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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
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 May 15, 1980, 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:

   2. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 3 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8104080002

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

FOR THE NUCLEAR REGULATORY COMMISSION

T.A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 27, 1981

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

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

Remove	Insert
ii	ii
vi	vi
14	14
34	34
-	48A (add)
54	54
55	55
-	60A (add)
62	62
63	63
81	81
91	91
122	122
130	130
150	150
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LIST OF TABLES

Amendment No. 3

Amendment No. 2

Bases:

2.3 The abnormal operational transients applicable to operation of the Monticello Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power level of 1670 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3-2-3 of the FSAR. The licensed maximum power level 1670 MWt represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The Doppler reactivity feedback coefficient has conservatively been derated to 90% of the expected value. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion assumed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. The early portion of the scram stroke accomplishes the desired effect by inserting sufficient negative reactivity to turn the transient around. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

TABLE 4.1.2

SCRAM INSTRUMENT CALIBRATION

MINIMUM CALIBRATION FREQUENCES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

INSTRUMENT CHANNEL	GROUP	CALIBRATION METHOD	MINIMUM FREQUENCY (2)
APRM	E	Heat Balance	Once every 3 days (4)
IRM	E	Heat Balance	See Note 1
High Reactor Pressure	D	Pressure Standard	Every 3 months
High Drywell Pressure	D	Pressure Standard	Every 3 months
Low Reactor Water	D	Pressure Standard	Every 3 months
High Water Level in Scram Discharge	D	Water Level	Every 3 months
Condenser Low Vacuum	D	Vacuum Standard	Every 3 months
High Steam Line Radiation	E	See Note 3	See Note 3
Main Steamline Isolation Valve Closure	D	Observation	Every Operating Cycle
Turbine Control Valve Fast Closure	D	Pressure Standard	Every 3 months
Turbine Stop Valve Closure	D	Observation	Every Operating Cycle
Recirculation Flow Meters & Flow Instrumentation	-	Pressure Standard	Every 3 months

Notes:

1. Perform calibration test during every startup and normal shutdown.
2. Calibration tests 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.
3. This instrument will be calibrated every three months by means of a build-in current source, and each refueling outage with a known radioactive source.
4. This calibration is performed by taking a heat balance and adjusting the APRM to agree with the heat balance. Alarms and trips will be verified and calibrated if necessary during weekly functional test.

***GROUPS:**

D. Passive type devices.

E. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
G. Safeguards Bus Voltage Protection 1. Whenever the safeguards auxiliary electrical power system is required to be operable by Specification 3.9, the Limiting Conditions for Operation for the instrumentation listed in Table 3.2.6 shall be met.	

Table 3.2.2 - Continued
Instrumentation That Initiates Emergency Core Cooling System

Function	Trip Setting	Min. No. of Operable or Operating Trip Systems(3)	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (3)	Required Conditions*
D. Diesel Generator					
1. Degraded or Loss of Voltage Essential Bus (5)					
2. Low Low Reactor Water Level	>6'6"<6'10"	2	4(4)	4	D.
3. High Drywell Pres-	<2 psig	2	4(4)	4	D.

NOTES:

1. High drywell pressure may be bypassed when necessary only by closing the manual containment isolation valves during purging for containment inerting or de-inerting. Verification of the bypass condition shall be noted in the control room log. Also need not be operable when primary containment integrity is not required.

2. One instrument channel is a circuit breaker contact and the other is an undervoltage relay.

Notes:

3. Upon discovery that minimum requirements for the number of operable or operating trip systems, or instrument channels are not satisfied action shall be initiated to:

   (a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

   (b) Place the plant under the specified required conditions using normal operating procedures.

4. All instrument channels are shared by both trip systems.

5. See Table 3.2.6.

* Required conditions when minimum conditions for operation are not satisfied.

A. Comply with Specification 3.5.A.

B. Comply with Specification 3.5.D.

C. Reactor pressure $\leq$150 psig.

D. Comply with Specification 3.9.B.

Table 3.2.6

Instrumentation for Safeguards Bus Degraded Voltage and Loss of Voltage Protection

Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems(1)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Channels Per Trip System (1)	Required Conditions*
1. Degraded Voltage Protection (3)	3885 ± 18V 10 ± 1 sec	1/bus	3	3	A
2. Loss of Voltage Protection (2)	2625 ± 175V ≤ 1 sec	2 bus	2	2	A

NOTE:

1. Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated to:

   a. Satisfy the requirements by placing the appropriate channels or systems in the tripped condition, or

   b. Place the plant under the specified required conditions using normal operating procedures.

2. One out of two twice logic.
3. Two out of three logic.

* Required conditions when minimum conditions for operation are not satisfied:

A. Cold shutdown within 24 hours.

Table 4.2.1 - Continued
Minimum Test and Calibration Frequency For Core Cooling
Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (2)	Sensor Check (3)
3. Steam Line Low Pressure	Once/month	Once/3 months	None
4. Steam Line High Radiation	Once/week (5)	Note 6	Once/shift
HPCI ISOLATION			
1. Steam Line High Flow	Once/month	Once/3 months	None
2. Steam Line High Temperature	Once/month	Once/3 months	None
RCIC ISOLATION			
1. Steam Line High Flow	Once/month	Once/3 months	None
2. Steam Line High Temperature	Note 1	Once/3 months	None
REACTOR BUILDING VENTIALTION			
1. Radiation Monitors (Plenum)	Note 1	Once/3 months	Once/shift
2. Radiation Monitors (Refueling Floor)	Note 1	Once/3 months	(4)
OFF-GAS ISOLATION			
1. Radiation Montiors (Air Ejectors)	Notes (1,5)	Note 6	Once/shift
RECIRCULATION PUMP TRIP			
1. Reactor High Pressure	Note 1	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/Day
2. Reactor Low Water Level (Note 7)	Once/month	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/shift

Amendment No. 3.

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)
SAFEGUARDS BUS VOLTAGE			
1. Degraded Voltage Protection	Note 1	Quarterly	Not applicable
2. Loss of Voltage Protection	Note 1	Once/Operating Cycle	Not applicable

NOTES:

(1) Initially once per month until exposure hours (M as defined on Figures 4.1.1) is 2.0×10^5, thereafter according to Figure 4.1.1 with an interval not greater than three months.

(2) Calibrate prior to normal shutdown and start-up and thereafter check once per shift and test once per week until no longer required. Calibration of this instrument prior to normal shutdown means adjustment ' 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 shift.

(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) This instrument will be calibrated every three months by means of a built in current source, and each refueling outage with a known radioactive source.

(7) Surveillance also to be performed on containment isolation function of this instrumentation at the specified intervals.

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

Percent of Rod Length Inserted	Seconds
5	0.398
20	0.954
50	2.120
90	3.80

C. Scram Insertion Times

During each Operating Cycle, each operable control rod shall be subjected to scram time tests from the fully withdrawn position. If testing is not accomplished during reactor power operation, the measured scram insertion times shall be extrapolated to the reactor power operation condition utilizing previously determined correlations.

The scram times for all control rods will be determined during each operating cycle. The weekly control rod exercise test serves as a periodic check against deterioration of the control rod system and also verifies the ability of the control rod drive to scram since if a rod can be moved with drive pressure, it will scram because of higher pressure applied during scram. Allowing for monthly exercising of one rod in any two by two array is consistent with the bases for local and overall core reactivity insertion rates assumed in the transient analyses discussed above. The frequency of exercising the control rods under the conditions of two or more control rods out of serv[illegible] provides even further assurance of the reliability of the remaining control rods.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds six, the allowable number of inoperable rods.

D. Control Rod Accumulators

The basis for this specification was not described in the FSAR and, therefore, is presented in its entirety. Requiring no more than one inoperable accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a $k_{eff}<1.0$ -- other repeating rod sequences with more rods withdrawn resulted in $k_{eff}>1.0$. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with inoperable accumulators will be spaced in one-in-nine array rather than grouped together.

E. Reactivity Anomalies

During each fuel cycle excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity is indicated by the integrated worth of control rods inserted into the core, referred to as the control rod inventory in the core. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of actual rod inventory at any base equilibrium core state to predicted rod inventory at that state. Rod inventory predictions can be normalized to actual initial steady state rod patterns to minimize calculational uncertainties. Experience with other operating BWR's indicates that the control rod inventory should be predictable to the equivalent of one per cent in reactivity.

B. Reactor Vessel Temperature and Pressure

1. During in-service hydrostatic or leak testing, the reactor vessel shell temperatures specified in 4.6.B.1 shall be at or above the higher of the temperatures shown on the two curves of Figure 3.6.2 where the dashed curve, "RPV Beltline Region," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

2. During heatup by non-nuclear means (except with the reactor vessel vented), cooldown following nuclear shutdown, or low level physics tests the reactor vessel shell and fluid temperatures specified in 4.6.A shall be at or above the higher of the temperatures of Figure 3.6.3 where the dashed curve, "RPV Beltline Region," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

3. During all operation with a critical reactor, other than for low level physics tests or at times when the reactor vessel is vented, the reactor vessel shell and fluid temperatures specified in 4.6.A shall be at or above the higher of the temperatures of Figure 3.6.4 where the dashed curve, "RPV Beltline Region," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

B. Reactor Vessel Temperature and Pressure

1. During in-service hydrostatic or leak testing when the vessel pressure is above 312 psig, the following temperatures shall be recorded at least every 15 minutes :

   a. Reactor vessel shell adjacent to shell flange.

   b. Reactor vessel bottom head.

2. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The material sample program shall conform to ASTM E 185-66. Samples shall be withdrawn at one fourth and three fourths service life. Analysis of the first sample shall include a quantitiative determination of the copper and phosphorous content.

3. Neutron flux wires shall be installed in the reactor vessel adjacent to the reactor vessel wall at the core midplane level. The wires shall be removed and tested during the first refueling outage to experimentally verify the calculated value of neutron fluence at one fourth of the beltline shell thickness that is used to determine the NDTT shift from Figure 3.6.1.

5. Snubbers may be added to safety related systems without prior License Amendment to Table 3.6.1 provided that a revision to Table 3.6.1 is included with the next license amendment request.

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

2. All hydraulic snubbers whose seal materials are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.

3. Once each Operating Cycle, a representative sample of 10 hydraulic snubbers or approximately 10% of the hydraulic snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement, lock up, and bleed. For each unit and subsequent unit found inoperable, an additional 10% or ten hydraulic snubbers shall be so tested until no more failures are found or all units have been tested. Snubbers designated in Table 3.6.1 as being especially difficult to remove or located in High Radiation Areas during shutdown are exempt from this requirement.

4. Snubbers may be reclassified as being in or out of High Radiation Areas during shutdown in Table 3.6.1 based on the most recent radiation survey provided that a revision to Table 3.6.1 is included with the next license amendment request.

D. Coolant Leakage

The former 15 gpm limit for leaks from unidentified sources was established assuming such a leakage was coming from the primary system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. From the crack size a leakage rate can be determined. For a crack size which gives a leakage of 5 gpm, the probability of rapid propagation is less than 10^{-5}. Thus, an unidentified leak of 5 gpm when assumed to be from the primary system had less than one chance in 100,000 of propagating, which provides adequate margin. A leakage of 5 gpm is detectable and measurable. The 24 hour period allowed for determination of leakage is also based on the low probability of the crack propagating.

The capacity of the drywell sump pumps is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

E. Safety/Relief Valves

Testing of all required safety/relief vavles each refueling outage ensures that any valve deterioration is detected. A tolerance value of 1% for safety/relief valve setpoints is specified in Section III of the ASME Boiler and Pressure Vessel Code. Analyses have been performed with all valves assumed set 1% higher (1108 psig + 1%) than the nominal setpoint; the 1375 psig code limit is not exceeded in any case.

The safety/relief valves are used to limit reactor vessel overpressure and fuel thermal duty.

The required safety/relief valve steam flow capacity is determined by analyzing the transient accompanying the main steam flow stoppage resulting from a postulated MSIV Closure from a power of 1670 Mwt. The analysis assumes a multiple-failure wherein direct scram (valve position) is neglected. Scram is assumed to be from indirect means (high flux). In this event, the safety/relief valve capacity is assumed to be 83.2% of the full power steam generation rate.

H. Shock Suppressors (Snubbers) (contd.)

Examination of defective snubbers at reactor facilities and material tests performed at several laboratories has shown that millable gum polyurethane deteriorates rapidly under the temperature and moisture conditions present in many snubber locations. Although molded polyurethane exhibits greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to precisely define an upper temperature limit for the molded polyrethane. Lab tests and in-plant experience indicate that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

To further increase the assurance of snubber reliability, functional tests should be performed once each operating cycle. These tests will incude stroking of the snubbers to verify proper piston movement, lockup and bleed. Ten percent or ten snubbers, whichever is less, represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Snubbers in High Radiation Areas or those especially difficult to remove need not be selected for functional tests provided operability was previously verified. Snubbers are considered especially difficult to remove if they (1) have a rated capacity greater than 50,000 lb, (2) are located greater than 5 feet above the adjacent platform, or (3) located greater than 3 feet below the adjacent platform.

C. Secondary Containment

1. Except as specified in 3.7.C.2 and 3.7.C.3, Secondary Containment Integrity shall be maintained during all modes of plant operation.

2. Secondary Containment Integrity is not required when all of the following conditions are satisfied:

   a. the reactor is subcritical and Specification 3.3.A is met.

   b. The reactor water temperature is below 212^{o} and the reactor coolant system is vented.

   c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A

   d. The fuel cask or irradiated fuel is not being moved within the reactor building.

3. With an inoperable secondary containment isolation damper, restore the inoperable damper to operable status or isolate the affected duct by use of a closed damper or blind flange within eight hours.

4. If Specifications 3.7.C.1 through 3.7.C.3 cannot be met, initiate a normal orderly shutdown and have the reactor in the Cold Shutdown condition within 24 hours. Alterations of the

C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

   a. Secondary containment capability to maintain at least a 1/4 inch of water vacuum under calm wind ($2 < u <$ mph) conditions with a filter train flow rate of $\leq$4,000 scfm, shall be demonstrated at each refueling outage prior to refueling. Verification that each automatic damper actuates to its isolation position shall be performed at each refueling outage and after maintenance, repair or replacement work is performed on the damper or its associated actuator, control circuit, or power circuit.

3.0 LIMITING CONDITIONS FOR OPRATION	4.0 SURVEILLANCE REQUIREMENTS
reactor core, operations with a potential for reducing the shutdown margin below that specified in specification 3.3.A, and handling of irradiated fuel or the fuel cask in the Secondary Containment are to be immediately suspended if secondary containment integrity is not maintained.	D. Primary Containment Isolation Valves
D. Primary Containment Isolation Valves	1. The primary containment isolation valve surveillance shall be performed as follows:
1. During reactor power operating conditions, all isolation valves listed in Table 3.7.1 and all primary system instrument line flow check valves shall be operable except as specified in 3.7.D.2.	a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.
	b. At least once per operating cycle the primary system instrument line flow check valves shall be tested for proper operation.
	c. At least once per quarter
	(1) All normally open power-operated isolation valves (except for the main steam line power-operated isolation valves) shall be fully closed and reopened.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
c. For the diesel generators to be considered operable, there shall be a minimum of 26,250 gallons of diesel fuel (7 days supply for 1 diesel generator at full load) in the diesel oil storage tank.	c. At least once each Operating Cycle during shutdown simulate a loss of offsite power in conjunction with an ECCS actuation test signal, and: 1. Verify de-energization of the emergency busses and load shedding from the emergency busses. 2. Verifying diesel starts from ambient conditions on the auto-start signal and is ready to accept emergency loads within ten seconds, energizes the emergency busses with permanently connected loads, energizes the auto-connected emergency loads in proper time sequence, and operates for greater than five minutes while its generator is loaded with the emergency loads. d. During the monthly generator test, the diesel fuel oil transfer pump and diesel oil service pump shall be operated. e. Once a month the quantity of diesel fuel available shall be logged. f. Once a month a sample of diesel fuel shall be taken and checked for quality.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
4. Station Battery System If one of the two 125 V battery systems or the 250 V battery system is made or found to be inoperable for any reason, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours unless such battery systems are sooner made operable.	4. Station Battery System a. Every week the specific gravity and voltage of the pilot cell and temperature of the adjacent cells and overall battery voltage shall be measured. b. Every three months the measurements shall be made of voltage of each cell to nearest 0.01 volt, specific gravity of each cell, and temperature of every fifth cell. c. Every refueling outage, the station batteries shall be subjected to a rated load discharge test. Determine specific gravity and voltage of each cell after the discharge.
5. 24V Battery Systems From and after the date that one of the two 24V battery systems is made or found to be inoperable for any reason, refer to Specificaton 3.2 for appropriate action.	5. 24V Battery Systems a. Every week the specific gravity and voltage of the pilot cell and temperature of adjacent cells and overall battery voltage shall be measured. b. Every three months the measurements shall be made of voltage of each cell to nearest 0.01 volt, specific gravity of each cell, and temperature of every fifth cell.

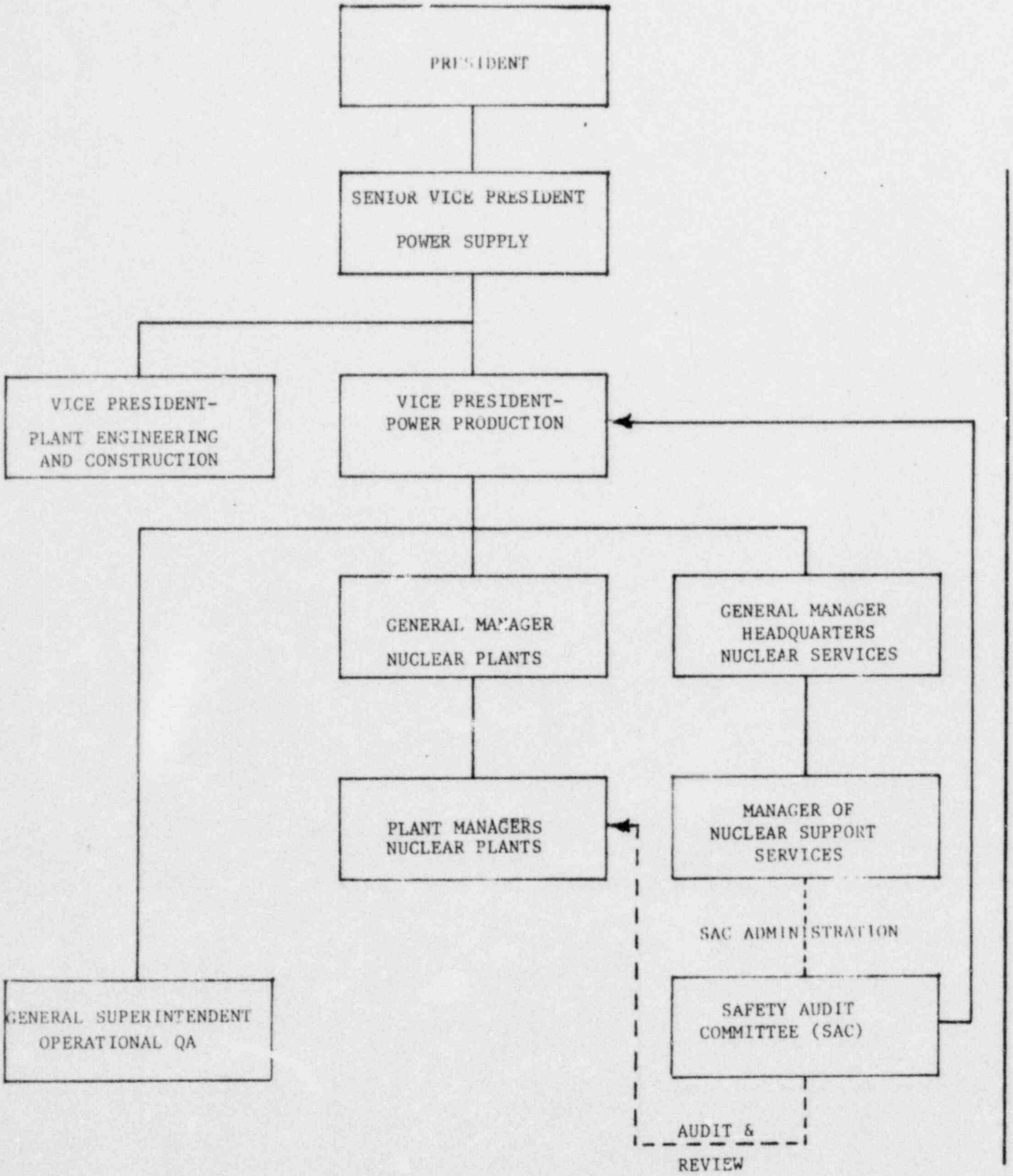
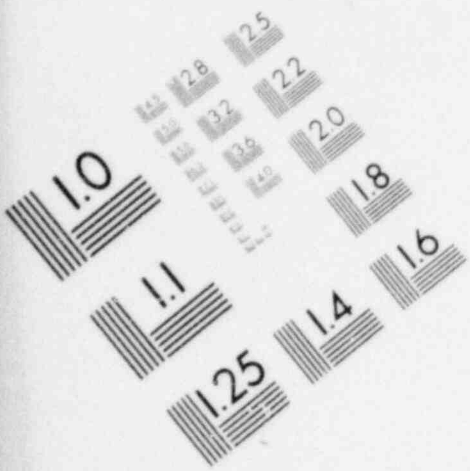


FIGURE 6.1.1

NSP CORPORATE ORGANIZATIONAL

RELATIONSHIP TO ON-SITE OPERATING ORGANIZATION

POOR ORIGINAL



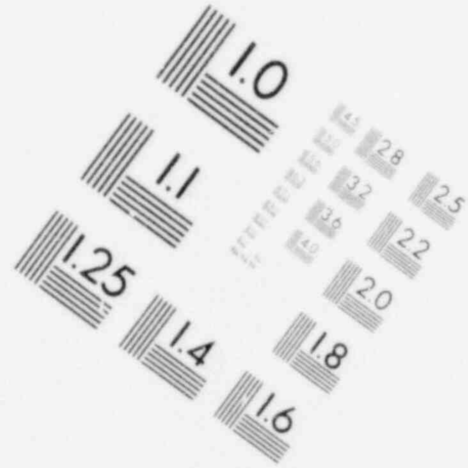
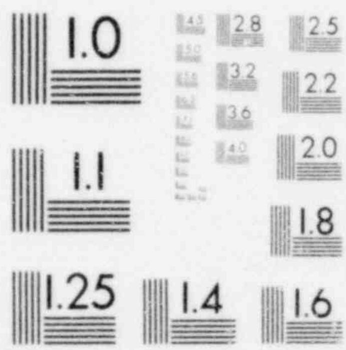


IMAGE EVALUATION TEST TARGET (MT-3)



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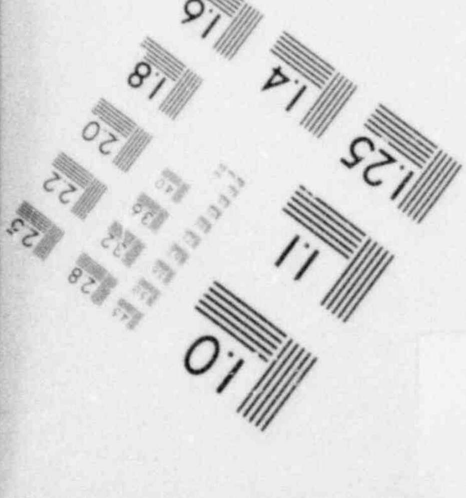


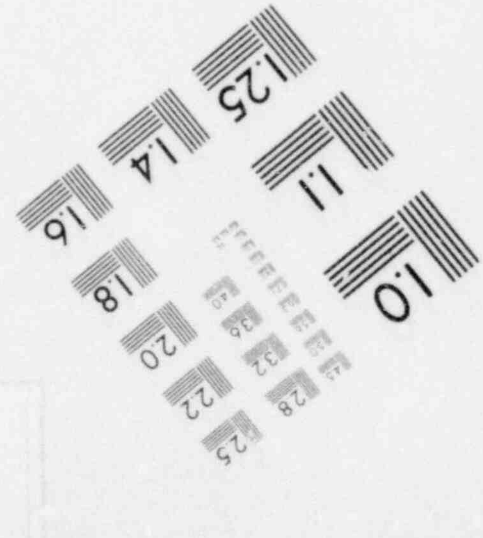
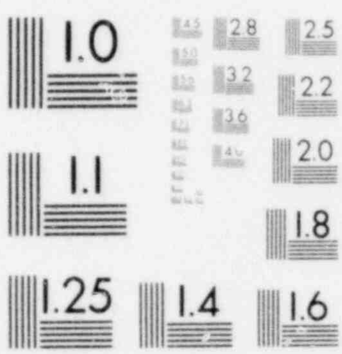


IMAGE EVALUATION
TEST TARGET (MT-3)



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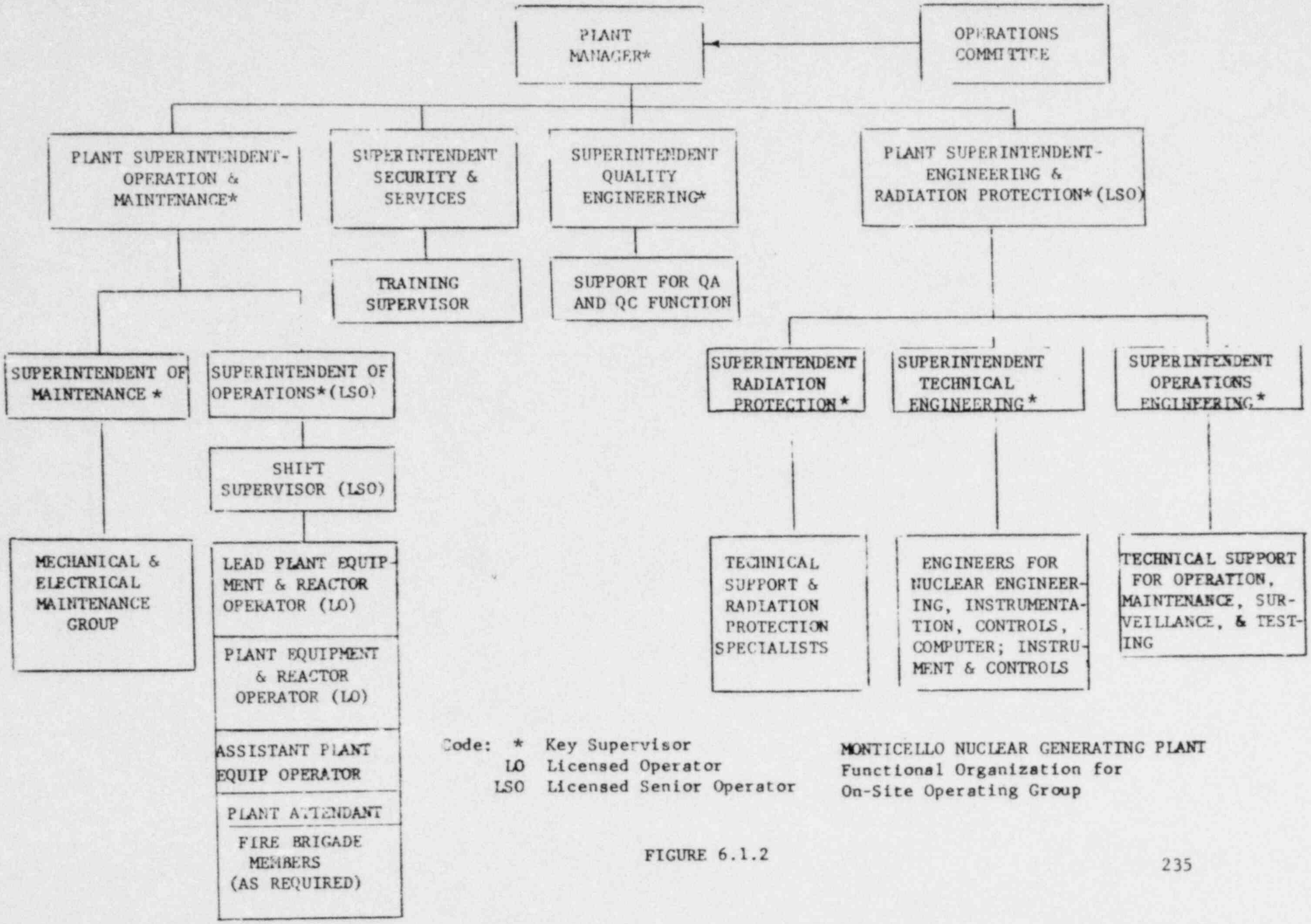

Code: * Key Supervisor
LO Licensed Operator
LSO Licensed Senior Operator

MONTICELLO NUCLEAR GENERATING PLANT
Functional Organization for
On-Site Operating Group

FIGURE 6.1.2

Amendment No. 3

6.2 Review and Audit

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

A. Safety Audit Committee (SAC)

The Safety Audit Committee provides the independent review of plant operations from a nuclear safety standpoint. Audits of plant operation are cnducted under the cogizance 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 the plant, appointed by the Vice President - Power Production. Other members shall be appointed by the Vice President - Power Production 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 events which are required by regulation or technical specifications to be reported to NRC in writing within 24 hours.

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.

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 Paragraph 4.4 of ANSI N18.7-1972, 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. **Periodic review of the audit program should be performed by the SAC at least twice a year to assure its adequacy.**

   c. Written reports of the audits shall be reviewed by the Vice President - Power Production, 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 Vice President - Power Production.

8. Records

Minutes shall be 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 Vice President - Power Production, the General Manager Nuclear Plants, 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.

A. Operations Committee (OC)

1. Membership

The Operations Committee shall consist of at least six (6) members drawn from the key supervisors of the on-site supervisory staff. The Plant Manager shall serve as Chairman of the OC and shall appoint a Vice Chairman from the OC membersip to act in his absence.

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 permanent members, including the Chairman or Vice Chairman

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

   a. Proposed tests and experiments and their results.

   b. Modifications to plant systems or equipment as described in Final 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 General Manager Nuclear Plants and to the Chairman of the Safety Audit Committee.

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

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

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

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

5. Authority

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

6. Records

Minutes shall be recorded for all meetings of the OC and shall identify all documentary material reviewed. The minutes shall be distributed to each member of the OC, the Chairman and each member of the Safety Audit Committee, the General Manager Nuclear Plants 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. Distributicn 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 off-site 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 General Manager Nuclear Plants, 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 be prepared. This report shall be submitted to the Commission, to the General Manager Nuclear Plants 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.