Exhibit B

Monticello Nuclear Generating Plant

License Amendment Request Dated August 15, 1996

Technical Specification Pages Marked Up
with Proposed Wording Changes

Exhibit B consists of the existing Technical Specification pages marked up with the proposed changes. Existing pages affected by this change are listed below:

Page

iv	vi	vii
22	31	69
89	99	102
164	188	190
198t	198y	200
227c	227d	227e
229b	229i	229ff
232	233	234
240	241	243
244a	246b	247a
	252	

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LIST OF TABLES

LIST OF TABLES (continued)

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.

1380

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 ~~1400~~ 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 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.

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.

3.2 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 ~~starting and running loads during a loss of coolant accident~~. 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

a steady state LOCA load that maintains adequate voltage at the 480 V essential MCCS.

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 thermal limits are never reached during the transients requiring control rod scram. The limiting operational transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection,~~ and MCPR remains above the Safety Limit (T.S.2.1.A).

Therefore

The CPR safety limit is never exceeded during any transient requiring control rod scram,

Basis 3.4 and 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 ~~for possible imperfect mixing of the chemical solution in the reactor water and dilution from the water in the cooldown circuit.~~ 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 (10CFR50.62) 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 10CFR50.62 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 CONDITION FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

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 cf 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. 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.

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

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. 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, dry-well 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 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). 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 the 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.

EXCEPT TESTING should be IAW ASTM D3803-1989.

Bases 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, ~~1983.~~ 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:

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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 radio-nuclides: 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.

from

3.8 and 4.8 Bases: (continued)

Specification 3.8.B.4.c is provided to ensure that the concentration of potentially explosive gas mixtures contained in the compressed storage subsystem is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. Maintaining the concentration of hydrogen below the flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

Specification 3.8.B.4.e is provided to limit the radioactivity which can be stored in one decay tank. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tanks contents, the resulting total body exposure to an individual at the site restricted area boundary will not exceed 20 mrem. A flow restrictor in the discharge line of the decay tanks prevents a tank ~~form~~ being discharged at an uncontrolled rate. In addition, interlocks prevent the contents of a tank from being released with less than 12 hours of holdup.

Specification 3.8.B.5 establishes a maximum activity at the steam jet air ejector. Restricting the gross radioactivity rate of noble gases form the main condenser provides reasonable assurance that the total body exposure to an individual at the restricted area boundary will not exceed the limits of 10 CFR Part 20 in the event this effluent is inadvertently discharged directly to the environment with minimal treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

Specification 3.8.B.6 requires the containment to be purged and vented through the standby gas treatment system except during inerting and deinerting operations. This provides for iodine and particulate removal from the containment atmosphere. During outages when the containment is opened for maintenance, the containment ventilation exhaust is directed to the monitored reactor building vent. Use of the 2-inch flow path prevents damage to the standby gas treatment system in the event of a loss of coolant accident during purging or venting. Use of the reactor building plenum and vent flow path for inerting and deinerting operations permits the control room operators to monitor the activity level of the resulting effluent by use of the Reactor Building Vent Wide Range Gas Monitors. In addition, the Reactor Building Plenum Monitors will automatically terminate releases if their release rate limits are exceeded. In the event that the reactor building release rate exceeds the Reactor Building Vent Wide Range Gas Monitor alarm settings, the monitors will alarm in the control room alerting the operators to take actions to limit the release of gaseous radioactive effluents. An analysis has been performed which shows that the control room operators would have in excess of two hours to take manual actions to terminate releases without exceeding the permissible levels of radiation exposure in 10 CFR Part 20 Section 20.105(a). The alarm settings for the reactor building vent wide range gas monitors are calculated in accordance with the NRC approved methods in the ODCM to ensure that alarms will alert control room operators prior to the limits of 10CFR Part 20, Section 20.105(a) being exceeded.

C. Solid Radioactive Waste

Specification 3.8.C.1 provides assurance that the solid radwaste system will be used whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criteria 60 of Appendix A to 10 CFR Part 50.

3.0 LIMITING CONDITION 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~~.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. 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	4.0 SURVEILLANCE REQUIREMENTS
H. Alternate Shutdown System	H. Alternate Shutdown System
1. The system controls on the ASDS panel shall be operable whenever that ~~systems~~ system/component is ~~controls are~~ required to be operable. ~~from the control room.~~	1. Switches on the alternate shutdown system panel shall be functionally tested once per operating cycle.
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;	2. The alternate shutdown system panel master transfer switch shall be verified to alarm in the control room when unlocked once per operating cycle.
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.	

TABLE 3.13.1
SAFETY RELATED FIRE DETECTION INSTRUMENTS

Fire Detection Zone	Location	Minimum Instruments Operable Heat	Flame	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
~~2E~~ 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	Reacto. 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
SAFETY RELATED FIRE DETECTION INSTRUMENTS

Detection

Fire Zone	Location	Minimum Instruments Operable Heat	Flame	Smoke
12A	Turbine Bldg. - 911' - 4.16 KV Switchgear			3
13C	Turbine Bldg. - 911' evel - 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.B.2~~ 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.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
Deleted	2. ~~Welds in austenitic stainless steel piping four inches or larger in diameter containing reactor coolant at a temperature above 200 degrees F during power operation, including reactor vessel attachments and appurtenances, shall be included in an augmented inspection program meeting the requirements of Generic Letter 88-01.~~

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
6.7.C.2	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.3.~~ 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.

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.

Shift Supervisor

The ~~Site Superintendent~~ (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 responsiblities 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 Vice President 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 responsiblity 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 on site 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.

General Superintendent Radiation Services

6. A fire brigade of at least five members shall be maintained on site 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 ~~Superintendent Radiation Protection~~ 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 ~~Superindent~~, 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 ~~Nothern~~ States Power management.

Northern

* 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.

Superintendent

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:

8 or 12-hour day, nominal 40-hour week

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-hour day, 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.

7. Authority

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

8. Records

Plant Manager

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 Vice President, Nuclear Generation, 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 distributiion 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 on-site 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 ~~General Manager Nuclear Plants~~ Vice President Nuclear Generation and to the Chairman of the Safety Audit 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 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

Vice President Nuclear Generation

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 by prepared. This report shall by 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.

General Superintendent Radiation Services.

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 ~~Superintendent, Radiation Protection.~~

   b. In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10CFR20, 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 ~~which~~ continuously indicates the radiation dose rate in the area.

   (2) A radiation monitoring device That ~~which~~ continuously integrates the radiation dose rate in the area and alarms when a preset ~~present~~ 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 ~~rate level in the area has been established~~ and personnel have been made knowledgeable of them.

   (3) An individual qualified in radiation protection procedures is ~~who is equipped~~ with a radiation dose rate monitoring device. This individual ~~shall be~~ responsible for providing positive radiation protection control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified ~~by the unit Health Physicist in the~~ 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.

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. Onsite Dose Calculation Manual (ODCM)

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 additionally 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. Security

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. ~~and Services.~~

G. Temporary Changes to Procedures

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 ~~Superintendent, Radiation Protection~~ for health physics procedures and the Superintendent, Security ~~and Services~~ for security procedures.

General Superintendent, Radiation Services

delete

B. Records Retained for Plant Life (continued)

11. ~~Records for Environmental Qualification which are covered under the provisions of paragraph 6.8.~~

~~12.~~ 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.

2. Environmental Special Reports

When radioactivity levels in samples exceed limits specified in Table 4.16.3 an Environmental Special Report shall be submitted within 30 days from the end of the affected calendar quarter. For certain cases involving long analysis time, determination of quarterly averages may extend beyond the 30 days period. In these cases the potential for exceeding the quarterly limits will be reported within the 30 day period to be followed by the Environmental Special Report as soon as practicable.

3. Other Environmental Reports (non-radiological, non-aquatic)

a. Environmental events that indicate or could result in a significant environmental impact causally related to plant operation. The following are examples: excessive bird impaction; onsite plant or animal disease outbreaks; unusual ~~morality~~ MORTALITY of any species protected by Endangered Species Act of 1973; increase in nuisance organisms or conditions; or excessive environmental impact caused by herbicide application to transmission corridors associated with the plant. This report shall be submitted within 30 days of the event and shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

b. Proposed changes, tests or experiments which may result in a significant increase in any adverse environmental impact which was not previously reviewed or evaluated in the Final Environmental Statement or supplements thereto. This report shall include an evaluation of the environmental impact of the proposed activity and shall be submitted 30 days prior to implementing the proposed change, test or experiment.

D. Special Reports

Unless otherwise indicated, special reports required by the Technical Specification shall be submitted within the time period specified for each report.

Exhibit C

Monticello Nuclear Generating Plant

License Amendment Request Dated August 15, 1996

Revised Monticello Technical Specification Pages

Exhibit C consists of revised Technical Specification pages that incorporate the proposed changes. The pages included in this exhibit are as listed below.

Page

iv	vi	vii
22	31	69
89	99	102
164	188	190
198t	198y	200
227c	227d	227e
229b	229i	229ff
232	233	234
240	241	243
244a	246b	247a
	252	

LIST OF TABLES

LIST OF TABLES (continued)

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.

Bases Continued:

3.2 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 480 V 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

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).

Basis 3.4 and 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 (10CFR50.62) 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 10CFR50.62 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 CONDITION FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

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. 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	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, dry-well 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 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 ASTM 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 the 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 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:

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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 analysed 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 venting if the limits of 3.8.B.1 are satisfied for other nuclides. The analysis shall be completed within 24 hours after sampling, however.

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

3.8 and 4.8 Bases: (continued)

Specification 3.8.B.4.c is provided to ensure that the concentration of potentially explosive gas mixtures contained in the compressed storage subsystem is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. Maintaining the concentration of hydrogen below the flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

Specification 3.8.B.4.e is provided to limit the radioactivity which can be stored in one decay tank. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tanks contents, the resulting total body exposure to an individual at the site restricted area boundary will not exceed 20 mrem. A flow restrictor in the discharge line of the decay tanks prevents a tank from being discharged at an uncontrolled rate. In addition, interlocks prevent the contents of a tank from being released with less than 12 hours of holdup. |

Specification 3.8.B.5 establishes a maximum activity at the steam jet air ejector. Restricting the gross radioactivity rate of noble gases form the main condenser provides reasonable assurance that the total body exposure to an individual at the restricted area boundary will not exceed the limits of 10 CFR Part 20 in the event this effluent is inadvertently discharged directly to the environment with minimal treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

Specification 3.8.B.6 requires the containment to be purged and vented through the standby gas treatment system except during inerting and deinerting operations. This provides for iodine and particulate removal from the containment atmosphere. During outages when the containment is opened for maintenance, the containment ventilation exhaust is directed to the monitored reactor building vent. Use of the 2-inch flow path prevents damage to the standby gas treatment system in the event of a loss of coolant accident during purging or venting. Use of the reactor building plenum and vent flow path for inerting and deinerting operations permits the control room operators to monitor the activity level of the resulting effluent by use of the Reactor Building Vent Wide Range Gas Monitors. In addition, the Reactor Building Plenum Monitors will automatically terminate releases if their release rate limits are exceeded. In the event that the reactor building release rate exceeds the Reactor Building Vent Wide Range Gas Monitor alarm settings, the monitors will alarm in the control room alerting the operators to take actions to limit the release of gaseous radioactive effluents. An analysis has been performed which shows that the control room operators would have in excess of two hours to take manual actions to terminate releases without exceeding the permissible levels of radiation exposure in 10 CFR Part 20 Section 20.105(a). The alarm settings for the reactor building vent wide range gas monitors are calculated in accordance with the NRC approved methods in the ODCM to ensure that alarms will alert control room operators prior to the limits of 10CFR Part 20, Section 20.105(a) being exceeded.

C. Solid Radioactive Waste

Specification 3.8.C.1 provides assurance that the solid radwaste system will be used whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criteria 60 of Appendix A to 10 CFR Part 50.

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

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 of 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 Instruments Operable Heat	Flame	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
SAFETY RELATED FIRE DETECTION INSTRUMENTS

Fire Detection Zone	Location	Minimum Instruments Operable Heat	Flame	Smoke
12A	Turbine Bldg. - 911' - 4.16 KV Switchgear			3
13C	Turbine bldg. - 911' evel - 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.

2. Deleted.

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 radio-nuclides 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.

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 responsiblities 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 Vice President 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 responsiblity 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 on site 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 on site 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.

7. Authority

The SAC shall be advisory to the Vice President, 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 Vice President, 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. <u>Operations Committee (OC)</u>

1. Membership

The Operations Committee shall consist of at least six (6) regular 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 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 Vice President Nuclear Generation and to the Chairman of the Safety Audit 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 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 Vice President 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 be prepared. This report shall by submitted to the Commission, to the Vice President 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 10CFR20, 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. Offsite Dose Calculation Manual (ODCM)

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 additionally 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. Security

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. Temporary Changes to Procedures

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. <u>Records Retained for Plant Life (continued)</u>

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.

2. Environmental Special Reports

When radioactivity levels in samples exceed limits specified in Table 4.16.3 an Environmental Special Report shall be submitted within 30 days from the end of the affected calendar quarter. For certain cases involving long analysis time, determination of quarterly averages may extend beyond the 30 days period. In these cases the potential for exceeding the quarterly limits will be reported within the 30 day period to be followed by the Environmental Special Report as soon as practicable.

3. Other Environmental Reports (non-radiological, non-aquatic)

a. Environmental events that indicate or could result in a significant environmental impact causally related to plant operation. The following are examples: excessive bird impaction; onsite plant or animal disease outbreaks; unusual mortality of any species protected by Endangered Species Act of 1973; increase in nuisance organisms or conditions; or excessive environmental impact caused by herbicide application to transmission corridors associated with the plant. This report shall be submitted within 30 days of the event and shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

b. Proposed changes, tests or experiments which may result in a significant increase in any adverse environmental impact which was not previously reviewed or evaluated in the Final Environmental Statement or supplements thereto. This report shall include an evaluation of the environmental impact of the proposed activity and shall be submitted 30 days prior to implementing the proposed change, test or experiment.

D. Special Reports

Unless otherwise indicated, special reports required by the Technical Specification shall be submitted within the time period specified for each report.