JAN [illegible] 1976

Docket No. 50-263

Northern States Power Company
ATTN: Mr. L. O. Mayer, Director
Nuclear Support Services
414 Nicollet Mall
Minneapolis, Minnesota 55401

Gentlemen:

In response to your request dated December 16, 1974, the Commission has issued the enclosed Amendment No. 15 to Provisional Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes in the Monticello Nuclear Generating Plant Technical Specifications that revise the reporting requirements. Changes to your proposal were necessary to meet our requirements. These have been discussed with your staff. The Technical Specifications are based on Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications," Revision 4.

We request that you use the formats presented in the Appendices to Regulatory Guide 1.16, Revision 4, for reporting operating information and that you report events of the type described under the section "Events of Potential Public Interest". Instructions for using these reporting formats are contained in Regulatory Guide 1.16 (a copy is enclosed for your use), and AEC report OOE-SS-001 titled "Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File" of which you were previously provided a copy. This report is modified by updated instructions dated December 8, 1975, which are enclosed. Copy requirements are summarized in Regulatory Guide 10.1, "Compilation of Reporting Requirements for Persons Subject to NRC Regulations", a copy of which is also enclosed. This guide will assist you in identifying reports that are required by the Commission's regulations set forth in Title 10 Code of Federal Regulations but are not contained in your technical specifications. Reports that are required by the regulations have not been repeated in your technical specifications.

Copies of the related Safety Evaluation and the Federal Register Notice also are enclosed.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Form AEC-318 (Rev. 9-53) AECM 0240

★ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

JAN 22 1976

Please note that we have discontinued the use of separate identifying numbers for changes to technical specifications. Sequential amendment numbers will be continued as in the past.

Sincerely,

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 15 to License DPR-22
2. Regulatory Guide 1.16
3. Updated Instructions
4. Regulatory Guide 10.1
5. Safety Evaluation
6. Federal Register Notice

cc w/enclosures:
See next page

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JAN 22 1975

cc w/enclosures:
Arthur Renquist, Esquire
Vice President - Law
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Gerald Charnoff
Shaw, Pittman, Potts &
Trowbridge
910 - 17th Street, N. W.
Washington, D. C. 20006

Howard J. Vogel, Esquire
Legal Counsel
2750 Dean Parkway
Minneapolis, Minnesota 55416

Steve Gadler
2120 Carter Avenue
St. Paul, Minnesota 55108

Mr. Daniel L. Ficker
Assistant City Attorney
638 City Hall
St. Paul, Minnesota 55102

Mr. Kenneth Dzugan
Environmental Planning
Consultant
St. Paul City Planning
421 Wabasha Street
St. Paul, Minnesota 55102

Sandra S. Gardebring
Special Assistant Attorney General
Counsel for Minnesota Pollution
Control Agency
1935 W. County Road B2
Roseville, Minnesota 55113

Anthony Z. Roisman, Esquire
Berlin, Roisman and Kessler
1712 N Street, N. W.
Washington, D. C. 20036

The Environmental Conservation
Library
Minneapolis Public Library
300 Nicollet Mall
Minneapolis, Minnesota 55401

Mr. D. S. Douglas, Auditor
Wright County Board of Commissioners
Buffalo, Minnesota 55313

cc w/enclosures and NSP filing
dtd. 12/16/74:
Warren H. Lawson, M. D.
Secretary and Executive Officer
State Department of Health
University Campus
Minneapolis, Minnesota 55440

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING STATION

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 15
License No. DPR-22

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

   A. The application for amendment by the Northern States Power Company (the licensee) dated December 16, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-22 is hereby amended to read as follows:

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SURNAME➤						
DATE➤						

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised.

3. This license amendment is effective 30 days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: JAN 22 1976

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 15

PROVISIONAL OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the existing pages of the Technical Specifications listed below with the attached revised pages bearing the same numbers, except as otherwise noted. Changed areas on these pages are shown by marginal lines:

Pages iii, v, vi and vii   Table of Contents
Pages viii and ix - Addition (as result of expansion of Table of Contents)
1
1A - Deleted
81
134
136
150
164
170A
189B
195
196
197
200
201
210
211
212
213
214
215
216
216A - Deleted
217
218
219 - Deleted
220 - Deleted

LIST OF FIGURES

LIST OF FIGURES

LIST OF TABLES

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. Deleted

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
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.	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.

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 NRC.

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 NRC in accordance with Specification 6.7.C.2.

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 NRC. These studies show that it requires thousands of stress cycles at

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

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° 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.

4.7 BASES

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.

3.11 REACTOR FUEL ASSEMBLIES

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.11-1. If at any time it is determined that the limiting value for APLHGR is being exceeded, action shall be taken immediately to restore operation to within the prescribed limits.

4.11 REACTOR FUEL ASSEMBLIES

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

1. The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.
2. Whenever the plant technical staff determines that more frequent surveillance of APLHGR is necessary, it shall specify an augmented surveillance program commensurate with reactor conditions.

6.2 Review and Audit

Organizational units for the review and audit of facility operations shall be constituted and have the responsibilities and authorities outlined below:

A. Safety Audit Committee (SAC)

The SAC must: verify that operation of the plant is consistent with company policy and rules, approved operating procedures and operating license provisions; review important proposed plant changes, tests and procedures; verify that unusual events are promptly investigated and corrected in a manner which reduces the probability of recurrence of such events; and detect trends which may not be apparent to a day-to-day observer.

Audits of selected aspects of plant operation shall be performed with a frequency commensurate with their safety significance and in such a manner as to assure that an audit of all nuclear safety related activities is completed within a period of two years. Periodic review of the audit programs should be performed by the SAC at least twice a year to assure that such audits are being accomplished in accordance with requirements of Technical Specifications. The audits shall be performed in accordance with appropriate written instructions or procedures and shall include verification of compliance with internal rules, procedures (for example: normal, off-normal, emergency, operating, maintenance, surveillance, test and radiation control procedures and the emergency and security plans), regulations involving nuclear safety and operating license provisions; training, qualification and performance of operating staff; and corrective actions following reportable occurrences or unusual events. A representative portion of procedures and records of the activities performed during the audit period shall be audited and, in addition, observations of performance of operating and maintenance activities shall be included. Written reports of such audits shall be reviewed at a scheduled meeting of the SAC and by appropriate members of management including those having responsibility in the area audited. Follow-up action, including reaudit of deficient areas, shall be taken when indicated.

1. Membership

   a. The SAC shall consist of at least five (5) persons.

b. The SAC Chairman shall be a NSP management representative appointed by the Group Vice President-Power Supply. Other SAC members shall be appointed by the Group Vice President-Power Supply or such other person as he may designate. The Chairman shall appoint a Vice Chairman from the SAC membership to act in his absence.

c. No more than two members of the SAC shall be from groups holding line responsibility for operation of the plant.

d. The SAC members should collectively have the capability required to review problems in the following areas: nuclear power plant operations, nuclear engineering, chemistry and radiochemistry, metallurgy, instrumentation and control, radiological safety, mechanical and electrical engineering, and other appropriate fields associated with the unique characteristics of the nuclear power plant involved. A minimum of four voting members shall have a minimum of a Bachelors Degree in Engineering or a scientific discipline and possess a minimum of three years of professional level experience in nuclear services, nuclear plant operation, or nuclear engineering. When the nature of a particular problem dictates, special consultants will be utilized, as necessary, to provide expert advice to the SAC.

2. Meeting Frequency

The SAC shall meet on call by the Chairman but not less frequently than twice a year.

3. Quorum

A quorum shall include a majority of the permanent members, including the Chairman or Vice Chairman. No more than a minority of the quorum shall be from groups holding line responsibility for the operation of the plant.

4. Responsibilities - The following subjects should be reported to and reviewed by the SAC:

a. Proposed tests and experiments, and their results, when such tests or experiments may constitute an unreviewed safety question as defined in Section 50.59(a)(c), Part 50, Title 10, Code of Federal Regulations.

b. Evaluations of proposed changes to procedures, equipment and systems completed under provisions of Section 50.59(a)(1), Part 50, Title 10, Code of Federal Regulations to verify that such proposed changes do not constitute unreviewed safety questions.

c. Proposed changes in procedures, equipment or systems which may involve a unreviewed safety question as defined in Section 50.59(a)(2), Part 50, Title 10, Code of Federal Regulations or changes referred to it by the operating organization.

d. Proposed changes in Technical Specifications or operating license.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, operating license requirements, or of internal procedures or instructions having safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment.

g. All events which are required by regulations or Technical Specifications to be reported to NRC in writing within 24 hours.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems or components.

i. Operations Committee proceedings and minutes to determine if matters considered by that Committee involve unreviewed or unresolved safety questions.

j. Training, qualification and performance of operating staff.

k. Disagreement between the recommendations of the Operations Committee and the Plant Manager.

l. Security and emergency plans and their implementing procedures.

m. Environmental Monitoring Program and its results.

n. Quality Assurance program and evaluate its adequacy.

Review of events covered under 4.e-4.h above shall include reporting to appropriate members of management on the results of investigations and recommendations to prevent or reduce the probability of recurrence.

5. Authority

The SAC shall be advisory to the Group Vice President-Power Supply.

f. All events which are required by regulations or Technical Specifications to be reported to NRC in writing within 24 hours.

g. Drills on emergency procedures (including plant evacuation) and adequacy of communication with off-site support groups.

h. All procedures required by these Technical Specifications, including implementing procedures of the Emergency Plan and the Security Plan shall be reviewed with a frequency commensurate with their safety significance but at an interval of not more than two years.

i. Perform special reviews and investigations, as requested by the Safety Audit Committee.

5. Authority

The OC shall be advisory to the Plant Manager. In the event of disagreement between the recommendations of the OC and the Plant Manager, the course determined by the Plant Manager to be the more conservative will be followed. A written summary of the disagreement will be sent to the General Superintendent of Nuclear Power Plant Operation and the Chairman of the SAC for review.

6. Records

Minutes shall be recorded for all meetings of the OC and shall identify all documentary material reviewed. The minutes shall be distributed to each member of the OC, the Chairman and each member of the Safety Audit Committee, the General Superintendent of Nuclear Power Plant Operation and others designated by OC Chairman or Vice Chairman.

7. Procedures

A written charter for the OC shall be prepared that contains:

a. Responsibility and authority of the group

b. Content and method of submission of presentations to the Operations Committee

c. Mechanism for scheduling meetings

d. Meeting agenda

e. Use of subcommittee

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

g. Distribution of minutes

6.3 Deleted

6.4 Action to be Taken if a Safety Limit is Exceeded

If a Safety Limit is exceeded, the reactor shall be shut down immediately. An immediate report shall be made to the General Superintendent of Nuclear Power Plant Operation, or his designated alternate in his absence, and reported as specified in Section 6.7. A complete analysis of the circumstances leading up to and resulting from the situation, together with recommendations by the Operations Committee, shall also be prepared. This report shall be submitted to the General Superintendent of Nuclear Power Plant Operation and the Chairman of the Safety Audit Committee.

Reactor operation shall not be resumed until authorized by the U. S. Nuclear Regulatory Commission.

5. Principal maintenance activities, including inspection, repairs and substitution or replacement of principal items of equipment pertaining to nuclear safety.

6. Records of changes to plant procedures and records of special tests and experiments.

7. Records of wind speed and direction.

8. Records of individual plant staff members showing qualifications, training and retraining.

9. Reportable Occurrences.

B. Records Retained for Plant Life

Records and logs relative to the following items shall be retained for the life of the plant:

1. Liquid and gaseous radioactive releases to the environs
2. Radiation exposures for all plant, visitor and contractor personnel
3. Off-site environmental monitoring surveys
4. Fuel accountability including new and spent fuel inventories and transfers, and fuel assembly histories
5. Radioactive shipments
6. Plant radiation and contamination surveys
7. Changes made to the plant as it is described in the Final Safety Analysis Report, reflected in updated, corrected and as-built drawings
8. Cycling beyond normal limits for those components that have been designed to operate safely for a limited number of cycles beyond such limits
9. Reactor coolant system in-service inspections
10. Minutes of meetings of the Safety Audit Committee

6.7 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.

A. Routine Reports

1. Startup Report

A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

2. Annual Operating Report 1/

Routine operating reports covering the operation of the unit during the previous calendar year should be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

The annual operating reports made by licensees shall provide a comprehensive summary of the operating experience gained during the year, even though some repetition of previously reported information may be involved. References in the annual operating report to previously submitted reports shall be clear.

Each annual operating report shall include:

a. A narrative summary of operating experience during the report period relating to safe operation of the facility, including safety-related maintenance not covered in item b.(5) below.

b. For each outage or forced reduction in power[2/] of over twenty percent of design power level where the reduction extends for greater than four hours:

(1) the proximate cause and the system and major component involved (if the outage or forced reduction in power involved equipment malfunction);

(2) a brief discussion of (or reference to reports of) any reportable occurrences pertaining to the outage or power reduction;

(3) corrective action taken to reduce the probability of recurrence, if appropriate;

(4) operating time lost as a result of the outage or power reduction (for scheduled or forced outages,[3/] use the generator off-line hours; for forced reductions in power, use the approximate duration of operation at reduced power);

(5) a description of major safety-related corrective maintenance performed during the outage or power reduction, including the system and component involved and identification of the critical path activity dictating the length of the outage or power reduction; and

---

2/ The term "forced reduction in power" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the load on the unit be reduced for corrective action immediately or up to and including the very next weekend. Note that routine preventive maintenance, surveillance and calibration activities requiring power reductions are not covered by this section.

3/ The term "forced outage" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the unit be removed from service for corrective action immediately or up to and including the very next weekend.

(6) a report of any single release of radioactivity or radiation exposure specifically associated with the outage which accounts for more than 10% of the allowable annual values.

c. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[4/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

d. Indications of failed fuel resulting from irradiated fuel examinations, including eddy current tests, ultrasonic tests, or visual examinations completed during the report period.

3. Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the appropriate Regional Office, to arrive no later than the tenth of each month following the calendar month covered by the report.

B. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

4/ This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

1. <u>Prompt Notification With Written Followup</u>

The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the appropriate Regional Office, or his designate no later than the first working day following the event, with a written followup report within two weeks. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

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 setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

Note: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under items 6.7.B.1.e., 6.7.B.1.f., or 6.7.B.2.a. below.

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

Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the technical specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under item 6.7.B.2.b. below.

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

Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation, greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if sub-critical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$ or occurrence of 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 system(s) used 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.

Note: For items 6.7.B.1.e. and 6.7.B.1.f. reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under items 6.7.B.2.b. and 6.7.B.2.c. below.

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 technical specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the 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 safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

Note: This item is intended to provide for reporting of potentially generic problems.

2. Thirty Day Written Reports

The reportable occurrences discussed below shall be the subject of written reports to the Director of the appropriate Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the 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 for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items 6.7.B.2.a. and 6.7.B.2.b. need not be reported except where test results themselves reveal a degraded mode as described above.

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 item 6.7.B.1.c. above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

C. Unique Reporting Requirements

1. Radioactive Effluent Releases

A statement of the quantities of radioactive effluent released from the plant, with data summarized on a monthly basis should be submitted in writing:

a. Gaseous Effluents

(1) Gross Radioactivity Releases

(a) Total gross radioactivity (in curies), primarily noble and activation gases.

(b) Maximum gross radioactivity release rate during any one-hour period.

(c) Total gross radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

(d) Percent of technical specification limit.

(2) Iodine Releases

(a) Total iodine radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

(b) Percent of technical specification limit for I-131 released.

(3) Particulate Releases

(a) Total gross radioactivity (β, γ) released (in curies) excluding background radioactivity.

(b) Gross alpha radioactivity released (in curies) excluding background radioactivity.

(c) Total gross radioactivity (in curies) of nuclides with half-lives greater than eight days.

(d) Percent of technical specification limit for particulate radioactivity with half-lives greater than eight days.

b. Liquid Effluents

(1) Total gross radioactivity (β, γ) released (in curies) excluding tritium and average concentration released to the unrestricted area.

(2) The maximum concentration of gross radioactivity (β, γ) released to the unrestricted area (averaged over the period of release).

(3) Total tritium and total alpha radioactivity (in curies) released and average concentration released to the unrestricted area.

(4) Total dissolved gas radioactivity (in curies) and average concentration released to the unrestricted area.

(5) Total volume (in liters) of liquid waste released.

(6) Total volume (in liters) of dilution water used prior to release from the unrestricted area.

(7) Total gross radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

(8) Percent of technical specification limit for total radioactivity.

c. Solid Waste

(1) The total amount of solid waste packaged (in cubic feet).

(2) The total estimated radioactivity (in curies) involved.

(3) Disposition including date and destination if shipped offsite.

d. Environmental Monitoring

(1) For each medium sampled, e.g., air, river bottom, surface water, soil, fish, include:

(a) Number of sampling locations,

(b) Total number of samples,

(c) Number of locations at which levels are found to be significantly above local backgrounds,

(d) Highest, lowest, and the annual average concentrations or levels of radiation for the sampling point with the highest average and description of the location of that point with respect to the site.

(2) If levels of radioactive materials in environmental media as determined by an environmental monitoring program indicate the likelihood of public intakes in excess of 1% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population groups, and assumptions upon which estimates are based shall be provided.

(3) If statistically significant variation of offsite environmental concentrations with time are observed, correlation of these results with effluent release shall be provided.

2. Other Reports

The following special reports shall be submitted in writing to the Office of Nuclear Reactor Regulation, NRC, Washington, D. C. 20555:

	Area	Specification Reference	Submittal
1.	Secondary Containment Leak Rate Tests (1)	4.7C	90 days after completion of the first refueling outage
2.	In-service Inspection Evaluation & Development	4.6F & 4.6F Bases	90 days after the "five-year" inspection has been completed (2)
3.	Failed Fuel Detection	3.2I Bases	5 years (2)
4.	Primary Coolant Leakage	4.6D Bases	Annual Report by March 31 of following year
5.	Instrument Line Flow	4.7D Bases	90 days after completion of first refueling outage

NOTE: (1) This summary technical report should include data on the wind speed, wind direction, outside and inside temperatures during the test, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of these data which demonstrate compliance with the specified leak rate limits. Surveillance testing at refueling outages after the first operating cycle should be reported in the semiannual operating reports.

(2) The summary technical report shall be submitted within the period of time listed based on the initial commercial service date as the starting point.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 15 TO PROVISIONAL OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

Introduction

By letter dated December 16, 1974, Northern States Power Company proposed changes to the Technical Specifications appended to Provisional Operating License No. DPR-22, for the Monticello Nuclear Generating Plant. The proposed changes involve changes to the reporting requirements.

Discussion

The proposed changes would be administrative in nature and would affect the conduct of operation. The proposed changes are intended to provide uniform license requirements. Areas covered by the proposed uniform specifications include reporting requirements and abnormal occurrence definition change.

In Section 208 of the Energy Reorganization Act of 1974 "abnormal occurrence" is defined as an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety. The term "abnormal occurrence" is reserved for usage by NRC. Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4, enumerates required reports consistent with Section 208. The proposed change to required reports identifies the reports required of all licensees not already identified by the regulations and those unique to this facility. The proposal would formalize present reporting and would delete any reports no longer needed for assessment of safety related activities.

Evaluation

The new guidance for reporting operating information does not identify any event as an "abnormal occurrence". The proposed reporting requirements also delete reporting of information no longer required and

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duplication of reported information. The standardization of required reports and desired format for the information will permit more rapid recognition of potential problems.

During our review of the proposed changes, we found that certain modifications to the proposal were necessary to have conformance with the desired regulatory position. These changes were discussed with the licensee's staff and have been incorporated into the proposal.

We have concluded that the proposal as modified improves the licensee's program for the reporting of the operating information needed by the Commission to assess safety related activities and is acceptable. The modified reporting program is consistent with the guidance provided by Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date:

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-263

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 15 to Provisional Operating License No. DPR-22, issued to Northern States Power Company (the licensee), which revised Technical Specifications for operation of the Monticello Nuclear Generating Plant (the facility) located in Wright County, Minnesota. The amendment is effective 30 days after its date of issuance.

The amendment revises the reporting requirements of the Technical Specifications for the facility.

The application for the amendment dated December 16, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made apprppriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

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For further details with respect to this action, see (1) the application for amendment dated December 16, 1974, (2) Amendment No. 15 to License No. DPR-22, and (3) the Commission's concurrently issued related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Environmental Conservation Library, Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this JAN 22 1976

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

DLZ

OFFICE ➤	OR:ORB #2	OR:ORB #2	OR:ORB #2			
SURNAME ➤	RMDiggs	BCBuckley:ah	DLZiemann			
DATE ➤	1/20/76	1/21/76	1/22/76			

ROUTING AND TRANSMITTAL SLIP

TO (Name, office symbol or location)	INITIALS / DATE	ACTION
1 OELD - f/concurrences	INITIALS DATE	CIRCULATE COORDINATION
2 DLZiemann - f/signatures	INITIALS DATE	FILE INFORMATION
3 Reba - for final checks	INITIALS DATE	NOTE AND RETURN PER CONVERSATION
4	INITIALS DATE	SEE ME SIGNATURE

REMARKS

Attached for your concurrence are five packages (Dresden Station, Quad Cities Station, Cooper, Pilgrim and Calvert Cliffs) of nine from ORB 2 which incorporate standard reporting requirement sections into the Appendix A Technical Specifications. One package, Pilgrim also revises the entire administrative controls section.

It is requested that, in the interest of review consistency, these packages (and the 4 future reporting requirements packages) be assigned to one OELD reviewer.

Questions may be directed to the PM for the particular case or to Mike Fletcher, coordinator for reporting (Exts. 7403, 7450)

11/3/75 No need for OELD concurrence this time this subject

Do NOT use this form as a RECORD of approvals, concurrences, disapprovals, clearances, and similar actions

FROM (Name, office symbol or location)	DATE / PHONE
DLZiemann DLZ	DATE 11-3-75 PHONE 7380

OPTIONAL FORM 41
AUGUST 1967
GSA FPMR (41CFR) 100-11.206

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