UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY
MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

REQUEST FOR AMENDMENT TO
OPERATING LICENSE DPR-22

REVISION ONE TO LICENSE AMENDMENT REQUEST DATED JULY 26, 1996
REACTOR COOLANT EQUIVALENT RADIOIODINE CONCENTRATION
AND CONTROL ROOM HABITABILITY

Northern States Power Company, a Minnesota corporation, requests authorization for changes to Appendix A of the Monticello Operating License as shown on the attachments labeled Exhibits A, B and C. Exhibit A describes the proposed changes, describes the reasons for the changes, and contains a Safety Evaluation, a Determination of Significant Hazards Consideration and an Environmental Assessment. Exhibit B contains current Technical Specification pages marked up with the proposed changes. Exhibit C contains the affected Monticello Technical Specifications pages with the proposed changes incorporated. Exhibit D and Exhibit E provides a summary of the analysis which supports the proposed change.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By William J Hill
William J Hill
Plant Manager
Monticello Nuclear Generating Plant

On this 11th day of April 1997 before me a notary public in and for said County, personally appeared William J Hill, Plant Manager, Monticello Nuclear Generating Plant, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, that he knows the contents thereof, and that to the best of his knowledge, information, and belief the statements made in it are true and that it is not interposed for delay.

Marvin R Engen
Marvin R Engen
Notary Public - Minnesota
Sherburne County
My Commission Expires January 31, 2000

MARVIN RICHARD ENGEN
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2000

9704180057 970411
PDR ADOCK 05000263
P PDR

EXHIBIT A

Monticello Nuclear Generating Plant

Revision One to License Amendment Request Dated July 26, 1996

Evaluation of Proposed Changes to the Technical Specifications
for Operating License DPR-22

I. Reason for Proposed Changes

Changes are proposed to Technical Specification sections 3.6.C, Coolant Chemistry, and 3/4.17.B, Control Room Emergency Filtration System. In addition, changes are submitted for the bases for these sections. The changes are proposed to establish Technical Specification requirements consistent with modified analysis inputs used for the evaluation of the radiological consequences of a postulated Main Steam Line Break (MSLB) accident, and of a postulated line break in the Reactor Water Cleanup (RWCU) system.

Main Steam Line Break Evaluation

The postulated MSLB accident involves an instantaneous circumferential break of a main steam line outside primary containment. Break flow is limited by a main steam line flow restrictor. Main Steam Isolation Valve (MSIV) closure is initiated due to low steam line pressure and a reactor scram is initiated when the MSIVs begin to close. The current licensing basis for the MSLB accident is provided in USAR Section 14.7.3.2. The analysis of the MSLB radiological consequences presented in the USAR determined that the offsite doses are well below the guidelines of 10CFR100.

Monticello has re-evaluated the radiological consequences of the MSLB accident taking into consideration current regulatory guidance and the analysis inputs of the current licensing basis. The current licensing basis analysis assumes a MSLB with the reactor at full power. During the re-analysis of this accident, it was identified that a postulated MSLB from a hot standby condition would provide a greater potential hazard than a full power MSLB. Hot standby is operation with the reactor critical in the startup mode at a power level sufficient to maintain reactor pressure and temperature. Hot standby is not a normal long term operating condition and the plant is in this condition only briefly during normal plant startups and shutdowns. The postulated MSLB from the hot standby condition results in a high rate of depressurization and rapid rising of water level to the main steam line inlet, thus maximum coolant mass is released through the break. The hot standby MSLB accident has a greater mass release and is thus a more conservative condition for analysis of the radiological consequences of this accident as compared to the full power condition.

The radiological consequences of the more conservative hot standby MSLB accident were analyzed using the current licensing basis analysis inputs for the source term, control room ventilation filter bypass leakage and control room filtration system efficiencies while taking into consideration the appropriate contemporary regulatory guidance and revised inputs for dose conversion factors and atmospheric dispersion factors. This analysis of the hot

standby MSLB determined that the calculated doses do not exceed the exposure guidelines of 10CFR100 and 10CFR50 Appendix A, General Design Criterion 19. These licensing basis analysis inputs provide a conservative analysis of the radiological consequences of the MSLB accident.

The licensing basis inputs used in the analysis of the hot standby MSLB provide a conservative evaluation of this design basis accident. However, the regulatory guidance for the analysis of these events has continued to evolve since the initial licensing of MNGP. MNGP recognizes the modifications to the guidance provided in the NRC Regulatory Guides as well as the guidance provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." MNGP has further analyzed the hot standby MSLB taking into consideration the appropriate contemporary regulatory guidance using updated analysis inputs . A summary of this analysis is provided as Exhibit D, "MNGP MSLBA Evaluation Summary." This further analysis similarly determined conformance to the guidelines of 10CFR100 and 10CFR50 Appendix A, General Design Criterion 19.

To provide additional margin to these exposure criteria for the hot standby MSLB radiological consequences analysis, the analysis inputs used a value for the reactor coolant radioiodine concentration of 2 μCi/gm and a value for Control Room Emergency Filtration system filter efficiency of 98%. The proposed changes to the plant Technical Specifications described in this license amendment request are conservative with respect to the radioiodine concentration and consistent with the Control Room Emergency Filtration Train iodine removal efficiency used as inputs for the MSLB evaluation summarized in Exhibit D.

<u>Reactor Water Cleanup High Energy Line Break Evaluation</u>

As the result of engineering evaluations conducted by Monticello, a discrepancy was identified and confirmed in the mass and energy release calculated for a postulated line break in the Reactor Water Cleanup (RWCU) piping. Plant operation with RWCU in service was shown to be acceptable and bounded by existing radiological analyses with an administrative limit imposed on the reactor coolant dose equivalent radioiodine concentration of 0.25 μCi/gm. This revision to our license amendment request dated July 26, 1996 proposes to incorporate this administrative limit into the plant Technical Specifications.

The limiting RWCU break mass flow rate used in the postulated RWCU line break evaluation had historically been a value of 244 lbm/sec based on break flow rates calculated during initial licensing of the plant. Monticello has re-evaluated the effect of postulated High Energy Line Breaks (HELBs). The break flow rate was determined to be 719 lbm/sec, approximately 3 times greater than the break flow previously used in the original evaluation.

An evaluation of the radiological consequences of the postulated limiting RWCU line break was completed. The release was assumed to occur at ground level without Standby Gas Treatment System (SGTS) filtration. Results of the evaluation show that the radiological consequences of the limiting postulated RWCU line break are well below the guidelines of 10 CFR Part 100 for offsite doses and below the guidelines of 10 CFR Part 50, Appendix A, GDC 19 for control room doses. This evaluation used the Monticello licensing basis source

term for reactor coolant radioiodine provided in the Monticello Updated Safety Analysis Report (USAR). The evaluation credits operator action to isolate the postulated RWCU line break, by closure of the RWCU containment isolation valves, ten (10) minutes after initiation of the postulated break. The motor operated containment isolation valves, which are remotely controlled by the control room operator, are conservatively credited as having a 29 second closure time.

Additional evaluation was performed which demonstrated that the radiological consequences for the full spectrum of postulated RWCU line breaks remains bounded by the radiological consequences of the Main Steam Line Break as evaluated in Section 4.4 of the NRC Safety Evaluation Report, dated March 18, 1970, supporting the Monticello provisional operating license. This evaluation used a source term based on a reactor coolant dose equivalent radioiodine concentration of 0.25 µCi/gm.

The atmospheric dispersion (X/Q) factors used in the analyses provided in Exhibit D and Exhibit E differ from those previously reviewed by the NRC staff. The atmospheric dispersion factors were determined using Monticello site meteorological data and the methodology of NUREG/CR-5055, *Atmospheric Diffusion for Control Room Habitability Assessments*, by J.V. Ramsdell, Pacific Northwest Laboratory, 1988; and Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, Revision 1, USNRC November 1982. The determination of the control room (X/Q) factors used the methodology of NUREG/CR-5055 for ground release from the Reactor Building plenum and the Turbine Building, and used the elevated stack release model from Regulatory Guide 1.145 for elevated releases from the plant Offgas Stack. The determination of offsite (X/Q) factors used the methodology of Regulatory Guide 1.145 for ground release from the Reactor Building plenum and the Turbine Building, and for the elevated releases from the plant Offgas Stack. The calculation which established these revised (X/Q) factors can be provided to the staff for review if desired.

II. Description of Proposed Changes

Pursuant to 10CFR50.90, Northern States Power Company hereby propose the following changes. Proposed changes are indicated by bolded text.

A. Reactor Coolant Chemistry Equivalent Radioiodine Concentration

1. Technical Specification Section 3.6, PRIMARY SYSTEM BOUNDARY, Specification 3.6.C.1, Coolant Chemistry, page 123.

   a) Specification 3.6.C.1 states:

      1. The steady state radioiodine concentration in the reactor coolant shall not exceed 5 microcuries of I-131 dose equivalent per gram of water.

   b) The specification is proposed to be changed to state:

      1. The steady state radioiodine concentration in the reactor coolant shall not exceed **0.25** microcuries of I-131 dose equivalent per gram of water.

2. Technical Specification Section 3.6 and 4.6 Bases, Section C, Coolant Chemistry, page 148.

The bases discussion contained in the first paragraph on page 148 is to be revised to reflect the analysis performed of the Main Steam Line Break Accident radiological consequences. The bases will reflect that the Main Steam Line Break Accident radiological consequences analysis demonstrated that the resulting dose consequences are well within the guidelines of 10CFR100 using the analysis input of 2 microcuries of Iodine-131 dose equivalent per gram of water in the reactor coolant for the steady state radioiodine concentration limit. The bases will reflect that the radiological consequences of a postulated high energy line break in the RWCU system outside the drywell are well within the guidelines of 10CFR100 using the evaluation input of 0.25 microcuries of Iodine-131 dose equivalent per gram of water in the reactor coolant for the steady state radioiodine concentration limit.

B. Control Room Habitability

1. Technical Specification Section 3.17, CONTROL ROOM HABITABILITY, Specification 3.17.B.2, Control Room Emergency Filtration System Performance Requirements, page 229w.

a) Specification 3.17.B.2 states:

2. Performance Requirements

a. Periodic Requirements

(1) The results of the in-place DOP tests at 1000 cfm ($\pm$10%) on HEPA filters shall show $\leq$1% DOP penetration.

(2) The results of in-place halogenated hydrocarbon tests at 1000 cfm ($\pm$10%) on charcoal banks show $\leq$1% penetration.

(3) The results of laboratory carbon sample analysis shall show $\geq$98% methyl iodide removal efficiency when tested at 80°C, 95% R.H.

b) The specification is proposed to be changed to state:

2. Performance Requirements

a. Periodic Requirements

(1) The ***combined*** results of the in-place DOP tests at 1000 cfm ($\pm$10%) for the HEPA filters shall show $\leq$ ***0.3***% DOP penetration.

(2) The results of in-place halogenated hydrocarbon tests at 1000 cfm (±10%) on charcoal banks show ≤ ***0.3***% penetration.

(3) The results of laboratory carbon sample analysis shall show ***the methyl iodide penetration ≤ 0.4% when tested at 30°C and 95% relative humidity.***

2. Technical Specification Section 3.17, CONTROL ROOM HABITABILITY, Specification 3.17.B.3, Post Maintenance Testing Requirements, page 229x.

   a) Specification 3.17.B.3 states:

      3. Post Maintenance Requirements

      a. After any maintenance or testing that could affect the HEPA filter or HEPA filter mounting frame leak tight integrity, the results of the in-place DOP tests at 1000 cfm (±10%) on HEPA filters shall show ≤ 1% DOP penetration.

      b. After any maintenance or testing that could affect the charcoal adsorber leak tight integrity, the results of in-place halogenated hydrocarbon tests at 1000 cfm (±10%) on charcoal adsorber banks shall show ≤1% penetration.

   b) The specification is proposed to be changed to state:

      3. Post Maintenance Requirements

      a. After any maintenance or testing that could affect the HEPA filter or HEPA filter mounting frame leak tight integrity, the ***combined*** results of the in-place DOP tests at 1000 cfm (±10%) on HEPA filters shall show ≤ ***0.3***% DOP penetration.

      b. After any maintenance or testing that could affect the charcoal adsorber leak tight integrity, the results of in-place halogenated hydrocarbon tests at 1000 cfm (±10%) on charcoal adsorber banks shall show ≤ ***0.3***% penetration.

3. Technical Specification Section 3.17, Section A, Control Room Ventilation System, Bases, page 229y. Technical Specification Section 3.17, Section B, Control Room Emergency Filtration System, Bases, page 229y. Technical Specification Section 4.17, Section B, Control Room Emergency Filtration System, Bases, page 229z.

   a) The bases for section 3.17.A, page 229y, are revised to reflect operation of the Control Room Ventilation system as modified to establish improved control room ventilation bypass leakage, which enhances the control room protective envelope. In addition, the bases are revised such that detailed design

information which does not directly support the Technical Specification is removed.

b) The bases for section 3.17.B, page 229y, are revised to reflect an emergency filtration system iodine removal efficiency of 98% as an input in the analysis performed of the Main Steam Line Break Accident radiological consequences. Paragraph two of this section is revised to be consistent with the proposed changes to the Technical Specifications which establish conservative testing criteria with respect to the iodine removal efficiency. Paragraph three of this section of the bases is proposed to be revised to reflect the inputs used in the analysis for control room dose calculations: 85% standby gas treatment system adsorption and filtration efficiency, and 98% control room emergency filtration system adsorption. The control room dose calculations confirmed that control room personnel whole body and organ doses remained within the guidelines of 10CFR50 Appendix A, GDC 19.

c) The bases for section 4.17.B, page 229z, are proposed to be revised to reflect revised guidance for the performance of filter testing for the Emergency Filtration Train System. The bases section is revised to reflect that the laboratory methyl iodide test of the carbon adsorber is to be performed in accordance with ASTM D 3803-89, "Standard Test Method for Nuclear-Grade Activated Carbon." The bases section is revised to reflect that the testing procedures for the in-place testing of the Emergency Filtration Train System HEPA filter and charcoal adsorber will be established using the applicable sections of ASME N510-1989, for procedural guidance.

The bases are revised to provide information concerning the intent of specification 4.17.B.2.a(1) and 4.17.B.3.a, concerning the manner in which the test results of the individual in-place DOP tests of the in series HEPA filters are to be combined to obtain an overall DOP penetration for the filter unit. In accordance with ASME N510-1989, section 10.5.10, when the housing contains more than one bank of HEPA filters in series which are required to be leak tested, the test procedure is to be performed for each bank of filters. The bases for specification 4.17.B are revised to reflect that the results of the individual test of the HEPA filters are to be multiplied to obtain the overall penetration for the unit. The individual test results obtained from in-place penetration testing for the HEPA filter upstream of the charcoal adsorber and of the HEPA filter downstream of the charcoal adsorber unit are to be multiplied together to determine the penetration of the combination of the two filters in series as a unit to satisfy the criteria of the specifications.

In addition, the bases are revised to establish consistency between the Technical Specification bases and Technical Specification 3.17.B.2.b(1). Technical Specification 3.17.B.2.b(1) states that the Control Room Emergency Filtration System shall be shown to be operable with a combined filter pressure drop of less than or equal to eight (8) inches of water. The Technical Specification bases for specification 3.17.B.2.b(1) on page 229z incorrectly states that a pressure drop of less than eight (8) inches of water is indicative that the filters

and adsorbers are not clogged by excessive amounts of foreign matter. The bases on page 229z are revised to state that a pressure drop of less than or equal to eight (8) inches of water is indicative of acceptable system performance.

III. Safety Assessment of the Proposed Changes

A. Coolant Chemistry Equivalent Radioiodine Concentration

Technical Specification Section 3.6, PRIMARY SYSTEM BOUNDARY, Specification 3.6.C.1, Coolant Chemistry, page 123.

Technical Specification Section 3.6 and 4.6 Bases, Section C, Coolant Chemistry, page 148.

A change is proposed to specification 3.6.C.1 and the associated bases to establish the limiting condition for operation such that the steady state radioiodine concentration in the reactor coolant shall not exceed 0.25 microcuries of Iodine-131 dose equivalent per gram of water. This change is proposed to reflect an analysis input used in the evaluation of the radiological consequences of a postulated line break in the Reactor Water Cleanup (RWCU) system.

During circulation, the reactor coolant acquires radioactive material due to activation of corrosion products and release of fission products from potential fuel leaks. The release of coolant during a design basis accident could release radioactive materials into the environment. A limit is established in the plant Technical Specifications on the maximum allowable level of radioactivity in the reactor coolant to ensure that in the event of a release of any radioactive material to the environment due to a postulated high energy line break outside of the primary containment up to and including a design basis Main Steam Line Break Accident, radiation doses are maintained within the guidelines of 10CFR100. The steady state radioiodine concentration in the reactor coolant is an initial input for analysis of the radiological consequences of an accident due to a Main Steam Line Break outside of containment. No fuel damage is postulated in the Main Steam Line Break Accident, and the release of radioactive material from the break ends shortly after the main steam isolation valves close completely.

MNGP has operated well within the current and proposed technical specification limits for reactor coolant concentration of dose equivalent Iodine-131. Data readily available from the last eight operating cycles, shows that the reactor coolant concentration of dose equivalent Iodine-131 has been a small fraction of the Technical Specification limit with a cycle average of 4.5×10^{-3} micro curies per gram over the last eight operating cycles. Even though the reactor core may contain no defective fuel, trace amounts of natural uranium in core construction materials and zircaloy cladding, as well as traces of enriched uranium on the external cladding surface, could be a source of fission products in the coolant during power operation.

The basis for the current Technical Specification limit for the reactor coolant radioiodine concentration of 5 microcuries per gram of dose equivalent Iodine-131 is derived from the analysis of the Main Steam Line Break accident performed by the Atomic Energy

Commission (AEC), predecessor to the NRC, in support of the issuance of the MNGP provisional operating license. License Amendment Number 9 to the Provisional Operating License, issued April 10, 1975, provided a change to the limiting condition for the reactor coolant activity and revised the technical specification bases to be consistent with the standard technical specifications in effect at the time of issuance of the amendment as well as the inputs used in the AEC analysis of the Main Steam Line Break Accident.

The Main Steam Line Break Accident radiological consequences have been reanalyzed. This analysis was performed using modified inputs consistent with the current regulatory guidance. The analysis conservatively used an input for the dose equivalent reactor coolant Iodine-131 concentration of 2 microcuries per gram. This analysis input provides a conservative assessment of the potential radiological consequences. The results of this analysis are provided in Exhibit D. The analysis demonstrated that the dose consequences from a potential Main Steam Line Break outside of primary containment are well within the guidelines of 10CFR100 and 10CFR50 Appendix A, General Design Criterion 19. The Technical Specification bases are to be revised to delete bases information which is not pertinent to the specification and to reflect the appropriate information which is consistent with the accident analysis.

The proposed change to the Technical Specification to limit the dose equivalent reactor coolant Iodine-131 concentration to 0.25 μCi/gm is conservative with respect to the evaluation input of 2 microcuries per gram dose equivalent reactor coolant Iodine-131 used for the evaluation of the radiological consequences of the postulated Main Steam Line Break Accident. Thus, the proposed change is acceptable with respect to establishing a limit on a process variable that is an initial condition for the postulated design basis Main Steam Line Break Accident.

Monticello has identified that a postulated High energy Line Break (HELB) in the RWCU system outside of containment, as evaluated with a revised mass release from the postulated break, is inconsistent with licensing basis information contained in the Monticello Updated Safety Analysis Report (USAR). The Monticello USAR states that the radiological consequences of a postulated RWCU system line break are bounded by those of the Main Steam Line Break Accident. An evaluation of the radiological consequences of the postulated limiting RWCU line break was completed. The release was assumed to occur at ground level without SGTS filtration. The evaluation credits operator action to isolate the postulated RWCU line break, by closure of the RWCU containment isolation valves, ten (10) minutes after initiation of the postulated break. The motor operated containment isolation valves, which are remotely controlled by the control room operator, are credited as having a 29 second closure time. The break flow is assumed as linearly proportional to the valve closure upon initiation of valve closure. This evaluation used a source term based on a reactor coolant dose equivalent radioiodine concentration of 0.25 μCi/gm. This source term is consistent with the change to Technical Specification 3.6.C.1 proposed by this License Amendment request. The results of this evaluation are provided in Exhibit E.

Reducing the allowable reactor coolant radioiodine concentration during plant operation to 0.25 μCi/gm will establish a limit on a process variable that is an initial condition for

the evaluation of a postulated high energy line break in the RWCU system. Evaluation of the radiological consequences of postulated RWCU breaks, using a source term based on a reactor coolant dose equivalent radioiodine concentration of 0.25 μCi/gm, demonstrates that the radiological consequences remain bounded by the Main Steam Line Break as evaluated in Section 4.4 of the NRC Safety Evaluation Report, dated March 18, 1970, supporting the Monticello provisional operating license. Monticello has imposed an administrative limit on dose equivalent radioiodine of 0.25 μCi/gm as an interim measure to provide assurance that the plant will operate in a manner consistent with the evaluation of this postulated event.

The evaluation credits operator action to isolate the postulated RWCU line break. The Monticello RWCU containment isolation valves receive an automatic closure signal on low reactor water level and high containment pressure. Automatic isolation of a postulated RWCU line break on low reactor water level will occur when the plant is operating near full power. At lower power levels, automatic RWCU isolation on low reactor water level may not mitigate the event consequences for all postulated break scenarios. For the postulated RWCU line break at reduced power levels, the excess capacity of the feedwater system will be able to compensate for the mass loss from the reactor vessel via the postulated break and steam generation being transferred to the power conversion systems. Performance of a mass balance on the reactor vessel has determined that for a reactor power of 77%, automatic isolation due to low reactor water level would occur at 10 minutes, coincident with the assumed operator action to initiate break isolation. The time for automatic isolation increases for lower power levels.

The re-evaluation of the postulated RWCU line break assumed operator action to isolate the postulated break. The postulated break is terminated by closure of the RWCU containment isolation valves. The motor operated containment isolation valves are credited as having a 29 second closure time. The closure time of the containment isolation valves credited in this evaluation is consistent with the limiting stroke time established for these valves in accordance with the plant Inservice Testing Program and is periodically tested per plant Technical Specification 4.15.B.

The indications credited for detection of the postulated RWCU line break rely on components which are not classified as safety related. Credit for nonsafety related equipment and operator actions for mitigating the consequences of this postulated RWCU line break is consistent with regulatory guidance provided in section B.3.b(4) of NRC Branch Technical Position SPLB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment." NRC Branch Technical Position SPLB 3-1 is provided as Appendix A to section 3.6.1 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." Section B.3.b(4) of NRC Branch Technical Position SPLB 3-1 states:

> All available systems, including those actuated by operator actions, may be employed to mitigate the consequences of postulated piping failure.

The control switches for the RWCU containment isolation valves are located in the control room and are readily available to the control room operators on the control room front panels. Isolation of the postulated break is not considered a complex evolution.

Multiple indications are available to the control room operator to provide indication of the postulated RWCU line break. These indications include annunciated alarms on high temperature in the area of RWCU room, RWCU system high/low pressure, and RWCU system filter/demineralizer trouble. Additional indications of a postulated RWCU line break would be provided by changes in feedwater system flow, reactor water level, and area radiation levels. In addition, alarms in the Radwaste Control Room annuciate due to loss of RWCU filter flow.

The ten (10) minute operator action time is considered to be conservative based on the very limited action required by the operator to close the containment isolation valves and the availability of control room indications to alert the operator to the postulated break. Thus, operator response to this postulated event will more likely be significantly less than ten (10) minutes. The use of a ten (10) minute operator response time to take manual actions in response to postulated events is consistent with Monticello's licensing basis for similar events. Monticello USAR sections 5.2.3.3, 6.2.3.2.2, 7.6.3.3.2.c, and 7.11.2 establish ten (10) minutes as the time period to be credited prior to initiation of operator actions.

Monticello has evaluated the thermal-hydraulic effects of the postulated RWCU system line break. The effect on Environmental Qualification of equipment was reviewed and it was concluded that the ability to safely shutdown the plant is maintained with the limiting postulated line break. Internal building flooding which could result from this postulated break was evaluated and found to have no adverse effects. The effect on the integrity of structures was evaluated and it was found that the break had no adverse effect on the Primary Containment or Reactor Building structures. The effect on the integrity of the RWCU Room structure was evaluated and it was determined that block walls associated with the room may fail, but safe shutdown equipment needed to mitigate the consequences of the postulated event would not be impacted by debris. Motor operated valve operability was examined and all affected valves were confirmed to be capable of performing their design basis functions.

The bases for the Technical Specifications state that radioiodine concentration can change rapidly in the reactor coolant during transient reactor operations such as reactor shutdown, reactor power changes and reactor startup if failed fuel is present. As specified in the Technical Specifications, additional reactor coolant samples shall be taken and analyzed for reactor operations in which steady state radioiodine concentrations in the reactor coolant indicate iodine release from the fuel. The capability to detect fuel element failures is inherent in the radiation monitors in the off-gas system.

The Monticello Technical Specification requirements established in specification 3.8/4.8.B "Gaseous Effluents" provide conservative limits on gaseous effluents consistent with the guidelines established in 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents." The gaseous effluents subject to the requirements of specification 3.8/4.8.B are directly related to the reactor coolant radioiodine concentration. Maintaining off-gas releases to within acceptable values provides

adequate operating constraints such that it is unlikely for transient reactor operations to result in radioiodine concentrations greater than the proposed Technical Specification value. Based on operating experience, if fuel defects were to occur, appropriate plant actions would be taken based upon off-gas release values which would ensure the integrity of the MSLB and RWCU line break safety analyses.

The proposed change to the plant Technical Specifications and Bases is consistent with the analyses performed of the Main Steam Line Break Accident and Reactor Water Cleanup line break radiological consequences. The proposed changes do not result in a significant increase in the probability or consequences of postulated accidents previously analyzed, an accident not previously analyzed, or in a significant reduction in the margin of safety.

B. Control Room Habitability

Technical Specification Section 3.17, CONTROL ROOM HABITABILITY, Specification 3.17.B.2, Control Room Emergency Filtration System Performance Requirements, page 229w.

Technical Specification Section 3.17, CONTROL ROOM HABITABILITY, Specification 3.17.B.3, Post Maintenance Testing Requirements, page 229x.

Technical Specification Section 3.17, Section A, Control Room Ventilation System, Bases, page 229y. Technical Specification Section 3.17, Section B, Control Room Emergency Filtration System, Bases, page 229y. Technical Specification Section 4.17, Section B, Control Room Emergency Filtration System, Bases, page 229z.

Changes are proposed to the plant Technical Specifications and bases to modify the limiting conditions for operation and surveillance requirements for the control room emergency filtration system. The changes are proposed to establish conservative limiting conditions and testing criteria with respect to the filter and charcoal adsorber efficiencies used in the analysis of control room operator doses for design basis accidents.

The function of Control Room Ventilation-Emergency Filtration Train (CRV-EFT) System is to maintain the environment of the Main Control Room, thereby ensuring it's habitability during normal and accident conditions. The CRV-EFT System is composed of two subsystems; the Control Room Ventilation (CRV) subsystem and the Emergency Filtration Train (EFT) subsystem. The function of the CRV portion of the system is to provide the control room and the first and second floors of the EFT building (Reactor Building Addition) with conditioned air to maintain acceptable temperature conditions during normal operation. The EFT subsystem provides for isolation of the control room and the first and second floors of the EFT building from outside air during a toxic chemical release or an accident where high levels of activity may be released. During a radiological accident, the EFT provides for immediate automatic pressurization of the Control Room with filtered air to minimize the activity, and therefore the radiological dose, inside the control room. The redundant air filtration units consist of the following components in series: a low efficiency filter, an electric heating element, a High

Efficiency Particulate Air (HEPA) filter, two 2-inch charcoal adsorber beds, a HEPA filter, and a centrifugal fan. The charcoal adsorber removes gaseous iodine, while the HEPA filters remove particulate matter.

The EFT system is designed to satisfy the criteria of NUREG 0737, Section III.D.3.4 which imposes the criteria of 10CFR50 Appendix A, General Design Criterion (GDC) 19. The EFT is designed to provide adequate radiation protection to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Section 6.7 of the MNGP USAR provides additional information on the CRV-EFT system.

The Main Steam Line Break Accident radiological consequences have been re-analyzed. The inputs used in this analysis provide a conservative assessment of the potential radiological consequences. The results of the analysis are provided in Exhibit D. The radiological analyses reflect an improvement for the control room emergency filtration system filter efficiency and reduced control room ventilation bypass leakage.

A modification to the control room emergency filtration system has been performed to establish the improved control room ventilation bypass leakage. With this modification, pressurization of the control room during normal system operation will not be a system function; however filtered pressurizing air will be supplied to the main control room upon automatic initiation of the EFT system. This modification enhances the isolation of the envelope supplied by the EFT system. In addition, it minimizes the potential for the introduction of non-filtered air during short term puff releases. The lack of normal pressurization for the control room does not have an adverse effect on control room habitability.

A 98% iodine removal efficiency for the EFT system was used as an input for the dose consequence analysis. A 98% removal efficiency is within the capability of the EFT system charcoal adsorber as demonstrated in previous surveillance tests. Changes are proposed to the limiting conditions for operation and surveillance requirements for the EFT system consistent with the analysis inputs for the iodine removal efficiency used in the postulated Main Steam Line Break and the RWCU high energy line break radiological consequences evaluation. The proposed changes are consistent with the applicable regulatory guidance provided in NRC Generic Letter 83-013; Regulatory Guide 1.52, Revision 2; and NUREG-1433.

A performance criterion of less than or equal to 0.4% methyl iodide penetration for laboratory testing is proposed for specification 3.17.B.2.a(3) consistent with the 98% charcoal adsorber efficiency. A safety factor of 5 was used in determining this test criterion consistent with applicable regulatory guidance for an EFT system with heaters in line with the process stream. The laboratory test of the EFT system charcoal adsorber is to be performed in accordance with American National Standard ASTM D 3803-89, "Standard Test Method for Nuclear-Grade Activated Carbon." The test conditions are proposed to be revised to 30°C and 95% relative humidity consistent with ASTM D3803-89 and NUREG-4960 section 5.2.3.3.f. These test conditions are more representative of the service conditions for the EFT system and establish a more

conservative test condition. ASTM D3803-89 has been determined to be a more appropriate and an enhanced specification governing this testing. The current Technical Specification test condition of 80°C is consistent with ASTM D3803-1979; however, surveillance test have been performed at both the 80°C and 30°C test conditions for several surveillance cycles. A review of the test data for the test conditions of 80°C and 30°C supports the 30°C test condition as a more conservative test condition. Guidance provided in ASTM D3803-89 states that the 30°C, 95% relative humidity methyl iodide test is the most reliable test method to establish iodide removal efficiency of any adsorbent.

The proposed changes to performance criteria and post maintenance testing criteria specified in technical specification requirements 3.17.B.2.a(1), 3.17.B.2.a(2), 3.17.B.3.a, and 3.17.B.3.b establish in-place testing criteria consistent with the regulatory guidance of NRC Generic Letter 83-013 for a system with a HEPA and charcoal adsorber efficiency of 98%. For in-place penetration testing of the HEPA filters with dioctyl phthalate (DOP), the HEPA filters upstream and downstream of the charcoal adsorbers are tested individually, with the individual tests then factored together to reflect the efficiency of the two HEPA filters in combination to satisfy the proposed criteria for Technical Specifications. The individual test results obtained from in-place penetration testing for the HEPA filter upstream of the charcoal adsorber and of the HEPA filter downstream of the charcoal adsorber unit are to be multiplied together to determine the penetration of the combination of the two filters in series as a unit to satisfy the criteria of the specifications. The bases of the technical specifications are proposed to be revised to reflect the revised testing criteria and test methods.

The bases for section 4.17.B, page 229z, are proposed to be revised to reflect revised guidance for the performance of in-place filter testing for the Emergency Filtration Train System. The bases section is revised to reflect that the testing procedures for the in-place testing of the Emergency Filtration Train System HEPA filter and charcoal adsorber will be established using the applicable sections of ASME N510-1989, for procedural guidance. Section 10, "HEPA FILTER BANK IN-PLACE TEST," and section 11 "ADSORBER BANK IN-PLACE TEST," of ASME N510-1989 provide guidance for the performance of the testing specified in Technical Specifications requirements 3/4.17.B.2.a(1), 3/4.17.B.2.a(2), 3/4.17.B.3.a, and 3/4.17.B.3.b. Section 11 of ASME N510-1989 is to be used as guidance for the performance of charcoal adsorber in-place testing with the following clarifications of sections 11.3, 11.4 and 11.5.8.

Section 11.3 of ASME N510-1989 states that sample points shall be downstream of a fan. Monticello experience has concluded that sampling down stream of the Emergency Filtration Train fan may result in a sample which is not representative of the charcoal adsorber flow stream. Thus the sample point is to be located upstream of the fan.

Section 11.4 of ASME N510-1989 states that R-11 is the preferred test gas with R-112 or R-112A as acceptable alternatives. Monticello reserves the ability to use alternate test gases that our found to be acceptable alternatives by the industry. Monticello currently employs R-11 as the test gas; however, concerns regarding the use of such halide gases may result in use of these gases not being feasible in the future.

Section 11.5.8 of ASME N510-1989 states that when the housing contains more than one bank of adsorbers in series to repeat the test procedure for each bank. Monticello intends to test the in series charcoal adsorbers as a unit. Testing of the in series charcoal adsorbers individually was not a requirement under ASME N510-1980 and is not feasible at Monticello.

In addition, the bases are revised to establish consistency between the Technical Specification bases and Technical Specification 3.17.B.2.b(1). Technical Specification 3.17.B.2.b(1) states that the Control Room Emergency Filtration System shall be shown to be operable with a combined filter pressure drop of less than or equal to eight (8) inches of water. By letter dated July 5, 1988, with subject "Revision 4 to License Amendment Request Dated April 3, 1984," Monticello responded to an NRC staff question regarding the appropriate criterion for specification 3.17.B.2.b.1 for the combined pressure drop across the HEPA filters and charcoal adsorber. Monticello stated the system was designed to have a pressure drop eight (8) inches of water. Amendment 65 was issued May 30, 1989 and specified an acceptance criterion of less than or equal to eight (8) inches of water for the combined filter pressure drop; however, the bases for the specification incorrectly discusses the acceptable value for the pressure drop in that it omits that values equal to 8 inches are acceptable. A review of design basis information concerning the Emergency Filtration Train confirms that the appropriate acceptance criterion for the combined pressure drop across the HEPA filters and charcoal adsorbers is less than or equal to eight (8) inches of water. The bases on page 229z are revised to state that a pressure drop of less than or equal to eight (8) inches of water is indicative of acceptable system performance.

The proposed changes to the Technical Specifications are provided to ensure testing is performed which provides a high level of assurance of the capability of the EFT system to perform as analyzed in the evaluation of the control room operator doses resulting from a postulated Main Steam Line Break. The analysis determined that control room operator doses remain below the regulatory guidelines of 10CFR50 Appendix A, GDC 19. The proposed changes to the Technical Specifications ensure continued compliance with NUREG-0737, Item III.D.3.4 which requires that nuclear power plants be equipped with a control room from which actions can be taken to operate the plant safely under normal and accident conditions. The testing criteria has been established in accordance with the applicable regulatory guidance while providing conservative margin to the analytical inputs used in the safety analyses, thus the proposed changes are acceptable. The proposed changes do not result in a significant increase in the probability or consequences of postulated accidents previously analyzed, an accident not previously analyzed, or in a significant reduction in the margin of safety.

IV. Determination of Significant Hazards Considerations

The proposed change to the Operating License has been evaluated to determine whether it constitutes a significant hazards consideration as required by 10CFR50.91 using standards provided in 10CFR50.92. This analysis is provided below:

A. **The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.**

A limit is established in the plant Technical Specifications for steady state radioiodine concentration in the reactor coolant to ensure that in the event of a release of radioactive material to the environment due to a postulated high energy line break up to and including a design basis Main Steam Line Break Accident, radiation doses are maintained within the guidelines of 10CFR100. The steady state radioiodine concentration in the reactor coolant is an input for analysis of the radiological consequences of an accident due to a Main Steam Line Break outside of containment and postulated high energy line breaks. In addition, requirements are established in the Technical Specifications for control room habitability. During an accident, the control room emergency filtration system provides filtered air to pressurize the Control Room to minimize the activity, and therefore the radiological dose, inside the control room.

A change is proposed for the steady state radioiodine concentration. This value is conservative with respect to the value used in the Main Steam Line Break dose consequences analysis and is consistent with the dose consequences evaluation of a postulated Reactor Water Cleanup (RWCU) line break. Changes are proposed to the limiting conditions for operation and surveillance requirements for the Control Room Emergency Filtration Train iodine removal efficiency. These changes are consistent with the inputs used in the analysis of the radiological consequences of the postulated RWCU line break and the Main Steam Line Break Accident. These proposed requirements maintain operating restrictions for analytical inputs used in the analysis of the Main Steam Line Break Accident. Evaluation of these events has demonstrated that the postulated radiological consequences will remain within the licensing basis established in the AEC Provisional Operating License Safety Evaluation Report, dated March 18, 1970, thus the proposed changes do not result in an increase in the consequences of previously evaluated accidents.

The analysis of the Main Steam Line Break Accident performed using a reactor coolant radioiodine concentration of 2 μCi/gm dose equivalent Iodine-131 and a control room ventilation filter efficiency consistent with the proposed Technical Specifications changes demonstrated that radiological consequences of the Main Steam Line Break are not changed significantly. The radiological consequences of the Main Steam Line Break Accident remain within the exposure guidelines of 10CFR100 and 10CFR50 Appendix A, General Design Criterion 19. The offsite dose consequences remain bounded by the licensing basis provided in the AEC Provisional Operating License Safety Evaluation Report, dated March 18, 1970. The control room doses calculated for the hot standby Main Steam Line Break Accident using the TID-14844 dose conversion factors remain bounded by the dose consequences of the comparable design basis loss of coolant accident.

The evaluation of the postulated RWCU line break, performed using a reactor coolant radioiodine concentration of 0.25 μCi/gm dose equivalent Iodine-131 and a control room ventilation filter efficiency consistent with the proposed Technical Specifications changes, demonstrated that the radiological consequences of this event remain within the exposure guidelines of 10CFR100 and 10CFR50 Appendix A, General Design

Criterion 19. The offsite dose consequences remain bounded by the Main Steam Line Break as established in the licensing basis provided in the AEC Provisional Operating License Safety Evaluation Report, dated March 18, 1970.

The proposed Technical Specification changes do not introduce new equipment operating modes, nor do the proposed changes alter existing system inter-relationships. The proposed changes do not introduce new failure modes. The system improvements to reduce bypass leakage during postulated accidents do not have an adverse effect on control room habitability. Therefore, this amendment will not cause a significant increase in the probability of an accident previously evaluated for the Monticello plant.

B. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed Technical Specification changes do not introduce new equipment operating modes, nor do the proposed changes alter existing system inter-relationships. Operator action to mitigate the consequences of the postulated RWCU line break is conservative based on the very limited action required by the operator to close the containment isolation valves and the availability of control room indications to alert the operator to the postulated break. The use of a ten (10) minute operator response time to take manual actions in response to postulated events is consistent with Monticello's licensing basis for similar events. The use of operator actions and all available equipment is consistent with current regulatory guidance for mitigating the consequences of postulated line breaks.

The proposed change to the specification for reactor coolant dose equivalent radioiodine is conservative with respect to the re-evaluation of the Main Steam Line Break Accident for the more conservative hot standby initial condition for the postulated accident. The proposed change to the specification for reactor coolant dose equivalent radioiodine is consistent with the postulated high energy line break of a Reactor Water Cleanup line. The proposed changes to the limiting conditions for operation and surveillance requirements for the control room emergency filtration train iodine removal efficiency are consistent with the inputs used in the evaluation of the radiological consequences of the postulated RWCU line break and the Main Steam Line Break Accident. The system improvements to reduce bypass leakage during postulated accidents do not have an adverse effect on control room habitability. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident.

C. The proposed amendment will not involve a significant reduction in the margin of safety.

Surveillance data has demonstrated the proposed requirements are within the current capability of the facility. The proposed changes maintain margins of safety. These proposed requirements maintain operating restrictions for analytical inputs used in the analysis of the bounding postulated high energy line break of a Reactor Water Cleanup line and the Main Steam Line Break Accident. The proposed change to the specification for reactor coolant dose equivalent radioiodine is conservative with respect to the re-evaluation of the Main Steam Line Break Accident for the more conservative hot standby

initial condition for the postulated accident. The proposed change to the specification for reactor coolant dose equivalent radioiodine is consistent with the postulated high energy line break of a Reactor Water Cleanup line. The evaluation of these postulated events determined that the radiological consequences remain within the exposure guidelines of 10CFR100 and of 10CFR50 Appendix A, General Design Criterion 19. The proposed changes to the limiting conditions for operation and surveillance requirements for the control room emergency filtration train iodine removal efficiency provide assurance that the system will perform at the filter efficiency as used in the evaluation of the radiological consequences of the postulated events. Therefore, the proposed amendment will not involve a significant reduction in the margin of safety.

V. Environmental Assessment

Northern States Power Company has evaluated the proposed changes and determined that:

1. The change does not involve a significant hazards consideration.

2 The changes do not involve a significant change in the type or significant increase in the amounts of any effluent that may be released offsite, or

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR Part 51, Section 51.22(b), an environmental assessment of the proposed changes is not required.

Exhibit B

Monticello Nuclear Generating Plant

Revision One to License Amendment Request dated July 26, 1996

Proposed Changes Marked Up on Existing
Technical Specification Pages

Exhibit B consists of the existing Technical Specification pages with the proposed changes marked up on those pages. Existing pages affected by this change are listed below:

Page

123
148
229w
229x
229y
229Z