The NRC staff finds these changes to be acceptable because they conform with the exemption granted on the same date as this safety evaluation.

3.5 Changes to the TS Bases

The licensee proposed changes to the TS Bases associated with the TS sections evaluated above. The TS Bases are not part of the TS (see 10 CFR Section 50.36(a)). The TS Bases

document is a licensee-controlled document. Accordingly, the NRC staff reviewed the licensee's proposed TS Bases changes and found that they reflect the proposed full-scope implementation of AST as evaluated above in Sections 3.1.

3.6 Summary of NRC Staff Evaluation

As delineated in Section 3.0, the NRC staff concludes that the proposed full-scope implementation of AST for DBA at MNGP has met the requirements and guidance set forth in Section 2.0 above. In addition, the NRC staff has reviewed the proposed TS changes and has found them acceptable as set forth in Section 3.4.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (71 FR 7808). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (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.

Principal Contributor: M. Hart
E. Forrest
B. Lee
P. Tam
L. Brown
K. Parczewski
S. R. Tammara

Date: December 7, 2006

Table 1
MNGP Accident Analysis Parameters

General	
Reactor power (1880 x 1.02), MWt	1918
Dose conversion factors	FGR11/FGR12
Breathing rate, offsite, m^3/s	
0-8 hours	3.5E-4
8-24 hours	1.8E-4
>24 hours	2.3E-4
Breathing rate, control room, m^3/s	3.5E-4
Control room isolation on LOCA signal	
No control room isolation or filtration assumed for non-LOCA DBAs	
Control room normal intake flow, cfm	7440
Control room emergency mode unfiltered infiltration, cfm	500
Control room unfiltered infiltration assumption for non-LOCA DBAs, cfm	1000
Control room minimum filtered pressurization, cfm	900
Control room volume, ft^3	27,000
Control room iodine filter efficiency, %	
Aerosols	98
Elemental	98
Organic	98
Control room occupancy factor	
0-24 hrs	1.0
1-4 days	0.6
4-30 days	0.4
SBGT Filter Efficiency, %	
Aerosols	98
Elemental	85
Organic	85
Offsite χ/Q values, sec/m^3	Table 3-4
Control Room χ/Q values, sec/m^3	Tables 3-1 to 3-3

Table 1 (cont.)

Loss of Coolant Accident (LOCA)

Containment Leakage Pathway	
Onset of gap release phase, min	2.0
Core release fractions and timing	RG 1.183, Table 1
Iodine species distribution	
Particulate	0.95
Elemental	0.0485
Organic	0.0015
Primary containment volume, ft^3	
Drywell	134,200
Torus airspace, minimum	103,340
Drywell to torus airspace mixing assumed > 2 hrs	
Primary containment leakrate, %/day	
0-24 hours (L_a)	1.2
24-72 hours ($0.61*L_a$)	0.732
Greater than 72 hours ($0.5*L_a$)	0.6
Secondary containment bypass, scfh	
0-24 hours (initial bypass)	35.2
24-72 hours (0.61* initial bypass)	21.47
Greater than 72 hours (0.5* initial bypass)	17.6
Secondary containment positive pressure period, min	5
Drywell natural deposition	Powers model, 10th-percentile
Drywell and torus spray removal	Not credited
Ground level release from reactor building wall from 0-5 minutes	
Elevated release from 5 minutes until termination of the accident	
MSIV Leakage Pathway	
Activity same as containment leakage case	
MSIV TS leak rate at 42 psig, scfh*	
One line	100
Total	200
*reduced to 61% of value after 24 hours, then 50% after 72 hours	
Aerosol and elemental iodine removal in piping (see text)	Credited
AEB 98-03 settling velocity used for aerosol deposition, single volume	
Cline model used for elemental iodine removal	

Table 1 (cont.)

Main condenser elemental and aerosol iodine deposition effective efficiency, %	
0-24 hours	98.62
24-72 hours	99.15
Greater than 72 hours	99.31
Release via turbine building HVAC exhaust vent	
ECCS Leakage Pathway	
Iodine species fraction	
Particulate/aerosol	0
Elemental	97
Organic	3
Suppression pool minimum liquid volume, ft^3	68,000
Estimated total ESF leakage, gpm	2.62
Flashing fraction for iodine	0.1
Ground level release from reactor building wall from 0-5 minutes	
Elevated release from 5 minutes until termination of the accident	

Control Rod Drop Accident (CRDA)

Core radial peaking factor	1.7
Fraction of core inventory in fuel rod gap	
Noble gases	0.1
Iodine	0.1
Alkali metals	0.12
Failed rods	850
Percent of failed rods that reach melt, %	1.06
Melted fuel inventory release fraction to coolant	
Noble gases	1.0
Iodine	0.5
Alkali metals	0.25
Fraction of activity released to vessel that enters main condenser	
Noble gases	1.0
Iodine	0.1
Remaining nuclides	0.01

Table 1 (cont.)

Fraction of activity released from main condenser	
Noble gases	1.0
Iodine	0.1
Remaining nuclides	0.01
Control room isolation	Not credited
Case 1 - SJAE operation	
Main condenser free volume, ft^3	68,550
Release rate from main condenser, %/day	1.0
Release duration, hours	24
SJAE flow rate, standard cubic feet per minute (scfm)	360.5
SJAE release holdup time, min	17
Elevated release through offgas stack	
Case 2 - MVP isolation	
Main condenser free volume, ft^3	68,550
Release rate from main condenser, %/day	1.0
Release duration, hours	24
MVP flow rate, scfm	2300
MVP isolation time, sec	10
MVP holdup time, min	0.38
Elevated release through offgas stack until MVP isolation	
Ground level release from turbine building post-MVP isolation	

Main Steamline Break

Reactor coolant activity, µCi/gm dose equivalent I-131	
Maximum equilibrium	0.2
Pre-accident spike	2.0
MSIV closure time, sec	10.5
Total mass release, lbm	91,834
Liquid release, lbm	76,295
Steam release, lbm	15,540

Table 1 (cont.)

All activity in water is assumed to be released	
Release duration (instantaneous ground level), sec	10.5
Iodine species	RG 1.183, Appendix D
Control room isolation	Not credited
Ground level release from turbine building	

Fuel-Handling Accident

Peaking factor	1.7
Fuel rods damaged	125
Decay period, hrs	24
Water depth, ft	≥23
Pool decontamination factor	
Iodine, effective	200
Noble gases	1
Particulate	Infinite
Fraction of core inventory in gap	
I-131	0.08
Kr-85	0.10
Other iodines	0.05
Other noble gases	0.05
Release period, hr	2
Control room isolation	Not credited
SBGT system operation	Not credited
Ground level release from limiting location - reactor building vent	

Table 2

MNGP Licensee Calculated Doses

TEDE (rem)

Event	0-2 hr EAB	30-day LPZ	30-day Control Room
LOCA	1.31	1.72	3.40
Dose Criterion	*25*	*25*	*5*
MSLB			
Equilibrium Activity	0.11	0.02	0.33
Dose Criterion	*2.5*	*2.5*	*5*
Pre-accident Iodine Spike	1.05	0.20	3.25
Dose Criterion	*25*	*25*	*5*
CRDA			
Case 1: SJAE Operation	1.73	0.79	1.70
Case 2: MVP Isolation	0.18	0.08	0.56
Dose Criterion	*6.3*	*6.3*	*5*
FHA	1.61	0.31	4.29
Dose Criterion	*6.3*	*6.3*	*5*

Table 3

Monticello Nuclear Generating Plant

χ/Q (sec/m^3) for Ground Level Releases to the Control Room Intake

Time Period	RB Nearest Wall	RB Vent	Turbine Building Vent
0 - 2 hours	1.00×10^{-2}	2.48×10^{-3}	2.51×10^{-3}
2 - 8 hours	7.09×10^{-3}	1.81×10^{-3}	1.73×10^{-3}
8 - 24 hours	2.75×10^{-3}	6.58×10^{-4}	6.86×10^{-4}
1 - 4 days	1.90×10^{-3}	4.67×10^{-4}	4.70×10^{-4}
4 - 30 days	1.42×10^{-3}	3.49×10^{-4}	3.52×10^{-4}

Table 4

Monticello Nuclear Generating Plant

χ/Q (sec/m^3) for Ground Level Releases to CR via Admin Building Intake

Time Period	RB Nearest Wall	RB Vent	Turbine Building Vent
0 - 2 hours	1.43×10^{-2}	2.47×10^{-3}	2.58×10^{-3}
2 - 8 hours	9.69×10^{-3}	1.76×10^{-3}	1.85×10^{-3}
8 - 24 hours	3.82×10^{-3}	6.31×10^{-4}	7.37×10^{-4}
1 - 4 days	2.65×10^{-3}	4.57×10^{-4}	4.90×10^{-4}
4 - 30 days	1.98×10^{-3}	3.41×10^{-4}	3.84×10^{-4}

Table 5

Monticello Nuclear Generating Plant

χ/Q (sec/m^3) for Elevated Releases to Control Room

Time Period	Control Room Intake	Admin Building Intake
Fumigation	3.37E-4	3.59E-4
0 - 2 hours	3.73E-6	4.02E-6
0 - 8 hours	5.62E-7	5.63E-7
8 - 24 hours	2.20E-7	2.13E-7
1 - 4 days	2.88E-8	2.58E-8
4 - 30 days	1.56E-9	1.25E-9

Table 6

Monticello Nuclear Generating Plant

χ/Q (sec/m^3) EAB and LPZ Analysis

Time Period	Elevated Release		Ground Level Release	
	EAB	LPZ	EAB	LPZ
Fumigation	1.11×10^{-4}	3.86×10^{-5}	---	---
0 - 2 hours	4.22×10^{-6}	3.79×10^{-6}	7.86×10^{-4}	1.53×10^{-4}
0 - 8 hours	---	2.14×10^{-6}	---	8.83×10^{-5}
8 - 24 hours	---	1.61×10^{-6}	---	6.71×10^{-5}
1 - 4 days	---	8.64×10^{-7}	---	3.70×10^{-5}
4 - 30 days	---	3.54×10^{-7}	---	1.57×10^{-5}

Monticello Nuclear Generating Plant

cc:

Jonathan Rogoff, Esquire
Vice President, Counsel & Secretary
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
2807 W. County Road 75
Monticello, MN 55362

Manager, Nuclear Safety Assessment
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

Commissioner
Minnesota Pollution Control Agency
520 Lafayette Road
St. Paul, MN 55155-4194

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
Suite 210
2443 Warrenville Road
Lisle, IL 60532-4351

Commissioner
Minnesota Department of Health
717 Delaware Street, S. E.
Minneapolis, MN 55440

Douglas M. Gruber, Auditor/Treasurer
Wright County Government Center
10 NW Second Street
Buffalo, MN 55313

Commissioner
Minnesota Department of Commerce
85 7th Place East, Suite 500
St. Paul, MN 55101-2198

Manager - Environmental Protection Division
Minnesota Attorney General's Office
445 Minnesota St., Suite 900
St. Paul, MN 55101-2127

Michael B. Sellman
President and Chief Executive Officer
Nuclear Management Company, LLC
700 First Street
Hudson, MI 54016

Nuclear Asset Manager
Xcel Energy, Inc.
414 Nicollet Mall, R.S. 8
Minneapolis, MN 55401