Docket Nos. 50-282, 50-306

JAN 9 1976

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

Gentlemen:

The Commission has issued the enclosed Amendment Nos. 9 and 4 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Units 1 and 2, respectively. These amendments consist of changes to the Technical Specifications and are in response to your request dated December 17, 1974.

The amendments incorporate into the Prairie Island Nuclear Generating Plant Technical Specifications changes to the reporting requirements. Changes to your proposal were necessary to meet our requirements. These have been discussed with your staff. The technical specifications are based on Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4.

We request that you use the formats presented in the Appendices to Regulatory Guide 1.16, Revision 4, for reporting operating information and that you report events of the type described under the section "Events of Potential Public Interest". Instructions for using these reporting formats are contained in Regulatory Guide 1.16 (a copy is enclosed for your use), and AEC report OOE-SS-001 titled "Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File" of which you were previously provided a copy. This report is modified by updated instructions dated December 3, 1975, which are enclosed. Copy requirements are summarized in Regulatory Guide 10.1, "Compilation of Reporting Requirements for Persons Subject to NRC Regulations", a copy of which is also enclosed. This Guide will assist you in identifying reports that are required by the Commission's regulations set forth in Title 10 Code of Federal Regulations but are not contained in your technical specifications. Reports that are required by the regulations have not been repeated in your technical specifications.

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

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SURNAME ➤						
DATE ➤						

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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

Sincerely,

Original signed by:
D...

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

Enclosures:
1. Amendment Nos. 9 and 4
2. Regulatory Guide 1.16
3. Updated Instructions
4. Regulatory Guide 10.1
5. Safety Evaluation
6. Federal Register Notice

cc w/enclosures:
See next page

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NORTHERN STATES POWER COMPANY

DOCKET NOS. 50-282 AND 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

AMENDMENT TO FACILITY OPERATING LICENSES

Amendment No. 9
License No. DPR-42

Amendment No. 4
License No. DPR-60

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

   A. The application for amendment by the Northern States Power Company (the licensee) dated December 17, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the licenses are amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License Nos. DPR-42 and DPR-60 is hereby amended to read as follows:

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DATE ➤						

"(2) Technical Specifications

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

3. These license amendments are effective 30 days after their date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
[illegible]

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

Attachment:
Changes to the
Technical Specifications

Date of Issuance: JAN [illegible]

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SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NOS. 9 AND 4

FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

The Technical Specifications contained in Appendix A of the above indicated licenses are revised by replacing the following pages with revised pages bearing the same numbers unless otherwise indicated. Changed areas on the revised pages are reflected by a marginal line.

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1.0 DEFINITIONS

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

A. (Deleted)

B. (Deleted)

C. <u>Containment System Integrity</u>

The containment system includes the steel containment vessel, the concrete shield building, and the auxiliary building special ventilation zone (ABSVZ). Containment system integrity exists when the containment vessel, shield building, and ABSVZ are closed and the following conditions are satisfied.

1. Non-automatic isolation valves are either locked closed or they are under direct administrative control and are capable of being closed within one minute following an accident.
2. Blind flanges required by Table TS.4.4-1 are installed.
3. The equipment hatch is closed and sealed.
4. Automatic containment isolation valves are operable.
5. At least one door in each personnel air lock is closed.
6. The shield building equipment opening is closed.
7. At least one door in each shield building air lock is closed.
8. Single doors in the ABSVZ boundary are either locked closed or they are under direct administrative control and are capable of being closed within six (6) minutes following an accident.
9. At least one door in each ABSVZ air lock type passage is closed.
10. The shield building ventilation system, the auxiliary building special ventilation system, isolation valves or dampers in the auxiliary building normal ventilation system, and the containment vacuum breaker system satisfy the operability requirements of Specification 3.6.

a. One diesel-driven cooling water pump may be inoperable for a period not to exceed seven days (total for both diesel-driven cooling water pumps during any consecutive 30 day period) provided

(1) the operability of the other diesel-driven pump and its associated diesel generator are demonstrated immediately and at least once every 24 hours thereafter,

(2) the engineered safety features associated with that pump are operable; and

(3) both off-site power supply paths from the grid to the 4Kv emergency buses are operable.

B. A reactor shall be placed in the cold shutdown condition if the requirements of Specification TS.3.7A cease to be satisfied. During startup operation or power operation, any of the following conditions of inoperability may exist for the times specified provided startup operation is discontinued until operability is restored.

1. One path from the grid to the plant 4kv bus may be inoperable for a period not to exceed seven days provided (a) both diesel generators and their associated diesel driven cooling water pumps are operable, and (b) all engineered safety features are operable.

2. One diesel generator may be out of service for a period not to exceed seven days (total for both diesel generators during any consecutive 30 day period) provided (a) the operability of the other diesel generator and its associated diesel driven cooling water pump are demonstrated immediately and at least once every 24 hours thereafter, (b) all engineered safety features are operable, and (c) both paths from the grid to the plant 4kv bus are operable.

3. One 4kv, 480V bus, or one battery charger may be out of service on each unit for a period not to exceed 8 hours provided its redundant counterpart is demonstrated to be operable and the safeguards equipment associated with its counterpart are operable, both diesel generators are operable, and both paths from the grid to the 4kv bus are operable.

4. One battery may be out of service for a period not exceeding 8 hours provided that the other battery and both battery chargers remain operable.

Basis

The intent of this specification is to provide assurance that at least one external source and one standby source of electrical power is always available to accomplish safe shutdown and containment isolation and to operate required engineered safeguards equipment following an accident.

Plant auxiliary power is normally supplied by the main auxiliary transformers backed up by three separate external power sources which have multiple off-site network connections; the reserve transformer from the 161kv portion of the plant substation; and the two cooling tower transformers, one of which is supplied from a tertiary winding on the substation auto transformer, and the other directly from the 345kv switchboard. Any one of the three sources is sufficient to supply all necessary accident and post-accident load requirements for one reactor, from any one of four network connections which will be augmented by an additional line by the time the second unit is completed.

Each source separately supplies the safeguards buses in such manner that items of equipment which are redundant to each other are supplied by separate sources and buses.

(1) An investigation to identify the causes for such release rates shall be made.

(2) A program to reduce such release rates to the design objectives shall be defined and initiated.

(3) These actions shall be reported in accordance with Specification 6.7.

b. If the experienced rate of release of radioactive material in liquid wastes when averaged over any three month period is such that these quantities, if continued at the same release rate for a year, would exceed 8 times the design objectives, a program shall be defined and initiated to reduce release rates and a report shall be submitted in accordance with Specification TS 6.7.

c. The rate of release of radioactive materials in liquid waste from the plant shall be controlled such that the instantaneous concentration of radioactivity in liquid waste prior to release to the Mississippi River does not exceed the values listed in 10 CFR Part 20, Appendix B, Table II, Column 2.

d. All radioactive liquid effluents released from the plant shall be reported in accordance with Specification 6.7.

2. Treatment and Monitoring

a. The equipment installed in the liquid radioactive waste system shall be maintained and operated to process, as a minimum, all liquids prior to their discharge when the radioactivity, exclusive of tritium and noble gases, released during any three-month period exceeds 1.25 curies for either unit.

(1) An investigation to identify the causes for such release rates shall be made.

(2) A program to reduce such release rates to the design objectives shall be defined and initiated.

(3) These actions shall be reported in accordance with Specification 6.7.

b. If the experienced rate of release of radioactive material in gaseous wastes, when averaged over any three-month period, is such that these quantities if continued at the same release rate for a year would exceed eight times the design objectives, a program shall be defined and initiated to reduce release rates and a report shall be submitted, in accordance with Specification 6.7.

c. The rate of release of radioactive materials in gaseous waste from the plant (except halogen and particulate radioisotopes with half lives greater than 8 days) shall be controlled such that the maximum release rate averaged over any one-hour period shall not exceed:

$$\sum_i \frac{Q_i}{(MPC)_i} = 1.1 \times 10^5 \ m^3/sec$$

d. Wind speed and direction shall be continuously recorded on site.

e. All radioactive gaseous effluents released from the plant shall be reported in accordance with Specification 6.7.

f. The radioactive gas contained in the gaseous waste system shall not be deliberately discharged to the environment during unfavorable wind conditions. For the purposes of this specification, unfavorable wind conditions are defined as wind from 5° west of north to 45° east of north at 10 miles per hour or less.

2. Treatment and Monitoring

a. During releases of radioactivity gaseous waste from the gaseous waste decay tanks to the auxiliary building exhaust stack, the following conditions shall be met:

(1) The effluent monitor and the stack sampling devices for halogens and particulates shall be operable. The normal response of the effluent monitor shall be verified by comparison with the pre-release sample analysis. The monitor shall be tested prior to any release of radioactive gas from a decay tank and shall be calibrated at refueling intervals. The calibration procedure shall consist of exposing the detector to a referenced calibration source in a controlled reproducible geometry. The source and geometry shall be referenced to the original monitor calibration which provides the applicable calibration curves.

(2) The gaseous waste from the decay tanks shall be filtered through the high efficiency particulate air filters and the charcoal adsorber provided.

b. (1) During normal conditions of plant operation, radioactive gaseous waste from the waste gas system shall be provided a minimum holdup of 60 days except for low radioactivity gaseous waste resulting from purge and fill operations associated with refueling and reactor startup.

(2) Holdup time less than that specified in 2.b.(1) above shall be reported in accordance with Specification 6.7.

(3) The maximum activity to be controlled in one gas decay tank shall not exceed 65,000 curies of Xe-133 equivalent.

the atmospheric steam dump is greater than twice the design objectives, the following actions shall be taken.

(1) An investigation to identify the causes for such releases shall be made.

(2) A program to reduce such releases to the design objectives shall be initiated.

(3) These actions shall be reported in accordance with Specification 6.7.

b. The total quantity of radioactivity released by the atmospheric steam dumps shall be reported in accordance with Specification 6.7.

2. Monitoring

a. The I-131 activity in the steam and water on the secondary side of each steam generator shall be determined as required in Specification Table TS.4.1-2B, Item 8.

b. Each time the atmospheric steam dump is used, the total amount of steam and water released shall be determined and the total amount of I-131 released shall be calculated based on the most recent activity measurements of the secondary steam and water.

c. If the total amount of I-131 released in one steam dump is greater than twice the design objective, the milk from dairy cows grazing in the downwind area shall be analyzed for a period of 5 days following the release. The downwind area shall include the 22-1/2-degree sector of a circle having its center at the plant and a 2-mile radius. The I-131 in the milk shall be determined each day following the dump, using instrumentation with a minimum I-131 detection limit of 1.5 pCi/l.

d. If the amount of I-131 exceeds 10 pCi/l, all milk produced at this location shall not be released for consumption until the I-131 concentration is below the detection limit.

J. Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the calculated power tilt shall be logged every two hours and after a load change greater than 10% of rated power.

Basis

Two criteria have been chosen as a design basis for fuel performance related to fission gas release, pellet temperature, and cladding mechanical properties. First the peak value of linear power density must not exceed 20.0 kW/ft.[1] Second, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.[2]

In addition to the above, the initial steady state conditions for the peak linear power for a loss-of-coolant accident must not exceed the values assumed in the accident evaluation. This interim limit is required in order for the maximum cladding temperature to remain below those limits established by the Interim Policy Statement for LOCA.[3] The interim limit has been modified to compensate for the effects of fuel densification.[4] The effects of fuel densification are such as to increase fuel-stored energy and cause local power spikes. The decrease in fuel temperatures and stored energy as a result of cladding creep down in reference[4] would permit higher power, but have not been allowed in this specification.

To aid in specifying the limits on power distribution the following hot channel factors are defined.

F_Q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux allowing for manufacturing tolerances on fuel pellets and rods. F_Q is the product of F_Q^N and F_Q^E.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances.

F_Q^N is the Nuclear Hot Channel Factor defined as the maximum local neutron flux in the core divided by the average neutron flux in the core.

It is anticipated that the data collected during the first 5 years of operation along with the overall operating experience will provide a suitable basis for determining the inspection program to be implemented for the lifetime of the facility. The results of this study together with the proposed lifetime program will be submitted to the NRC in accordance with Table TS.6.7-1, Item 1.

b. An initial leakage rate test will be performed at a pressure of 23 psig (P_t) and a second test at 46 psig (P_a).

c. The design basis accident leakage rate (L_a) shall be 0.25 weight percent per 24 hours at pressure P_a.

2. Initial and periodic type B (except airlocks) and type C tests of penetrations (Table TS.4.4-1) shall be performed at a pressure of 46 psig (P_a) in accordance with the provisions of Appendix J, Section III.B and Section III.C, and Specification 4.4.A.5. The airlocks shall be tested initially and at six-month intervals at 46 psig by pressurizing the inner volume. In addition, when containment system integrity is required, each airlock shall be tested every 3 days if it is in use by pressurizing the intergasket space to 10 psig.

3. Tests of each shield building and auxiliary building special ventilation zone (ABSVZ) will be in accordance with the provisions of Appendix J, Section IV B and the following conditions:

a. Each shield building shall be functionally tested initially and periodically with the same frequency as the primary containment. Tests will be performed at pressures ≤ -2 inches of water gage. Tests results are acceptable if in-leakage is less than 75% of that in Figure TS.4.4-1.

b. The auxiliary building special ventilation zone shall be functionally tested initially and periodically with the same frequency as the primary containment. Test results are acceptable if one fan will maintain the zone at a negative pressure with an opening in the ABSVZ boundary of at least 10 square feet.

c. For initial tests of the shield building for each unit and the ABSVZ, the negative pressures will be determined from measurements of several locations in each region, and will include an allowance for anticipated plant operating and environmental conditions. For subsequent tests, the pressures may be determined from other instrumentation, determined to be representative of the negative pressure in each region, as established and reported in accordance with Table TS.6.7-1, Item 2.

4. Type A, tests will be considered to be satisfactory if the acceptance criteria delineated in Appendix J, Section III.A are met.

5. Type B and C tests will be considered to be satisfactory if the combined leakage rate of all components subjected to Type B and C tests does not exceed 60% of L_a and if the following conditions are met.

   a. For pipes connected to systems that are in the ABSVZ (Designated ABSVZ in Table TS.4.4-1) the total leakage past isolation valves shall be less than 0.1 weight percent per 24 hours at pressure P_a.

   b. For pipes connected to systems that are exterior to both the shield building and the ABSVZ (designated EXTERIOR in Table TS.4.4-1) the total leakage past isolation valves shall be less than 0.01 weight percent per 24 hours at pressure P_a.

   c. For airlocks, the leakage shall be less than the design leakage reported in "Supplement No. 1 to Unit 1 Reactor Containment Building Integrated Leak Rate Test-June, 1973", dated June 6, 1974.

6. The retest schedules for Type A, B, and C tests will be in accordance with Section III-D of Appendix J. Each shield building shall be retested in accordance with the Type A test schedule for its containment. The auxiliary building special ventilation zone shall be retested in accordance with the Type A test schedule for Unit 1 containment.

7. Type A, B and C tests will be in accordance with Section IV of Appendix J. Inspection and reporting requirements of each shield building test shall be the same as for Type A tests. The auxiliary building special ventilation zone shall have the same inspection and reporting requirements as for the Type A tests of Unit 1.

4.10 ENVIRONMENTAL MONITORING OF RADIOACTIVE EFFLUENT

Applicability

Applies to the periodic monitoring and recording of radioactive effluents found in the plant environs.

Objective

To ascertain the radioactive releases are being maintained as low as practicable and within allowable values.

Specification

A. Procedures and equipment for surveillance and control of radioactive releases to the environment shall be maintained and employed consistent with the design objectives of Specification 3.9.

B. The environmental surveillance program described in Table TS.4.10-1 shall be conducted.

C. A report of the results obtained from the program described in Table TS.4.10-1 shall be submitted in accordance with Specification 6.7.

Basis

The operational program of environmental monitoring described in Section 2.8 of the FSAR will have been in progress for more than two years before initial plant startup. The number and distribution of sampling locations and the various types of measurements, together with the pre-operational background data, will provide verification of the effectiveness of plant effluent control and indication of measurable changes in the activity of the environment.

The frequency and types of off-site sample collection and analysis program presented in Table TS.4.10-1 are more comprehensive than that outlined in the Safety Guide - "Monitoring and reporting of effluents and environmental levels" (Released for comment June 23, 1971). These guidelines include:

6.2 REVIEW AND AUDIT

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

A. Safety Audit Committee (SAC)

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

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

1. Membership

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

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

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

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

2. Meeting Frequency

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

3. Quorum

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

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

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

b. Proposed tests and experiments, and their results, when such tests or experiments may constitute an unreviewed safety question, as defined in Paragraph 50.59(a)(2), Part 50, Title 10, Code of Federal Regulations.

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

d. Proposed changes in Technical Specifications or operating licenses.

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

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

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

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

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

j. Training, qualification and performance of operating staff.

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

l. Security and emergency plans and their implementing procedures.

m. Environmental Monitoring Program and its results.

n. Quality Assurance program and evaluation of its adequacy.

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

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

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

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

5. Authority

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

6. Records

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

7. Procedures

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

a. Responsibility and authority of the group

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

c. Mechanism for scheduling meetings

d. Provision for meeting agenda

6.3 REPORTABLE OCCURRENCE ACTION

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

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

6.4 SAFETY LIMIT VIOLATION

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

Operation shall not be resumed until authorized by the Commission.

(3) Written procedures to assure the adequate fitting of respirators; and the testing of respiratory protective equipment for operability immediately prior to use.

(4) Written procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair and storage.

(5) Written operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.

(6) Bioassays and/or whole body counts of individuals (and other surveys, as appropriate) to evaluate individual exposures and to assess protection actually provided.

e. The licensee uses equipment approved by the U. S. Bureau of Mines/National Institute for Occupational Safety and Health (NIOSH) under appropriate Approval Schedules as set forth in Table TS.6.5-1 below. Equipment not approved under U. S. Bureau of Mines/NIOSH Approval Schedules may be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U. S. Bureau of Mines/NIOSH approved equipment of the same type, as specified in Table TS.6.5-1 below.

f. Unless otherwise authorized by the Commission, the licensee does not assign protection factors in excess of those specified in Table TS.6.5-1 below in selecting and using respiratory protective equipment.

4. These Specifications with respect to the provisions of 20.103 shall be superseded by adoption of proposed changes to 10CFR20, Section 20.103, which would make this Specification unnecessary.

PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION	MODES 1/	PROTECTION FACTORS 2/ PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE 3/	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES/NATIONAL INSTITUTE FOR OCCUPATIONAL SAFETY AND HEALTH APPROVALS
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask 4/ 7/	NP	5	30 CFR Part 11 Subpart K
Facepiece, full 7/	NP	100	30 CFR Part 11 Subpart K
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			30 CFR Part 11 Subpart J
Facepiece, half-mask	CF	100	30 CFR Part 11 Subpart J
Facepiece, full	CF	1000	30 CFR Part 11 Subpart J
Facepiece, full 7/	D	100	30 CFR Part 11 Subpart J
Facepiece, full	PD	1000	30 CFR Part 11 Subpart J
Hood	CF	5/	(6)
Suit	CF	5/	
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full 7/	D	100	30 CFR Part 11 Subpart H
Facepiece, full	PD	1000	30 CFR Part 11 Subpart H
Facepiece, full	R	100	30 CFR Part 11 Subpart H
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above.	30 CFR Part 11 § 11.63(b)

1/, 2/, 3/, 4/, 5/, 6/, 7/, (These notes are on the following pages) of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide. See also footnote 5/, below, concerning supplied-air suits and hoods.

4/ Under chin type only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix B, Table I, Column 1 of 10CFR20.

5/ Appropriate protection factors must be determined taking account of the design of the suit or hood and its permeability to the contaminant under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

6/ No approval schedules currently available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

7/ Only for shaven faces.

NOTE 1: Protection factors for respirators, as may be approved by the U. S. Bureau of Mines and/or National Institute for Occupational Safety and Health (NIOSH) according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this Table. The protection factors in this Table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U. S. Bureau of Mines and/or NIOSH in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table I of 10CFR, Part 20 are based on internal dose to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.

6.6 PLANT OPERATING RECORDS

A. Records Retained for Five Years

Records and logs relative to the following items shall be retained for at least five years:

1. Normal plant operation including such items as power level, period of operation at each level, fuel exposure and shutdowns.

2. Written shift supervisory and reactor logs.

3. Periodic checks, inspections, tests and calibrations of components and systems, as related to these Technical Specifications.

4. Reviews of changes made to procedures or equipment or reviews of tests and experiments.

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

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

7. Records of wind speed and direction.

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

9. Reportable Occurrences.

B. Records Retained for the Life of the Plant

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

1. Liquid and airborne radioactive releases to the environs.

2. Radiation exposures for all plant personnel, visitor and contract personnel.

3. Off-site environmental monitoring surveys.

4. Fuel accountability including new and spent fuel inventories and transfers and fuel assembly histories.

5. Radioactive shipments.

6. Plant radiation and contamination surveys

7. Changes made to the plant as it is described in the Final Safety Analysis Report, reflected in updated, corrected and as-built drawings.

8. Cycling beyond normal limits for those components that have been designed to operate safely for a limited number of cycles beyond such limits.

9. Reactor coolant system in-service inspections.

10. Minutes of meetings of the Safety Audit Committee.

6.7 Reporting Requirements

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

A. Routine Reports

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

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

2. Annual Operating Report.[1/] Routine operating reports covering the operation of the unit during the previous calendar year should be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

   The annual operating reports made by licensees shall provide a comprehensive summary of the operating experience gained during the year, even though some repetition of

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

previously reported information may be involved. References in the annual operating report to previously submitted reports shall be clear.

Each annual operating report shall include:

(a) A narrative summary of operating experience during the report period relating to safe operation of the facility, including safety-related maintenance not covered in item A.2.(b)(5) below.

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

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

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

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

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

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

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

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

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

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

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

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

B. Reportable Occurrences

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

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

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

(a) Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

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

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

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

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

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

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

(e) Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

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

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

(g) Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

(h) Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

(i) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

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

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

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

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

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

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

(d) Abnormal degradation of systems other than those specified in item B.1(c) above designed to contain radioactive material resulting from the fission process.

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

C. Environmental Reports

The following reports relating to environmental activities shall be submitted to the Director of the Regulatory Operations Regional Office. They are included in this Appendix A Technical Specification section until the Appendix B Technical Specifications for the Prairie Island Nuclear Generating Plant have been revised to include the radioactive effluent environmental monitoring program and reporting requirements.

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

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

SPECIAL REPORTS

	Area	Reference (a)	Submittal Date
1.	In-service Inspection, Evaluation and Development	TS.4.2	December 31, 1978
2.	Containment System Special Analyses (b)	TS 3.6.C. TS 4.4.A.3 TS 3.6.D.	3 months before second fuel cycle startup of Unit 1
3.	Fuel Surveillance (c)	SER p. 4-3	3 months after first re-fueling outage of Unit 1
4.	Leakage Detection Analysis (d)	TS 3.1.C.	3 months before second fuel cycle startup of Unit 1

NOTES:

(a) TS refers to these Technical Specifications. SER refers to the NRC Safety Evaluation Report for Prairie Island, dated September 28, 1972.

(b) Analyses and tests will include those necessary to determine for a spectrum of plant operating and environmental conditions:

(1) the locations and limiting settings for the shield building and ABSVZ pressure instruments required by Specification 4.4.3.;

(2) the locations, limiting settings, and surveillance requirements for the primary containment and shield building temperature instruments required by Specification 3.6.C.; and

(3) the locations, limiting settings, and surveillance requirements for the primary containment shell temperature instruments required by Specification 3.6.D.

(c) Surveillance will be made of the fuel assemblies to determine effects caused by phenomena such as fuel densification and shall include visual observation of at least one high power density fuel assembly.

(d) The report will include reactor coolant system leakage data, those evaluations necessary to determine the limiting settings, and surveillance requirements for the means of leakage detection required by Specification 3.1.C.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 9 AND 4 TO FACILITY LICENSE NOS. DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

Introduction

By letter dated December 17, 1974, the Northern States Power Company proposed changes to the Technical Specifications appended to Facility Operating License Nos. DPR-42 and DPR-60, for the Prairie Island Nuclear Generating Plant Units 1 and 2. The proposed changes involve changes to the reporting requirements.

Discussion

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

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

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Evaluation

The new guidance for reporting operating information does not identify any event as an "abnormal occurrence". The proposed reporting requirements also delete reporting of information no longer required and duplication of reported information. The standardization of required reports and desired format for the information will permit more rapid recognition of potential problems. Similar changes are being approved for all power reactor licensees, so all licensees will have the same requirements presented in a uniform manner.

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

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

Environmental Consideration

We have determined that the 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) than an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of the amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the changes 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 changes 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: JAN

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

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 9 and 4 to Facility Operating License Nos. DPR-42 and DPR-60, issued to the Northern States Power Company (the licensee), which revised Technical Specifications for operation of Units 1 and 2 of the Prairie Island Nuclear Generating Plant (the facilities) located in Goodhue County, Minnesota. The amendments are effective 30 days after their date of issuance.

The amendments revise the reporting requirements of the Technical Specifications for the facilities.

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 these amendments is not required since the amendments do not involve a significant hazards consideration.

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 statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of these amendments.

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For further details with respect to this action, see (1) the application for amendments dated December 17, 1974, (2) Amendment Nos. 9 and 4 to License Nos. DPR-42 and DPR-60, respectively, and (3) the Commission's concurrently issued related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at The Environmental Conservation Library of the Minneapolis Public 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 Operating Reactors.

Dated at Bethesda, Maryland, this JAN

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

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

DLZ

OFFICE ➤	OR:ORB #2	OR:ORB #2	OELD	OR:ORB #2		
SURNAME ➤	RMDiggs	BCBuckley	see att. note	DLZiemann		
DATE ➤	1/22/76	1/22/76	1/ /76	1/23/76		

ROUTING AND TRANSMITTAL SLIP

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

REMARKS

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

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

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

11/3/75 No need for OELD concurrence this time on this subject. [illegible]

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

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

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

648—16—81694-1 552-103 GPO 5041-101