AMENDMENT REQUEST DATED - DECEMBER 16, 1974

EXHIBIT B

This exhibit consists of the following pages revised to incorporate the proposed Technical Specification changes:

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INTRODUCTION

These Technical Specifications are prepared in accordance with the requirements of 10 CFR 50.36 and apply to the Monticello Nuclear Generating Plant, Unit No. 1. The bases for these Specifications are included for information and understandability purposes.

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the Specifications may be achieved.

A. Abnormal Occurrence

1. Prompt Notification with Written Followup

a. Failure of the reactor protection system, or other systems subject to limiting safety system settings, to initiate the required protective function by the time a monitored parameter reaches the value specified as the limiting safety system setting in the Appendix A Technical Specifications, or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the Appendix A Technical Specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary or primary containment.

d. Reactivity anomalies involving disagreement with predicted value of reactivity balance under steady state conditions greater than or equal to \$1.00; a calculated reactivity balance indicating shutdown margin less conservative than specified in the Appendix A Technical Specifications; short term reactivity increases that correspond to a reactor period of less than 5 seconds, or if subcritical, an unplanned reactivity insertion of more than 50¢; or any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

g. Conditions arising from natural or man-made events, that, as a direct result of the event, require plant shutdown, operation of safety systems, or other protective measures required by Appendix A Technical Specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the SAR or in the bases for the Appendix A Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the SAR or Appendix A Technical Specifications bases or discovery during plant life of conditions not specifically considered in the SAR or Appendix A Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

2. Thirty Day Written Reports

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Appendix A Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition of operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems, other than those specified in T.S. 1.0.A.1.c above, designed to contain radioactive material resulting from the fission process.

B. Alteration of the Reactor Core

The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. (Normal operating functions such as control rod movement using the normal drive mechanism, tip scans, SRM and IRM detector movements, etc., are not to be considered core alterations.)

C. Hot Standby

Hot Standby means operation with the reactor critical in the startup mode at a power level just sufficient to maintain reactor pressure and temperature.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

E. Reactivity Anomalies

At a specific steady state base condition of the reactor actual control rod inventory will be periodically compared to a normalized computed prediction of the inventory. If the difference exceeds one per cent, Δk, reactor power operation shall not be permitted until the cause has been evaluated and appropriate corrective action has been completed.

F. If Specifications 3.3.A through D above are not met, an orderly shutdown shall be initiated and have reactor in the cold shutdown condition within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

E. Reactivity Anomalies

During the startup test program and at each startup following refueling outages, the actual rod inventory shall be compared to a normalized computed prediction of the inventory. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the actual rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.

Bases 3.6 And 4.6 - Continued:

The initial NDT Temperature of the vessel shell material opposite the reactor core region is 0°F. The initial NDT temperature in the main dome flanges, and the shell and head material connecting to these flanges is 10°F, and elsewhere is 40°F. The design life of the reactor vessel is 40 years and the maximum fast neutron exposure at 40 years is calculated to be 5.4×10^{17} nvt. The NDT temperature limit curve in Figure 4.6.1 uses the "worst case" curve of the FSAR to establish the NDT temperature shift and is therefore conservative. The expected NDT temperature shift for this vessel at 5.4×10^{17} nvt is expected to be 0°F. Figure 4.6.1 also incorporates a 60°F factor of safety. This factor is based upon the requirements of the ASME code and the considerations which resulted in these requirements. Therefore, the specification provides for "worst case" data as well as 60°F of margin to provide assurance that operation in the non-ductile region will not occur.

The reactor vessel head flange and the vessel flange in combination with the double "O" ring type seal are designed to provide a leak-tight seal when bolted together. When the vessel head is placed on the reactor vessel, only that portion of the head flange near the inside of the vessel rests on the vessel flange. As the head bolts are replaced and tensioned, the vessel head is flexed slightly to bring together the entire contact surfaces adjacent to the "O" rings of the head and vessel flange. Both the head and the headflange have an NDT temperature of 10°F, and they are not subject to any appreciable neutron radiation exposure. Therefore, the minimum vessel head and head flange temperature for bolting the head flange and vessel flange is established as 10°F + 60°F or 70°F.

Numerous data are available relating integrated flux and the change in nil-ductility transition temperature (NDTT) in various steels. The most conservative data has been used in Specification 3.3. The integrated flux at the vessel wall is calculated from core physics data and will be measured using flux monitors installed inside the vessel. The measurements of the neutron flux at the vessel wall will be used to check and if necessary correct, the calculated data to determine an accurate NDTT.

In addition, vessel material samples will be located within the vessel to monitor the affect of neutron exposure on these materials. The samples include specimens of base metal, weld zone metal, heat affected zone metal, and standard specimens. These samples will receive neutron exposure more rapidly than the vessel wall material and therefore will lead the vessel in integrated neutron flux exposure. These samples will provide further assurance that the shift in NDTT used in the specification is conservative. An analysis and report will be submitted to the AEC on all such surveillance specimens removed from the reactor vessels in accordance with 10 CFR Part 50, Appendix H. These reports shall include the information specified in ASTM E-185-66, "Recommended Practices for Surveillance Tests on Structural Materials in Nuclear Reactor," and information obtained on the level of integrated fast neutron irradiation received by the specimens and actual vessel material.

Bases Continued 3.6 and 4.6:

D. Coolant Leakage

The former 15 gpm limit for leaks from unidentified sources was established assuming such leakage was coming from the primary system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. From the crack size a leakage rate can be determined. For a crack size which gives a leakage of 5 gpm, the probability of rapid propagation is less than 10^{-5}. Thus, an unidentified leak of 5 gpm when assumed to be from the primary system had less than one chance in 100,000 of propagating, which provides adequate margin. A leakage of 5 gpm is detectable and measurable. The 24 hour period allowed for determination of leakage is also based on the low probability of the crack propagating.

The capacity of the drywell sump pumps is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

E. Safety/Relief Valves

Testing of all safety/relief valves each refueling outage ensures that any valve deterioration is detected. A tolerance value of 1% for safety/relief valve setpoints is specified in Section III of the ASME Boiler and Pressure Vessel Code. Analyses have been performed with all valves assumed set 1% higher (1080 psig + 1%) than the nominal setpoint; the 1375 psig code limit is not exceeded in any case.

The safety/relief valves are used to limit reactor vessel overpressure and fuel thermal duty.

The required safety/relief valve steam flow capacity is determined by analyzing the transient accompanying the mainsteam flow stoppage resulting from a postulated MSIV Closure from a power of 1670 MW_t. The analysis assumes a multiple-failure wherein direct scram (valve position) is neglected. Scram is assumed to be from indirect means (high flux). In this event, the safety/relief valve capacity is assumed to be 71% of the full power steam generation rate.

Bases Continued 3.6 and 4.6:

Design confirmation and construction adequacy will be demonstrated during the plant startup and power ascension test program. As part of this program, cold and hot vibration tests on certain reactor vessel internals will be performed. The tests, described in a letter to Dr. P. A. Morris, dated March 5, 1970, are designed to obtain confirmatory data on the design features of Monticello as compared to Dresden Unit 2 design. Thus, the basis for the Monticello vibration test program is predicated on obtaining satisfactory data which confirms common design features from earlier BWR plants such as Dresden Unit 2. In the event that data from these earlier plants are not available before routine power operation of Monticello, the matter will be reviewed by the AEC.

The program outlined in Table 4.6.1 is limited to inspections of the primary coolant system. It is anticipated that the data collected during the first five years of operation will provide a suitable basis to evaluate the need for inspecting other portions of the facility (such as the main steam lines downstream of the main steamline isolation valves). These data along with the overall operating experiences will be reviewed to determine the inspection program to be implemented for the lifetime of the facility. The results of this study together with the proposed lifetime inspection program will be submitted to the AEC in accordance with Specification 6.7.B.

The special inspection of the main feed and steam lines is to provide added protection against pipe whip. The Group I welds are selected on the basis of an analysis that shows these welds are the highest stress welds and that due to their physical location, a break would result in the least interference and maximum energy upon impact with the drywell. These welds are the only ones which offer any significant risk and will be included in future inspections as determined by the study described above.

Group II welds are selected because without regard for the operating stress levels and interfering equipment, they have sufficient theoretical energy to penetrate and would propel the pipe toward the containment. They are therefore included in the first inspection. Upon consideration of impact angle, interfering equipment and distance pipe travels, no substantial risk is involved and no extra inspection is needed.

In addition, extensive visual inspection for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsored by the AEC. These studies show that it requires thousands of stress cycles at

C. Secondary Containment

1. Secondary containment integrity, shall be maintained during all modes of plant operation except when all of the following conditions are met.

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

   b. The reactor water temperature is below 212^{o} and the reactor coolant system is vented.

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

C. Secondary Containment

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

   a. Secondary containment capability to maintain at least a 1/4 inch of water vacuum under calm wind ($2 < u < 5$ mph) conditions with a filter train flow rate of $\leq 4,000$ scfm, shall be demonstrated at each refueling outage prior to refueling.

Bases Continued:

The acceptable values for local leak rate tests have been specified in terms of standard cubic feet per hour (scf/hr) for purposes of clarity. Following is the list of equivalent values given in terms of an allowable percentage of the allowable operational leak rate (L_{to}).

17.2 scf/hr = 5% L_{to}
@ 41 psig

34.4 scf/hr = 10% L_{to}
@ 41 psig

103.2 scf/hr = 30% L_{to}
@ 41 psig

where L_{to} = .75 L_t (the maximum allowable leak rate)
and L_t = 1.2 weight percent of the contained air at the test pressure of 41 psig.

Results of loss of coolant accident analyses indicate that fission products would not be released directly to the environs because of leakage through the main line isolation valves due to holdup in the steam system complex. Although this effect shows that an adequate margin exists with regard to release of fission products, the results of leak tests on the main steam line isolation valves will be closely followed in order to determine the adequacy of these valves to perform their intended function.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

Bases Continued:

4.7 The containment is penetrated by a large number of small diameter instrument lines. A program for the periodic testing (see Specification 4.7.D) and examination of the valves in these lines has been developed and a report covering this program was submitted to the AEC on July 27, 1973.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

3.0 LIMITING CONDITIONS OF OPERATION	4.0 SURVEILLANCE REQUIREMENTS
a. Investigate to identify the causes for such release rates. b. Define and initiate a program to reduce such release rates to the as low as practical levels. c. Provide a report describing these actions within 30 days. 10. At least one of the two stack monitors, including the charcoal cartridge and particulate filter, shall be operable at all times that the stack is releasing effluents to the environs. 11. If both stack monitors are made or found inoperable, the reactor shall be placed in the hot standby condition within 24 hours. 12. Except as specified in 3.8.A.13, the off-gas stack and reactor building vent monitors shall have automatic isolation set points consistent with Specification 3.8.A.1 and alarm set points consistent with Specification 3.8.A.2. 13. If operation is necessary with the Off-gas Holdup System recombiners bypassed, the off-gas stack monitors shall serve only an alarm function.	5. A determination shall be made of the total I-131 released weekly. An analysis shall be performed on a sample at least monthly for I-133 and I-135. 6. A determination shall be made of the total radioactive particulates with half-lives greater than 8 days released weekly. The particulate filters shall be removed and analyzed for gross beta particulate radioactivity with half-lives greater than 8 days. Monthly, a composite of those filters used during the month shall be prepared and analyzed for the principal gamma emitting radionuclides. 7. Analysis for Sr-89 and Sr-90 shall be made quarterly. Gross alpha radioactivity shall be determined quarterly.

c. Mechanism for scheduling meetings

d. Meeting agenda

e. Use of subcommittees

f. Review and approval, by members, of OC actions

g. Distribution of minutes

6.3 Abnormal Occurrence Action

In the event of an abnormal occurrence as defined in the Appendix A Technical Specifications, the Commission shall be notified and/or a report submitted pursuant to the requirements of T.S.6.7.A.

Each abnormal occurrence shall be reported to the Operations Committee, either by copy of the report previously submitted to the Commission or by a separate investigation report. The Operations Committee shall review the report and recommend further action if necessary. Copies of the report and Operations Committee minutes documenting their review shall be submitted to the Safety Audit Committee and the General Superintendent of Nuclear Power Plant Operation.

6.4 Safety Limit Violation

If a safety limit is exceeded, the reactor shall be shut down and the Commission shall be notified immediately. It shall also be promptly reported to the General Superintendent of Nuclear Power Plant Operation and the Chairman of the Safety Audit Committee, or their designated alternates. A safety limit violation report shall be prepared. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) the basis for corrective action taken to preclude recurrence. The report shall be reviewed by the Operations Committee. The safety limit violation report shall be submitted to the Commission, the General Superintendent of Nuclear Power Plant Operation, and the Safety Audit Committee with two weeks of the event.

Operation shall not be resumed until authorized by the Commission.

10. Reactor coolant system in-service inspections

11. Minutes of meetings of the Safety Audit Committee

6.7 Reporting Requirements

A. Routine Reports and Event Reports

The operating information to be reported to the USAEC in addition to the reports required by Title 10, Chapter 1, Code of Federal Regulations, shall be in accordance with the Regulatory position of Regulatory Guide 1.16, Revision 2, "Reporting of Operating Information - Appendix A Technical Specifications," with the following exceptions:

1. In case of conflict between definitions or requirements in Regulatory Guide 1.16 and these Appendix A Technical Specifications, the latter will take precedent. In those cases where the Regulatory Guide definitions of abnormal occurrences contain interpretive notes on what is reportable, those interpretations will apply to the definitions listed in T. S. 1.0.A.

2. Regulatory Guide 1.16, Revision 2 (R.G. 1.16-2), para. C.1.b(2): replace "five percent" with "20%."

3. R.G. 1.16-2, para. C.1.b(3): Exposure reporting will be for individuals receiving exposures greater than 100 mrem in the reporting period and assignment of exposures to various duty functions will be based on best estimates from normally used personnel monitoring devices.

4. R.G. 1.16-2, para. C.1.b(4): The reporting recommended under sub-paragraphs (a)-(d) will not be required.

5. R.G. 1.16-2, para. C.2.a: The prompt notification shall be as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram or facsimile transmission to the Director of the appropriate Regulatory Operations Regional Office, or his designate no later than the first working day following the event, with a written follow-up report within two weeks.

6. R.G. 1.16-2, para. C.2.a(1): change to read "Failure of the reactor protection system, or other systems subject to limiting safety system settings, to initiate the required protective function by the time a monitored parameter reaches the value specified as the limiting safety system setting in the Appendix A Technical Specifications, or failure to complete the required protective function."

7. R.G. 1.16-2, para. C.2.a(3): change to read "Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary or primary containment."

8. R.G. 1.16-2, para. C.2.a(4): Delete this definition and reporting requirement.

9. R.G. 1.16-2, para. C.2.b: Change the reporting time requirement to 30 days following the event.

10. R.G. 1.16-2, para. C.2.b(2): change to read "Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition of operation.

11. R.G. 1.16-2, para. C.2.b(3): Delete this definition and reporting requirement and re-number C.2.b(4) as C.2.b(3). Add new paragraph C.2.b(4) which reads "Abnormal degradation of systems other than those specified in C.2.a(3) above, designed to contain radioactive material resulting from the fission process."

12. R.G. 1.16-2, para. C.2.c: The events listed under this paragraph are not abnormal occurrences and are not required to be reported.

B. Special Reports

The following special reports shall be submitted in writing to the Director of the Regulatory Operations Regional Office within the time period specified for each report:

Area	Reference	Submittal Date
1. Primary Containment Leak Rate Tests	4.7A	90 days after each integrated leak rate test
2. In-Ser ice Inspection Evaluation & Development	4.6F & 4.6F Bases	October 1, 1976
3. Failed Fuel Detection	3.2 Bases	July 1, 1976

C. Environmental Reports

The following reports relating to environmental activities shall be submitted to the Director of the Regulatory Operations Regional Office and are included in this Appendix A Technical Specification section until an Appendix B Technical Specification has been issued for the Monticello Nuclear Generating Plant:

1. A Semiannual Radioactive Effluents Report shall be submitted within 60 days after January 1 and July 1 of each year. The report will meet the intent of Regulatory Guide 1.21, Revision 1, and will include a summary of the quantities of radioactive liquid and gaseous effluents and solid wastes released from the plant during the previous six months of operation.

2. An Annual Radiological Environmental Monitoring Report shall be submitted by April 1 of the subsequent year. The report will meet the intent of Regulatory Guide 4.1 (1/18/73) and will include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities. In the event that some results are not available within the 90 day period, the report will be submitted noting and explaining the reasons for the missing results which will be submitted as soon as possible in a supplementary report.

3. An Annual Environmental Monitoring and Ecological Studies Program Report shall be submitted by August 1 of the subsequent year. The report will include summaries, interpretations, and statistical evaluation of the results of the non-radiological environmental surveillance activities.

AEC DISTRIBUTION FOR PART 50 DOCKET MATERIAL
(TEMPORARY FORM)

CONTROL NO 12737

FILE:

FROM: Northern States Power, Minneapolis, Minn. 55401, Mr. L.O. Mayer	DATE OF DOC 12-16-74	DATE REC'D 12-18-74	LTR x	TWX	RPT	OTHER
TO: E.G. Case	ORIG 3 signed	CC	OTHER	SENT AEC PDR XXX; SENT LOCAL PDR XXX		
CLASS / UNCLASS XXX / PROP INFO	INPUT XXX	NO CYS REC'D 40	DOCKET NO: 50-263			

DESCRIPTION:

Ltr re our 10-21-74 ltr....trans the following ....

ENCLOSURES:

Amdt to the OL, consist of rev & addl pgs, tables & figs to the tech specs....requesting a change to their tech specs...concerning.. conformance with Reg Guide 1.16

(40 cys encl rec'd)

PLANT NAME: Monticello

FOR ACTION/INFORMATION 12-18-74 JB

BUTLER (L) W/ Copies	SCHWENCER (L) W/ Copies	ZIEMANN (L) W/ 6 Copies	REGAN (E) W/ Copies
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