December 24, 1 3

Mr. Roger O. Anderson, Director
Nuclear Energy Engineering
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT RE: REVISION OF STATEMENT ON SHIFT LENGTH AND OTHER MISCELLANEOUS CHANGES (TAC NO. M96380)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 104 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications in response to your application dated August 15, 1996, as supplemented March 19 and October 12, 1998.

The amendment revises the Technical Specifications so that either 8 or 12 hour shifts will be considered "normal" and 40 hours will be considered a "nominal" week, changes the wording for surveillances required "once per shift" to "once per 12 hours," clarifies the "once per hour" wording related to fire watch patrols, and makes a number of other clarifications and typographical corrections.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
ORIGINAL SIGNED BY
Carl F. Lyon, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-263
Enclosures: 1. Amendment No. 104 to DPR-22
2. Safety Evaluation
cc w/encl: See next page
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AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-22 - MONTICELLO

Docket File (50-263)
PUBLIC
PD31-1 Reading
E. Adensam (EGA1)
C. Jamerson
C. F. Lyon
OGC
G. Hill (2)
R. Eckenrode
L. Marsh
W. Beckner
D. Desaulniers
ACRS
MKunowski, RIII
SEDB (TLH3)

Mr. Roger O. Anderson, Director
Northern States Power Company

Monticello Nuclear Generating Plant

cc:

J. E. Silberg, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N. W.
Washington DC 20037

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
2807 W. County Road 75
Monticello, Minnesota 55362

Plant Manager
Monticello Nuclear Generating Plant
ATTN: Site Licensing
Northern States Power Company
2807 West County Road 75
Monticello, Minnesota 55362-9637

Robert Nelson, President
Minnesota Environmental Control
Citizens Association (MECCA)
1051 South McKnight Road
St. Paul, Minnesota 55119

Commissioner
Minnesota Pollution Control Agency
520 Lafayette Road
St. Paul, Minnesota 55119

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Commissioner of Health
Minnesota Department of Health
717 Delaware Street, S. E.
Minneapolis, Minnesota 55440

Darla Groshens, Auditor/Treasurer
Wright County Government Center
10 NW Second Street
Buffalo, Minnesota 55313

Kris Sanda, Commissioner
Department of Public Service
121 Seventh Place East
Suite 200
St. Paul, Minnesota 55101-2145

Adonis A. Neblett
Assistant Attorney General
Office of the Attorney General
445 Minnesota Street
Suite 900
St. Paul, Minnesota 55101-2127

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 104
License No. DPR-22

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

    A. The application for amendment by Northern States Power Company (the licensee) dated August 15, 1996, as supplemented March 19 and October 12, 1998, 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-22 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 104 , 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 issuance, with full implementation within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Carl F. Lyon

Carl F. Lyon, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 24, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 104

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

REMOVE	INSERT
i	i
ii	ii[1]
iii	iii[1]
iv	iv
v	v
vi	vi
vii	vii
22	22
31	31
61	61
62	62
63a	63a
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164	164
188	188
190	190
198t	198t
200	200
223	223
227b-227e	227b-227e
229b	229b
229c	229c
229ff	229ff
229i	229i

[1]Font change only for document consistency. No other changes to these pages.

ATTACHMENT TO LICENSE AMENDMENT NO. 104 (Continued)

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

REMOVE	INSERT
229u	229u
232	232
233	233
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Bases 2.2:

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1335 psig as measured in the vessel steam space is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value was derived from the design pressures of the reactor pressure vessel, coolant piping, and recirculation pump casing. The respective design pressures are 1250 psig at 575°F, 1148 psig at 562°F, and 1380 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and the USAS Piping Code Section B31.1 for the reactor coolant system piping. The ASME Code permits pressure transients up to 10 percent over the vessel design pressure (110% x 1250 = 1375 psig) and the USAS Code permits pressure transients up to 20 percent over the piping design pressure (120% x 1148 = 1378 psig).

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig and temperature of 575°F; this is more than a factor of 1.5 below the yield strength of 42,300 psi at this temperature. At the pressure limit of 1375 psig, the general membrane stress increases to 29,400 psi, still safely below the yield strength.

The reactor coolant system piping provides a comparable margin of protection at the established pressure safety limit.

Table 3.1.1 - Continued

e. The high drywell pressure scram functions in the Startup and Run modes when necessary during purging for containment inerting or de-inerting only by closing the manual containment isolation valves. Verification of the bypass condition shall be noted in the control room log.

f. One instrument channel for the functions indicated in the table to allow completion of surveillance testing, provided that:

1. Redundant instrument channels in the same trip system are capable of initiating the automatic function and are demonstrated to be operable either immediately prior or immediately subsequent to applying the bypass.

2. While the bypass is applied, surveillance testing shall proceed on a continuous basis and the remaining instrument channels initiating the same function are tested prior to any other. Upon completion of surveillance testing, the bypass is removed.

Table 4.2.1
Minimum Test and Calibration Frequency for Core Cooling,
Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
ECCS INSTRUMENTATION			
1. Reactor Low-Low Water Level	Once/3 months (Note 5)	Every Operating Cycle - Transmitter Once/3 months -Trip Unit	Once/12 hours
2. Drywell High Pressure	Once/3 months	Once/3 months	None
3. Reactor Low Pressure (Pump Start)	Once/3 months	Once/3 months	None
4. Reactor Low Pressure (Valve Permissive)	Once/3 months	Once/3 months	None
5. Undervoltage Emergency Bus	Refueling Outage	Refueling Outage	None
6. Low Pressure Core Cooling Pumps Discharge Pressure Interlock	Once/3 months	Once/3 months	None
7. Loss of Auxiliary Power	Refueling Outage	Refueling Outage	None
8. Condensate Storage Tank Level	Refueling Outage	Refueling Outage	None
9. Reactor High Water Level	Once/3 months (Note 5)	Every Operating Cycle - Transmitter Every 3 months - Trip Unit	Once/12 hours
ROD BLOCKS			
1. APRM Downscale	Once/3 months (Note 5)	Once/3 months	None
2. APRM Flow Variable	Once/3 months (Note 5)	Once/3 months	None
3. IRM Upscale	Notes (2,5)	Note 2	Note 2
4. IRM Downscale	Notes (2,5)	Note 2	Note 2
5. RBM Upscale	Once/3 months (Note 5)	Once/3 months	None
6. RBM Downscale	Once/3 months (Note 5)	Once/3 months	None
7. SRM Upscale	Notes (2,5)	Note 2	Note 2
8. SRM Detector Not-Full-In Position	Notes (2,9)	Note 2	None
9. Scram Discharge Volume-High Level	Once/3 months	Refueling Outage	None
MAIN STEAM LINE (GROUP 1) ISOLATION			
1. Steam Tunnel High Temperature	Refueling Outage	Refueling Outage	None
2. Steam Line High Flow	Once/3 months	Once/3 Months	Once/12 hours

Table 4.2.1 Continued Minimum Test and Calibration Frequency for Core Cooling, Rod Block and Isolation Instrumentation			
Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
3. Steam Line Low Pressure	Once/3 months	Once/3 months	None
4. Reactor Low Low Water Level	Once/3 months (Note 5)	Every Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/12 hours
CONTAINMENT ISOLATION (GROUPS 2 & 3)			
1. Reactor Low Water Level (Note 10)	-	-	-
2. Drywell High Pressure (Note 10)	-	-	-
HPCI (GROUP 4) ISOLATION			
1. Steam Line High Flow	Once/3 months	Once/3 months	None
2. Steam Line High Temperature	Once/3 months	Once/3 months	None
RCIC (GROUP 5) ISOLATION			
1. Steam Line High Flow	Once/3 months	Once/3 months	None
2. Steam Line High Temperature	Once/3 months	Once/3 months	None
REACTOR BUILDING VENTILATION & STANDBY GAS TREATMENT			
1. Reactor Low Low Water Level	Once/3 months (Note 5)	Every Operating Cycle - Transmitter Once/3 months - Trip Unit	Once/12 hours
2. Drywell High Pressure (Note 10)	-	-	-
3. Radiation Monitors (Plenum)	Once/3 months	Once/3 months	Once/day
4. Radiation Monitors (Refueling Floor)	Once/3 months	Once/3 months	Note 4
RECIRCULATION PUMP TRIP AND ALTERNATE ROD INJECTION			
1. Reactor High Pressure	Once/3 months (Note 5)	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/Day
2. Reactor Low Low Water Level	Once/3 months (Note 5)	Once/Operating Cycle- Transmitter Once/3 Months-Trip Unit	Once 12 hours
SHUTDOWN COOLING SUPPLY ISOLATION			
1. Reactor Pressure Interlock	Once/3 months	Once/3 Months	None

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling,
Rod Block and Isolation Instrumentation

NOTES:

(1) (Deleted)

(2) Calibrate prior to normal shutdown and start-up and thereafter check once per 12 hours and test once per week until no longer required. Calibration of this instrument prior to normal shutdown means adjustment of channel trips so that they correspond, within acceptable range and accuracy, to a simulated signal injected into the instrument (not primary sensor). In addition, IRM gain adjustment will be performed, as necessary, in the APRM/IRM overlap region.

(3) Functional tests, calibrations and sensor checks are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

(4) Whenever fuel handling is in process, a sensor check shall be performed once per 12 hours.

(5) A functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action.

(6) (Deleted)

(7) (Deleted)

(8) Once/shutdown if not tested during previous 3 month period.

(9) Testing of the SRM Not-Full-In rod block is not required if the SRM detectors are secured in the full-in position.

(10) Uses contacts from scram system. Tested and calibrated in accordance with Tables 4.1.1 and 4.1.2.

increases core voiding, a negative reactivity feedback. High pressure sensors initiate the pump trip in the event of an isolation transient. Low level sensors initiate the trip on loss of feedwater (and the resulting MSIV closure). The recirculation pump trip is only required at high reactor power levels, where the safety/relief valves have insufficient capacity to relieve the steam which continues to be generated after reactor isolation in this unlikely postulated event, requiring the trip to be operable only when in the RUN mode is therefore conservative.

The ATWS high reactor pressure and low-low water level logic also initiates the Alternate Rod Injection System. Two solenoid valves are installed in the scram air header upstream of the hydraulic control units. Each of the two trip systems energizes a valve to vent the header and causes rod insertion. This greatly reduces the long term consequences of an ATWS event.

Voltage sensing relays are provided on the safeguards bus to transfer the bus to an alternate source when a loss of voltage condition or a degraded voltage condition is sensed. On loss of voltage this transfer occurs immediately. The transfer on degraded voltage has a time delay to prevent transfer during the starting of large loads. The degraded voltage setpoint corresponds to the minimum acceptable safeguards bus voltage for a steady state LOCA load that maintains adequate voltage at the 480V essential MCCS. An allowance for relay tolerance is included.

Safety/relief valve low-low set logic is provided to prevent any safety/relief valve from opening when there is an elevated water leg in the respective discharge line. A high water leg is formed immediately following valve closure due to the vacuum formed when steam condenses in the line. If the valve reopens before the discharge line vacuum breakers act to return water level to normal, water clearing thrust loads on the discharge line may exceed their design limit. The logic reduces the opening setpoint and increases the blowdown range of three non-APRS valves following a scram. A 15-second interval between subsequent valve actuations is provided assuming one valve fails to

Bases 4.2:

The instrumentation in this section will be functionally tested and calibrated at regularly scheduled intervals. Although this instrumentation is not generally considered to be as important to plant safety as the Reactor Protection System, the same design reliability goals are applied. As discussed in Section 4.1 Bases, monthly or quarterly testing is generally specified unless the testing must be conducted during refueling outages. Quarterly calibration is specified unless the calibration must be conducted during refueling outages. Where applicable, sensor checks are specified on a once/12 hours or once/day basis.

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3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

Any four rod group may contain a control rod which is valved out of service provided the above requirements and Specification 3.3.A are met.

D. Control Rod Accumulators

Control rod accumulators shall be operable in the Startup, Run, or Refuel modes except as provided below.

1. In the Startup or Run Mode, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

   (a) Inoperable accumulator, or

   (b) Directional control valve electrically disarmed while in a non-fully inserted position.

   If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator.

D. Control Rod Accumulators

Once per 12 hours check the status in the control room of the required Operable accumulator pressure and level alarms.

consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit (T.S.2.1.A). This requires the negative reactivity insertion in any local region of the core and in the overall core to be equivalent to at least the scram reactivity curve used in the transient analysis. The required average scram times for three control rods in all two by two arrays and the required average scram times for all control rods are based on inserting this amount of negative reactivity at the specified rate locally and in the overall core. Under these conditions, the CPR safety limit is never exceeded during any transient requiring control rod scram, and therefore MCPR remains above the Safety Limit (T.S.2.1.A).

Bases 3.4/4.4:

A. The design objective of the standby liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron which produces a concentration of boron in the reactor core in less than 125 minutes sufficient to bring the reactor from full power to a 3% delta k subcritical condition considering the hot to cold reactivity swing, xenon poisoning and an additional 25% boron concentration margin to allow for leakage and imperfect mixing.

The time requirement (125 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak.

The ATWS Rule (10 CFR 50.62) requires the addition of a new design requirement to the generic SLC System design basis. Changes to flow rate, solution concentration or boron enrichment to meet the ATWS Rule do not invalidate the original system design basis. Paragraph (c)(4) of 10 CFR 50.62 states that:

> "Each boiling water reactor must have a Standby Liquid Control System (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution" (natural boron enrichment).

The described minimum system parameters (equivalent to 24 gpm, 10.7% concentration and 55 atom percent Boron-10 enrichment) will ensure an equivalent injection capability that meets the ATWS rule requirement.

Boron enrichment concentration, solution temperature, and volume (including check of tank heater and pipe heat tracing system) are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. A reliability analysis indicates that the plant can be operated safely in this manner for ten days. For additional margin, the allowable out of service time has been reduced to seven days.

The only practical time to test the standby liquid control system is during a refueling outage and by initiation from local stations. Components of the system are checked periodically as described above and make a functional test of the entire system on a frequency of less than once each refueling outage unnecessary. A test of explosive charges from one manufacturing batch is made to assure that the replacement charges for the tested system are satisfactory. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

The relief valves in the standby liquid control system protect the system piping and positive displacement pumps which are nominally designed for 1500 psi from overpressure. The pressure relief valves discharge back to the standby liquid control solution tank.

3.0 LIMITING CONDITIONS FOR OPERATION

3. One of the following conditions of inoperability may exist for the period specified:

   a. One Core Spray subsystem may be inoperable for 7 days, or

   b. One RHR pump may be inoperable for 30 days, or

   c. One low pressure pump or valve (Core Spray or RHR) may be inoperable with an ADS valve inoperable for 7 days, or

   d. One of the two LPCI injection paths may be inoperable for 7 days, or

   e. Two RHR pumps may be inoperable for 7 days, or

   f. Both of the LPCI injection paths may be inoperable for 72 hours, or

   g. HPCI may be inoperable for 14 days, provided RCIC is operable, or

   h. One ADS valve may be inoperable for 14 days, or

   i. Two or more ADS valves may be inoperable for 12 hours.

4. If the requirements or conditions of 3.5.A.1, 2 or 3 cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be placed in a condition in which the affected equipment is not required to be operable within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

4. Perform the following tests:

Item	Frequency
Motor Operated Valve Operability	Pursuant to Specification 4.15.B
ADS Valve Operability	Each Operating Cycle

Note: Safety/relief valve operability is verified by cycling the valve and observing a compensating change in turbine bypass or control valve position.

Item	Frequency
ADS Inhibit Switch Operability	Each Operating Cycle
Perform a simulated automatic actuation test (including HPCI transfer to the suppression pool and automatic restart on subsequent low reactor water level)	Each Operating Cycle

5. Perform the following test on the Core Spray Δp Instrumentation:

Check	Once/day
Test	Once/month
Calibrate	Once/3 months

3.0 LIMITING CONDITIONS FOR OPERATION

2. (a) The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour except as specified in 3.6.C.2.b.

Conductivity	5 μmho/cm
Chloride ion	0.1 ppm

(b) For reactor startups the maximum value for conductivity shall not exceed 10 μmho/cm and the maximum value for chloride ion concentration shall not exceed 0.1 ppm for the first 24 hours after placing the reactor in the power operating condition.

3. Except as specified in 3.6.C.2.b above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 lbs. per hour.

Conductivity	5 μmho/cm
Chloride ion	0.5 ppm

4. If Specifications 3.6.C.1 through 3.6.C.3 are not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

2. During startup and at steaming rates below 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for conductivity and chloride content.

3.(a) With steaming rates greater than or equal to 100,000 lbs. per hour, a reactor coolant sample shall be taken at least every 96 hours and when the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes) and analyzed for conductivity and chloride ion content.

(b) When the continuous conductivity monitor is inoperable, during power operation, a reactor coolant sample should be taken once per 12 hours and analyzed for conductivity and chloride ion content.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, reactor coolant system leakage, based on sump monitoring, shall be limited to:

    a. 5 gpm Unidentified Leakage

    b. 2 gpm increase in Unidentified Leakage within any 24 hour period

    c. 20 gpm Identified Leakage

    d. no pressure boundary leakage

2. With reactor coolant system leakage greater than 3.6.D.1.a or 3.6.D.1.c above, reduce the leakage rate to within acceptable limits within four hours or initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.

3. With an increase in Unidentified Leakage in excess of the rate specified in 3.6.D.1.b, identify the source of increased leakage within four hours or initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.

4. If any Pressure Boundary Leakage is detected when the corrective actions outlined in 3.6.D.2 and 3.6.D.3 above are taken, initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, the following surveillance program shall be carried out:

    a. Unidentified and Identified Leakage rates shall be recorded once per 12 hours using primary containment floor and equipment drain sump monitoring equipment.

2. The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

    a. Primary containment atmosphere particulate monitoring systems-performance of a sensor check once per 12 hours, a channel functional test at least monthly and a channel calibration at least once per cycle.

    b. Primary containment sump leakage measurement system-performance of a sensor check once per 12 hours and a channel calibration test at least once per cycle.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

4. Pressure Suppression Chamber-Drywell Vacuum Breakers

   a. When primary containment integrity is required, all eight drywell-suppression chamber vacuum breakers shall be operable and positioned in the closed position as indicated by the position indication system, except during testing and except as specified in 3.7.A.4.b through 3.7.A.4.d below.

   b. Any drywell-suppression chamber vacuum breaker may be nonfully closed as indicated by the position indication and alarm system provided that drywell to suppression chamber differential pressure decay does not exceed that shown on Figure 3.7.1

   c. Up to two drywell-suppression chamber vacuum breakers may be inoperable provided that: (1) the vacuum breakers are determined to be fully closed and at least one position alarm circuit is operable or (2) the vacuum breaker is secured in the closed position or replaced by a blank flange.

   d. Drywell-suppression chamber vacuum breakers may be cycled, one at a time, during containment inerting and deinerting operations to assist in purging air or nitrogen from the suppression chamber vent header.

4.0 SURVEILLANCE REQUIREMENTS

4. Pressure Suppression Chamber-Drywell Vacuum Breakers

   a. Operability and full closure of the drywell-suppression chamber vacuum breakers shall be verified by performance of the following:

      (1) Monthly each operable drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle.

      (2) Once each operating cycle, drywell to suppression chamber leakage shall be demonstrated to be less than that equivalent to a one-inch diameter orifice and each vacuum breaker shall be visually inspected. (Containment access required)

      (3) Once each operating cycle, vacuum breaker position indication and alarm systems shall be calibrated and functionally tested. (Containment access required)

      (4) Once each operating cycle, the vacuum breakers shall be tested to determine that the force required to open each valve from fully closed to fully open does not exceed that equivalent to 0.5 psi acting on the suppression chamber face of the valve disc. (Containment access required.)

Bases 4.7 (Continued):

B. Standby Gas Treatment System, and C. Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain the design negative pressure within the secondary containment provides an adequate test of the reactor building isolation valves and the standby gas treatment system. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system operational capability. Secondary Containment Capability Test data obtained under non-calm conditions is to be extrapolated to calm wind conditions using information provided in "Summary Technical Report to the United States Atomic Energy Commission, Directorate of Licensing, on Secondary Containment Leak Rate Test", submitted by letter dated July 23, 1973, and as described in NSP letter to the NRC dated August 18, 1995, with subject, "Revision 2 to License Amendment Request Dated June 8, 1994, Standby Gas Treatment and Secondary Containment Technical Specifications."

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Standby gas treatment system inplace testing procedures will be established utilizing applicable sections of ANSI N510-1989 standard as a procedural guideline only. If painting, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis should be performed as required for operational use. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52 Revision 2 (March 1978) except testing should be IAW D3803-1989. The charcoal adsorber efficiency test procedures will allow for the removal of a representative sample. The 30°C, 95% relative humidity test per ASTM D 3803-89 is the test method to establish the methyl iodine removal efficiency of adsorbent. The sample will be at least two inches in diameter and a length equal to the thickness of the bed. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system will be replaced. High efficiency particulate filters are installed before and after the charcoal filters to prevent clogging of the carbon adsorbers and to minimize potential release of particulates to the environment. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by inplace testing with DOP as the testing medium. Any HEPA filters found defective will be replaced with filters qualified pursuant to regulatory guide position C.3.d of Regulatory Guide 1.52 Revision 2 (March 1978). Once per operating cycle demonstration of HEPA filter pressure drop, operability of inline heaters at rated power, automatic initiation of each standby gas treatment system circuit, and leakage tests after maintenance or testing which could affect leakage, is necessary to assure system performance capability.

Bases 4.7 (Continued):

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

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

E. Combustible Gas Control System

The Combustible Gas Control System (CGCS) is functionally tested once every six months to ensure that the recombiner trains will be available if required. In addition, calibration and maintenance of essential components is specified once each operating cycle.

TABLE 4.8.4 - RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM (Continued)
(Page 2 of 2)

Notes:

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal. Note (a) of Table 4.8.3 is applicable.

b. Grab samples taken at the discharge of the plant stack and reactor building vent are generally below minimum detectable levels for most nuclides with existing analytical equipment. For this reason, isotopic analysis data, corrected for holdup time, for samples taken at the steam jet air ejector may be used to calculate noble gas ratios.

c. Whenever the steady state radioiodine concentration is greater than 10 percent of the limit of Specification 3.6.C.1, daily sampling of reactor coolant for radioactive iodines of I-131 through I-135 is required. Whenever a change of 25% or more in calculated Dose Equivalent I-131 is detected under these conditions, the iodine and particulate collection devices for all release points shall be removed and analyzed daily until it is shown that a pattern exists which can be used to predict the release rate. Sampling may then revert to weekly. When samples collected for one day are analyzed, the corresponding LLD's may be increased by a factor of 10. Samples shall be analyzed within 48 hours after removal.

d. To be representative of the average quantities and concentrations of radioactive materials in particulate form in gaseous effluents, samples should be collected in proportion to the rate of flow of the effluent streams.

e. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

f. Nuclides which are below the LLD for the analyses shall be reported as "less than" the LLD of the nuclide and should not be reported as being present at the LLD level for that nuclide. The "less than" values shall not be used in the required dose calculations. When unusual circumstances result in LLD's higher than reported, the reasons shall be documented in the semiannual effluent report.

g. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period sampled.

h. H^3 analysis shall not be required prior to purging if the limits of 3.8.B.1 are satisfied for other nuclides. However, the H^3 analysis shall be completed within 24 hours after sampling.

i. In lieu of grab samples, continuous monitoring with bi-weekly analysis using silica-gel samplers may be provided.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

2. Both diesel generators are operable and capable of feeding their designated 4160 volt buses.

3.(a) 4160V Buses #15 and #16 are energized.

(b) 480V Load Centers #103 and #104 are energized.

4. All station 24/48, 125, and 250 volt batteries are charged and in service, and associated battery chargers are operable.

B. When the mode switch is in Run, the availability of electric power shall be as specified in 3.9.A, except as specified in 3.9.B or the reactor shall be placed in the cold shutdown condition within 24 hours.

1. Transmission Lines

From and after the date that incoming power is available from only one line, reactor operation is permissible only during the succeeding seven days unless an additional line is sooner placed in

3.0 LIMITING CONDITIONS FOR OPERATION

3.13 FIRE DETECTION AND PROTECTION SYSTEMS

Applicability:

Applies to instrumentation and plant systems used for fire detection and protection of the nuclear safety-related structures, systems, and components of the plant.

Objective:

To insure that the structures, systems, and components of the plant important to nuclear safety are protected from fire damage.

Specification:

A. Fire Detection Instrumentation

1. Except as specified below, the minimum fire detection instrumentation for each fire detection zone shown in Table 3.13.1 shall be operable whenever equipment in that fire detection zone is required to be operable.

2. If specification 3.13.A.1 cannot be met, within one hour establish a fire watch patrol to inspect the zone(s) with inoperable instruments once per hour (+ 25%). Restore the minimum number of instruments to operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of the inoperability and the plans and schedule for restoring the instruments to operable status.

4.0 SURVEILLANCE REQUIREMENTS

4.13 FIRE DETECTION AND PROTECTION SYSTEMS

Applicability:

Applies to the periodic testing of instrumentation and plant systems used for fire detection and protection of the nuclear safety related structures, systems, and components.

Objective:

To verify the operability of instrumentation and plant systems used for fire detection and protection of nuclear safety related structures, systems, and components.

Specification:

A. Fire Detection Instrumentation

1. Fire detection instrumentation in each of the zones in Table 3.13.1 shall be demonstrated operable every six months by performance of functional tests.

2. Alarm circuitry associated with the fire detector instruments in each of the zones in Table 3.13.1 shall be demonstrated operable every six months.

3.0 LIMITING CONDITIONS FOR OPERATION

F. Halon Systems

1. The cable spreading room Halon system shall be operable with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure.

2. If specification 3.13.F.1 cannot be met, within one hour establish a continuous fire watch with backup fire suppression equipment in the cable spreading room. Restore the system to operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of the inoperability and the plans and schedule for restoring the system to operable status.

G. Penetration Fire Barriers

1. All penetration fire barriers in fire area boundaries shall be operable whenever safe shutdown equipment in that fire area is required to be operable.

2. If Specification 3.13.G.1 cannot be met, a continuous fire watch shall be established on at least one side of the affected penetration(s) within one hour or verify the operability of fire detectors on at least one side of the non-functional fire barrier and establish an hourly (+ 25%) fire watch patrol. Restore the inoperable penetration fire barriers to Operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of the inoperability and the plans and schedule for restoring the barriers to Operable status.

4.0 SURVEILLANCE REQUIREMENTS

F. Halon Systems

1. The cable spreading room Halon system shall be demonstrated operable as follows:

   a. Each valve (manual, power operated, or automatic) in the flow path that is not electrically supervised, locked, sealed or otherwise secured in position, shall be verified to be in its correct position every month.

   b. Verify Halon storage tank weight and pressure every six months.

   c. Perform a system functional test every 18 months which includes verifying the system, including associated ventilation dampers, actuates manually and automatically, upon receipt of a test signal.

   d. Perform an air flow test every 3 years through headers and nozzles to assure no blockage.

   e. Visually examine headers and nozzles every 18 months. An air flow test shall be performed upon evidence of obstructions of any Halon system nozzle.

G. Penetration Fire Barriers

1. A visual inspection of penetration fire barriers in fire area boundaries protecting safe shutdown equipment shall be conducted every 18 months.

2. Following repair or maintenance of a penetration fire barrier a visual inspection of the seal shall be conducted.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
H. Alternate Shutdown System 1. The system controls on the ASDS panel shall be operable whenever that system/component is required to be operable. 2. If system controls required to be operable by Specification 3.13.H.1 are made or found inoperable, restore the inoperable system control to operable within 7 days, or perform one of the following; a. Provide equivalent shutdown capability and within 60 days restore the inoperable system controls to operable; or b. Establish a continuous fire watch in the cable spreading room and the back-panel area of the control room and within 60 days restore the inoperable system controls to operable; or c. Verify the operability of the fire detectors in the cable spreading room and the back-panel area of the control room and establish a hourly fire watch patrol and within 60 days restore the inoperable system controls to operable; or d. Place the reactor in a condition where the systems for which the system controls at the ASDS are inoperable are not required to be operable within 24 hours. 3. The alternate shutdown system panel master transfer switch shall be locked in the normal position except when in use, being tested or being maintained.	H. Alternate Shutdown System 1. Switches on the alternate shutdown system panel shall be functionally tested once per operating cycle. 2. The alternate shutdown system panel master transfer switch shall be verified to alarm in the control room when unlocked once per operating cycle.

TABLE 3.13.1

SAFETY RELATED FIRE DETECTION INSTRUMENTS

Fire Detection Zone	Location	Minimum Heat	Instruments Flame	Operable Smoke
1A	"B" RHR Room			3
1B	"A" RHR Room			3
1C	RCIC Room			3
1E	HPCI Room			2
1F	Reactor Building-Torus Compartment			11
2A	Reactor Bldg. 935′ elev - TIP Drive Area			1
2B	Reactor Bldg. 935′ elev - CRD HCU Area East			10
2C	Reactor Bldg. 935′ elev - CRD HCU Area West			11
2G/2H	Reactor Bldg. 935′ - LPCI Injection Valve Area			1
3B	Reactor Bldg. 962′ elev - SBLC Area			2
3C	Reactor Bldg. 962′ elev - South			5
3D	Reactor Bldg. 962′ elev - RBCCW Pump Area			4
4A	Reactor Bldg. 985′ elev - South			4
4B	Reactor Bldg. 985′ elev - RBCCW Hx Area			5
4D	SBGT System Room			2
5A	Reactor Bldg. 1001′ elev - South			7
5B	Reactor Bldg. 1001′ elev - North			3
5C	Reactor Bldg. - Fuel Pool Cooling Pump Area			1
6	Reactor Building 1027′ elev			5
7A	Battery Room			1
7B	Battery Room			1
7C	Battery Room			1
8	Cable Spreading Room			7

TABLE 3.13.1 (Continued)

SAFETY RELATED FIRE DETECTION INSTRUMENTS

Fire Detection Zone	Location	Minimum Heat	Instruments Flame	Operable Smoke
12A	Turbine Bldg. - 911' - 4.16 KV Switchgear			3
13C	Turbine Bldg. - 911' elev - MCC 133 Area			1
14A	Turbine Bldg. - 931' - 4.16 KV Switchgear			2
15A/15C	#12 DG Room & Day Tank Room		3	
15B/15D	#11 DG Room & Day Tank Room		3	
16	Turbine Bldg. 931' elev - Cable Corridor			3
17	Turbine Bldg. 941' elev - Cable Corridor			3
19A	Turbine Bldg. 931' elev - Water Treatment Area			5
19B	Turbine Bldg. 931' elev - MCC 142-143 Area			1
19C	Turbine Bldg. 931' elev - FW Pipe Chase			1
20	Heating Boiler Room	1		
23A	Intake Structure Pump Room			3
31A	1st Floor - Reactor Building Addition - Division I			3
31B	1st Floor - Reactor Building Addition - Division II			15
32A	2nd Floor - Reactor Building Addition - Division I			6
32B	2nd Floor - Reactor Building Addition - Division II			4
33	3rd Floor - Reactor Building Addition			5

Table 3.14.1

Instrumentation for Accident Monitoring

Function	Total No. of Instrument Channels	Minimum No. of Operable Channels	Required Conditions*
Reactor Vessel Fuel Zone Water Level	2	1	A, B
Safety/Relief Valve Position (One Channel Pressure Switch and One Channel Thermocouple Position Indication per Valve)	2	1	A, C
Drywell Wide Range Pressure	2	1	A, B
Suppression Pool Wide Range Level	2	1	A, B
Suppression Pool Temperature	2	1	A, D
Drywell High Range Radiation	2	1	A, D
Drywell and Suppression Pool Hydrogen and Oxygen Monitor	2	1	A, B
Offgas Stack Wide Range Radiation	2	1	A, D
Reactor Bldg Vent Wide Range Radiation	2	1	A, D

* Required Conditions

A. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the total number of channels, either restore the inoperable channels to operable status within seven days, or prepare and submit a special report to the Commission pursuant to Technical Specification 6.7.D within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status.

Table 3.14.1 (Continued)

Instrumentation for Accident Monitoring

* Required Conditions (continued)

B. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, the minimum number of channels shall be restored to operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours and Cold Shutdown within the following 24 hours.

C. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, the torus temperature shall be monitored once per 12 hours (+25%) to observe any unexplained temperature increase which might be indicative of an open SRV; the minimum number of channels shall be restored to operable status within 30 days or be in at least Hot Shutdown within the next 12 hours and Cold Shutdown within the following 24 hours.

D. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, initiate the preplanned alternate method of monitoring the appropriate parameters in addition to submitting the report required in (A) above.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

B. Inservice Testing

1. Inservice Testing of Quality Group A, B, and C pumps and valves shall be performed in accordance with the requirements for ASME Code Class 1, 2 and 3 pumps and valves, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g) except where relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55(a)(g)(6)(i), or where alternate testing is justified in accordance with Generic Letter 89-04.

2. Nothing in the ASME Boiler and Pressure Vessel code shall be construed to supersede the requirements of any Technical Specification.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3. Deviations are permitted from the required sampling schedule if samples are unobtainable due to hazardous conditions, seasonable unavailability, or to malfunction of automatic sampling equipment. If the latter occurs, every effort shall be made to complete corrective action prior to the end of the next sampling period.

4. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of Table 4.16.3 when averaged over any calendar quarter, submit a special report to the Commission within 30 days from the end of the affected calendar quarter pursuant to Specification 6.7.C.2. When more than one of the radionuclides in Table 4.16.3 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots > 1.0$$

When radionuclides other than those in Table 4.16.3 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.8.A.2, 3.8.B.2, or 3.8.B.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiation Environmental Monitoring Report.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.17 CONTROL ROOM HABITABILITY

Applicability:

Applies to the control room ventilation system equipment necessary to maintain habitability.

Objectives:

To assure the control room is habitable both under normal and accident conditions.

Specification:

A. Control Room Ventilation System

1. Except as specified in 3.17.A.2 and 3.17.A.3 below, both trains of the control room ventilation system shall be operable, whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, or during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel.

2.a With one control room ventilation train inoperable, restore the inoperable train to operable status within 30 days.

2.b If 2.a is not met, then be in hot shutdown within the next 12 hours following the 30 days and in cold shutdown within 24 hours following the 12 hours.

2.c If 2.a is not met during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel then immediately place the operable control room ventilation train in operation or immediately suspend these activities.

4.17 CONTROL ROOM HABITABILITY

Applicability:

Applies to the periodic testing requirements of systems required to maintain control room habitability.

Objectives:

To verify the operability of equipment related to control room habitability.

Specification:

A. Control Room Ventilation System

1. Once per 12 hours check control room temperature.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

A. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for the safe operation and maintenance of the plant. During periods when the Plant Manager is unavailable, this responsibility may be delegated to other qualified supervisory personnel.

The Shift Supervisor (or, a designated individual during periods of absence from the control room and shift supervisor's office) shall be responsible for the control room command function.

B. Offsite and Onsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include positions for activities affecting plant safety.

1. Lines of authority, responsibility and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, function descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements are documented in corporate and plant procedures, or the Updated Safety Analysis Report or the Operational Quality Assurance Plan.

2. The President, NSP Nuclear Generation shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining and providing technical support to the plant to ensure nuclear safety. This position has the responsibility for the Fire Protection Program.

3. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

C. Plant Staff

1. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.1.1.

2. At least one licensed operator shall be in the control room when fuel is in the reactor.

3. At least two licensed operators shall be present in the control room during cold startup, scheduled reactor shutdown, and during recovery from reactor trips.

4. An individual qualified in radiation protection procedures shall be onsite when fuel is in the reactor.

5. All alterations of the reactor core shall be directly supervised by a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

6. A fire brigade of at least five members shall be maintained onsite at all times.* The fire brigade shall not include the three members of the shift organization required for safe shutdown of the reactor from outside the control room.

7. The General Superintendent, Operations shall be formerly licensed as a Senior Reactor Operator or hold a current Senior Reactor Operator License.

8. At least one member of plant management holding a current Senior Reactor Operator License shall be assigned to the plant operations group on a long term basis (approximately two years). This individual will not be assigned to a rotating shift.

D. Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the General Superintendent Radiation Services who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents, and (3) the General Superintendent, Operations who shall meet the requirement of ANSI N18.1-1971 except that NRC license requirements are as specified in Specification 6.1.C.7. The training program shall be under the direction of a designated member of Northern States Power management.

* Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements.

E. A training program for individuals serving in the fire brigade shall be maintained under the direction of a designated member of Northern States Power management. This program shall meet the requirement of Section 27 of the NFPA Code - 1976 with the exception of training scheduling. Fire brigade training shall be scheduled as set forth in the training program.

F. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel. Procedures shall include the following provisions:

1. Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8 or 12-hour day, nominal 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

   a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

   b. Overtime should be limited for all nuclear plant staff personnel so that total work time does not exceed 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, not more than 84 hours in any seven day period, all excluding shift turnover time. Individuals should not be required to work more than 15 consecutive days without two consecutive days off.

   c. A break of at least eight hours including shift turnover time should be allowed between work periods.

   d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

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. <u>Safety Audit Committee (SAC)</u>

The Safety Audit Committee provides the independent review of plant operations from a nuclear safety standpoint. Audits of plant operation are conducted under the cognizance of the SAC.

1. Membership

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

   b. The SAC Chairman shall be an NSP representative, not having line responsibility for operation of the plant, appointed by the President, NSP Nuclear Generation. Other members shall be appointed by the President, NSP Nuclear Generation or by 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. A SAC member may appoint an alternate to serve in his absence, with concurrence of the Chairman. No more than one alternate shall serve on the SAC at any one time. The alternate member shall have voting rights.

2. Qualifications

   a. The SAC members should collectively have the capability required to review activities in the following areas: nuclear power plant operations, nuclear engineering, chemistry and radiochemistry, metallurgy, instrumentation and control, radiological safety, mechanical and electrical engineering, quality assurance practices, and other appropriate fields associated with the unique characteristics of the nuclear power plant.

f. Investigation of all Reportable Events and Events requiring Special Reports to the Commission.

g. Revisions to the Facility Emergency Plan, the Facility Security Plan, and the Fire Protection Program.

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

i. Other nuclear safety matters referred to the SAC by the Operations Committee, plant management or company management.

j. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components.

k. Reports of special inspections and audits conducted in accordance with specification 6.3.

l. Changes to the Offsite Dose Calculation Manual (ODCM).

m. Review of investigative reports of unplanned releases of radioactive material to the environs.

6. Audit - The operation of the nuclear power plant shall be audited formally under the cognizance of the SAC to assure safe facility operation.

   a. Audits of selected aspects of plant operation, as delineated in ANSI N18.7-1976 as modified by the Operational Quality Assurance Plan, shall be performed with a frequency commensurate with their nuclear safety significance and in a manner to assure that an audit of all nuclear safety-related activities is completed within a period of two years. The audits shall be performed in accordance with appropriate written instructions and procedures.

   b. Audits of aspects of plant radioactive effluent treatment and radiological environmental monitoring shall be performed as follows:

      1. Implementation of the Offsite Dose Calculation Manual and quality controls for effluent monitoring at least once every two years.
      2. Implementation of the Process Control Program for solidification of radioactive waste at least once every two years.
      3. The Radiological Environmental Monitoring Program and the results thereof, including quality controls, at least once every year.

   c. Periodic review of the audit program should by performed by the SAC at least twice a year to assure its adequacy.

   d. Written reports of the audits shall be reviewed by the President, NSP Nuclear Generation, by the SAC at a scheduled meeting, and by members of Management having responsibility in the areas audited.

7. Authority

The SAC shall be advisory to the President, NSP Nuclear Generation.

8. Records

Minutes shall by prepared and retained for all scheduled meetings of the Safety Audit Committee. The minutes shall be distributed within one month of the meeting to the President, NSP Nuclear Generation, the Plant Manager, each member of the SAC, and others designated by the Chairman or Vice Chairman. There shall be a formal approval of the minutes.

9. Procedures

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

a. Subjects within the purview of the group.

b. Responsibility and authority of the group.

c. Mechanisms for convening meetings.

d. Provisions of use of specialists or subgroups.

e. Authority to obtain access to the nuclear power plant operating record files and operating personnel when assigned audit functions.

f. Requirements for distribution of reports and minutes prepared by the group to others in the NSP Organization.

B. Operations Committee (OC)

1. Membership

The Operations Committee shall consist of at least six (6) regular members drawn from the key supervisors of the onsite supervisory staff. The Plant Manager shall serve as Chairman of the OC and shall appoint a regular member to act as Vice Chairman in his absence. Alternates to the regular members shall be designated in writing by the Chairman, or Vice Chairman in the Chairman's absence, to serve on a temporary basis. No more than two alternates shall participate as voting members of the Operations Committee at any one time.

2. Meeting Frequency

The Operations Committee will meet on call by the Chairman or as requested by individual members and at least monthly.

3. Quorum

A quorum shall include a majority of the membership, including the Chairman or Vice Chairman.

4. Responsibilities - The following subjects shall by reviewed by the Operations Committee:

a. Proposed tests and experiments and their results.

b. Modifications to plant systems or equipment as described in the Updated Safety Analysis Report and having nuclear safety significance or which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposals which would effect permanent changes to normal and emergency operating procedures and any other proposed changes or procedures that are determined by the Plant Manager to affect nuclear safety.

d. Proposed changes to the Technical Specifications or operating license.

e. All reported or suspected violations of Technical Specifications, operating license requirements, administrative procedures, or operating procedures. Results of investigations, including evaluation and recommendations to prevent recurrence, will be reported, in writing, to the President, NSP Nuclear Generation and to the Chairman of the Safety Audit Committee.

f. Investigation of all Reportable Events and Events requiring Special Reports to the Commission.

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

h. All procedures required by these Technical Specifications, including implementing procedures of the Emergency Plan and the Security Plan (except as exempted in Section 6.5.F), shall be reviewed with a frequency commensurate with their safety significance but at an interval of not more than two years.

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

j. Review of investigative reports of unplanned releases of radioactive material to the environs.

k. All changes to the Process Control Program (PCP) and the Offsite Dose Calculation Manual (ODCM).

5. Authority

The OC Shall be advisory to the Plant Manager. In the event of disagreement between the recommendations of the OC and the Plant Manager, the course determined by the Plant Manager to be the more conservative will be followed. A written summary of the disagreement will be sent to the President, NSP Nuclear Generation 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 President, NSP Nuclear Generation and others designated by OC Chairman or Vice Chairman.

7. Procedures

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

a. Responsibility and authority of the group.

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

c. Mechanism for scheduling meetings

d. Meeting agenda

e. Use of subcommittee

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

g. Distribution of minutes

6.3 Special Inspections and Audits

A. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite Northern States Power Company personnel or an outside fire protection consultant.

B. An inspection and audit by an outside qualified fire protection consultant shall be performed at intervals no greater than three years.

6.4 Action to be Taken if a Safety Limit is Exceeded

If a Safety Limit is exceeded, the reactor shall be shut down immediately. An immediate report shall be made to the Commission and to the President, NSP Nuclear Generation or his designated alternate in his absence. A complete analysis of the circumstances leading up to and resulting from the situation, together with recommendations by the Operations Committee, shall also by prepared. This report shall be submitted to the Commission, to the President, NSP Nuclear Generation and the Chairman of the Safety Audit Committee within 14 days of the occurrence.

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

B. Radiological

1.a. A Radiation Protection Program, consistent with the requirements of 10 CFR 20, shall be developed and followed. The Radiation Protection Program shall consist of the following:

(1) A Radiation Protection Plan, which shall be a complete definition of radiation protection policy and program

(2) Procedures which implement the requirements of the Radiation Protection Plan

The Radiation Protection Plan and implementing procedures, with the exception of those non-safety related procedures governing work activities exclusively applicable to or performed by health physics personnel, shall be reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager. Health physics procedures not reviewed by the Operations Committee shall be reviewed and approved by the General Superintendent Radiation Services.

b. In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit.[1] Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

(1) A radiation monitoring device that continuously indicates the radiation dose rate in the area.

(2) A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been determined and personnel have been made knowledgeable of them.

(3) An individual qualified in radiation protection procedures with a radiation dose rate monitoring device. This individual is responsible for providing positive radiation protection control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation protection procedures or the applicable Radiation Work Permit.

c. The above procedure shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition doors shall be locked or attended, to prevent unauthorized entry into these areas and the keys or key devices for locked doors shall be maintained under the administrative control of the Plant Manager.

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1. Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the Radiation Work Permit issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas. This footnote applies only to high radiation areas of 1000 mrem/hr or less.

E. <u>Offsite Dose Calculation Manual (ODCM)</u>

The ODCM shall be approved by the Commission prior to initial implementation. Changes to the ODCM shall satisfy the following requirements:

1. Shall by submitted to the Commission with the Semi-Annual Radioactive Effluent release report for the period in which the change(s) were made effective. This submittal shall contain:

   a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with a revision date, together with appropriate analyses or evaluations justifying the change(s).

   b. a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. documentation of the fact that the change has been reviewed and found acceptable by the Operations Committee.

2. Shall become effective upon review and acceptance by the Operations Committee.

F. <u>Security</u>

Procedures shall be developed to implement the requirements of the Security Plan and the Security Contingency Plan. These implementing procedures, with the exception of those non-safety related procedures governing work activities exclusively applicable to or performed by security personnel, shall be reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager. Security procedures not reviewed by the Operations Committee shall be reviewed and approved by the Superintendent, Security.

G. <u>Temporary Changes to Procedures</u>

Temporary changes to those procedures which are required to be reviewed by the Operations Committee described in A, B, C, D, E and F above, which do not change the intent of the original procedures may be made with the concurrence of two members of the unit management staff, at least one of whom holds a Senior Operator License. Such changes should be documented, reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager within one month. Temporary changes to health physics and security procedures not reviewed by the Operations Committee shall be reviewed by the General Superintendent, Radiation Services for health physics procedures and the Superintendent, Security for security procedures.

B. Records Retained for Plant Life (continued)

11. Records of the service lives of all safety-related snubbers, including the date at which the service life commences and associated installation and maintenance records.

B. Reportable Events

The following actions shall be taken for Reportable Events:

a. The Commission shall be notified by a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50 and,

b. Each Reportable Event shall be reviewed by the Operations Committee and the results of this review shall be submitted to the Safety Audit Committee and the President, NSP Nuclear Generation.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated August 15, 1996, as supplemented March 19 and October 12, 1998, the Northern States Power Company (NSP or the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The proposed amendment revises the TS so that either 8 or 12 hour shifts will be considered "normal" and 40 hours will be considered a "nominal" week, changes the wording for surveillances required "once per shift" to "once per 12 hours," clarifies the "once per hour" wording related to fire watch patrols, and makes a number of other clarifications and typographical corrections.

The October 12, 1998, submittal provided additional clarifications and provided new TS pages. This information was within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards considerations determination.

2.0 EVALUATION

2.1 Change in Shift Length

The licensee proposes to change TS 6.1.F.1 for the normal working hours of unit staff who perform safety-related functions from "a normal 8-hour day, 40-hour week" to "a normal 8 or 12-hour day, nominal 40-hour week." The change, in effect, would enable the licensee to establish unit staff work schedules that average 40 hours per week using 8- or 12-hour shifts.

The NRC staff has previously approved the use of 12-hour shifts at other U.S. commercial nuclear power plants and has found no evidence of adverse effects on plant safety resulting from the use of such shifts. All other provisions of the specification concerning consecutive hours of work, overtime, and breaks remain unchanged. Thus, the proposed change does not alter the intent of the existing specification with respect to the number of hours that should normally be worked per week, and TS 6.1.F.1 will continue to provide adequate assurance that routine heavy use of overtime will not be necessary to provide adequate shift coverage.

The NRC staff finds the proposed change to be consistent with Section 5.2.2.e of NUREG-1433, Rev.1, "Standard Technical Specifications, General Electric Plants, BWR/4," (STS). NUREG-1433, Rev.1 provides licensees the option of specifying either an 8-hour or a

12-hour shift. Although the staff notes that the licensee has alternatively proposed a specification that allows both 8- and 12- hour shifts, the staff finds no safety concern with the proposed specification which affords the licensee the flexibility to establish shift lengths of either 8 or 12 hours. Accordingly, the staff concludes that the proposed change to TS 6.1.F.1 is acceptable.

2.2 Hourly Fire Watch Patrol Interval

With regard to fire detection instrumentation, TS 3.13.A.2 requires that, "If specification 3.13.A.1 cannot be met, within one hour establish a fire watch patrol to inspect the zone(s) with inoperable instruments at least once per hour." The licensee proposes to change "at least once per hour" to read "once per hour (+25%)." Deleting the words "at least" and adding "(+25%)" will require patrols on an hourly basis while providing flexibility to complete patrols within a 15-minute window. Similarly, TS 3.13.G.2 requires hourly fire watch patrols when penetration fire barriers are inoperable. The licensee proposes to change the frequency from "hourly" to "hourly (+25%)." The changes to clarify the frequency of the hourly fire watch patrols by defining the intervals as 1 hour with a margin of 15 minutes are consistent with other TS surveillance frequencies that allow margins of 25%. The changes are acceptable to the staff.

2.3 Surveillance Intervals

Changing the shift length from 8 to 12 hours affects the frequency of surveillance tests required once per shift. There is not a clear definition of the term "shift" in the current TS or Updated Safety Analysis Report (USAR). Since the number of shifts per day has been reduced from three to two, the actual number of times the surveillances are performed has also been reduced. The concern with frequencies or time periods that are not clearly defined was noted and addressed during development of NUREG-1433, Rev.1. To avoid confusion in the STS, each completion time and frequency is specifically noted with incremental time periods. The licensee proposes to resolve the issue where applicable by eliminating reference to "once per shift" and adopting the actual value "once per 12 hours" as the required surveillance frequency.

The term "sensor check" as defined in TS 1.0 and used throughout the Monticello TS is equivalent to the STS term "channel check." According to STS Bases Surveillance Requirement (SR) SR 3.3.5.1.1, "Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred...The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO." Similar justification is used throughout STS Bases Section B 3.3, "Instrumentation."

On pages 61 and 62, TS Table 4.2.1, "Minimum Test and Calibration Frequency for Core Cooling Rod Block and Isolation Instrumentation," the licensee proposes to change the sensor check frequency for the following sensors from "once/shift" to "once/12 hours":

ECCS [Emergency Core Cooling System] Instrumentation reactor low-low water level and reactor high water level,
Main Steam Line (Group I) Isolation steam line high flow and reactor low low water level,
Reactor Building Ventilation & Standby Gas Treatment reactor low low water level, and
Recirculation Pump Trip and Alternate Rod Injection reactor low low water level.

The changes are consistent with the channel check frequencies of STS 3.3.5.1, "ECCS Instrumentation," STS 3.3.6.1, "Primary Containment Isolation Instrumentation," and STS 3.3.4.2, "Anticipated Transient without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation," and are acceptable to the staff.

On page 63a, Table 4.2.1 Note 2, the licensee proposes to change the phrase "thereafter check once per shift" to "thereafter check once per 12 hours." The note applies to certain Table 4.2.1 Rod Block instrument channel sensor checks during startup and shutdown. The change is consistent with the channel check frequency of STS 3.3.2.1, "Control Rod Block Instrumentation," and is acceptable to the staff.

On page 63a, Table 4.2.1 Note 4, the licensee proposes to change the frequency of the sensor check from "once per shift" to "once per 12 hours." The note applies during fuel handling operations to the refueling floor radiation monitors for Reactor Building Ventilation & Standby Gas Treatment. The change is consistent with the channel check frequency of STS 3.3.6.1, "Primary Containment Isolation Instrumentation," and is acceptable to the staff.

On page 72, in 4.2 Bases, the licensee proposes to change the last sentence to read, "Where applicable, sensor checks are specified on a once/12 hours or once/day basis," instead of "once/shift or one[sic]/day basis." The change is consistent with the changes requested for Table 4.2.1 above and are acceptable to the staff.

On page 82, TS 4.3.D, the licensee proposes to change the frequency to check the status in the control room of the required Operable control rod accumulator pressure and level alarms from "once a shift" to "once per 12 hours." The change is consistent with the similar STS 3.3.1.1, "Reactor Protection System Instrumentation," scram discharge volume channel check frequency and is acceptable to the staff.

On page 125, TS 4.6.C.3(b), the licensee proposes to change the reactor coolant sample frequency during power operation when the continuous conductivity monitor is inoperable, from "at least once per shift" to "once per 12 hours." According to the bases for Section 4.6, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. In accordance with TS 4.6.C.3(a), the routine reactor coolant sample frequency is at least once/96 hours, even when the continuous conductivity monitor indicates abnormal conductivity. Changing the frequency from once/shift to once/12 hours when the continuous conductivity monitor is inoperable will not affect the ability to detect long-term changes in the chloride ion content and is acceptable to the staff.

On page 126, TS 4.6.D.1.a requires that "Unidentified and Identified Leakage rates shall be recorded once per shift not to exceed 12 hours." The licensee proposes to change the frequency to "once per 12 hours." The change does not affect the periodicity already allowed by the TS and is acceptable to the staff.

On page 126, TS 4.6.D.2.a requires a sensor check of the primary containment atmosphere particulate monitoring systems "at least once per 12 hours." Also, TS 4.6.D.2.b requires a sensor check of the primary containment sump leakage measurement system "at least once per shift not to exceed 12 hours." The licensee proposes to change the frequencies to "once per 12 hours" to be consistent. The change does not affect the periodicity already allowed by the TS. The change is consistent with the channel check frequency of STS 3.4.6, "Reactor Coolant System Leakage Detection Instrumentation," and is acceptable to the staff.

On page 229c, Table 3.14.1, "Instrumentation for Accident Monitoring," Required Condition C requires "When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, the torus temperature shall be monitored at least once per shift to observe any unexplained temperature increase which might be indicative of an open SRV [safety relief valve]." The licensee proposes to change the frequency to "once per 12 hours (+25%)." The change to specify 12 hours does not change the capability of the licensee to detect an open SRV, and the change to clarify the frequency by adding a 25% margin is consistent with other TS surveillance frequencies that allow 25% margins. The changes are acceptable to the staff.

On page 229u, TS 4.17.A.1 requires "At least once per shift, check control room temperature." The check is to assure control room ventilation system operability. The licensee proposes to change the frequency to "once per 12 hours." The frequency is consistent with the channel check frequencies of STS 3.3.7.1, "Main Control Room Environmental Control Instrumentation." Also, since the control room is continuously manned, operators would note any abnormal temperatures. The change is acceptable to the staff.

2.4 <u>Typographical Errors</u>

The licensee proposes to correct the following typographical errors:

On pages 61 and 62, in the title block of Table 4.2.1, add a comma after the words "Core Cooling."

On page 72, in TS 4.2 Bases, correct "one/day" to "once/day."

On page 198t, in Note i of Table 4.8.4 - Radioactive Gaseous Waste Sampling and Analysis Program, change "silica-jel" to "silica-gel." In Note e, change "radio-nuclides" to "radionuclides."

On Bases page 198y, the change "form" to "from" has been made by a previous Bases revision.

On page 223, three errors should be corrected. In TS 3.13 section Applicability, add a period at the end of the sentence. In TS 4.13 section Applicability, capitalize the first word of the sentence (Applies). In TS 4.13.A.2, correct the spelling of "circuity" to "circuitry."

On page 227b, in TS 3.13.F.1, delete the phrase "a. Each valve (manual, power operated, or." The phrase was inadvertently copied from adjoining section TS 4.13.F.1.a by Amendment No. 61 to Facility Operating License No. DPR-22, dated March 29, 1989. Also, in TS 3.13.G.2 correct the word "if" in the phrase "verify the operability if fire detectors" to "of," so the phrase reads "verify the operability of fire detectors."

On page 227e, in Table 3.13.1, zone 13C, change "evel" to "elev."

On page 229u, in TS 3.17.A.2.c, correct the spelling of "activies" to "activities."

On page 232, in TS 6.1.B.1, change "responsiblities" to "responsibilities." In TS 6.1.B.2, change "responsiblity" to "responsibility."

On page 233 in TS 6.1.D, change "Nothern" to "Northern," to read "Northern States Power management." Also, correct "Superindent" to "Superintendent." In TS 6.1.C.4 and 6.1.C.6, change "on site" to "onsite."

On page 244a in TS 6.5.B.1.b (2), change "present" to "preset," to read "alarms when a preset integrated dose is received."

On page 252 spelling of "mortality" had been corrected previously. No change necessary.

The corrections are editorial and are acceptable to the staff.

2.5 Clarifications and Corrections

The licensee proposes to make the following minor clarifications and corrections:

On page i, the licensee proposes to correct page number 41 to 42 and page number 25a to 25b in the Table of Contents. Current page 41 was changed to page 25b under the licensee's TS Bases Control Program on April 30, 1998. As a result, the notation "next page is 42" was added to the bottom of page 40. The changes are editorial and are acceptable to the staff.

On page iv, the licensee proposes to correct page number 229 to 228b. The staff found that this error had already been corrected by letter dated July 14, 1989, from J. Stefano (NRC) to T. Parker (NSP); therefore, no change is required. The licensee proposes to correct page number 246 to 246c. The licensee proposes to change 3.16 and 4.16.A from "Sampling and Analysis" to "Sample Collection & Analysis," and 3.16 and 4.16.C from "Interlaboratory Comparison" to "Interlaboratory Comparison Program." The licensee proposes to change 3.17 and 4.17.A from "Ventilation System" to "Control Room Ventilation System," and 3.17 and 4.17.B from "Emergency Ventilation System" to "Control Room Emergency Ventilation System." In its October 12, 1998, submittal, the licensee withdrew its requests to delete headings for 3.14 and 4.14 Bases and to add a 6.8 Environmental Qualification heading. The changes are editorial and are acceptable to the staff.

On page v, the licensee proposes to delete reference to Figure 3.5.1. The figure was deleted by Amendment No. 97 to Facility Operating License No. DPR-22, dated September 17, 1996, but the Table of Contents was not updated. The change is editorial and is acceptable to the staff.

On page vi, the licensee proposes to correct page number 57 to 56. The change is editorial and is acceptable to the staff. The licensee proposes to change Table No. 4.6-1 to 4.6.1. The staff notes that the correct table number on page 132a is 4.6-1; therefore, no change is required.

On page vii, the licensee proposes to correct page number 227c to 227d. The licensee proposes to add "(LLD)" to the title of Table 4.16.2, to read, "REMP - Maximum Values for the Lower Limits of Detection (LLD)." The changes are editorial and are acceptable to the staff.

On page 22, the licensee proposes to correct the recirculation pump casing design pressure in Bases Section 2.2 from 1400 psig to 1380 psig. The error was discovered by the licensee during the Design Basis Document review program. Changing the Bases number will make it consistent with the original design value and is acceptable to the staff.

On page 31, Table 3.1.1, "Reactor Protection System (Scram) Instrument Requirements," Allowable Bypass Condition e, the licensee proposes to revise the wording of the first sentence to read "The high drywell pressure scram functions in the Startup and Run modes when necessary during purging for containment inerting and de-inerting only by closing the manual containment isolation valves." The original wording was hard to understand. Rewording the sentence does not change its intent or meaning and is acceptable to the staff.

On page 69, Bases Section 3.2, the licensee proposes to change the sentence, "The degraded voltage setpoint corresponds to the minimum acceptable safeguards bus voltage for starting and running loads during a loss of coolant accident," to "The degraded voltage setpoint corresponds to the minimum acceptable safeguards bus voltage for a steady state LOCA [loss-of-coolant accident] load that maintains adequate voltage at the 480 V essential MCCs [motor control centers]." During the Design Basis Document review program, the licensee determined that the degraded bus voltage setpoint basis was uniquely determined for Monticello using the steady-state LOCA loading condition, not starting loads, as currently stated in this Bases section. This setpoint basis was previously reviewed and approved by letter and enclosed safety evaluation dated March 20, 1985, from V. Rooney (NRC) to D. Musolf (NSP); therefore, the change is acceptable to the staff.

On page 89, Bases 3.3 and 4.3 Section C incorrectly states that the turbine stop valve closure with bypass valve failure is the limiting transient for Monticello. This is no longer correct, as documented in USAR Appendix 14A, Section 5.1.1. The licensee proposes to correct the Bases to read, "Under these conditions, the CPR [critical power ratio] safety limit is never exceeded during any transient requiring control rod scram, and therefore MCPR [minimum CPR] remains above the Safety Limit (T.S.2.1.A)." The change is acceptable to the staff.

On page 99, the licensee proposes to revise the first paragraph of Basis 3.4 and 4.4 Section A to be consistent with USAR Section 6.6.1.1, the Operations Manual, and Design Bases Documents Modification 87M022 and a Bases calculation for the Standby Liquid Control System. The water in the shutdown cooling system was included in the original volume calculation; therefore, there is no dilution when the shutdown cooling system is added to the standby liquid control system. The last sentence of the paragraph is corrected to read, "and an additional 25% boron concentration margin to allow for leakage and imperfect mixing." The change is acceptable to the staff.

On page 102, the licensee proposes to revise the note in TS 4.5.A.4 to read, "and observing a compensating change in turbine bypass or control valve position." The change will correctly reflect the system response for an SRV actuation in the case where the turbine is on line and the bypass valve would not actuate because the transient would be handled by the functioning control valve. The change reflects actual system response and is acceptable to the staff.

On page 164, the licensee proposes to revise TS 3.7.A.4.a to read, "When primary containment integrity is required." The addition of the word "integrity" clarifies that it is the integrity of the primary containment that is important in these conditions. The change is editorial and is acceptable to the staff.

On page 188, Bases Section 4.7.B and C, the licensee proposes to revise the sentence "Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52 Revision 2 (March 1978)" to add "except testing should be IAW ASTM D3803-1989." The change adds the specification number that is used for adsorbent testing. American Society for Testing and Materials (ASTM) D3803-1989 is endorsed by the staff (see NRC Information Notice 87-32, "Deficiencies in the Testing of Nuclear-Grade Activated Charcoal," and SECY-97-299 dated December 24, 1997). The clarification does not reduce any requirement and is acceptable to the staff.

On page 190, Bases Section 4.7.D, the licensee proposes to correct the date of the report submitted to the AEC from "1983" to "1973." The correct date was verified on the original document; therefore, the change is editorial and is acceptable to the staff.

On page 200, the licensee proposes to revise TS 3.9.B to read "the availability of electric power shall be as specified in 3.9.A, except as specified in 3.9.B, or the reactor shall be placed in the cold shutdown condition within 24 hours." The current wording is "except as specified in 3.9.B.1, 3.9.B.2, 3.9.B.3 and 3.9.B.4 or the reactor shall be placed in the cold shutdown condition within 24 hours." The revision clarifies that 3.9.B.5 (availability requirement for the 24V batteries) should be included with the other 3.9.B sections. The revision does not change any existing requirement and is acceptable to the staff.

On page 227c, the licensee proposes to revise TS 3.13.H.1 to read, "The system controls on the ASDS [alternate shutdown system] panel shall be operable whenever that system/component is required to be operable" instead of "The system controls on the ASDS panel shall be operable whenever that systems [sic] controls are required to be operable from the control room." The change clarifies the operability requirement of the system. The revision does not change any existing requirement and is acceptable to the staff.

On page 227d, Table 3.13.1, "Safety Related Fire Detection Instruments," the licensee proposes to correct the fire zone listed as 2E to 2G/2H, and on page 227e, Table 3.13.1, correct the fire zone listed as 15A to 15A/C and the fire zone listed as 15B to 15B/D. The actual zones were verified by the licensee. In addition, change the column heading, "Fire Zone," to read, "Fire Detection Zone." The column heading is being changed because of slight differences between layout of detectors in physical and functional zones. Table 3.13.1 is based on locations of the fire detectors in their zones. Adding the word "detection" is for clarification. The corrections and clarification do not change any existing requirements and are acceptable to the staff.

On page 229b, Table 3.14.1, "Instrumentation for Accident Monitoring," the licensee proposes to correct Required Condition A to read, "submit a special report to the Commission pursuant to Technical Specification 6.7.D," instead of, "pursuant to Technical Specification 6.7.B.2." This change corrects a branching error in Required Condition A. TS 6.7.D is a requirement for special reports. There is no TS 6.7.B.2. The correction is acceptable to the staff.

On page 229ff, the licensee proposes to delete TS 4.15.A.2. The two welds at Monticello that were identified as susceptable to intergranular stress corrosion cracking (IGSCC) in accordance with NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," were replaced in 1989, as documented by Licensee Event Report 89-23, dated October 10, 1989, and letter and enclosed safety evaluation from W. Long (NRC) to T. Parker (NSP), dated December 7, 1989. TS 4.15.A.2 is no longer applicable and should have been deleted after the welds were replaced. The change is acceptable to the staff.

On page 229i, the licensee proposes to correct TS 4.16.A.4 to read, "submit a special report to the Commission within 30 days from the end of the affected calendar quarter pursuant to Specification 6.7.C.2" instead of "pursuant to Specification 6.7.C.3." This change corrects a branching error in TS 4.16.A.4. TS 6.7.C.2 is the requirement for Environmental Special Reports applicable to TS 4.16.A.4. The correction is acceptable to the staff.

On page 244a, the licensee proposes to revise the wording of TS 6.5.B.1.b(1), (2), and (3) to match the wording of Section 2.4 (bulleted paragraphs) of Regulatory Guide (RG) 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants." The proposed wording is, "(1) A radiation monitoring device that continuously indicates the radiation dose rate in the area. (2) A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been determined and personnel have been made knowledgeable of them. (3) An individual qualified in radiation protection procedures with a radiation dose rate monitoring device. This individual is responsible for providing positive radiation protection control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation protection procedures or the applicable Radiation Work Permit." An exception to the RG 8.38 wording is that the licensee proposes to keep the Section (3) wording "shall perform periodic radiation surveillance" rather than "should perform periodic radiation surveillance." The "shall" is consistent with the wording in NUREG-1433, Rev.1, "Standard Technical Specifications, General Electric Plants, BWR/4," and is more conservative than "should." The proposed revision does not change any existing requirements and is acceptable to the staff.

On page 247a, the licensee proposes to delete current TS 6.6.B.11. Current Section 12 would become new Section 11 due to renumbering. The requirement to retain environmental qualification records for the life of the plant was superseded by 10 CFR 50.49 and removed by Amendment No. 59 to Facility Operating License No. DPR-22, dated February 16, 1989. TS 6.6.B.11 should have been deleted at that time. The change is acceptable to the staff.

2.6 Administrative Titles

On pages 232, 233, 234, 240, 243, 244a, and 246b, the licensee proposes to update various administrative titles. "Site Superintendent" is updated to "Shift Supervisor," "Superintendent, Radiation Protection" is updated to "General Superintendent, Radiation Services," "General Manager Nuclear Plants" is updated to "Plant Manager" or "Vice President Nuclear Generation," and Superintendent, Security and Services" is updated to "Superintendent, Security." The staff found no titles on page 234, so no change was necessary on that page. Also, on pages 232, 237, 239, 240, 242, and 250, "Vice President Nuclear Generation" is changed to "President, NSP Nuclear Generation," and on pages 241 and 243, "General Manager Nuclear Plants" is updated to "President, NSP Nuclear Generation." The proposed changes are administrative, reflect current organization titles, and do not reduce the level of plant responsibility. No requirements are revised by the changes, and they are acceptable to the staff.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (63 *FR* 53951). The amendment also changes recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment for the above items.

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the *Federal Register* on December 24, 1998 (63 *FR* 71316) on those items relating to administrative clarifications, corrections, and title changes, and typographical corrections.

Accordingly, based on the environmental assessment, the Commission has determined that these proposed changes will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: D. Desaulniers
F. Lyon

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