UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	
Northern States Power Company	)	Docket No. 50-263
	)	
(Monticello Nuclear Generating Plant, Unit 1)	)	
	)	

ORDER PROVIDING FOR PREHEARING CONFERENCE, REOPENED HEARING, AND RELATED ANNOUNCEMENT

In behalf of the board, the chairman ORDERS:

A. That a prehearing conference shall be held in the United States Federal Courthouse, 316 North Robert Street, St. Paul, Minnesota, at Courtroom 4 (7th floor) on Tuesday, August 4, 1970, at 10:00 a.m.

B. That the hearing shall reopen at the same place beginning Wednesday, August 5, 1970, at 9:00 a.m.

C. That this order reopening the hearing and providing for a prehearing conference shall be published promptly in the Federal Register and shall be the subject of a public announcement by the Commission's Division of Public Information.

ATOMIC SAFETY AND LICENSING BOARD

By Valentine B. Deale
Valentine B. Deale, Chairman

July 17, 1970

H Smith

PPR

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

7-17-70

In the Matter of)
)
NORTHERN STATES POWER COMPANY) Docket No. 50-263
(Monticello Nuclear Generating)
Plant Unit 1))

CERTIFICATE OF SERVICE

I hereby certify that copies of (1) ORDER AND MEMORANDUM PERTAINING TO DISCOVERY AND DEFINITION OF CONTENTIONS AND SCHEDULING OF PREHEARING CONFERENCE AND REOPENING OF HEARING and (2) ORDER PROVIDING FOR PREHEARING CONFERENCE, REOPENED HEARING, AND RELATED ANNOUNCEMENT, both dated July 17, 1970 in the captioned matter have been served on the following by deposit in the United States mail, first class or air mail, this 17th day of July 1970:

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cc: Mr. Beale
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PDR

UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

In the Matter of

NORTHERN STATES POWER COMPANY

Monticello Nuclear Generating Plant, Unit 1

Docket No. 50-263

APPLICANT'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW IN THE FORM OF A PROPOSED INITIAL DECISION ORDERING THE ISSUANCE OF A PROVISIONAL OPERATING LICENSE

PRELIMINARY STATEMENT

1. On August 1, 1966, Northern States Power Company (Applicant) filed with the Atomic Energy Commission (AEC or Commission) an application for a license to construct and operate the Monticello Nuclear Generating Plant having a boiling water nuclear reactor designed to operate at power levels of up to 1670 megawatts thermal. Following a review of the August 1, 1966, application, including eight amendments thereto, by the Commission's regulatory staff (staff) and the Advisory Committee on Reactor Safeguards (ACRS), a public hearing was held before an Atomic Safety and Licensing Board to consider whether a provisional construction permit should be issued by the Commission. There were no intervenors and the hearing was an uncontested proceeding. Pursuant to an order by that Board in its Initial Decision dated June 19, 1967, the Commission

issued a provisional construction permit authorizing the construction of the Monticello Nuclear Generating Plant, Unit 1, on the Mississippi River in Wright County, Minnesota. Applicant proceeded to construct the plant.

2. On November 7, 1968, Applicant submitted Amendment No. 9 to the application which superseded in their entirety the application for a construction permit and the previous eight amendments. Amendment No. 9 requested a license to operate the plant at its rated power level of 1670 megawatts thermal and included the Applicant's Final Safety Analysis Report (FSAR). This was thereafter supplemented by Amendments 10 through 28 to the application. Amendment 28 to the application dated July 21, 1970 requested extension of the construction permit to February 1, 1971, and by Order dated July 31, 1970 the staff granted such extension.

3. Following review by the staff and the ACRS of the updated application for an operating license, the Commission, pursuant to the Atomic Energy Act of 1954, as amended (Act), and its own regulations, announced by publication in the Federal Register on March 11, 1970 (35 Fed. Reg. 4344), that a public hearing would be held before this Atomic Safety and Licensing Board (Board) to consider whether a provisional operating license should be issued to Applicant. The published Notice of Hearing specified seven issues for this Board to consider at the hearing

in arriving at its determination. 1/

1/ The Commission's notice of hearing published on March 11, 1970, at 35 Fed. Reg. 4344, specified the following issues to be considered at the hearing:

1. Whether the applicant has submitted to the Commission all technical information required by Provisional Construction Permit No. CPPR-31, the Act, and the rules and regulations of the Commission to complete the application for the provisional operating license;

2. Whether construction of Unit 1 has proceeded and there is reasonable assurance that it will be completed, in conformity with Provisional Construction Permit No. CPPR-31 the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

3. Whether there is reasonable assurance (i) that the activities authorized by the provisional operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with rules and regulations of the Commission;

4. Whether the applicant is technically and financially qualified to engage in the activities authorized by the provisional operating license in accordance with the rules and regulations of the Commission;

5. Whether the applicant has furnished to the Commission proof of financial protection in accordance with 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements", of the Commission's regulations;

6. Whether there is reasonable assurance that Unit 1 will be ready for initial loading with nuclear fuel within 90 days from the date of issuance of the provisional operating license; and

7. Whether issuance of the provisional operating license under the terms and conditions proposed will be inimical to the common defense and security or to the health and safety of the public.

4. The Notice of Hearing set the time and place of the hearing and provided for a prehearing conference. It also explained how interested persons could petition for leave to intervene in the proceedings as parties and how persons wishing to express their views at the hearing could do so without becoming intervening parties.

5. As scheduled in the Notice of Hearing, this Board held a prehearing conference on April 7, 1970, at Buffalo, Minnesota. The public hearing also had been scheduled for Buffalo, but was held in the United States Federal Courthouse in St. Paul in response to requests of, and to facilitate attendance at the hearing by, interested citizens in the Twin Cities area.[2/] Public hearings were held on April 28 - May 1, 1970, June 15 - June 18, 1970, and August 5 - August 7, 1970.

6. This is a contested proceeding within the meaning of section 2.4(n) of the Commission's Rules of Practice. The parties to this proceeding are Applicant, the staff, and the following three intervenors from whom petitions for leave to intervene were received by this Board, and granted:[3/]

2/ Board Order, April 8, 1970, Tr. p. 208.

3/ A petition for leave to intervene filed by Mr. John Pegors on behalf of Clean Air-Clean Water Unlimited was denied by this Board for failing to state his contentions in reasonably specific detail as required by section 2.714(a) of the Commission's Rules of Practice. The organization was granted the right to make a limited appearance. Tr. p. 209.

a. Minnesota Environmental Control Citizens Association (MECCA);

b. Mr. Michael Donahue, a resident of Elk River, Minnesota (Donahue);[4] and

c. Messrs. Kenneth Dzugan, Theodore Pepin, and George Burnett, graduate students at the University of Minnesota (Dzugan).

7. Mr. John P. Badalich appeared pursuant to section 2.715(c) of the Commission's Rules of Practice to make an unsworn statement on behalf of the Minnesota Pollution Control Agency.[5] Pursuant to section 2.715(a) of the Rules of Practice, limited appearances were granted by this Board to allow presentation of unsworn statements by the City of St. Paul[6] and by eleven individuals on their own behalf and on behalf of organizations they represented.[7]

CERTIFICATION OF QUESTIONS TO ATOMIC SAFETY AND LICENSING APPEAL BOARD

8. By letter dated April 20, 1970, addressed to this Board, Dzugan requested that

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4/ Prior to the second phase of hearings and by telegram dated June 14, 1970, Mr. Donahue withdrew as a party to the proceeding.

5/ Tr. pp. 333-35.

6/ Tr. pp. 353-61.

7/ Tr. pp. 335-52, 362-92, 467-69.

a. Applicant's records of the corrective action and disposition of nonconformance in materials at vendors' shops,

b. Applicant's records of the base line inspection, and

c. all AEC onsite inspection reports

be made available to them. Thereafter, and prior to the commencement of the public hearing on April 28, 1970, Applicant made available to Dzugan Applicant's records which had been requested. The AEC inspection reports were exclusively in the possession of the staff.

9. Intervenor MECCA, by letter of April 21, 1970, requested this Board to subpoena the AEC's "original inspection documents of the construction and quality assurance of the plant". The Board, on April 24, 1970, issued a subpoena duces tecum to AEC's Director of Regulation directing the production of "the original reports, or complete copies thereof, of the inspections made by the Division of Compliance" in connection with the Monticello Nuclear Generating Plant, Unit 1. On June 1, 1970, the staff transmitted to this Board and the parties the subpoenaed inspection reports with the following four categories of information deleted:

Category 1 - the names of persons, other than AEC personnel, who provided information during the inspections;

Category 2 - references to AEC internal memoranda,

instructions, including inspection techniques, and meetings;

Category 3 - references to other identified facilities;

Category 4 - information of a proprietary nature.

MECCA objected to the deletions in all four categories "with the possible exception" of the proprietary information category. Dougan objected to any deletions from the inspection reports.

10. On July 6, 1970 this Board certified to the Atomic Safety and Licensing Appeal Board the following two questions:

Question 1 - "In the present matter, is the board entitled to make its own judgment as to the propriety of the Director of Regulation's deletions in the subpoenaed inspection reports?"

Question 2 - "On the assumption of an affirmative answer to question 1, where does the greater merit lie as between the Director of Regulation's position with respect to the deletions and the board's position?"

In a Commentary accompanying the questions this Board described the character of the deletions. The category 1 deletions "pertain to the names and titles or positions of persons other than AEC personnel who provided information during the AEC inspections". While the deletions may not be proper in the Board's view, no substantive information was included in the category 1 deletions.

11. The category 2 deletions, this Board noted, resulted in six types of deletions:

Type (a) "Mere reference to a fact that a meeting was held with minimal information, if any, about time and location of meeting, its attendees, and purpose or subject of meeting."

Type (b) "Nomenclature of AEC inspection category applicable to the inspection covered by report or referred to therein."

Type (c) "Mere reference to a report, memorandum, or letter with no more than a notation of the subject matter, date, author, and addressee."

Type (d) "Actual report of a meeting of AEC and non-AEC personnel concerned with particular problems relating to plant construction."

Type (e) "Bare references to internal instructions covering inspection procedures."

Type (f) "Indications of what items will be subject to review in one AEC inspection undertaking and of what items had been the subject of review in another one."

Types (a), (b), (c) and (e), this Board maintained in its Commentary, are mere references to materials and not the materials themselves. Types (d) and (f) contain substantive information

which ought to have been made available in this proceeding in the view of this Board although this Board acknowledges the "validity of maintaining in confidence data regarding special inspection plans and procedures." This Board notes too that Types (d) and (f) do not include descriptions of AEC inspection procedures; nor do they contain AEC inspection manuals.

12. The category 3 deletions, this Board observed in its Commentary, "amount to an indiscriminate elimination of references to any nuclear power plant facility other than the applicant's". The deletions, while improper in the Board's view, contain no details of inspection procedures or AEC inspection manuals, although they do in a few instances contain substantive data pertaining to other facilities.

13. This Board also noted that the category 4 deletions included proprietary information of certain of Applicant's contractors and vendors. They contain no details of AEC inspection procedures. Applicant twice offered to make the information available to intervenors under conditions which would not limit their cross-examination privileges but which would protect the value of the proprietary information to the owners thereof. On both occasions the intervenors rejected Applicant's offers.

14. The deletions from the inspection reports present, in this Board's judgment, issues of general principle, as discussed in this Board's certification to the Atomic Safety and Licensing Appeal Board. The deletions did not, however, prevent the development of an adequate record to support the findings made

herein. At no time after the availability of the inspection reports, with the deletions, was there any showing of the reasonableness, need or necessity of the discovery sought and denied by the deletions in the particular context of this proceeding.

15. While this Board observed that an evaluation of the AEC inspection effort might be affected by the unavailability to the parties of AEC inspection manuals, the Board notes that the AEC inspectors who performed the inspections and who wrote the reports were available for cross-examination throughout the proceeding by the parties and by this Board and that no request for discovery of such inspection manuals was made by any party to the proceeding.

16. In any event, this Initial Decision pursuant to 10 CFR Section 50.57(c) of the Commission's regulations is subject to review by the Atomic Safety and Licensing Appeal Board before it becomes a final decision of the Commission and the Appeal Board will have an opportunity to consider the adequacy of the record in light of the deletions in the inspection reports determined by the Director of Regulation and which are the subject of this Board's Certification of Questions to the Appeal Board.

MOTION FOR INTERIM AUTHORITY TO LOAD FUEL AND CONDUCT LOW POWER STARTUP TESTING

17. On April 12, 1970, prior to the commencement of the public hearing, Applicant moved this Board to order the

issuance of an interim provisional operating license authorizing initial fuel loading and low power startup testing at power levels up to a maximum of five megawatts thermal. That motion was denied by this Board on May 1, because, in this Board's view, the record was not at that time sufficiently complete to allow consideration of the necessary findings to support such an order. Applicant renewed its motion for a fuel loading and low power testing license at the second phase of hearings on June 17, when the record was substantially more complete than it had been on May 1. The Applicant, the staff and the intervenors were permitted to submit proposed findings and conclusions with respect to a low power license. Findings were received from Applicant, the staff and Dzugan. MECCA did not file any such proposed findings and conclusions. The Board's disposition of Applicant's motion is considered in its Order and Initial Decision dated _____, in this proceeding.

CONTENTS OF APPLICATION AND RECORD OF PROCEEDING

18. The application and the record of the proceeding contain much detailed information about the plant, including data and information about the site and the basis of its suitability, the design and construction of the plant, quality assurance and quality control programs, engineered safeguards, design features not fully developed and evaluated at the time construction was authorized, emergency procedures, proposed technical specifications governing operation of the plant, the

Applicant's technical and financial qualifications, and the plant's bearing upon the common defense and security. At the time the construction permit was issued for the plant a number of design features were identified by the staff and the ACRS as areas requiring further information to be developed and submitted. These areas, relating to flood protection, effluent control during periods of minimum river flow, seismic design, tornado protection, reactor vessel stress analysis, isolation valve testing, and onsite emergency power supply, have all been included in Applicant's FSAR and the staff has concluded that Applicant has submitted all technical information required. 8/

19. Applicant's testimony on direct examination was primarily in the form of documentary evidence. These documents included a document dated March 19, 1970, entitled, "Applicant's Summary of the Application for the Provisional Operating License for the Monticello Nuclear Generating Plant No. 1," (Applicant's Summary) and a document dated March 26, 1970, entitled, "Financial Qualifications of Northern States Power Company -- Testimony of G. F. Johnson." Most of the staff's direct testimony was also documentary, consisting of a document dated March 18, 1970, entitled, "Safety Evaluation by the Division of Reactor Licensing" (Staff Safety Evaluation) and Supplement No. 1 thereto dated

8/ Applicant's Summary, pp. 21-22; Staff Safety Evaluation, p. 37; Tr. pp. 464-466, 469-476.

March 30, 1970. These documents were supplemented by oral testimony to include consideration of Amendments 26 and 27 and Applicant's emergency plan, and to reflect the ongoing staff inspections of the construction of the plant. The staff and Applicant also offered in evidence a document entitled, "Record for Hearing -- Correspondence", and the documents referenced therein, for the purpose of including in the hearing record the record of the application pursuant to section 2.743(g) of the Commission's rules of evidence. In addition Applicant and the staff presented evidence under cross-examination by the intervenors, by oral redirect and rebuttal evidence, and in response to questions asked by the Board. Applicant and the staff also introduced testimony in response to questions and comments by limited appearors. Of the intervenors, MECCA alone presented direct oral evidence and was cross-examined by Applicant and the staff.

20. Pursuant to the National Environmental Policy Act of 1969 [9] and the Commission's published statement of general policy for implementation of that Act, [10] the Commission has issued a detailed statement on the environmental considerations involved in the Monticello Plant. [11]

9/ Public Law 91-190.

10/ Appendix D, 10 CFR Part 50, 35 Fed. Reg. 5463 (April 2, 1970).

11/ Statement on Environmental Considerations Relating to Proposed Operation by Northern States Power Company of the Monticello Nuclear Generating Plant (Unit 1), April 24, 1970.

The Plant Site

21. The site of the Monticello Nuclear Generating Plant, Unit 1, consists of 1325 acres located partially in Sherburne County (on the east bank of the Mississippi River) and partially in Wright County (on the west bank of the River). The plant is located in Wright County. The site is about 22 miles southeast of St. Cloud (1960 population 33,815) and 30 miles northwest of Minneapolis. The nearest residence is off-site, approximately 2750 feet from the plant. The area surrounding the site is primarily agricultural. A low population zone with a radius of one mile includes a population of about 25. The minimum exclusion zone radius is 1600 feet. 12/ The plant design takes into account meteorological, hydrological, ground water, and soil conditions, as well as the possibility of credible earthquakes, wind storms, tornadoes, and floods. 13/

Features of the Plant

22. The nuclear steam supply system is a General Electric boiling water reactor design which is identical in most features to Commonwealth Edison Company's Dresden Unit 2, recently licensed by AEC for operation, and is similar to other operating boiling water reactors. 14/ The reactor is a single-cycle, forced circulation, boiling water reactor producing steam

12/ Applicant's Summary, p. 3; Staff Safety Evaluation, p. 5.

13/ Applicant's Summary, pp. 3-6; Staff Safety Evaluation, pp. 5-9.

14/ Applicant's Summary, p. 7; Staff Safety Evaluation, p. 11.

for direct use in the steam turbine. The reactor will be fueled with slightly enriched uranium dioxide pellets sealed in Zircaloy fuel rods. Reactivity control is provided by movable control rods and variable recirculation flow.[15] The primary containment system, consisting of a steel drywell and a steel pressure suppression chamber, is designed to accommodate the pressures and temperatures which would result from, or occur subsequent to, a failure equivalent to a double-ended, circumferential rupture of a reactor coolant recirculation system line resulting in the loss of reactor water at the maximum rate. The primary safety functions of the secondary containment, consisting of the reactor building and the standby gas treatment system, are to minimize ground level release of airborne radioactive materials, and to provide for controlled, filtered, elevated release of the reactor building atmosphere under postulated design basis accident conditions. The reactor building provides secondary containment during periods when the primary containment system is in service, and primary containment during periods when the primary containment is open.[16]

23. In addition to the primary and secondary containment systems, the plant has a number of safety features designed for limiting the consequences of accidents, including the highly unlikely loss-of-coolant accident. The principal safety features

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15/ Applicant's Summary, pp. 7-9; Staff Safety Evaluation, pp. 12-14.

16/ Applicant's Summary, pp. 9-12; Staff Safety Evaluation, pp. 22-25.

include the emergency core cooling systems, [17] the reactor standby gas treatment system, [18] a reactor protection system designed to automatically shutdown the reactor when pre-established safety limits are reached, [19] and a standby liquid control system which provides backup reactivity shutdown capability in the unlikely event that shutdown cannot be accomplished by control rods alone. [20]

24. The reactor primary coolant system includes the reactor pressure vessel, the two-loop reactor coolant recirculation system, and the main steam piping. The water circulating in the primary system is used both to cool the reactor core within the pressure vessel and to produce steam for the production of electrical power. [21]

25. Dzugan questioned Applicant about the capability to detect a loose object in the core which might interrupt the coolant flow patterns. Applicant testified that the velocities of the coolant at the bottom of the vessel were too slow to carry objects of significant size up into the

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17/ Applicant's Summary, pp. 12-14; Staff Safety Evaluation, pp. 25-28.

18/ Applicant's Summary, pp. 11-12; Staff Safety Evaluation, pp. 24-25.

19/ Applicant's Summary, pp. 15-16; Staff Safety Evaluation, p. 28.

20/ Applicant's Summary, p. 8; Staff Safety Evaluation, p. 13.

21/ FSAR, §4-1.

core region. The testimony indicated that small "postage stamp size" pieces of metal might be carried up into the core, but would cause only local coolant blockage around a specific fuel rod. Such blockage could conceivably interrupt the coolant flow enough to cause the rod to fail, but failure of one or two rods is of little significance in terms of release of radioactivity into the primary coolant. Applicant also noted that calculations and tests show that the flow through a channel would have to be blocked by 80 or 90 percent to produce fuel clad failure. Should coolant activity exceed specified limits, the steam line radiation monitors would detect it and cause the reactor to shut down. 22/

26. To ensure the integrity of reactor systems, including the primary coolant system, the components are fabricated and inspected in accordance with applicable engineering codes and standards which include provisions for detailed quality control measures taken during fabrication. Testimony by Applicant, on cross-examination, revealed that each weld of the pressure vessel, for example, was inspected by various scientific methods about 17 times, that the vessel was fabricated and inspected in such a manner as to be completely certified and stamped under the A.S.M.E. code for nuclear vessels, and that Applicant's subequent inspection of the accessible welds on the installed vessel yielded no questionable welds. 23/

22/ Tr. pp. 1150-52.
23/ Tr. p. 764-77.

Liquid Effluents

27. Liquid wastes generated by normal operation of the plant are collected and processed through a radwaste system which removes radioactive contaminants by filtration and/or mixed deep-bed ion exchange demineralization. Radioactivity is also reduced through decay during storage in holdup tanks. Liquid wastes with high levels of radioactivity are processed and normally returned for reuse within the plant. Low level radioactivity liquid waste is processed, stored, sampled, analyzed, diluted, and periodically released into the Mississippi River under carefully controlled batch-by-batch conditions to ensure that allowable concentration limits are not exceeded, even during periods of extremely low river water level.[24/]

28. MECCA and Dzugan indicated concern that radioactive effluents were being discharged upstream from the drinking water inlets of St. Paul and Minneapolis. These are the nearest public water intakes, being 33 miles and 37 miles, respectively, from the plant. Testimony by Applicant and the staff indicated that the annual average concentration of radioactivity in the effluents was not expected to be more than a few percent of AEC's limits in Part 20 of its regulations and, considering a further dilution factor experienced during the downstream flow, the radioactive

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24/ Applicant's Summary, pp. 18-19,20; Staff Safety Evaluation, pp. 32-37, 41-42; Tr. pp. 470-71.

concentrations at the public water intake would be well below allowable Part 20 limits.[25/] The consequences of the worst possible accidental release from the liquid radwaste storage tanks to the river at the plant site boundary result in a concentration of radioactivity less than 10 CFR Part 20 limits.[26/]

Gaseous Effluents

29. Radioactive gases generated during normal operation of the plant are stored to provide radioactive decay time, filtered, diluted, and finally released through the plant off-gas stack which provides further dilution in the atmosphere. Releases are carefully monitored and controlled to ensure that the radiation dose at the theoretical point of highest exposure offsite, i.e. at the site boundary, will be below the limits of Part 20 of AEC's regulations. Exposures further away from the site boundary will be still less.[27/] A continuous monitoring system automatically terminates release when preset limits are reached.[28/]

[25/] Staff Safety Evaluation, pp. 34-35; Tr. pp. 647-50, 660-62.

[26/] Deposition Tr. p. 301; FSAR p. 9-2.7.

[27/] Tr. p. 659.

[28/] Applicant's Summary, pp. 18-19, 20; Staff Safety Evaluation pp. 37-42.

30. Most of the gaseous isotopes such as Krypton-85 which decay into biologically important radioactive daughters decay into radioactive particulates during the 30-minute hold-up time and the high-efficiency filters collect the radioactive particulates with an efficiency of 99.97%. Undecayed noble gases would not be stopped by the filters and are released to the atmosphere but their contribution to offsite radiation exposure is insignificant.[29/] Applicant has determined that, based on natural dispersion of gaseous effluent in the atmosphere and the meteorological and topographical characteristics of the site area, an annual average stack release rate of 0.48 curies per second (ci/sec) can be accommodated without exceeding AEC's regulatory dose limits in 10 CFR Part 20.[30/] The staff, using more conservative assumptions, would limit the annual average stack release to 0.27 curies per second.[31/] The expected composition of the effluent gases and the decay schemes of each radioisotope are considered in calculating the allowable discharges to the atmosphere.[32/]

31. Applicant's and the staff's dose calculations take into consideration the entire spectrum of meteorological conditions at the plant site and gaseous effluents can be released even during periods of the most adverse and unstable atmospheric conditions.[33/]

29/ Tr. pp. 1822-1824.

30/ FSAR p. 9-3.1; Tr. pp. 566-568.

31/ Tr. pp. 781-782.

32/ Tr. pp. 1074-1097, 1822-1826.

33/ Tr. pp. 986-991, 1818-1819.

32. Dzugan cross-examined Applicant and the staff at some length based on a concern that the offsite dosage would increase because of a possible buildup of radioactivity over the years from normal plant releases. Technical testimony by the applicant thereby elicited indicated that accumulation, resulting primarily from deposition of particulate materials with long half lives, constituted a negligible contribution to offsite dose. Technical testimony by the staff indicated that their calculation did take into account the accumulation of fission products with long half lives. In any event any such accumulation would be promptly detected by the Applicant's radiological monitoring program.[34/]

33. The Applicant's offsite dosage calculations are performed using a mathematical model or formula derived from empirical observations. Dzugan questioned Applicant about the accuracy of the model used by Applicant in predicting offsite dosages under various meteorological conditions. Applicant expressed confidence in the model and testified that the model used is an analytical model developed at AEC's Hanford laboratory on the basis of experimental results observed over many years. Applicant explained that an error analysis is performed at the time a model is developed. An error analysis is not

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34/ Tr. pp. 564-581, 651-672, 781-797.

performed on every project because the extremely conservative engineering and modeling assumptions used make it unnecessary to do so. The model used by Applicant has been verified with the use of meteorological, off-gas, and dose measurements at the Brookhaven National Laboratory. During operation of the plant, the calculated models are verified by actual measurement under Applicant's radiation monitoring program.35/

34. In any event the Technical Specification limit is based on the staff's more conservative calculational method. Dzugan asked if Applicant took into account in the dosage calculations the effect of the differing chemical properties of the various molecules which have taken up tritium. Applicant testified that the possible chemical alterations in molecules which incorporate tritium had been considered by the International Commission on Radiological Protection when it established the standards for tritium uptake -- the effects are negligible and are of no importance in determining the radiological significance of tritium.36/

35/ Tr. pp. 949-971, 977-984, 1820-22.

36/ Tr. pp. 971-972, 2003-2004.

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35/ Tr. pp. 949-971, 977-984, 1820-22.

36/ Tr. pp. 971-972, 2003-2004.

Design Basis Accident

35. In determining the safety of the reactor design, detailed safety evaluations and analyses were made by Applicant and the staff, and reviewed by the ACRS, to determine the capability of the design to mitigate the consequences of a design basis accident should it occur.[37] Design basis accidents are the worst possible accidents postulated for the reactor.[38] With regard to the calculations of radiation doses which might be received at various distances from the plant in the event of a design basis accident, MECCA, referring to a 1967 report prepared by the U.S. Public Health Service,[39] contended that Applicant had not evaluated the consequences of an accident wherein 100% of the reactor core is melted. MECCA requested that an operating license not be granted until such an evaluation is made.[40] The evidence, however, indicates that the staff's evaluation and Applicant's evaluation of the radiological consequences of a loss or coolant accident at the plant take into consideration the fission products which would result from a 100% core meltdown notwithstanding the fact that a 100% core meltdown is precluded by the incorporation of highly redundant networks of engineered safeguards to

37/ Applicant's Summary, pp. 1-2; Staff Safety Evaluation, pp. 43-

38/ Tr. pp. 483-35.

39/ Public Health Evaluation, Monticello Nuclear Generating Plant, May 10, 1967; Tr. p. 735.

40/ Tr. pp. 688-89, 697.

cool the core in the event of a loss of coolant accident. These safeguards also thereby assure the integrity of the containment system for mitigating the release of fission products to the atmosphere.[41] Safety evaluations by Applicant and the staff, introduced into evidence, demonstrate that the doses which could result from a design basis accident are well within the guideline values of Part 100 of the Commission's regulations.[42] Such doses would not be expected to cause biological injury to persons in the vicinity of the plant.[43]

Plant Security

36. Section 50.13 of the Commission's regulations provides that:

> "An applicant for a license to construct and operate a production or utilization facility, or for an amendment to such license, is not required to provide for design features or other measures for

41/ Tr. pp. 536-39, 641-47, 1886-1918, 2000-1.

42/ Applicant's Summary, p. 20; Staff Safety Evaluation, p. 40. Although the accident doses calculated by Applicant and the staff were both well within the Part 100 guideline values, the staff's calculated doses were higher than those of the Applicant. Testimony by both parties explained that the calculations involve the assignment of many parametric values related to the size of the source of radioactivity, transport and behavior mechanisms of radioactive materials, meteorological conditions, and dose conversion factors. In nearly all cases the staff used more conservative parametric values leading to a higher calculated accident dose. The staff witness emphasized the conservatism of the staff's approach and suggested that Applicant's parametric values, leading to lower calculated doses, were probably the more realistic. Tr. pp. 420-29, 476-483.

43/ Tr. pp. 1896-1902.

> the specific purpose of protection against the effects of (a) attacks and destructive acts, including sabotage, directed against the facility by an enemy of the United States, whether a foreign government or other person, or (b) use or deployment or weapons incident to U.S. defense activities."

Protection against possible industrial sabotage including the provision of appropriate industrial security measures, is an appropriate matter for consideration at the operating license stage.[44/] Access to the plant will be safeguarded by a number of fences. Gates in the security fence will be locked when unattended. The locks and keys at the plant site are part of a non-commercial keyway system established by the lock manufacturer specifically for Applicant. Protection of plant facilities will also be available from local law enforcement authorities and National Guard personnel, when appropriate. The design of the plant structures and equipment which are important to the safety of the plant include allowance for the effects of floods, tornadoes, and earthquakes. These design measures taken together with the inaccessibility to the reactor vessel and primary system piping during operation and the redundant safeguard systems inherently provide a substantial degree of protection against any public safety consequences of possible industrial sabotage. The plant security measures make any such sabotage a very low probability event.[45/]

44/ In the Matter of Florida Power & Light Company 2 CCH At. E. L. Rep., ¶11259.

45/ Tr. 1351-1880.

Emergency Plans

37. Applicant has prepared a plan describing the emergency organization and the arrangements to be effected in the unlikely event of an accident which might affect the general public. Emergency communications have been installed to provide uninterrupted liaison between onsite personnel and offsite support groups and agencies. Applicant has made emergency arrangements with responsible agencies of the State of Minnesota and with appropriate local officials, and has made emergency medical arrangements with a local hospital for treatment of contaminated patients.46/

38. Applicant's emergency plan had been submitted as a part of the FSAR. Applicant introduced as an exhibit in these proceedings detailed procedures which supplement the emergency plan and which will become a chapter of the plant operations manual. In response to questioning by this Board, the staff testified that the detailed procedures conformed to the staff-approved emergency plan, and, further, that the plan and the procedures meet the emergency planning guidelines of the Commission's proposed amendment to 10 CFR Part 50 of its regulations, and meet

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46/ FSAR §13-44; Applicant's Summary, pp. 33-34; Staff Safety Evaluation, pp. 50-51; Tr.pp. 539-50, 551-52, 844-45, 901-904, 1298-1307.

the intent of a draft document which the staff had prepared for the use of applicants in developing their emergency plans. The staff testified that Applicant's emergency plan had been reviewed to determine that its various elements were sufficient to provide reasonable assurance that measures can and will be taken in the event of an emergency to adequately protect the health and safety of the public and to prevent damage to property. The staff also testified that the detailed implementation procedures described the assignment of emergency duties of plant personnel and off-site groups, define classes of emergencies and the range of possible accidents, define the action of responsible individuals both within and without Applicant's organization in responding to the emergency and evacuating off-site personnel, provide details for post-accident monitoring of effluents and the environment by the Applicant's staff and the Minnesota Department of Health, describe the communications network for on-site and off-site communications, state the rule of local authorities if evacuation becomes necessary, describe the plan for traffic control, including detour plan; and include an expanded reentry procedure which specifies the criteria for reentry of affected areas.47/

47/ Tr. pp. 1210-16, 1298-1307.

39. Applicant's calculated doses resulting from a design basis accident are well below the guideline doses set forth in 10 CFR Part 100 of AEC regulations and would not require evacuation of any people outside of the exclusion area. The exclusion area is a fenced area within the reactor site over which Applicant has complete control. Even AEC's more conservative dose calculations would not predict the need for more than limited evacuation of the low population zone which is an area within a radius one mile from the plant. About 25 people live within a one-mile radius of the plant. It is inconceivable that an accident could occur which would require evacuation of people living beyond the low population zone. Evacuation plans have been formulated and will be coordinated by the offices of the Wright County Sheriff, the Sherburne County Sheriff, and the Monticello area Civil Defense Coordinator.[48/]

40. Applicant testified that the procedures for testing the emergency plan had been prepared and the emergency plan would be tested prior to loading of fuel into the reactor. The emergency plan testing consists of five separate tests including postulated airborne

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48/ Tr. pp. 1919-26.

releases, postulated liquid releases, in-plant testing of the evacuation system, testing of the communication system, and a test evacuation of all people on the site. The tests will include actual contact with all involved on-site personnel and off-site government officials. Testing of the communications system includes simulating loss of telephone contact and testing a specially installed radio system with backup battery supply.[49/]

Environmental Monitoring

41. The applicant initiated in June 1968, an environmental radiation monitoring program to determine and evaluate the effects of the plant's operation on the environment. The program will continue through plant startup and operation, and includes the collection and analyses of samples of air, water, soil, vegetation, milk and aquatic life. Studies to date have been conducted in cooperation with the Minnesota Department of Health, and the applicant has taken into account the recommendations of the Fish and Wildlife Service, U. S. Department of the Interior. Annual reports of the monitoring program are widely distributed to Federal and State agencies and are available to other interested parties. Applicant is

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49/ Tr. pp. 1974-85.

also conducting a companion ecological monitoring program dedicated to the study of the aquatic environment on a six-mile stretch of the Mississippi River in the vicinity of the plant. The first program includes the study of concentration of radioactive materials in aquatic life, and the ecological program will include monitoring and analysis of the effects of thermal discharges on the aquatic environment.[50] Applicant testified that it does not presently plan to conduct studies on the effects of radioactivity on the terrestrial wildlife in the vicinity of the plant, primarily because similar terrestrial studies conducted by others, Applicant and the U. S. Public Health Service, yielded nothing to indicate abnormal concentration of radioisotopes in terrestrial animal life.[51]

42. A representative of the U. S. Fish and Wildlife Service, testifying on behalf of the staff, said that present knowledge supported the conclusion that fish and other less advanced animals are less sensitive to radiation than man, and that the concentrations of radioactivity required to injure fish and wildlife are much higher than the maximum permissible concentrations specified in Commission regulations.[52] Experience

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50/ Tr. pp. 502-3, 505-7, 509-11, 809-32, 834-6.

51/ Tr. pp. 511-12, 558-60.

52/ Tr. pp. 507-9, 840-5.

derived from environmental monitoring programs conducted in the vicinity of other operating nuclear power plants indicates no significant concentration of radioactivity in aquatic organisms.[53/]

Construction in Accordance with Applicant's Construction Permit

43. The Commission's Division of Compliance has followed closely the progress of plant construction by means of a series of onsite inspections and conferences with Applicant's and vendor's personnel. The inspection activities, conducted both at the site and at the fabrication shops included review and audit of Applicant's quality assurance and quality control programs, inspection of quality control records, observation of construction work in progress, review of construction procedures, observation of major testing, review of functional testing programs, and review of preparations for facility operations. As a result of these inspections and conferences, the staff has concluded that "there is reasonable assurance that Unit 1 will be completed in conformance with Provisional Construction Permit No. CPPR-31, the application, as amended, the provisions of the Act, and the rules and regulations of the Commission."[54/]

53/ Tr. pp. 552-7, 832-4.

54/ Supplement No. 1 to Staff Safety Evaluation, p. 19; Tr. pp. 1713-17.

44. During the course of construction of the containment drywell a crack was discovered, on January 18, 1968, in the containment vessel at a location where an insert plate was welded to the shell. The evidence shows that an extensive program was employed to isolate the cracking, establish its cause, and to make the necessary repairs. The cracking was found to be surface type cracking caused by the presence of hydrogen, high residual shrinkage stresses, discontinuities at the surface and high hardness. Non-destructive testing methods showed no indication of subsurface cracking in areas where surface cracking had been detected or in areas which were free of surface cracks. The cracks were repaired and the containment was inspected and tested to assure that no cracking resulted from the repair procedure and that the cracks were properly repaired. The repair and evaluation procedures were independently reviewed and found acceptable by the Hartford Steam Boiler and Inspection Company and the regulatory staff.[55/]

45. Applicant was cross-examined at some length by Dzugan about the details of testing and inspection of the reactor vessel, containment, and associated piping. Testimony of Applicant and the staff demonstrated the adequacy of Applicant's inspection programs and indicated

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55/ Tr. pp. 1793-4, 1934-68.

that the components of the primary coolant system have been constructed, tested, and inspected in accordance with the applicable codes and specifications.[56/]

46. MECCA and Dzugan contend that the plant construction has not been adequately tested and inspected by the staff in that the staff does not do any "independent" testing and in that the AEC inspection program does not provide for resident inspectors at the reactor site. The record shows that the AEC inspection program is directed toward verifying, on a sampling basis, that the completed facility conforms to the application and the AEC regulations. At periodic visits to the reactor site and to selected vendor shops, the AEC, among other things:

a. reviews the applicable quality assurance and quality control programs and their implementation;

b. reviews quality control records, including material test reports and non-destructive test records;

c. observes construction work in progress and construction procedures, including concrete placement, equipment installation and non-destructive testing;

56/ Supplement No. 1 to Staff Safety Evaluation; Tr. pp. 764-77, 876-901, 1986-1999.

d. witnesses major construction tests; and

e. reviews operating organization and reviews test and operating plans and procedures.

47. While the AEC inspectors do not physically perform the tests, the inspectors do independently review and evaluate the Applicant's records including non-destructive testing documentation. They make independent judgments of test results and the validity of test procedures. For example, the results of the integrated leak rate tests of the primary containment conducted by the Applicant in March and in April were not considered acceptable by AEC inspectors and the test had to be performed again.57/

48. The AEC inspection program is adequate and would not be frustrated by incorrect records whether falsely or negligently generated. No evidence was adduced in support of the proposition that resident inspectors would necessarily do a superior job of inspection than frequent, random inspection visits. Nor was any evidence adduced to challenge the conclusions of the inspectors, who were present and available during the course of the hearing for examination, that the plant construction conforms with the construction permit, the application and Commission's

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57/ Supplement No. 1 to Staff Safety Evaluation, p. 11; Tr. pp. 1806-1808.

requirements. Nor was any evidence adduced that there is any unsafe feature of the plant. The random nature of the inspection program, the substantial number of contractors and Applicant personnel who would have to be involved in a coverup of incorrect information, the substantial verification of records provided by construction testing and preoperational testing, and the experience and competence of the AEC inspectors substantially minimize the possibility of any falsification of the records.[58] Furthermore, the AEC requirements for quality assurance programs[59] and conformance to codes and standards as well as the Applicant's vested interest in a reliable plant[60] collectively involve the application of a complex system of checks and tests by numerous independent organizations to assure the integrity of the design and construction of the plant. While the plant construction is not fully complete, (a number of unresolved items have been identified and all tests have not been completed) the evidence, without contradiction, supports the finding that the plant construction has proceeded in conformance with the construction

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58/ Tr. pp. 1808-1811.

59/ 10 CFR Part 50, Appendix B.

60/ Tr. pp. 1986-1999, 2135-36.

permit, the application and the rules and regulations of the Commission.

49. This Board is not required to find that the identified unresolved items[61] are all resolved prior to issuing this Initial Decision. It is sufficient for the Board to find that construction "has proceeded and there is reasonable assurance that it will be completed" in conformance with the construction permit, the application and the Commission's requirements. The issuance of the license ordered herein is subject to verification of completion by the Division of Compliance.

61/ Tr. pp. 1717-23, 2031-44.

Applicant's Technical Qualifications

50. Applicant has gained considerable nuclear experience in the construction and operation of the Pathfinder Atomic Power Plant. The supervisory staff chosen to manage operations at the Monticello plant is composed of formerly licensed reactor operators at the Pathfinder plant and the qualifications of the key supervisory and professional personnel meet the "Proposed Standards for Selection and Training of Personnel for Nuclear Power Plants," Draft No. 9, July 3, 1969, prepared by the American Nuclear Society Standards Committee. 62/

Applicant's Financial Qualifications

51. The Applicant estimates an average annual cost of $8.8 million for each of the first five years of operation. The record indicates that the Applicant's operating revenues will be ample to cover these costs and to engage in the activities which would be authorized by the full power provisional operating license. 63/

Financial Protection and Indemnity Requirements

52. The Applicant has satisfied the present financial protection requirements under 10 CFR Part 140 of the Commission's regulations by furnishing to the Commission proof of financial protection in the amount of $1,000,000, as needed for the period

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62/ Applicant's Summary, pp. 30-32; Staff Safety Evaluation, pp. 51-52; Tr. 1900-1930; 2001-2002.

63/ Financial Qualifications of Northern States Power Company -- Testimony of G. F. Johnson; Staff Safety Evaluation, pp. 56-

fuel is stored unused on the site, in the form of a Nuclear Energy Liability Insurance Association policy No. NF-174, and by entering into Indemnity Agreement No. B-42 with the Commission applicable to fuel storage. The Applicant has obtained letters from the Nuclear Energy Liability Insurance Association and Mutual Atomic Energy Liability Underwriters committing to provide an aggregate financial protection of up to $82 million, the maximum amount required by the Commission's regulations for a full power license for a facility of this size.[64/] In accordance with section 50.57(a)(5) of the Commission's regulations, the Commission will not issue an operating license until Applicant has obtained the amount of financial protection required by Part 140. There is reasonable assurance that Applicant will obtain such required financial protection.

Common Defense and Security

53. The activities to be conducted under the provisional operational license will be within the jurisdiction of the United States, and all of the directors and principal officers of the Applicant are United States citizens. The Applicant is not owned, controlled, or dominated by an alien, a foreign corporation or a foreign government. The activities

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64/ Financial Qualifications of Northern States Power Company -- Testimony of G. F. Johnson; Staff Safety Evaluation, pp. 56-57.

to be conducted do not involve any restricted data, but Applicant has agreed to safeguard any such data which might become involved in accordance with the Commission's regulations. Special nuclear material for use as fuel in the proposed facility will be subject to Commission regulations and will be obtained from sources of supply available for civilian purposes. 65/

REVIEW OF APPLICATION BY REGULATORY STAFF AND ACRS

54. Since the filing in November 1968, the application and the amendments thereto have been under constant and thorough review and evaluation by the regulatory staff. During the evaluation, which was conducted in accordance with current Commission regulatory criteria and policies, the regulatory staff has held numerous meetings with the applicant to discuss and clarify the information submitted in the amendments. 66/ The regulatory staff made use of studies by independent experts in its evaluation of such plant safety aspects as air dispersion of gaseous effluents (Air Resources Environmental Laboratory, Environmental Science Services Administration), 67/ site hydrology (Geological Survey, U. S. Department of the Interior), 68

65/ Applicant's Summary, p. 35; Staff Safety Evaluation, p. 56.

66/ Staff Safety Evaluation, p.2.

67/ Staff Safety Evaluation, Appendix B.

68/ Staff Safety Evaluation, Appendix C.

ecological effects (Fish and Wildlife Service, U. S. Department of the Interior),[69] reactor vessel stress analysis (Teledyne Materials Research),[70] structural design adequacy (Nathan M. Newmark Consulting Engineers),[71] and site seismology (U. S. Coast and Geodetic Survey).[72]

55. The ACRS has also reviewed the amended application for an operating license and, after identifying several items for resolution by Applicant and the staff and making several recommendations, concluded that the plant can be operated at power levels of up to 1670 megawatts thermal without undue risk to the health and safety of the public. The items identified by the ACRS have been considered by the regulatory staff in its evaluation of the application, and Applicant has agreed to implement the recommendations of the ACRS.[73]

69/ Staff Safety Evaluation, Appendix D.

70/ Staff Safety Evaluation, Appendix E.

71/ Staff Safety Evaluation, Appendix F.

72/ Staff Safety Evaluation, p. 7.

73/ Applicant's Summary, pp. 23-27; Staff Safety Evaluation, pp. 54-55. The ACRS reported on the suitability of the Monticello site in a letter dated May 11, 1966; on the construction permit application in a report dated April 13, 1967; on the operating license application through amendment 24 in a report dated January 10, 1970; and on changes to the reactor vessel nozzle safeends, as described in amendments 26 and 27 in a report dated June 16, 1970. Tr. pp. 465-6, 469-76, 1223-27.

56. The results of the regulatory staff's review and evaluation of the application are contained in the regulatory staff's safety evaluation which has been made available to the public and which has been admitted into evidence in this proceeding. The regulatory staff concluded in its safety evaluation that, with respect to a provisional operating license authorizing operation at power levels up to 1670 megawatts thermal: 74/

a. The applicant has submitted to the Commission all technical information required by Provisional Construction Permit No. CPPR-31, the Atomic Energy Act of 1954, as amended (Act), and the rules and regulations of the Commission to complete the application for the provisional operating license;

b. The construction of Unit 1 has proceeded, and there is reasonable assurance that it will be completed in conformity with Provisional Construction Permit No. CPPR-31, the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

c. There is reasonable assurance (i) that the activities authorized by the

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74/ Staff Safety Evaluation, pp. 60-61.

provisional operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

d. The applicant is technically and financially qualified to engage in the activities authorized by the provisional operating license in accordance with the rules and regulations of the Commission;

e. The applicant has furnished to the Commission proof of financial protection in accordance with 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations; and

f. The issuance of the provisional operating license under the terms and conditions proposed will not be inimical to the common defense and security or to the health and safety of the public.

The regulatory staff also concluded and testified in its Supplement No. 1 to AEC Regulatory Staff Safety Evaluation that there is reasonable assurance that the plant will be ready for initial loading with nuclear fuel within 90 days from the date of issuance of the provisional operating license.

CONCLUSIONS

57. Based on the Board's review of the entire record in this proceeding and the foregoing findings of fact and discussions, the Board concludes that:

a. The applicant has submitted to the Commission all technical information required by Provisional Construction Permit No. CPPR-31, the Act, and the rules and regulations of the Commission to complete the application for the interim provisional operating license;

b. Construction of Unit 1 has proceeded, and there is reasonable assurance that it will be completed, in conformity with Provisional Construction Permit No. CPPR-31, the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

c. There is reasonable assurance (i) that the activities authorized by the interim provisional operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

d. The applicant is technically and financially qualified to engage in the activities authorized by the interim provisional operating license in accordance with the rules and regulations of the Commission;

e. The applicant has furnished or will timely furnish to the Commission proof of financial protection in accordance with 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements", of the Commission's regulations;

f. There is reasonable assurance that Unit 1 will be ready for initial loading with nuclear fuel within 90 days from the date of issuance of the interim provisional operating license; and

g. Issuance of the interim provisional operating license under the terms and conditions proposed will not be inimical to the common defense and security or to the health and safety of the public.

ORDER

58. Pursuant to the Act and the Commission's Regulations, IT IS ORDERED THAT THE Director of Regulation issue to Northern States Power Company a provisional operating license, containing Technical Specifications as described in the hearing record, authorizing operation of the Monticello Nuclear Generating Plant, Unit 1, at power levels up to a maximum of 1670 megawatts thermal, upon verification by the Commission's Division of Compliance that the Monticello Nuclear Generating Plant, Unit 1, has been completed in conformity with Provisional Construction Permit No. CPPR-31, the application, as amended, the provisions of the Act, and the rules and regulations of the Commission, and upon receipt by the Director of Reactor Licensing of proof that Applicant has provided financial protection in the amount required by the Commission's regulations. IT IS FURTHER ORDERED in accordance with Section 50.57(e) of the Commission's regulations, that this Initial Decision shall become effective ten days after its issuance subject to (i) the review thereof and further decision by the Atomic Safety and Licensing Appeal Board, upon exceptions filed by any party, and (ii) such order as the Atomic Safety and Licensing Appeal Board may enter upon such Exceptions or upon its own motion within forty-five days after the issuance of this Initial Decision.

ATOMIC SAFETY AND LICENSING BOARD

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Valentine B. Deale, Chairman

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John C. Geyer

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Eugene Greuling