DEC 3 0 1981

Docket Nos. 50-282
and 50-306

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DDiIanni

Mr. L. O. Mayer, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Mayer:

The Commission has issued the enclosed Amendment Nos. 53 and 47 to Facility Operating Licenses Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, respectively. The amendments complete our response to your application dated August 27, 1976. The remaining items were addressed in Amendments 21 and 15 issued June 6, 1977.

The amendments revise the Appendix A Technical Specifications concerned with the operation of the steam exclusion dampers, the applicability of the containment pressure and temperature limits and the deletion of special tests which have been completed.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by

Dominic DiIanni, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 53 to DPR-42
2. Amendment No. 47 to DPR-60
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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ACD CMT for RBC 12/30/81

MB-No legal objection to form of amendment

OFFICE	ORB#3 DL PMKreutzer	ORB#3 DL DDIianni/dd	ORB#3 DL RAClark	AD:OR:DL TMNovak	OELD MBLUME		
SURNAME	12/16/81	12/15/81	12/15/81	12/15/81	12/18/81		
DATE							

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

DISTRIBUTION:
Docket File
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Docket No. 50-282 and 50-306

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: NORTHERN STATES POWER COMPANY, Prairie Island Nuclear Generating Plant, Units Nos. 1 and 2.

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).

☒ Other: Amendment Nos. 53 and 47.
Referenced documents have been provided PDR.

Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE →	ORB#3:DL					
SURNAME →	PMKreutzer/pn					
DATE →	12/31/81					

NRC FORM 102 7—79

Northern States Power Company

cc:

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Bernard M. Cranum
Bureau of Indian Affairs, DOI
15 South 5th Street
Minneapolis, Minnesota 55402

Mr. R. L. Tannner
County Auditor
Red Wing, Minnesota 55066

U. S. Environmental Protection Agency
Federak Activities Branch
Region V Office
ATTN: Regional Radiation
Representative
230 South Dearborn Street
Chicago, Illinois 60604

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
License No. DPR-42

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

   A. The application for amendment by Northern States Power Company (the licensee) dated August 27, 1976, 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. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-42 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 53 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for CM Trammell
Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: DEC 30 1981

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 47
License No. DPR-60

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

   A. The application for amendment by Northern States Power Company (the licensee) dated August 27, 1976, 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. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 47, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

CM Trammell

for Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: DEC 30 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. DPR-42

AMENDMENT NO. 47 TO FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
TS.3.4-2	TS.3.4-2
TS.3.6-3	TS.3.6-3
Table TS.4.1-1, (pg 3 of 6)	Table TS.4.1-1, (pg 3 of 6)
TS.4.4-6	TS.4.4-6
TS.4.4-9	TS.4.4-9
Table TS.6.7-1	-

e. For Unit 1 operation motor operated valves MV32242 and MV32243 shall have valve position monitor lights operable and shall be locked in the open position by having the motor control center supply breakers manually locked open. For Unit 2, corresponding valve conditions shall exist.

f. Essential features including system piping, valves, and interlocks directly associated with the above components are operable.

g. Manual valves in the above systems that could (if one is improperly positioned) reduce flow below that assumed for accident analysis shall be locked in the proper position for emergency use. During power operation, changes in valve position will be under direct administrative control.

3. Steam Exclusion System

Both isolation dampers in each ventilation duct that penetrates rooms containing equipment required for a high energy line rupture outside of containment shall be operable or one damper in each duct with an inoperable isolation damper shall be closed.

4. Radiochemistry

The iodine-131 activity of the water on the secondary side of either steam generator for that reactor does not exceed 0.30 uCi/cc.

B. If, during startup operation or power operation, any of the conditions of Specification 3.4.A., except as noted below for 2.a, 2.b or 4 cannot be met, startup operations shall be discontinued and if operability cannot be restored within 48 hours, the affected reactor shall be placed in the cold shutdown condition using normal operating procedures.

With regard to Specifications 2.a or 2.b, if a turbine driven AFW pump is not operable, that AFW pump shall be restored to operable status within 72 hours or the affected reactor shall be cooled to less than 350°F within the next 12 hours. If a motor driven AFW pump is not operable, that AFW pump shall be restored to operable status within 72 hours or one unit shall be cooled to less than 350°F within the next 12 hours.

Basis

A reactor shutdown from power requires removal of decay heat. Decay heat removal requirements are normally satisfied by the steam bypass to the condenser and by continued feedwater flow to the steam generators. Normal feedwater flow to the steam generators is provided by operation of the turbine-cycle feedwater system.

9. The valves and actuation circuits that isolate the auxiliary building normal ventilation system following an accident shall be considered operable for containment integrity if the ventilation system can be manually isolated within 6 minutes following an accident and the inoperable components are repaired within 7 days.

10. The Auxiliary Building Special Ventilation System shall be considered operable only if the Turbine Building roof exhauster fans can be deenergized within 30 minutes of a loss of coolant accident.

B. Containment Internal Pressure

If the internal pressure of the containment vessel exceeds 2 psig whenever containment integrity is required, the condition shall be corrected within eight (8) hours or procedures shall be initiated immediately to establish reactor conditions for which containment integrity is not needed.

C. Containment and Shield Building Air Temperature

If the average temperature of the air in the containment vessel exceeds 44°F above the average temperature of the air in the Shield Building whenever containment integrity is required, the condition shall be corrected within eight hours or procedures shall be initiated immediately to establish reactor conditions for which containment integrity is not needed.

D. Containment Shell Temperature

If the containment shell temperature becomes less than 30°F whenever containment integrity is required, the condition shall be corrected within eight hours or procedures shall be initiated immediately to establish reactor conditions for which containment integrity is not needed.

TABLE TS.4.1-1
(Page 3 of 5)

Channel Description	Check	Calibrate	Functional Response Test	Functional Response Test	Remarks
*19. Radiation Monitoring	*D	R	M	NA	Includes all channels used for leak detection per Spec. 3.1.D. and effluent release monitoring per Spec. 3.9 and 5.5.
20. Boric Acid Make-up Flow Channel	NA	R	NA	NA	
21. Containment Sump Level	NA	R	R	NA	Includes Sumps A, B, and C
22. Accumulator Level and Pressure	S	R	R	NA	
23. Steam Generator Pressure	S	R	M	NA	
24. Turbine First Stage Pressure	S	R	M	NA	
25. Emergency Plan Radiation Instruments	*M	R	M	NA	Includes those named in the emergency procedure (referenced in Spec. 6.5 A.6)
26. Protection Systems Logic Channel Testing	NA	NA	M	NA	Includes auto load sequencers
27. Turbine Overspeed Protection Trip Channel	NA	R	M	NA	
28. Deleted					
29. Deleted					
30. Environmental Monitors	M	NA	NA	NA	Includes those used per Spec. 4.10
31. Seismic Monitors	R	R	NA	NA	Includes those reported in Item 4 of Table TS.6.7-1
32. Coolant Flow-RTD Bypass Flowmeter	S	R	M	NA	
33. CRDM Cooling Shroud Exhaust Air Temperature	S	NA	R	NA	FSAR page 3.2-56
34. Reactor Gap Exhaust Air Temperature	S	NA	R	NA	FSAR page 5.4-2

E. Containment Isolation Valves

During each refueling shutdown, the containment isolation valves, shield building ventilation valves, and the auxiliary building normal ventilation system isolation valves shall be tested for operability by applying a simulated accident signal to them.

F. Post Accident Containment Ventilation System

During each refueling shutdown, the operability of system recirculating fans and valves, including actuation and indication, shall be demonstrated.

G. Containment and Shield Building Air Temperature

Prior to establishing reactor conditions requiring containment integrity, the average air temperature difference between the containment and its associated Shield Building shall be verified to be within acceptable limits.

H. Containment Shell Temperature

Prior to establishing reactor conditions requiring containment integrity, the temperature of the containment vessel wall shall be verified to be within acceptable limits.

Basis

The containment system consists of a steel containment vessel, a concrete shield building, the auxiliary building special ventilation zone (ABSVZ), a shield building ventilation system, and an auxiliary building special ventilation system. In the event of a loss-of-coolant accident, a vacuum in the shield building annulus will cause most leakage from the containment vessel to be mixed in the annulus volume and recirculated through a filter system before its deferred release to the environment through the exhaust fan that maintains vacuum. Some of the leakage goes to the ABSVZ from which it is exhausted through a filter. A small fraction bypasses both filter systems.

The freestanding containment vessel is designed to accommodate the maximum internal pressure that would result from the Design Basis Accident.[1] For initial conditions typical of normal operation, 120°F and 15 psia, an instantaneous double-ended break with minimum safeguards results in a peak pressure of less than 46 psig at 268°F.

The containment will be strength-tested at 51.8 psig and leak-tested at 46.0 psig to meet acceptance specifications.

The safety analysis[2][3] is based on a conservatively chosen reference set of assumptions regarding the sequence of events relating to activity release and attainment and maintenance of vacuum in the shield building annulus and the auxiliary building special ventilation zone, the effectiveness of filtering, and the leak rate of the containment vessel as a function of time. The effects of variation in these assumptions, including that for leak rate, has been investigated thoroughly. A summary of the items of conservatism involved in the reference calculation and the magnitude of their effect upon off-site dose demonstrates the collective effectiveness of conservatism in these assumptions.

blocked off. Until these trays can be installed, to guarantee a representative adsorbent sample, procedures should allow for the removal of a tray containing the oldest batch of adsorbent in each train, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. One sample will be submitted for laboratory analysis and the other held as a backup. If test results are unacceptable, all adsorbent in the train will be replaced. Adsorbent in the tray removed for sampling will be renewed. Any HEPA filters found defective will be replaced. Replacement charcoal adsorber and HEPA filters will be qualified in accordance with the intent of Regulatory Guide 1.52 - Rev. 1 June 1976.

If significant painting, fire, or chemical release occurs such that the HEPA filters or charcoal adsorbers could become contaminated from the fumes, chemicals, or foreign material, the same tests and sample analysis will be performed as required for operational use.

Operation of each train of the system for 10 hours every month will demonstrate operability of the system and remove excessive moisture which may build up on the adsorber.

Periodic checking of the inlet heaters and associated controls for each train will provide assurance that the system has the capability of reducing inlet air humidity so that charcoal adsorber efficiency is enhanced.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

A minimum containment shell temperature of 30°F has been specified to provide assurance that an adequate margin above NDTT exists. Evaluation of data collected during the first fuel cycle of Unit No. 1 shows that this limit can be approached only when the plant is in cold shutdown. Requiring containment shell temperature to be verified to be above 30°F prior to plant heatup from cold shutdown provides assurance that this temperature is above NDTT prior to establishing conditions requiring containment integrity[7].

A maximum temperature differential between the average containment and annulus air temperatures of 44°F has been specified to provide assurance that offsite doses in the event of an accident remain below those calculated in the FSAR. Evaluation of data collected during the first fuel cycle of Unit No. 1 shows that this limit can be approached only when the plant is in cold shutdown. Requiring this temperature differential to be verified to be less than 44°F prior to plant heatup from cold shutdown provides assurance that this parameter is within acceptable limits prior to establishing conditions requiring containment integrity[7].

References

(1) FSAR, Section 5, and Appendix 14-C
(2) FSAR, Section 14, and Appendix G
(3) Safety Evaluation Report, Sections 6.2 and 15.0
(4) FSAR, Section 14
(5) FSAR, Section 14.3.6
(6) Letter to NSP from AEC dated November 29, 1973
(7) NSP Report, "Prairie Island Containment Systems Special Analyses," dated April 9, 1976.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 53 AND 47 TO FACILITY
LICENSE NOS. DPR-42 AND DPR-60

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NOS. 1 AND 2

DOCKET NOS. 50-282 AND 50-306

Introduction

By letter dated August 27, 1976 Northern States Power Company (NSP), the licensee, requested amendments to Facility License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Units Nos. 1 and 2 (PINGP). The requested amendments proposed changes in the Technical Specifications (TSs) in the following areas.

1. TS 2.3.A.2h Power Range Flux Rates
2. TS 3.3.A.1h(3) RHR Valve Position Indication
3. TS 3.3.D.2 Cooling Water Header Operability
4. TS 3.4.A.8 Steam Exclusion Dampers
5. TSs 3.6, 4.4 and Table TS 4.1-1 Containment and Annulus temperatures
6. TS 4.5.B.2 Containment Fan Motor Operability tests
7. TS 5.5.D Environmental Monitoring
8. Table TS 6.7-1 Special tests

Items 1, 2, 3, 6, 7 and the first part of Item 8 have been addressed by Amendment Nos. 21 and 15 issued by our letter dated June 6, 1977. This Safety Evaluation report addresses the remaining open areas that NSP requested. They are item 4, "TS 3.4.A.8 Steam Exclusion Dampers", item 5, "TS 3.6, 4.4 and Table TS 4.1-1 Containment and Annulus Temperatures" and the remaining parts of Item 8 "Table 6.7-1 Special tests" that have not been covered by Amendments 21 and 15.

Discussion and Evaluation

I. Item 4 "TS 3.4.A.8 Steam Exclusion Dampers" (Presently existing under TS 3.4.A.3)

The licensee has proposed a change in the wording of the TS concerned with the steam exclusion dampers. Currently the TS reads as follows:

> "Both isolation dampers in each ventilation duct that penetrates rooms containing equipment required for a high energy line rupture outside of containment shall be operable or at least one damper in each duct shall be closed."

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The licensee proposes to change this TS to read:

> "Both isolation dampers in each ventilation duct that penetrates rooms containing equipment required for a high energy line rupture outside of containment shall be operable or one damper in each duct with an inoperable isolation damper shall be closed."

As currently phrased in the TS, the specification can be interpreted to require closing at least one damper in all ducts if one duct has an inoperable isolation damper. This interpretation is certainly not the intent of the TS. We agree with the licensee that the proposed change does clarify the intent of the existing TS i.e., if a duct contains an inoperable damper, then the operable damper in the affected duct should be closed and in no way has any adverse safety implications. On this basis we have concluded that the proposed change to the TS is acceptable.

II. Item 5 "TSs 3.6, 4.4 and Table TS 4.1-1 Containment and Annulus Temperatures"

The licensee's proposed changes under this item consist of the following:

(A) Specifications 3.6.B, 3.6.C and 3.6.D now require that during power operation limits exist on the internal pressure and average air temperature of the containment vessel and the containment shell temperature. The proposed change would require these limits to be met whenever containment integrity is required instead of only during power operation. We find the licensee's proposed change is more conservative in that the proposed change would envelope other operational modes as well as power operation, since containment integrity is required for all operating modes except when the reactor is in cold shutdown, the reactor is in a refueling mode or the fuel inside containment has never been irradiated. Therefore, we concur with the licensee that if limits of the internal pressure and the average air temperature in the containment vessel and the containment shell temperature are exceeded for all modes of operations including power operation then conditions shall be corrected within eight hours or procedures shall be implemented immediately to establish reactor conditions for which containment is not needed. On this basis we conclude that the proposed change to the TS is acceptable.

(B) The licensee has proposed to delete the wording in TS 3.6.C and 3.6.D which refer to the special test program conducted during the first fuel cycle of Unit No. 1 and associated surveillance requirement in Table TS 4.1-1. During the first fuel cycle of Unit No. 1, a special test program was conducted to determine the average air temperatures from air temperature measurements at a number of locations in the shield building and the containment building. Measurements were made monthly to verify that the containment-annulus differential temperature and the shell temperature were within acceptable limits.

Evaluation of data collected during the first fuel cycle has shown that the measured temperatures in the containment, annulus and containment shell are within the limits established in the TS. The objective

of the special test program had been accomplished, and this requirement no longer exists. Therefore, on this basis the licensee proposed to delete the special test program as stated in TS 3.6.C and 3.6.D, and to delete items 28 and 29 from Table TS-4.1-1 which deals with the inspection of the instrument used in the special program. We have reviewed the information and the data taken during the first fuel cycle provided by the licensee and find that the program objectives have been accomplished. On this basis, we conclude that the change that deletes the special test program as stated in TS 3.6.C and 3.6.D and the deletion of items 28 and 29 from Table TS-4.1-1 are acceptable.

(C) The licensee has proposed additional surveillance requirements to the TS as a result of his review of data from the special test program conducted during the first fuel cycle of Unit No. 1. Under TS 4.4 the licensee proposes to add the following:

Containment and Shield Building Air Temperature

Prior to establishing reactor conditions requiring containment integrity, the average air temperature difference between the containment and its associated Shield Building shall be verified to be within acceptable limits.

Containment Shell Temperature

Prior to establishing reactor conditions requiring containment integrity, the temperature of the containment vessel wall shall be verified to be within acceptable limits.

Evaluation of data collected during the first fuel cycle of Unit No. 1 showed that the proposed limiting temperature differential of 44°F between the average containment and annulus air temperatures and the existing limiting containment shell temperature of 30°F for plant operation can be approached only when the plant is in cold shutdown. Accordingly, the licensee proposed the above additional surveillances to monitor containment air and shell temperatures and annulus air temperature prior to plant heatup from cold shutdown. The objective of these conservative surveillances are to provide assurance that the above cited parameters are within acceptable limits prior to establishing conditions requiring containment integrity.

We have reviewed the information and data taken by the licensee during the first fuel cycle. We concur with the licensee that the proposed limiting temperature differential of 44°F between the average containment and annulus air temperatures and the existing limiting containment shell temperature of 30°F for plant operation can be approached only when the plant is in cold shutdown. Therefore, we conclude that the above proposed surveillance requirement added to the Technical Specifications is acceptable.

III. Item 8 Table TS 6.7-1 Special Tests

The licensee has proposed to delete Table 6.7-1 Special Reports. Currently, this table requires that the licensee submit three special reports to the NRC of studies that were to be conducted during the first fuel cycle of Unit 1. The special reports covering these studies consist of

1. Containment System Special Analysis
2. Fuel Surveillance
3. Leakage Detection Analysis.

The reports covering these studies resulted from requirements established in the Safety Evaluation Report for Prairie Island dated September 28, 1972 at the time of issuance of the Prairie Island Unit No. 1 operating license. The licensee has satisfied these requirements by letters dated March 31, April 9, and May 26, 1976 transmitting the special reports titled: "Coolant Leakage Detection System Performance at the Prairie Island Nuclear Generating Plant", "Prairie Island Containment System Special Analysis" and "Unit 1 - End of Cycle 1 Spent Fuel Inspection".

The report titled "Coolant Leakage Detection System Performance at the Prairie Island Nuclear Generating Plant" shows that the leakage detection systems at PINGP No. 1 and 2 meet the General Design Criterion 30 of 10 CFR 50 Appendix A. We agree with the licensee that instruments and techniques used in the surveillance program provide an effective means of detecting abnormal reactor coolant leakage. Based on our review of the licensee submittal we find that the leakage detection analysis performed by the licensee is acceptable. We thus conclude that Item 4 "Leakage Detection Analysis" in Table TS 6.7-1 can be deleted.

The report titled "Prairie Island Containment System Special Analysis" shows that temperatures measured in the containment, in the annulus and at the containment shell are within the limits established in the TS. Based on our review of this report, we find that the objectives of the special program have been met. We thus conclude that Item 2, Containment System Special Analysis in Table TS 6.7-1 can be deleted.

The report titled "Unit 1 - End of Cycle 1 Spent Fuel Inspection" shows that surveillance of highest power density fuel assemblies during the first refueling cycle of Unit 1 showed no evidence of fuel damage or densification. In addition, fuel surveillance personnel from the fuel vendor Westinghouse Corp. have reviewed the results of the inspection and concur that no safety significant items were observed. We have reviewed the licensee's report and find that his conclusions are acceptable. We thus conclude that the licensee has satisfied the requirements of Item 3 Fuel Surveillance in Table TS 6.7-1 and therefore it can be deleted.

In conclusion, the above Safety Evaluation addresses all of the remaining areas existing in Table TS 6.7-1. Thus, based on these evaluations Table TS 6.7-1 can be deleted.

Environmental Consideration

We have determined that the amendments do 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 amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: DEC 3 0 1981

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 53 to Facility Operating License No. DPR-42, and Amendment No. 47 to Facility Operating License No. DPR-60 issued to Northern States Power Company (the licensee), which revised Technical Specifications for operation of Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of the date of issuance.

The amendments revise the Appendix A Technical Specifications concerned with the operation of the steam exclusion dampers, the applicability of the containment pressure and temperature limits and the deletion of special tests which have been completed.

The application for the amendments 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 appropriate 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 amendments. Prior public notice of the amendments was not required since the amendments do not involve a significant hazards consideration.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 53 to Facility Operating License No. DPR-42, and Amendment No. 47 to Facility Operating License No. DPR-60 issued to Northern States Power Company (the licensee), which revised Technical Specifications for operation of Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of the date of issuance.

The amendments revise the Appendix A Technical Specifications concerned with the operation of the steam exclusion dampers, the applicability of the containment pressure and temperature limits and the deletion of special tests which have been completed.

The application for the amendments 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 appropriate 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 amendments. Prior public notice of the amendments was not required since the amendments do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendments dated August 27, 1976, (2) Amendment Nos. 53 and 47 to License Nos. DPR-42 and DPR-60, and (3) the Commission's 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, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A 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 Licensing.

Dated at Bethesda, Maryland, this 30th day of December, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles M. Trammell

Charles M. Trammell, Acting Chief
Operating Reactors Branch #3
Division of Licensing

DEC 21 1981

RECEIVED DEC 31 1981 US NUCLEAR REGULATORY COMMISSION DOCUMENT MANAGEMENT BR TIDC

Docket No.: STN 50-483

Mr. D. F. Schnell
Vice President - Nuclear
Union Electric Company
Post Office Box 149
St. Louis, Missouri 63166

Dear Mr. Schnell:

Subject: Order Extending Construction Completion Date - Callaway Plant, Unit No. 1

In response to your letter, dated July 22, 1981, the Nuclear Regulatory Commission has issued an Order extending the construction completion date for the Callaway Plant, Unit No. 1 from February 28, 1982 to December 31, 1983. Construction Permit No. CPPR-139 was issued on April 16, 1976.

A copy of the Order, and the staff safety evaluation are enclosed for your information and use. The original of the Order has been forwarded to the Office of Federal Register for publication.

Sincerely,

Original signed by
Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosures:
1. Order
2. Staff Evaluation

cc w/encls.: See next page

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Division of Planning
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P. O. Box 809, State Capitol Building
Jefferson City, Missouri 65101

Honorable Paul H. Murphy
Presiding Judge
Callaway County Court
Fulton, Missouri 65251

Mayor Samuel J. Birk
Route 1, Box 243
Morrison, Missouri 65051

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT NO. 1

DOCKET NO. STN 50-483

ORDER EXTENDING THE CONSTRUCTION COMPLETION DATE

Union Electric Company is the holder of Construction Permit No. CPPR-139 issued on April 16, 1976 for construction of the Callaway Plant, Unit No. 1 on a site in Callaway County, Missouri.

By letter, dated July 22, 1981, Union Electric Company filed a request for the extension of the latest construction completion date for the Callaway Plant from February 28, 1982 to December 31, 1983. The requested extension is required because of changes to Union Electric's construction program delaying scheduled completion for twelve months resulting from, according to Union Electric: (1) a change in the Missouri public utility law which prohibits the inclusion of construction work in progress in the rate base; and (2) projected load growth being less than originally anticipated. The aforementioned state law change, according to Union Electric, affected the amount of funds available for construction of the Callaway Plant. Additional reasons for the extension include successive strikes by the laborers' and the operating engineers' unions which together lasted nine weeks, and design changes to the plant initiated to satisfy NRC requirements resulting from the accident at Three Mile Island. Included in the new requirements are two new buildings, a Technical Support Center and an Emergency Operations Facility, and associated data systems which are to be

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operational prior to fuel loading. The time required for design and construction of these two facilities will extend beyond the earlier fuel load date of April 1981. In addition, the productivity of construction labor has been lower than anticipated and is requiring more manhours than estimated at the initial planning stage due, in part, to increased quality control requirements.

This action involves no significant hazards consideration, good cause has been shown for the delays, and the requested extension is for a reasonable time, the bases for which are set forth in the staff's evaluation of the request for extension.

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

The NRC staff evaluation and the order and the request for extension of the construction permit are available for public inspection at the Commission's Public Document Room located at 1717 H Street, N. W., Washington, D. C. 20555 and at the Fulton City Library, 709 Market Street, Fulton, Missouri and at the Olin Library of Washington University, Skinker and Lindell Boulevards, St. Louis, Missouri 63130.

IT IS HEREBY ORDERED THAT the latest construction completion date for CPPR-139 be extended from February 28, 1982 to December 31, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by
Darrell G. Eisenhut

Date of Issuance: DEC 21 1981

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

**SEE PREVIOUS FOR CONCURRENCE.

OFFICE			DL:LB#1	OELD	DL:LB#1	DL:ADL	DL:DIR
SURNAME	Date of Issuance:		GEdison/tg	RLessy**	BJYoungblood	RLIedesco	DGEisenhut
DATE	retyped:12/14		11/28/81	12/09/81	12/14/81	12/ /81	12/21/81

Libby, Roy Lessy Changed. 12/14/81 MC

Changes on Approval

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT NO. 1

DOCKET NO. STN 50-483

ORDER EXTENDING THE CONSTRUCTION COMPLETION DATE

Union Electric Company is the holder of Construction Permit No. CPPR-139 issued on April 16, 1976 for construction of the Callaway Plant, Unit No. 1 on a site in Callaway County, Missouri.

By letter, dated July 22, 1981, Union Electric Company filed a request for the extension of the latest construction completion date for the Callaway Plant from February 28, 1982 to December 31, 1983. The requested extension is required because of changes to Union Electric's construction program delaying ~~first~~, a ~~delay of 12 months in~~ scheduled completion for twelve months resulting from, according to Union Electric, (1) a change in the Missouri public utility law which prohibits the inclusion of construction work in progress in the rate base, and ~~also because~~ (2) projected load growth being less than originally anticipated. ~~The changes in the Missouri public utility law resulted in a decrease in the amount of cash generated internally and an increase in the amount of allowance for funds used during construction included in the Company's earnings.~~ The aforementioned state law changes ~~change in earnings structure~~ according to Union Electric, affected the amount of funds available for construction of the Callaway Plant. ~~The second~~ Additional reasons for the extension ~~a schedule change~~ is a result of several factors, including successive strikes by the laborers' and the operating engineers' unions which together lasted nine weeks, and design changes to the plant initiated to satisfy NRC requirements resulting from the accident at Three Mile Island. Included in the new requirements are two new buildings, a Technical Support Center and an Emergency Operations Facility, and associated data systems which are to be

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operational prior to fuel loading. The time required for design and construction of these two facilities will extend beyond the earlier fuel load date of April 1981. In addition, the productivity of construction labor has been lower than anticipated and is requiring more manhours than estimated at the initial planning stage; ~~this appears to be caused by~~ due in part to increased ~~regulatory~~ requirements ~~on~~ quality control.

This action involves no significant hazards consideration, good cause has been shown for the delays, and the requested extension is for a reasonable time, the bases for which are set forth in the staff's evaluation of the request for extension.

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

The NRC staff evaluation and the order and the request for extension of the construction permit are available for public inspection at the Commission's Public Document Room located at 1717 H Street, N. W., Washington, D. C. 20555 and at the Fulton City Library, 709 Market Street, Fulton, Missouri and at the Olin Library of Washington University, Skinker and Lindell Boulevards, St. Louis, Missouri 63130.

IT IS HEREBY ORDERED THAT the latest construction completion date for CPPR-139 be extended from February 28, 1982 to December 31, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

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SURNAME	Date of Issuance:		GEdison/lg MRushbrook	RLessy	BJYoungblood	RLTedesco	DGEisenhut
DATE			11/28/81	12/9/81	12/ /81	12/ /81	12/ /81

EVALUATION OF REQUEST FOR EXTENSION OF
CONSTRUCTION PERMIT NO. CPPR-139
FOR THE CALLAWAY PLANT, UNIT 1
DOCKET NO. STN 50-483

A. INTRODUCTION

Union Electric Company is the holder of Construction Permit No. CPPR-139 issued by the Nuclear Regulatory Commission on April 16, 1976 for construction of the Callaway Plant, Unit 1. The plant is presently under construction at a site located in Callaway County, Missouri approximately 30 miles south of Columbia, Missouri. In accordance with Section 185 of the Atomic Energy Act of 1954, as amended, 42 U. S. C. Section 2235, and in accordance with the Commission's regulations, 10 CFR Section 50.55, the Construction Permit states the earliest and latest dates for the completion of construction. By letter dated July 22, 1981, the permittee advised the NRC staff that construction could not be completed by the latest date presently specified, namely February 28, 1982.

The permittee has therefore requested in the July 22, 1981 letter that the Construction Permit be extended to December 31, 1983. In accordance with 10 CFR Section 50.55 (b), the staff, having found good cause shown, and for the reasons stated below, is extending the latest completion date to December 31, 1983.

This evaluation contains the following Sections: Section B, the specification of "good cause" shown by the permittee for an extension, i.e., the specific delays which the permittee has cited in support of its request for an extension; Section C, the staff's independent judgment as to the "reasonable time" necessary from the present forward to compensate for each factor of delay; Section D, a finding as to significant hazards and environmental impact consideration, and Section E, a conclusion and recommendation for an Order.

B. Specified Delays

The permittee stated in the July 21, 1981 letter that the following factors led to the overall delay in the completion of construction of the facility.

1. A change in Missouri public utlility law prohibited the inclusion of construction work in the rate base, which affected the amount of funds available for construction.

2. Between 1973 and 1977 the projected load growth for the company decreased from 5.6% to 4.4% per year.

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3. A number of additions and modifications to design have been made late in the construction schedule to meet recent and changing NRC requirements in the aftermath of TMI. The construction schedule has been adversely affected by the additional work to implement plant modifications required as a result of reassessment made following the TMI accident. This includes the construction of two new buildings, the Technical Support Center and the Emergency Operations Facility.

4. A significant drop in labor productivity has occurred due to increased regulatory requirements in the area of quality control.

5. Successive strikes by two labor unions lasted a total of nine weeks during the spring of 1980.

C. REASONABLE COMPENSATION TIME

We concur with the applicant that the construction permit should be extended an additional 22 months to account for schedule delays and contingencies. We find that the justification for this extension is primarily a combination of less funds available for construction, lower demand growth rate for electric power, and increased regulatory requirements which require changes in construction as well as lowering labor producivity.

We have also examined the construction times for other commercial pressurized water reactors constructed during the same period. We find that the construction time for this facility is comparable to other plants and reasonable. Because of this and the above reasons, we find December 31, 1983 acceptable as the latest date for completion of construction for this facility.

D. SIGNIFICANT HAZARDS AND ENVIRONMENTAL IMPACT CONSIDERATION

We find that because the request is merely for an extension of time to complete work already reviewed and approved, no significant hazard consideration is involved in granting the request; thus, prior notice of this action is not required. It is also concluded that there will be no environmental impact attributable to the proposed action other than that already predicted and described in the Commission's Draft Environmental Statement issued in September 1981. Pursuant to 10 CFR 51.5(d)(4), an environmental impact statement, or negative declaration and environmental impact appraisal, need not be prepared in connection with this action.

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E. CONCLUSION AND RECOMMENDATIONS

For the reasons stated herein, the staff concludes that issuance of an Order extending the latest construction completion date for construction of the Callaway Plant, Unit 1, Construction Permit No. CPPR-139, to December 31, 1983 is reasonable and so ordered.

/s/

G. E. Edison, Project Manager
Licensing Branch No. 1
Division of Licensing

/s/

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Dated: DEC 21 1981

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DATE	12/15/81	12/16/81					

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EVALUATION OF REQUEST FOR EXTENSION OF
CONSTRUCTION PERMIT NO. CPPR-139
FOR THE CALLAWAY PLANT, UNIT 1
DOCKET NO. STN 50-483

A. INTRODUCTION

Union Electric Company is the holder of Construction Permit No. CPPR-139 issued by the Nuclear Regulatory Commission on April 16, 1976 for construction of the Callaway Plant, Unit 1. The plant is presently under construction at a site located in Callaway County, Missouri approximately 30 miles south of Columbia, Missouri. In accordance with Section 185 of the Atomic Energy Act of 1954, as amended, 42 U. S. C. Section 2235, and in accordance with the Commission's regulations, 10 CFR Section 50.55, the Construction Permit states the earliest and latest dates for the completion of construction. By letter dated July 22, 1981, the permittee advised the NRC staff that construction could not be completed by the latest date presently specified, namely February 28, 1982.

The permittee has therefore requested in the July 22, 1981 letter that the Construction Permit be extended to December 31, 1983. In accordance with 10 CFR Section 50.55 (b), the staff, having found good cause shown, and for the reasons stated below, is extending the latest completion date to December 31, 1983.

This evaluation contains the following Sections: Section B, the specification of "good cause" shown by the permittee for an extension, i.e., the specific delays which the permittee has cited in support of its request for an extension; Section C, the staff's independent judgment as to the "reasonable time" necessary from the present forward to compensate for each factor of delay; Section D, a finding as to significant hazards consideration, and Section E, a conclusion and recommendation for an Order.

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3. A number of additions and modifications to design have been made late in the construction schedule to meet recent and changing NRC requirements in the aftermath of TMI. The construction schedule has been adversely affected by the additional work to implement plant modifications required as a result of reassessment made following the TMI accident. This includes the construction of two new buildings, the Technical Support Center and the Emergency Operations Facility.

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D. SIGNIFICANT HAZARDS CONSIDERATION

We find that because the request is merely for an extension of time to complete work already reviewed and approved, no significant hazard consideration is involved in granting the request; thus, prior notice of this action is not required.

E. CONCLUSION AND RECOMMENDATIONS

For the reasons stated herein, the staff concludes that issuance of an Order extending the latest construction completion date for construction of the Callaway Plant, Unit 1, Construction Permit No. CPPR-139, to December 31, 1983 is reasonable and so ordered.

G. E. Edison, Project Manager
Licensing Branch No. 1
Division of Licensing

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

OFFICE ▶			DL:LB#1	DL:LB#1			
SURNAME ▶	Dated:		GEdison/lg	BJYoungblood			
DATE ▶			12/1/81	12/14/81			

DISTRIBUTION FOR ORDER EXTENDING CONSTRUCTION COMPLETION DATE FOR CALLAWAY, UNIT NO. 1

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