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

LICENSE AMENDMENT REQUEST DATED MAY 5, 1986

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 Exhibit A and Exhibit B. Exhibit A describes the proposed changes, reasons for the changes, and a significant hazards evaluation. Exhibit B is a copy of the Monticello Technical Specifications incorporating the proposed changes.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By ______________________
David Musolf
Manager-Nuclear Support Services

On this 5th day of May, 1986 before me a notary public in and for said County, personally appeared David Musolf, Manager-Nuclear Support Services, 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 is is not interposed for delay.

Dody A Brose
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DODY A. BROSE
NOTARY PUBLIC — MINNESOTA
HENNEPIN COUNTY
My Commission Expires Dec. 26, 1989

Exhibit A

Monticello Nuclear Generating Plant

License Amendment Request Dated May 5, 1986

Description and Evaluation of Proposed
Change to Appendix A of Operating License DPR-22

Pursuant to 10 CFR Part 50, Section 50.90, the holders of Operating License DPR-22 hereby propose the following changes to Appendix A, Technical Specifications:

Miscellaneous Technical Specification
Improvements and Clarifications

Proposed Changes

1. Definition of Core Alteration

Revise the definition of "Alteration of the Reactor Core" on page 1 by adding the words, "with the vessel head removed and fuel in the vessel," to the end of the first sentence.

Reason: Literal interpretation of the existing definition would require movements to be considered core alterations even when fuel is not present within the vessel. This is not necessary and it creates conflicts with other Specifications. For example, Specification 3.10.B, which uses this definition, would require SRM operability when no fuel is in the vessel.

2. 2.3 Bases Correction

Delete the partial sentence in the first line of the first paragraph of the Section 2.3 Bases on page 17.

Reason: These words should have been deleted with a previous License Amendment Request. NSP neglected to request this change earlier through an oversight. The partial sentence is no longer necessary.

3. Table 3.1.1, Startup Mode Operability Requirements

Move the reference to Note 4 from the "Refuel" column to the "Trip Function" column on page 28 so the note is applicable to all modes and add the following note to the table on page 30:

9. High reactor pressure and main steam line radiation high radiation are not required to be operable when the reactor vessel head is unbolted.

Add a reference to Note 9 to the table entries for high reactor pressure and main steam line high radiation on pages 28 and 29.

Reason: The Startup Mode operability requirements listed for high drywell pressure, high reactor pressure, and main steam line high radiation are unnecessarily restrictive for some activities such as low power physics tests. It is desirable to eliminate these unnecessary requirements. With the vessel head unbolted, high reactor pressure and steam line radiation functions are not necessary. High drywell pressure functions are not necessary when containment integrity is not required.

4. Table 4.1.1 and Table 4.2.1, Variable Surveillance Frequencies, and Associated Bases

Delete Note 1 of Table 4.1.1 on page 33 and Note 1 of Table 4.2.1 on page 63a. Delete Figure 4.1.1 and correct the List of Figures to reflect deletion of this Figure. Delete all references to Note 1 on both tables and replace with a requirement for monthly surveillance. Delete those portions of the 4.1 and 4.2 Bases on pages 41 - 45 and 72 - 76 which refer to variable surveillance frequencies.

Reason: Note 1 of these Tables allows certain surveillance intervals to be lengthened up to a maximum of three months by application of Figure 4.1.1. Lengthening of surveillance intervals is based on the number of unsafe failures that are experienced over a period of time. Several years ago NRC Inspection and Enforcement personnel asked NSP not to use Figure 4.1.1 to lengthen surveillance intervals. We are reluctant to do so also, since this would require periodic changes in test intervals and records. As a result, monthly testing has always been conducted even though the Technical Specifications would permit less frequent testing. Because this is a potentially confusing situation, we ask that the Technical Specifications be revised to eliminate the option to extend surveillance intervals.

5. SRM Not-Full-In Rod Block Interlock Conflicts

Add a Note 9 to Table 4.2.1 on page 63a as follows:

> 9. Testing of the SRM Not-Full-In rod block is not required if the SRM detectors are secured in the full-in position.

Add a reference to Note 9 on page 61 under item 8 of Rod Blocks. Change the item to read, "SRM Detector Not-Full-In Position instead of, "...Not in Start-Up Position." Change the sensor check requirement from "Note 2" to "None."

Reason: The existing testing requirement for the SRM Not-Full-In rod block interlock conflicts with normal CRD maintenance work. Literal interpretation of the test requirements could be inconsistent with the normal and prudent practices of rerouting the SRM cables to allow CRD maintenance and securing the SRM detectors in the full-in position. Also, it is not possible to perform the required sensor check of the interlock. It is an on-off device, not an analog signal subject to sensor checks.

Because of the need to reroute the drive cables to allow normal CRD work, and the lack of space under the vessel, it is preferable not to perform detector withdrawals for testing. Such testing would damage the reconfigured cable. Restoration of original cable configuration to allow testing would result in additional personnel exposure, additional wear, and risk of damage to cable and connector assemblies. Verifying that the detectors are full-in and securing the detector drive power in "off" enforces the condition under which the interlock is satisfied.

Although NRC inspectors have accepted this procedure in the past for the reasons stated, it is desirable to revise the Technical Specification wording to agree with practice.

6. Clarification of Containment Isolation Instrumentation Surveillance on Table 4.2.1, Pages 61 and 62

Expand the headings for main steam, HPCI, and RCIC isolation by adding a reference to the containment isolation group on pages 61 and 62 and add a new category for Group 2 and Group 3 containment isolation. Delete Note 7 on page 63a and delete all references to Note 7 in the Table. Add a new Note 10 on page 63a as follows:

10. Uses contacts from scram system. Tested and calibrated in accordance with Tables 4.1.1 and 4.1.2.

Add a reference to Note 10 on page 62 for containment isolation Group 2 reactor low water level and drywell high pressure surveillance.

Reason: The existing Technical Specification surveillance requirements for containment isolation functions is misleading and imprecise. The proposed changes will clarify and expand the table to more clearly list the containment isolation logic inputs and also note that some of the signals for Group 2 are derived from the scram logic.

Note 7 was added as part of an earlier Technical Specification change. The intent was to have the Table cover containment isolation surveillance. The earlier change was not as precise as it should have been.

7. Bases for Specification 4.0

Add a Bases section to explain the derivation of the general surveillance testing requirements in Section 4.0 of the Technical Specifications and add information to assist in understanding and interpreting this section.

The proposed wording is derived from the NRC Standard Technical Specifications. Additional clarification of the surveillance interval tolerance is derived from clarifying information contained in NRC Inspection Report 50-263/85012(DRP) dated July 19, 1985.

Reason: This addition to the Bases will help in understanding and interpreting Section 4.0 of the Technical Specifications. This section was recently added to provide general requirements for the Surveillance Program. The proposed wording summarizes the interpretations related to surveillance intervals and surveillance scheduling established with NRC inspectors over a period of many years.

8. Table 3.2.5, Note 1, ATWS Instrumentation Requirements

Revise Note 1 of Table 3.2.5 on page 60 to read:

1. When one of the two trip systems is made or found to be inoperable, restore the inoperable trip system to operable status within 14 days or place the plant in the specified required condition within the next eight hours. When both trip systems are inoperable, place the plant in the specified required condition within eight hours unless at least one trip system is sooner made operable.

Reason: The existing Note 1 is inconsistent with the requirements for minimum number or operable or operating trip systems in the Table. A loss of one trip system would require a plant shutdown since it is not possible to place a trip system in a tripped condition without actually causing actuation of the logic (this is 1 of 2 logic). As long as the remaining trip system is operable, this is an unnecessary requirement. One operable trip system is sufficient to initiate the required protective action. We believe this was an oversight on our part when this Technical Specification change was first proposed by NSP.

9. Control Rod Accumulator Operability Clarification

Delete the last paragraph of Specification 3.3.D on page 82, and redesignate items 1 and 2 under 3.3.D as items 3.3.D.1(a) and (b). Reword the opening paragraph as follows:

Control rod accumulators shall be operable in the in the Startup, Run, or Refuel modes except as provided below:

Add Specification 3.3.D.2 as follows:

2. In the Refuel Mode, a rod accumulator may be inoperable provided:

   (a) All fuel is removed from the cell containing the associated control rod, or
   (b) The one-rod-out refuel interlock for the associated rod drive is operable.

Reason: A literal interpretation of the last paragraph of this Specification could preclude normal CRD maintenance where the rod out refuel interlock is not bypassed and fuel is not removed from the cell. This could force cell unloading for all drive changeouts. This was not the intent of the Specification. The Specification was intended to apply only to the situation of multiple CRD removal for extended core and control rod drive maintenance, controlled by Specification 3.10.E, where the rod out interlock is bypassed for withdrawn rods. It is not impractical to unload fuel cells during actual refueling with the vessel head removed. It is impractical, however, to require cell unloading for situations requiring drive maintenance after the core is fully reloaded and the head has been replaced.

10. HPCI, APRS, and RCIC Operability Conditions

Revise Specifications 3.5.D.1, 3.5.E.1, and 3.5.F.1 on pages 108, 110, and 111 respectively so that operability of these systems is not required above 150 psig during reactor coolant system leakage and hydrostatic tests by revising the Operability condition to read, "...whenever the reactor pressure is above 150 psig and irradiated fuel is in the reactor vessel, except during reactor vessel hydrostatic or leakage tests." Also, reformat pages 109 and 110 to move the headings for the APRS sections to the top of page 110.

Reason: A literal interpretation of the current requirements for HPCI, APRS, and RCIC system operability conflicts with the requirement to perform reactor coolant system leakage tests following each refueling outage and reactor coolant system hydrostatic tests at ten-year intervals and following major system repairs or modifications. The Technical Specifications currently require these systems to be operable when irradiated fuel is in the vessel and reactor pressure is greater than 150 psig. Since these systems are designed to operate from a source of steam, they cannot be made operable during leakage or hydrostatic tests when the vessel is flooded and reactor coolant temperature is below saturation temperature.

NRC Region III Inspection and Enforcement and NRC NRR personnel were contacted during the last outage to obtain concurrence with the logical interpretation that these systems are in fact not required to be operable during leakage and hydrostatic tests. The Technical Specifications for many other boiling water reactor plants contain similar conflicts. The propose wording change will eliminate the need for this interpretation.

If review and approval of this proposed change has not been completed prior to the next required Monticello reactor coolant system leakage or hydrostatic test, it would be our intention to rely on the previous NRC clarification and interpretation of this requirement which would allow such testing to proceed.

11. Clarification of Primary Containment Requirements

Reword Specification 3.7.A.1 on page 156 as follows, "When irradiated fuel is in the reactor vessel and either the reactor coolant temperature is greater than 212 F or work is being done which has the potential to drain the vessel, the following requirements shall be met except as permitted by Specification 3.5.G.4 ...."

Reason: Literal interpretation of existing Technical Specification 3.7.A.1 would not allow draining of the suppression chamber when irradiated fuel was not in the reactor vessel and work which would, or had the potential, to drain the reactor vessel was in progress. This is due to the omission, in this case, of the standard wording, "when irradiated fuel is in the reactor vessel." All other Monticello primary containment and ECCS Technical Specifications contain this provision.

12. Reactor Coolant System Venting and Requirements for Secondary Containment

In Specification 3.7.C.2.5 on page 169, delete the phrase, "...and the reactor coolant system is vented."

Reason: The requirements for the reactor to be vented as a condition for not requiring secondary containment conflicts with normal and reasonable activities during outages. For example, reactor vents must be closed to perform vessel leakage and hydrostatic testing. At other times it is prudent to close reactor vents for radiological protection purposes.

There is no basis for relating secondary containment requirements to reactor venting. Closing reactor vents during an outage when secondary containment was not established could be interpreted as a violation of the Technical Specifications even though the event would have no safety implications.

The NRC Standard Technical Specifications do not include this or similar limitations on reactor venting.

13. Extended Core and CRD Maintenance Technical Specification Conflicts

Delete Specification 3.10.E.2 and redesignate Specification 3.10.E.1 and 3.10.E. Reword the first portion of the Specification to read, "More than one control rod may be withdrawn from the reactor core during outages provided that, except for momentary switching to the startup mode for interlock testing, the reactor mode switch is locked in the refuel position. The refueling interlock ...." Change "withdrawn control rod" to "control rod" in two locations.

Reason: Existing Specification 3.10.E.2 is totally redundant to Specification 3.10.B and therefore unnecessary and possibly confusing.

There is a conflict between the existing Specification 3.10.E.1 and Specification 4.10.A. Also, a literal interpretation of the existing Specification would prohibit rod withdrawal for normal operation and testing.

Specification 4.10.A requires weekly checks of the refueling interlocks until core alterations are completed and they are no longer required. Core alterations, as defined in Section 1.0, occur throughout most of an outage. During this time is is normal to have control rods removed from the core for maintenance in accordance with Specification 3.10.E.1. The conflict is that Specification 3.10.E.1 requires the mode switch to be locked in "Refuel," but the weekly check of refueling interlocks requires switching momentarily to the "Startup" mode.

The lead-in statement of the existing Technical Specification does not limit its applicability. The unfortunate wording, taken literally, would require that the mode switch be locked in refuel for any rod withdrawal from the core. Although obviously not the intent, this would prohibit more than one rod from ever being withdrawn.

14. Accident Monitoring Instrumentation Operability Conditions

Revise Specification 3.14 on page 229a to require operability of accident monitoring instrumentation, "...whenever irradiated fuel is in the reactor vessel and reactor coolant water temperature is greater than 212 F...." Revise the notes to Table 3.14.1 on page 229c to require placing the plant in the cold shutdown condition within eight hours when required conditions of instrument operability are not satisfied.

Reason: The existing wording for Specification 3.14 requires operability of this instrumentation in the startup and run modes. During outages, the mode switch must be placed often in the startup position to perform tests or other normal operations. However, accident monitoring instrumentation may be inoperable during an outage due to normal activities or conditions. For example, SRV removal would render SRV position indication inoperable and vessel draining would render vessel level instrumentation inoperable.

There is no reason for accident instrumentation operability to be based on mode switch position. A more desirable wording for the Technical Specification is to make it consistent with other accident mitigation system operability requirements (i. e. above 212 F) and the NRC Standard Technical Specifications.

15. Tables 3.14.1 and 4.14.1, Suppression Pool Temperature Monitoring Instrumentation

Add the Suppression Pool Temperature Monitoring instrumentation to Table 3.14.1 on page 229b and Table 4.14.1 on page 229d.

Reason: This instrumentation was added as required by the Mark I Containment Long Term improvement program to accurately monitor suppression pool average temperature. The operability and surveillance requirements proposed for this instrumentation are consistent with other Technical Specifications for accident monitoring instrumentation.

The Suppression Pool Temperature Monitoring System (SPOTMOS) installed at Monticello was described in Volume 1, Section 5, of the Monticello Plant Unique Analysis Report submitted on December 15, 1982. This design was reviewed and approved by the NRC and a Safety Evaluation Report was issued on September 11, 1985.

In the event of inoperability of SPOTMOS, alternate methods of monitoring suppression pool temperature are available until operability of SPOTMOS can be restored. These methods include use of an alternate multipoint recorder and temperature sensors in the suppression pool (original temperature monitoring system).

16. Clarification of Table 4.14.1 Sensor Checks

Provide additional notes on Table 4.14.1 to clarify sensor check requirements for reactor water level, SRV valve position pressure switches, and SRV valve position thermocouples as follows:

(2) Once/month sensor check will consist of verifying that the pressure switches are not tripped.

(3) Once/month sensor check will consist of verifying fuel zone level indicates off scale high.

(4) Following every Safety/Relief Valve actuation it will be verified that recorder traces or computer logs indicate sensor responses.

Add a reference to Note 2 for SRV position pressure switches. Add a reference to Note 3 for reactor vessel fuel zone water level, and add a reference to Note 4 for SRV position pressure switches and thermocouples.

Reason: Literal interpretation of the requirements for sensor checks of the SRV position pressure switches would require operation of the SRV's once per month. It would also require establishing an abnormal reactor level to perform a sensor check of the fuel zone level instrument.

Table 14.1.1 notes have been revised to clarify the intent of these sensor checks, which is to require a verification of sensor operation following each SRV actuation and once each month to verify that the fuel zone indicator is off scale high and the SRV pressure switches are not tripped.

These sensor checks do not comply literally with the definition of "Sensor Check" in Section 1.0 of the Technical Specifications and it is important that notes to the Table clearly specify the intended requirements.

Determination of Significant Hazards Considerations

The proposed changes to Appendix A of the Operating License have been evaluated to determine whether they constitute a significant hazards consideration as required by 10 CFR Part 50, Section 50.91 using the standards provided in Section 50.92. This evaluation is provided below:

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

   The proposed amendment would 1) clarify the definition of Core Alteration, 2) correct a typographical error in the Section 2.3 Bases, 3) correct and clarify the Startup Mode operability requirements for high drywell pressure, high reactor pressure, and main steam line high radiation, 4) delete the obsolete provisions of the Technical Specifications which allow surveillance intervals to be extended, 5) correct conflicts with the SRM-Not-Full-In rod block interlock and CRD maintenance, 6) correct and clarify the surveillance requirements for containment isolation instrumentation, 7) provide an additional section to the Bases related to general surveillance requirements, 8) correct the action statements for ATWS instrumentation to correspond with the as-installed logic, 9) clarify CRD accumulator operability requirements, 10) correct the HPCI, RCIC, and APRS operability requirements to permit reactor coolant system leakage and hydrostatic testing, 11) clarify the requirements for containment integrity when no fuel is in the reactor, 12) correct and clarify the relationship between secondary containment requirements and reactor venting, 13) clarify the requirements for extended CRD maintenance, 14) correct and clarify the operability conditions for accident monitoring instrumentation, 15) add Technical Specification limiting conditions for operation and surveillance requirements for suppression pool temperature monitoring instrumentation, and 16) clarify the meaning of sensor checks for safety/relief valve position pressure switches and reactor fuel zone water level instrumentation.

   With the exception of item 2, which corrects a typographical error, and item 15, which adds Technical Specification requirements for a new instrumentation system, all of these changes have the intent of eliminating conflicts and interpretation problems in the Technical Specifications.

These items were identified during a detailed review of the Technical Specifications by senior SRO licensed members of the Monticello technical staff. This review was made to fulfill a commitment made to NRC Region III and NRC NRR management following the discovery during the last refueling and maintenance outage of a number of conflicts in the Technical Specifications.

With the exception of items 2 and 15, these changes improve the clarity and logic of the wording in the Technical Specifications. While some relief from impossible or unreasonable restrictions is granted in several instances (e.g. HPCI will no longer be required operable during hydrostatic tests - but because the vessel is filled solid with subcooled water during these tests it is an impossible condition to impose), the requested changes will not, in any significant way, change the way the plant is operated or maintained.

Item 2 is a purely administrative change. Item 15 adds new requirements for an instrumentation system installed to meet the requirements of the NRC approved Mark I Containment Long Term Program and NRC Regulatory Guide 1.97, Revision 2. It is a new instrumentation system which will enhance the information available to plant operators during normal and postulated accident conditions.

Since the requested changes will not, in any significant way, affect any aspect of plant operation or maintenance or relax, in any significant way, valid limitations placed on systems and equipment, they will not increase the probability of consequences of any previously evaluated accident.

2. <u>The proposed amendment will not create the possibility of a new or difference kind of accident from any accident previously evaluated.</u>

As discussed above, item 2 is an administrative change which corrects a typographical error. Item 15 adds additional requirements to the Technical Specifications for a new instrumentation system. The remainder of the requested changes make desirable clarifications and remove conflicts from the Technical Specifications. Since the requested changes will not, in any significant way, affect any aspect of plant operation or maintenance or relax, in any significant way, valid limitations placed on systems and equipment, they will not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

   As discussed above, the proposed changes involve the correction of a typographical error, adding limiting conditions for operation and surveillance requirements for a new monitoring instrument, clarifications, changes which remove conflicts between various sections of the Technical Specifications, and a number of changes which eliminate impossible or unreasonable limitations on plant systems and components. This latter group of changes may be considered relief from restrictions imposed by the Technical Specifications, but in every case the proposed change will not in any significant way, change any aspect of plant operation or maintenance or relax, in any significant way, valid limitations placed on systems and equipment. Therefore no proposed change significantly reduces any margin of safety as described in the Technical Specifications or Updated Safety Analysis Report.

The Commission has provided guidance concerning the application of the Standards for determining whether a significant hazards consideration exists by providing certain examples of amendments that are considered not likely to involve significant hazards considerations. These examples were published in the Federal Register on March 6, 1986.

Item 2 of this application is representative of a purely administrative change presented as NRC example (i). Items 4 and 15 of this application are similar to NRC example (ii) since they consist of additional limitations, restrictions, or controls not presently in the Technical Specifications. The remaining items are similar to NRC example (i) since they can be best described as corrections of errors, correction of nomenclature, and changes necessary to achieve consistency.

Exhibit B

License Amendment Request Dated May 5, 1986
Docket No. 50-263 License No. DPR-22

Exhibit B consists of revised pages for the Monticello Nuclear Generating Plant Technical Specifications showing the proposed changes:

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