

Exhibit B

Technical Specification Pages Marked Up
with Proposed Changes

Monticello Nuclear Generating Plant

Supplement 2 to License Amendment Request Dated
August 15, 1996

Exhibit B consists of the existing Technical Specification pages marked up with the proposed changes from submittals on August 15, 1996, March 19, 1998, and of Supplement 2. Pages affected by this change are listed below:

Page

i, ii, iii, iv, v
vi, vii, 22, 31, 61, 62
63a, 69, 72, 82, 89
99, 102, 125, 126, 164
188, 190, 198t, 200
223, 227b, 227c, 227d, 227e
229b, 229c, 229ff, 229i, 229u
232, 233, 234, 237, 239
240, 241, 242, 243, 244a, 246b
247a, 250

TABLE OF CONTENTS

	<u>Page</u>
1.0 DEFINITIONS	1
2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS	6
2.1 and 2.3 Fuel Cladding Integrity	6
2.1 Bases	10
2.3 Bases	14
2.2 and 2.4 Reactor Coolant System	21
2.2 Bases	22
2.4 Bases	24
3.0 LIMITING CONDITIONS FOR OPERATION AND 4.0 SURVEILLANCE REQUIREMENTS	25a
4.0 Surveillance Requirements	25a 25b
3.1 and 4.1 Reactor Protection System	26
3.1 Bases	35
4.1 Bases	34 42
3.2 and 4.2 Protective Instrumentation	45
A. Primary Containment Isolation Functions	45
B. Emergency Core Cooling Subsystems Actuation	46
C. Control Rod Block Actuation	46
D. Other Instrumentation	46a
E. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation	47
F. Recirculation Pump Trip Initiation and Alternate Rod Injection Initiation	48
G. Safeguards Bus Voltage Protection	48
H. Instrumentation for S/RV Low-Low Set Logic	48
I. Instrumentation for Control Room Habitability Protection	48
3.2 Bases	64
4.2 Bases	72
3.3 and 4.3 Control Rod System	76
A. Reactivity Limitations	76
B. Control Rod Withdrawal	77
C. Scram Insertion Times	81
D. Control Rod Accumulators	82
E. Reactivity Anomalies	83
F. Scram Discharge Volume	83A
G. Required Action	83A
3.3 and 4.3 Bases	84

Supplement 2

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Supplement 2
(No changes -)
(reprinted only)

	<u>Page</u>	
3.4 and 4.4	Standby Liquid Control System	93
A.	System Operation	93
B.	Boron Solution Requirements	95
C.		96
	3.4 and 4.4 Bases	99
3.5 and 4.5	Core and Containment/Spray Cooling System	101
A.	ECCS Systems	101
B.	RHR Intertie Return Line Isolation Valves	103
C.	Containment/Spray Cooling System	104
D.	RCIC	105
E.	Cold Shutdown and Refueling Requirements	106
F.	Recirculation System	107
	3.5/4.5 Bases	110
3.6 and 4.6	Primary System Boundary	121
A.	Reactor Coolant Heatup and Cooldown	121
B.	Reactor Vessel Temperature and Pressure	122
C.	Coolant Chemistry	123
D.	Coolant Leakage	126
E.	Safety/Relief Valves	127
F.	Deleted	
G.	Jet Pumps	128
H.	Snubbers	129
	3.6 and 4.6 Bases	145
3.7 and 4.7	Containment Systems	156
A.	Primary Containment	156
B.	Standby Gas Treatment System	166
C.	Secondary Containment	169
D.	Primary Containment Isolation Valves	170
E.	Combustible Gas Control System	172
	3.7 Bases	175
	4.7 Bases	183

3.8 and 4.8 Radioactive Effluents	192
A. Liquid Effluents	192
B. Gaseous Effluents	197
C. Solid Radioactive Waste	198e
D. Dose from All Uranium Fuel Cycle Sources	198f
3.8 and 4.8 Bases	198u
3.9 and 4.9 Auxiliary Electrical Systems	199
A. Operational Requirements for Startup	199
B. Operational Requirements for Continued Operation	200
1. Transmission Lines	200
2. Reserve Transformers	201
3. Standby Diesel Generators	201
4. Station Battery System	203
5. 24V Battery Systems	203
3.9 Bases	204
4.9 Bases	205
3.10 and 4.10 Refueling	206
A. Refueling Interlocks	206
B. Core Monitoring	207
C. Fuel Storage Pool Water Level	207
D. Movement of Fuel	207
E. Extended Core and Control Rod Drive Maintenance	208
3.10 and 4.10 Bases	209
3.11 and 4.11 Reactor Fuel Assemblies	211
A. Average Planar Linear Heat Generation Rate	211
B. Linear Heat Generation Rate	212
C. Minimum Critical Power Ratio	213
3.11 Bases	216
4.11 Bases	218
3.12 and 4.12 Sealed Source Contamination	219
A. Contamination	219
B. Records	221
3.12 and 4.12 Bases	222

3.13 and 4.13 Fire Detection Protection Systems

- A. Fire Detection Instrumentation 223
- B. Fire Suppression Water System 224
- C. Hose Stations 226
- D. Yard Hydrant Hose Houses 227
- E. Sprinkler Systems 227a
- F. Halon Systems 227b
- G. Penetration Fire Barriers 227b
- H. Alternate Shutdown System 227c

- 3.13 Bases 228
- 4.13 Bases ~~229~~ 228b

3.14 and 4.14 Accident Monitoring Instrumentation 229a

~~3.14 and 4.14 Bases 229e~~

3.15 and 4.15 Inservice Inspection and Testing 229f

- 3.15 and 4.15 Bases 229g

3.16 and 4.16 Radiation Environmental Monitoring Program 229h

- A. ~~Sampling and Analysis~~ 229h
- B. Land Use Census 229j
- C. Interlaboratory Comparison Program 229k

- 3.16 and 4.16 Bases 229t

3.17 and 4.17 Control Room Habitability 229u

- A. Ventilation System 229u
- B. Emergency Filtration System 229v

- 3.17 Bases 229y
- 4.17 Bases 229z

5.0 DESIGN FEATURES 230

- 5.1 Site 230
- 5.2 Reactor 230
- 5.3 Reactor Vessel 230
- 5.4 Containment 230
- 5.5 Fuel Storage 231
- 5.6 Seismic Designs 231

6.0 ADMINISTRATIVE CONTROLS 232

- 6.1 Organization 232
- 6.2 Review and Audit 237
- 6.3 Special Inspection and Audits 243
- 6.4 Action to be taken if a Safety Limit is Exceeded 243
- 6.5 Plant Operating Procedures 244
- 6.6 Plant Operating Records 246
- 6.7 Reporting Requirements 248

~~6.8 Environmental Qualification 253~~

Do not add
(Supplement 2)

LIST OF FIGURES

<u>Figure No.</u>		<u>Page No.</u>
3.4.1	Sodium Pentaborate Solution Volume-Concentration Requirements	97
3.4.2	Sodium Pentaborate Solution Temperature Requirements	98
3.5.1	Single Loop Operation Surveillances Power/Flow Curve	109
3.6.1	Core Beltline Operating Limits Curve Adjustment vs. Fluence	133
3.6.2	Minimum Temperature vs. Pressure for Pressure Tests	134
3.6.3	Minimum Temperature vs. Pressure for Mechanical Heatup or Cooldown Without the Core Critical	135
3.6.4	Minimum Temperature vs. Pressure for Core Operation	136
4.6.2	Chloride Stress Corrosion Test Results @ 500°F	137
3.7.1	Differential Pressure Decay Between the Drywell and Wetwell	191
3.8.1	Monticello Nuclear Generating Plant Site Boundary for Liquid Effluents	198g
3.8.2	Monticello Nuclear Generating Plant Site Boundary for Gaseous Effluents	198h

LIST OF TABLES

<u>Table No.</u>	<u>Page</u>
3.1.1 Reactor Protection System (Scram) Instrument Requirements	28
4.1.1 Scram Instrument Functional Tests - Minimum Functional Test Frequencies for Safety Instrumentation and Control Circuits	32
4.1.2 Scram Instrument Calibration - Minimum Calibration Frequencies for Reactor Protection Instrument Channels	34
3.2.1 Instrumentation that Initiates Primary Containment Isolation Functions	49
3.2.2 Instrumentation that Initiates Emergency Core Cooling Systems	52
3.2.3 Instrumentation that Initiates Rod Block	51 56
3.2.4 Instrumentation that Initiates Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation	59
3.2.5 Instrumentation that Initiates a Recirculation Pump Trip and Alternate Rod Injection	60
3.2.6 Instrumentation for Safeguards Bus Degraded Voltage and Loss of Voltage Protection	60a
3.2.7 Instrumentation for Safety/Relief Valve Low-Low Set Logic	60b
3.2.8 Other Instrumentation	60d
3.2.9 Instrumentation for Control Room Habitability Protection	60e
4.2.1 Minimum Test and Calibration Frequency for Core Cooling, Rod Block and Isolation Instrumentation	61
4.6-1 4.6.1 Snubber Visual Inspection Interval	132a
3.8.1 Radioactive Liquid Effluent Monitoring Instrumentation	198i
3.8.2 Radioactive Gaseous Effluent Monitoring Instrumentation	198k
4.8.1 Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements	198m
4.8.2 Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements	198n
4.8.3 Radioactive Liquid Waste Sampling and Analysis Program	198p
4.8.4 Radioactive Gaseous Waste Sampling and Analysis Program	198s

Supplement 2

4.6.1

Supplement 2

LIST OF TABLES (continued)

<u>Table No.</u>		<u>Page</u>
3.13.1	Safety Related Fire Detection Instruments	227c 227d
3.14.1	Instrumentation for Accident Monitoring	229b
4.14.1	Minimum Test and Calibration Frequency for Accident Monitoring Instrumentation	229d
4.16.1	Radiation Environmental Monitoring Program (REMP) Sample Collection and Analysis	229-1
4.16.2	REMP - Maximum Values for the Lower Limits of Detection (LLD)	229q
4.16.3	REMP - Reporting Levels for Radioactivity Concentrations in Environmental Samples	229s
6.1.1	Minimum Shift Crew Composition	236

Supplement 2

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 ($110\% \times 1250 = 1375$ psig) and the USAS Code permits pressure transients up to 20 percent over the piping design pressure ($120\% \times 1148 = 1378$ 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.

This page reflects changes proposed July 5, 1995 titled "Surveillance Test Interval/Allowed Outage Time Extension Program - Part 2."

Table 4.2.1 Minimum Test and Calibration Frequency for Core Cooling, Rod Block and Isolation Instrumentation			
Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
<u>ECCS INSTRUMENTATION</u>			
1. Reactor Low-Low Water Level	Once/3 months (Note 5)	Every Operating Cycle - Transmitter Once/3 months - Trip Unit	Once/Shift <i>12 hours</i>
2. Drywell High Pressure	Once/3 months	Once/3 months	None
3. Reactor Low Pressure (Pump Start)	Once/3 months	Once/3 months	None
4. Reactor Low Pressure (Valve Permissive)	Once/3 months	Once/3 months	None
5. Undervoltage Emergency Bus	Refueling Outage	Refueling Outage	None
6. Low Pressure Core Cooling Pumps Discharge Pressure Interlock	Once/3 months	Once/3 months	None
7. Loss of Auxiliary Power	Refueling Outage	Refueling Outage	None
8. Condensate Storage Tank Level	Refueling Outage	Refueling Outage	None
9. Reactor High Water Level	Once/3 months (Note 5)	Every Operating Cycle - Transmitter Every 3 months - Trip Unit	Once/Shift <i>12 hours</i>
<u>ROD BLOCKS</u>			
1. APRM Downscale	Once/3 months (Note 5)	Once/3 months	None
2. APRM Flow Variable	Once/3 months (Note 5)	Once/3 months	None
3. IRM Upscale	Notes (2,5)	Note 2	Note 2
4. IRM Downscale	Notes (2,5)	Note 2	Note 2
5. RBM Upscale	Once/3 months (Note 5)	Once/3 months	None
6. RBM Downscale	Once/3 months (Note 5)	Once/3 months	None
7. SRM Upscale	Notes (2,5)	Note 2	Note 2
8. SRM Detector Not-Full-In Position	Notes (2,9)	Note 2	None
9. Scram Discharge Volume-High Level	Once/3 months	Refueling Outage	None
<u>MAIN STEAM LINE (GROUP 1) ISOLATION</u>			
1. Steam Tunnel High Temperature	Refueling Outage	Refueling Outage	None <i>12 hours</i>
2. Steam Line High Flow	Once/3 months	Once/3 Months	Once/Shift

This page reflects changes proposed July 5, 1995 titled "Surveillance Test Interval/Allowed Outage Time Extension Program - Part 2."

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

Table 4.2.1 - Continued

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

NOTES:

- (1) (Deleted)
- (2) Calibrate prior to normal shutdown and start-up and thereafter check once per ~~shift~~ ^{12 hours} and test once per week until no longer required. Calibration of this instrument prior to normal shutdown means adjustment of channel trips so that they correspond, within acceptable range and accuracy, to a simulated signal injected into the instrument (not primary sensor). In addition, IRM gain adjustment will be performed, as necessary, in the APRM/IRM overlap region.
- (3) Functional tests, calibrations and sensor checks are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.
- (4) Whenever fuel handling is in process, a sensor check shall be performed once per ~~shift~~ ^{12 hours.}
- (5) A functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action.
- (6) (Deleted)
- (7) (Deleted)
- (8) Once/shutdown if not tested during previous 3 month period.
- (9) Testing of the SRM Not-Full-In rod block is not required if the SRM detectors are secured in the full-in position.
- (10) Uses contacts from scram system. Tested and calibrated in accordance with Tables 4.1.1 and 4.1.2.

Bases 3.2 (Continued):

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.

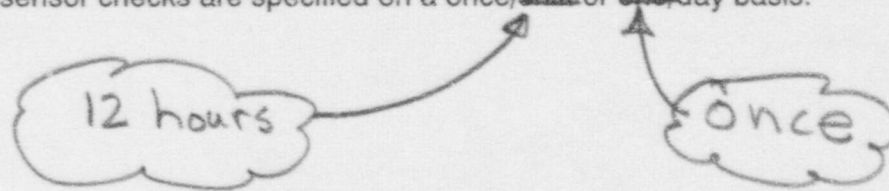
Voltage sensing relays are provided on the safeguards bus to transfer the bus to an alternate source when a loss of voltage condition or a degraded voltage condition is sensed. On loss of voltage this transfer occurs immediately. The transfer on degraded voltage has a time delay to prevent transfer during 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 480V essential MCCs.

Bases 4.2:

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



3.0 LIMITING CONDITIONS FOR OPERATION

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

D. Control Rod Accumulators

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

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

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

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

4.0 SURVEILLANCE REQUIREMENTS

per 12 hours

D. Control Rod Accumulators

Once ~~a shift~~ check the status in the control room of the required Operable accumulator pressure and level alarms.

Bases 3.3/4.3 (Continued):

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,

8-15-96
Supplement 2

To allow for leakage and imperfect mixing.

Bases 3.4/4.4:

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

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

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

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

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

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

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

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

3.0 LIMITING CONDITION FOR OPERATION

- 3. One of the following conditions of inoperability may exist for the period specified:
 - a. One Core Spray subsystem may be inoperable for 7 days, or
 - b. One RHR pump may be inoperable for 30 days, or
 - c. One low pressure pump or valve (Core Spray or RHR) may be inoperable with an ADS valve inoperable for 7 days, or
 - d. One of the two LPCI injection paths may be inoperable for 7 days, or
 - e. Two RHR pumps may be inoperable for 7 days, or
 - f. Both of the LPCI injection paths may be inoperable for 72 hours, or
 - g. HPCI may be inoperable for 14 days, provided RCIC is operable, or
 - h. One ADS valve may be inoperable for 14 days, or
 - i. Two or more ADS valves may be inoperable for 12 hours.
- 4. If the requirements or conditions of 3.5.A.1, 2 or 3 cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be placed in a condition in which the affected equipment is not required to be operable within 24 hours

4.0 SURVEILLANCE REQUIREMENTS

4. Perform the following tests:

<u>Item</u>	<u>Frequency</u>
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 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

OR CONTROL

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

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

3-19-98

3.0 LIMITING CONDITIONS FOR OPERATION

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

Conductivity 5 umho/cm
Chloride ion 0.1 ppm

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

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

Conductivity 5 umho/cm
Chloride ion 0.5 ppm

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

4.0 SURVEILLANCE REQUIREMENTS

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

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

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

delete

once per 12 hours

3.0 LIMITING CONDITIONS FOR OPERATION

4.0 SURVEILLANCE REQUIREMENTS

D. Coolant Leakage

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, reactor coolant system leakage, based on sump monitoring, shall be limited to:
 - a. 5 gpm Unidentified Leakage
 - b. 2 gpm increase in Unidentified Leakage within any 24 hour period
 - c. 20 gpm Identified Leakage
 - d. no pressure boundary leakage
2. With reactor coolant system leakage greater than 3.6.D.1.a or 3.6.D.1.c above, reduce the leakage rate to within acceptable limits within four hours or initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.
3. With an increase in Unidentified Leakage in excess of the rate specified in 3.6.D.1.b, identify the source of increased leakage within four hours or initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.
4. If any Pressure Boundary Leakage is detected when the corrective actions outlined in 3.6.D.2 and 3.6.D.3 above are taken, initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.

3.6/4.6

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

a. Unidentified and Identified Leakage rates shall be recorded once per ~~shift~~ ~~not to exceed~~ 12 hours using primary containment floor and equipment drain sump monitoring equipment.

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

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

b. Primary containment sump leakage measurement system- performance of a sensor check ~~at least~~ once per ~~shift~~ ~~not to exceed~~ 12 hours and a channel calibration test at least once per cycle.

delete

delete

3.0 LIMITING CONDITIONS FOR OPERATION

4. Pressure Suppression Chamber-Drywell Vacuum Breakers

integrity

- a. When primary containment is required, all eight drywell-suppression chamber vacuum breakers shall be operable and positioned in the closed position as indicated by the position indication system, except during testing and except as specified in 3.7.A.4.b through 3.7.A.4.d below.
- b. Any drywell-suppression chamber vacuum breaker may be nonfully closed as indicated by the position indication and alarm system provided that drywell to suppression chamber differential pressure decay does not exceed that shown on Figure 3.7.1
- c. Up to two drywell-suppression chamber vacuum breakers may be inoperable provided that: (1) the vacuum breakers are determined to be fully closed and at least one position alarm circuit is operable or (2) the vacuum breaker is secured in the closed position or replaced by a blank flange.
- d. Drywell-suppression chamber vacuum breakers may be cycled, one at a time, during containment inerting and deinerting operations to assist in purging air or nitrogen from the suppression chamber vent header.

4.0 SURVEILLANCE REQUIREMENTS

4. Pressure Suppression Chamber-Drywell Vacuum Breakers

- a. Operability and full closure of the drywell-suppression chamber vacuum breakers shall be verified by performance of the following:
 - (1) Monthly each operable drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle.
 - (2) Once each operating cycle, drywell to suppression chamber leakage shall be demonstrated to be less than that equivalent to a one-inch diameter orifice and each vacuum breaker shall be visually inspected. (Containment access required)
 - (3) Once each operating cycle, vacuum breaker position indication and alarm systems shall be calibrated and functionally tested. (Containment access required)
 - (4) Once each operating cycle, the vacuum breakers shall be tested to determine that the force required to open each valve from fully closed to fully open does not exceed that equivalent to 0.5 psi acting on the suppression chamber face of the valve disc. (Containment access required)

Bases 4.7 (Continued):

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

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

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

except testing should be IAW ASTM D 3803-1989.

Bases 4.7 (Continued):

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

8-15-96
Supplement 2

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

Notes:

- a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal. Note (a) of Table 4.8.3 is applicable.
- b. Grab samples taken at the discharge of the plant stack and reactor building vent are generally below minimum detectable levels for most nuclides with existing analytical equipment. For this reason, isotopic analysis data, corrected for holdup time, for samples taken at the steam jet air ejector may be used to calculate noble gas ratios.
- c. Whenever the steady state radioiodine concentration is greater than 10 percent of the limit of Specification 3.6.C.1, daily sampling of reactor coolant for radioactive iodines of I-131 through I-135 is required. Whenever a change of 25% or more in calculated Dose Equivalent I-131 is detected under these conditions, the iodine and particulate collection devices for all release points shall be removed and analyzed daily until it is shown that a pattern exists which can be used to predict the release rate. Sampling may then revert to weekly. When samples collected for one day are analyzed, the corresponding LLD's may be increased by a factor of 10. Samples shall be analyzed within 48 hours after removal.
- d. To be representative of the average quantities and concentrations of radioactive materials in particulate form in gaseous effluents, samples should be collected in proportion to the rate of flow of the effluent streams.
- e. The principal gamma emitters for which the LLD specification will apply are exclusively the following 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.

3.8/4.8

g

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CB-15-96

3.0 LIMITING CONDITION FOR OPERATION

4.0 SURVEILLANCE REQUIREMENTS

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- 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

add

3.0 LIMITING CONDITIONS FOR OPERATION

4.0 SURVEILLANCE REQUIREMENTS

3.13 FIRE DETECTION AND PROTECTION SYSTEMS

4.13 FIRE DETECTION AND PROTECTION SYSTEMS

Applicability:

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

Applicability:

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

Objective:

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

Objective:

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

Specification:

Specification:

A. Fire Detection Instrumentation

A. Fire Detection Instrumentation

1. Except as specified below, the minimum fire detection instrumentation for each fire detection zone shown in Table 3.13.1 shall be operable whenever equipment in that fire detection zone is required to be operable.
2. If specification 3.13.A.1 cannot be met, within one hour establish a fire watch patrol to inspect the zone(s) with inoperable instruments ~~at least~~ once per hour. Restore the minimum number of instruments to operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of the inoperability and the plans and schedule for restoring the instruments to operable status.

1. Fire detection instrumentation in each of the zones in Table 3.13.1 shall be demonstrated operable every six months by performance of functional tests.
2. Alarm ~~circuitry~~ associated with the fire detector instruments in each of the zones in Table 3.13.1 shall be demonstrated operable every six months.

delete

circuitry

(+25%)

3-19-98

3.0 LIMITING CONDITIONS FOR OPERATION

4.0 SURVEILLANCE REQUIREMENTS

F. Halon Systems

- 1. The cable spreading room Halon system shall be operable with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure.
~~a. Each valve (manual, power operated, or full charge pressure.~~
- 2. If specification 3.13.F.1 cannot be met, within one hour establish a continuous fire watch with backup fire suppression equipment in the cable spreading room. Restore the system to operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of the inoperability and the plans and schedule for restoring the system to operable status.

Delete

G. Penetration Fire Barriers

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

of (+25%)

3.13/4.13

F. Halon Systems

- 1. The cable spreading room Halon system shall be demonstrated operable as follows:
 - a. Each valve (manual, power operated, or automatic) in the flow path that is not electrically supervised, locked, sealed or otherwise secured in position, shall be verified to be in its correct position every month.
 - b. Verify Halon storage tank weight and pressure every six months.
 - c. Perform a system functional test every 18 months which includes verifying the system, including associated ventilation dampers, actuates manually and automatically, upon receipt of a test signal
 - d. Perform an air flow test every 3 years through headers and nozzles to assure no blockage.
 - e. Visually examine headers and nozzles every 18 months. An air flow test shall be performed upon evidence of obstruction of any Halon system nozzle.

G. Penetration Fire Barriers

- 1. A visual inspection of penetration fire barriers in fire area boundaries protecting safe shutdown equipment shall be conducted every 18 months.
- 2. Following repair or maintenance of a penetration fire barrier a visual inspection of the shall be conducted.

Amendment No. 7, AB, 61

227b REV

~~CONFIDENTIAL~~

3.0 LIMITING CONDITIONS FOR OPERATION

4.0 SURVEILLANCE REQUIREMENTS

II. Alternate Shutdown System

II. Alternate Shutdown System

1. The system controls on the ASDS panel ^{system/component} shall be operable whenever that ~~system~~ ~~is~~ ~~controls~~ are required to be operable, ~~from the control room.~~
2. If system controls required to be operable by Specification 3.13.H.1 are made or found inoperable, restore the inoperable system control to operable within 7 days, or perform one of the following;
 - a. Provide equivalent shutdown capability and within 60 days restore the inoperable system controls to operable; or
 - b. Establish a continuous fire watch in the cable spreading room and the back-panel area of the control room and within 60 days restore the inoperable system controls to operable; or
 - c. Verify the operability of the fire detectors in the cable spreading room and the back-panel area of the control room and establish a hourly fire watch patrol and within 60 days restore the inoperable system controls to operable; or
 - d. Place the reactor in a condition where the systems for which the system controls at the ASDS are inoperable are not required to be operable within 24 hours.
3. The alternate shutdown system panel master transfer switch shall be locked in the normal position except when in use, being tested or being maintained.

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.

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8-15-96

TABLE 3.13.1
SAFETY RELATED FIRE DETECTION INSTRUMENTS

Detection

Fire Zone	Location	Minimum Instruments Operable		
		Heat	Flame	Smoke
1A	"B" RHR Room			3
1B	"A" RHR Room			3
1C	RCIC Room			2
1E	HPCI Room			11
1F	Reactor Building-Torus Compartment			1
2A	Reactor Bldg. 935' elev - TIP Drive Area			10
2B	Reactor Bldg. 935' elev - CRD HCU Area East			11
2C	Reactor Bldg. 935' elev - CRD HCU Area West			1
2E 2G/2H	Reactor Bldg. 935' - LPCI Injection Valve Area			2
3B	Reactor Bldg. 962' elev - SBLC Area			5
3C	Reactor Bldg. 962' elev - South			4
3D	Reactor Bldg. 962' elev - RBCCW Pump Area			4
4A	Reactor Bldg. 985' elev - South			5
4B	Reactor Bldg. 985' elev - RBCCW Hx Area			2
4D	SBG T System Room			7
5A	Reactor Bldg. 1001' elev - South			3
5B	Reactor Bldg. 1001' elev - North			1
5C	Reactor Bldg. - Fuel Pool Cooling Pump Area			5
6	Reactor Building 1027' elev			1
7A	Battery Room			1
7B	Battery Room			1
7C	Battery Room			7
8	Cable Spreading Room			

8-15-96
 Supplement 2

Detection
 Fire Zone

TABLE 3:13.1
 SAFETY RELATED FIRE DETECTION INSTRUMENTS

Fire Zone	Location	Minimum Instruments Operable		
		Heat	Flame	Smoke
12A	Turbine Bldg. - 911' - 4.16 KV Switchgear			3
13C	Turbine Bldg. - 911' ^{elev} - MCC 133 Area			1
14A	Turbine Bldg. - 931' - 4.16 KV Switchgear			2
15A /ISC	#12 DG Room & Day Tank Room		3	
15B /ISD	#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

elev (Supplement 2)

8-15-96

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~~ 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.

6.7.D

3.14/4.14

3-19-98

Table 3.14.1 (continued)

Instrumentation for Accident Monitoring

* Required Conditions (continued)

- B. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, the minimum number of channels shall be restored to operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours and Cold Shutdown within the following 24 hours.
- C. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, the torus temperature shall be monitored ~~at least once per shift~~ to observe any unexplained temperature increase which might be indicative of an open SRV; the minimum number of channels shall be restored to operable status within 30 days or be in at least Hot Shutdown within the next 12 hours and Cold Shutdown within the following 24 hours.
- D. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, initiate the preplanned alternate method of monitoring the appropriate parameters in addition to submitting the report required in (A) above.

once per 12 hours (+25%)

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.

8-15-96

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, reasonable 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:

6.7.C.2

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

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

3-19-98

3.0 LIMITING CONDITIONS FOR OPERATION

4.0 SURVEILLANCE REQUIREMENTS

3.17 CONTROL ROOM HABITABILITY

4.17 CONTROL ROOM HABITABILITY

Applicability:

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

Applicability:

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

Objectives:

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

Objectives:

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

Specification:

Specification:

A. Control Room Ventilation System

A. Control Room Ventilation System

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

- 1. ~~At least once per shift,~~ check control room temperature.

Once per 12 hours

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

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

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

activities

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 Site Superintendent~~ ^{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.

(Supplement 2) responsibilities

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

Supplement 2

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

(Supplement 2) responsibility

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.

8-15-96
Supplement 2

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~~ ^{(Supplement 2) (onsite)} when fuel is in the reactor.
5. All alterations of the reactor core shall be directly supervised by a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.
6. A fire brigade of at least five members shall be maintained ~~on site~~ ^{(Supplement 2) (onsite)} at all times.* The fire brigade shall not include the three members of the shift organization required for safe shutdown of the reactor from outside the control room.
7. The General Superintendent, Operations shall be formerly licensed as a Senior Reactor Operator or hold a current Senior Reactor Operator License.
8. At least one member of plant management holding a current Senior Reactor Operator License shall be assigned to the plant operations group on a long term basis (approximately two years). This individual will not be assigned to a rotating shift.

General Superintendent
Radiation Services

- 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 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.

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

8-15-96

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 days,
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.

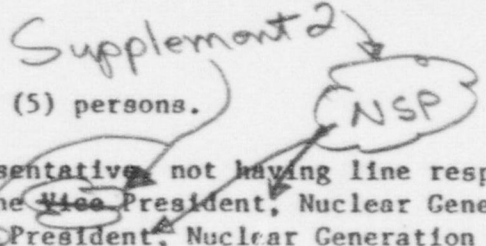
Organizational units for the review and audit of facility operations shall be constituted and have the responsibilities and authorities outlined below:

A. Safety Audit Committee (SAC)

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

1. Membership

- a. The SAC shall consist of at least five (5) persons.
- b. The SAC Chairman shall be an NSP representative not having line responsibility for operation of the plant, appointed by the Vice President, Nuclear Generation. Other members shall be appointed by the Vice President, Nuclear Generation or by such other person as he may designate. The Chairman shall appoint a Vice Chairman from the SAC membership to act in his absence.
- c. No more than two members of the SAC shall be from groups holding line responsibility for operation of the plant.
- d. A SAC member may appoint an alternate to serve in his absence, with concurrence of the Chairman. No more than one alternate shall serve on the SAC at any one time. The alternate member shall have voting rights.



2. Qualifications

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

Amendment No. 3, 16, 46

- f. Investigation of all Reportable Events and Events requiring Special Reports to the Commission.
 - g. Revisions to the Facility Emergency Plan, the Facility Security Plan, and the Fire Protection Program.
 - h. Operations Committee minutes to determine if matters considered by that Committee involve unresolved or unresolved safety questions.
 - i. Other nuclear safety matters referred to the SAC by the Operations Committee, plant management or company management.
 - j. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components.
 - k. Reports of special inspections and audits conducted in accordance with specification 6.3.
 - l. Changes to the Offsite Dose Calculation Manual (ODCM).
 - m. Review of investigative reports of unplanned releases of radioactive material to the environs.
6. Audit - The operation of the nuclear power plant shall be audited formally under the cognizance of the SAC to assure safe facility operation.
- a. Audits of selected aspects of plant operation, as delineated in ANSI N18.7-1976 as modified by the Operational Quality Assurance Plan, shall be performed with a frequency commensurate with their nuclear safety significance and in a manner to assure that an audit of all nuclear safety-related activities is completed within a period of two years. The audits shall be performed in accordance with appropriate written instructions and procedures.
 - b. Audits of aspects of plant radioactive effluent treatment and radiological environmental monitoring shall be performed as follows:
 - 1. Implementation of the Offsite Dose Calculation Manual and quality controls for effluent monitoring at least once every two years.
 - 2. Implementation of the Process Control Program for solidification of radioactive waste at least once every two years.
 - 3. The Radiological Environmental Monitoring Program and the results thereof, including quality controls, at least once every year.
 - c. Periodic review of the audit program should be performed by the SAC at least twice a year to assure its adequacy.
 - d. Written reports of the audits shall be reviewed by the ~~Vice~~ President Nuclear Generation, by the SAC at a scheduled meeting, and by members of Management having responsibility in the areas audited.

Supplement 2

NSP

8-15-96
Supplement 2

Supplement 2

7. Authority

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

8. Records

Minutes shall be prepared and retained for all scheduled meetings of the Safety Audit Committee. The minutes shall be distributed within one month of the meeting to the Vice President, 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 distribution of reports and minutes prepared by the group to others in the NSP Organization.

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8-15-96
Supplement 2

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.

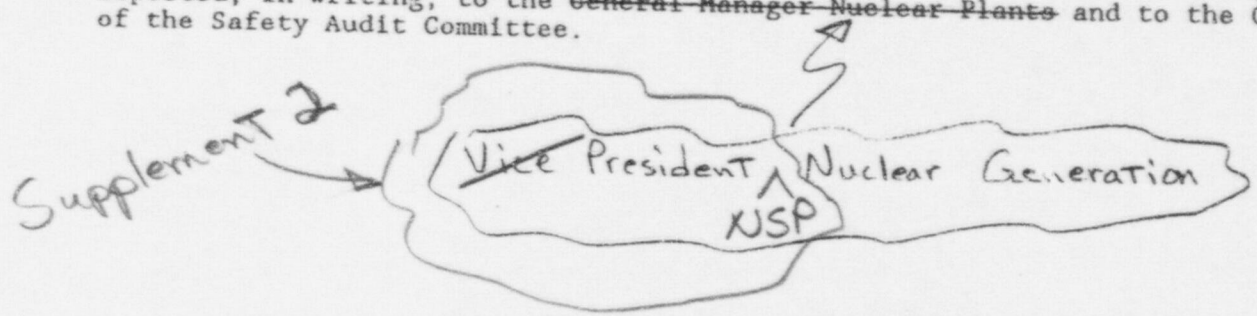
3. Quorum

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

4. Responsibilities - The following subjects shall be 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~~ and to the Chairman of the Safety Audit Committee.

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241
REV 119-9/12/89

- f. Investigation of all Reportable Events and Events requiring Special Reports to the Commission.
- g. Drills on emergency procedures (including plant evacuation) and adequacy of communication with off-site support groups.
- h. All procedures required by these Technical Specifications, including implementing procedures of the Emergency Plan and the Security Plan (except as exempted in Section 6.5.F), shall be reviewed with a frequency commensurate with their safety significance but at an interval of not more than two years.
- i. Perform special reviews and investigations, as requested by the Safety Audit Committee.
- j. Review of investigative reports of unplanned releases of radioactive material to the environs.
- k. All changes to the Process Control Program (PCP) and the Offsite Dose Calculation Manual (ODCM).

5. Authority

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

6. Records

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

7. Procedures

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

- a. Responsibility and authority of the group.
- b. Content and method of submission of presentations to the Operations Committee.

Supplement 2
NSP
NSP

- 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.

Supplement 2

NSP

Vice President Nuclear Generation

6.4 Action to be Taken if a Safety Limit is Exceeded

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

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

General Superintendent
Radiation Services.

8-15-96

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:

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etermined

- (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 ~~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 control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified ~~by the unit Health Physicist in the Radiation Work Permit.~~

radiation
protection

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.

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E. On-site 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:

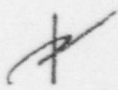
- 1. Shall be submitted to the Commission with the Semi-Annual Radioactive Effluent release report for the period in which the change(s) were made effective. This submittal shall contain:
 - a. sufficiently detailed information to totally support the rationale for the change without benefit of additionally 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. 



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General Superintendent, Radiation Services

8-15-96

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B. Records Retained for Plant Life (continued)

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

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

§

B. Reportable Events

The following actions shall be taken for Reportable Events:

- a. The Commission shall be notified by a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50 and,
- b. Each Reportable Event shall be reviewed by the Operations Committee and the results of this review shall be submitted to the Safety Audit Committee and the Vice President Nuclear Generation.

NSP
Supplement 2

Exhibit C

Revised Technical Specification Pages

Monticello Nuclear Generating Plant

Supplement 2 to License Amendment Request Dated
August 15, 1996

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

Page

i, ii, iii, iv, v
vi, vii, 22, 31, 61, 62
63a, 69, 72, 82, 89
99, 102, 125, 126, 164
188, 190, 198t, 200
223, 227b, 227c, 227d, 227e
229b, 229c, 229ff, 229i, 229u
232, 233, 234, 237, 239
240, 241, 242, 243, 244a, 246b
247a, 250

TABLE OF CONTENTS

		<u>Page</u>
1.0	DEFINITIONS	1
2.0	SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS	
2.1 and 2.3	Fuel Cladding Integrity	6
2.1	Bases	10
2.3	Bases	14
2.2 and 2.4	Reactor Coolant System	21
2.2	Bases	22
2.4	Bases	24
3.0	LIMITING CONDITIONS FOR OPERATION AND 4.0 SURVEILLANCE REQUIREMENTS	
4.0	Surveillance Requirements	25a
4.0	Bases	25b
3.1 and 4.1	Reactor Protection System	26
3.1	Bases	35
4.1	Bases	42
3.2 and 4.2	Protective Instrumentation	45
A.	Primary Containment Isolation Functions	45
B.	Emergency Core Cooling Subsystems Actuation	46
C.	Control Rod Block Actuation	46
D.	Other Instrumentation	46a
E.	Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation	47
F.	Recirculation Pump Trip Initiation and Alternate Rod Injection Initiation	48
G.	Safeguards Bus Voltage Protection	48
H.	Instrumentation for S/RV Low-Low Set Logic	48
I.	Instrumentation for Control Room Habitability Protection	48
3.2	Bases	64
4.2	Bases	72
3.3 and 4.3	Control Rod System	76
A.	Reactivity Limitations	76
B.	Control Rod Withdrawal	77
C.	Scram Insertion Times	81
D.	Control Rod Accumulators	82
E.	Reactivity Anomalies	83
F.	Scram Discharge Volume	83a
G.	Required Action	83a
3.3 and 4.3	Bases	84

TABLE OF CONTENTS (Cont'd)

		<u>Page</u>
3.4 and 4.4	Standby Liquid Control System	93
	A. System Operation	93
	B. Boron Solution Requirements	95
	C.	96
	3.4 and 4.4 Bases	99
3.5 and 4.5	Core and Containment/Spray Cooling Systems	101
	A. ECCS Systems	101
	B. RHR Intertie Return Line Isolation Valves	103
	C. Containment Spray/Cooling System	104
	D. RCIC	105
	E. Cold Shutdown and Refueling Requirements	106
	F. Recirculation System	107
	3.5/4.5 Bases	110
3.6 and 4.6	Primary System Boundary	121
	A. Reactor Coolant Heatup and Cooldown	121
	B. Reactor Vessel Temperature and Pressure	122
	C. Coolant Chemistry	123
	D. Coolant Leakage	126
	E. Safety/Relief Valves	127
	F. Deleted	
	G. Jet Pumps	128
	H. Snubbers	129
	3.6 and 4.6 Bases	145
3.7 and 4.7	Containment Systems	156
	A. Primary Containment	156
	B. Standby Gas Treatment System	166
	C. Secondary Containment	169
	D. Primary Containment Isolation Valves	170
	E. Combustible Gas Control System	172
	3.7 Bases	175
	4.7 Bases	183

TABLE OF CONTENTS (Cont'd)

	<u>Page</u>
3.8 and 4.8 Radioactive Effluents	192
A. Liquid Effluents	192
B. Gaseous Effluents	197
C. Solid Radioactive Waste	198e
D. Dose from All Uranium Fuel Cycle Sources	198f
3.8 and 4.8 Bases	198u
3.9 and 4.9 Auxiliary Electrical Systems	199
A. Operational Requirements for Startup	199
B. Operational Requirements for Continued Operation	200
1. Transmission Lines	200
2. Reserve Transformers	201
3. Standby Diesel Generators	201
4. Station Battery System	203
5. 24V Battery Systems	203
3.9 Bases	204
4.9 Bases	205
3.10 and 4.10 Refueling	206
A. Refueling Interlocks	206
B. Core Monitoring	207
C. Fuel Storage Pool Water Level	207
D. Movement of Fuel	207
E. Extended Core and Control Rod Drive Maintenance	208
3.10 and 4.10 Bases	209
3.11 and 4.11 Reactor Fuel Assemblies	211
A. Average Planar Linear Heat Generation Rate	211
B. Linear Heat Generation Rate	212
C. Minimum Critical Power Ratio	213
3.11 Bases	216
4.11 Bases	218
3.12 and 4.12 Sealed Source Contamination	219
A. Contamination	219
B. Records	221
3.12 and 4.12 Bases	222

TABLE OF CONTENTS (Cont'd)

	<u>Page</u>
3.13 and 4.13 Fire Detection Protection Systems	223
A. Fire Detection Instrumentation	223
B. Fire Suppression Water System	224
C. Hose Stations	226
D. Yard Hydrant Hose Houses	227
E. Sprinkler Systems	227a
F. Halon Systems	227b
G. Penetration Fire Barriers	227b
H. Alternate Shutdown System	227c
3.13 Bases	228
4.13 Bases	228b
3.14 and 4.14 Accident Monitoring Instrumentation	229a
3.14 and 4.14 Bases	229e
3.15 and 4.15 Inservice Inspection and Testing	229f
3.15 and 4.15 Bases	229g
3.16 and 4.16 Radiation Environmental Monitoring Program	229h
A. Sample Collection & Analysis	229h
B. Land Use Census	229j
C. Interlaboratory Comparison Program	229k
3.16 and 4.16 Bases	229t
3.17 and 4.17 Control Room Habitability	229u
A. Control Room Ventilation System	229u
B. Control Room Emergency Filtration System	229v
3.17 Bases	229y
4.17 Bases	229z
5.0 DESIGN FEATURES	230
5.1 Site	230
5.2 Reactor	230
5.3 Reactor Vessel	230
5.4 Containment	230
5.5 Fuel Storage	231
5.6 Seismic Designs	231
6.0 ADMINISTRATIVE CONTROLS	232
6.1 Organization	232
6.2 Review and Audit	237
6.3 Special Inspection and Audits	243
6.4 Action to be taken if a Safety Limit is Exceeded	243
6.5 Plant Operating Procedures	244
6.6 Plant Operating Records	246c
6.7 Reporting Requirements	248

LIST OF FIGURES

<u>Figure No.</u>		<u>Page</u>
3.4.1	Sodium Pentaborate Solution Volume-Concentration Requirements	97
3.4.2	Sodium Pentaborate Solution Temperature Requirements	98
3.6.1	Core Beltline Operating Limits Curve Adjustment vs. Fluence	133
3.6.2	Minimum Temperature vs. Pressure for Pressure Tests	134
3.6.3	Minimum Temperature vs. Pressure for Mechanical Heatup or Cooldown Without the Core Critical	135
3.6.4	Minimum Temperature vs. Pressure for Core Operation	136
4.6.2	Chloride Stress Corrosion Test Results @ 500°F	137
3.7.1	Differential Pressure Decay Between the Drywell and Wetwell	191
3.8.1	Monticello Nuclear Generating Plant Site Boundary for Liquid Effluents	198g
3.8.2	Monticello Nuclear Generating Plant Site Boundary for Gaseous Effluents	198h

LIST OF TABLES

<u>Table No.</u>		<u>Page</u>
3.1.1	Reactor Protection System (Scram) Instrument Requirements	28
4.1.1	Scram Instrument Functional Tests - Minimum Functional Test Frequencies for Safety Instrumentation and Control Circuits	32
4.1.2	Scram Instrument Calibration - Minimum Calibration Frequencies for Reactor Protection Instrument Channels	34
3.2.1	Instrumentation that Initiates Primary Containment Isolation Functions	49
3.2.2	Instrumentation that Initiates Emergency Core Cooling Systems	52
3.2.3	Instrumentation that Initiates Rod Block	56
3.2.4	Instrumentation that Initiates Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation	59
3.2.5	Instrumentation That Initiates a Recirculation Pump Trip and Alternate Rod Injection	60
3.2.6	Instrumentation for Safeguards Bus Degraded Voltage and Loss of Voltage Protection	60a
3.2.7	Instrumentation for Safety/Relief Valve Low-Low Set Logic	60b
3.2.8	Other Instrumentation	60d
3.2.9	Instrumentation for Control Room Habitability Protection	60e
4.2.1	Minimum Test and Calibration Frequency for Core Cooling, Rod Block and Isolation Instrumentation	61
4.6.1	Snubber Visual Inspection Interval	132a
3.8.1	Radioactive Liquid Effluent Monitoring Instrumentation	198i
3.8.2	Radioactive Gaseous Effluent Monitoring Instrumentation	198k
4.8.1	Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements	198m
4.8.2	Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements	198n
4.8.3	Radioactive Liquid Waste Sampling and Analysis Program	198p
4.8.4	Radioactive Gaseous Waste Sampling and Analysis Program	198s

LIST OF TABLES (Cont'd)

<u>Table No.</u>		<u>Page</u>	
3.13.1	Safety Related Fire Detection Instruments	227d	
3.14.1	Instrumentation for Accident Monitoring	229b	
4.14.1	Minimum Test and Calibration Frequency for Accident Monitoring Instrumentation	229d	
4.16.1	Radiation Environmental Monitoring Program (REMP) Sample Collection and Analysis	229-l	
4.16.2	REMP - Maximum Values for the Lower Limits of Detection (LLD)	229q	
4.16.3	REMP - Reporting Levels for Radioactivity Concentrations in Environmental Samples	229s	
6.1.1	Minimum Shift Crew Composition	236	

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 ($110\% \times 1250 = 1375$ psig) and the USAS Code permits pressure transients up to 20 percent over the piping design pressure ($120\% \times 1148 = 1378$ psig).

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

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

Table 3.1.1 - Continued

- e. The high drywell pressure scram functions in the Startup and Run modes when necessary during purging for containment inerting or de-inerting only by closing the manual containment isolation valves. Verification of the bypass condition shall be noted in the control room log.
- f. One instrument channel for the functions indicated in the table to allow completion of surveillance testing, provided that:
 - 1. Redundant instrument channels in the same trip system are capable of initiating the automatic function and are demonstrated to be operable either immediately prior or immediately subsequent to applying the bypass.
 - 2. While the bypass is applied, surveillance testing shall proceed on a continuous basis and the remaining instrument channels initiating the same function are tested prior to any other. Upon completion of surveillance testing, the bypass is removed.

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

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

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

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

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

NOTES:

- (1) (Deleted)
- (2) Calibrate prior to normal shutdown and start-up and thereafter check once per 12 hours and test once per week until no longer required. Calibration of this instrument prior to normal shutdown means adjustment of channel trips so that they correspond, within acceptable range and accuracy, to a simulated signal injected into the instrument (not primary sensor). In addition, IRM gain adjustment will be performed, as necessary, in the APRM/IRM overlap region.
- (3) Functional tests, calibrations and sensor checks are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.
- (4) Whenever fuel handling is in process, a sensor check shall be performed once per 12 hours.
- (5) A functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action.
- (6) (Deleted)
- (7) (Deleted)
- (8) Once/shutdown if not tested during previous 3 month period.
- (9) Testing of the SRM Not-Full-In rod block is not required if the SRM detectors are secured in the full-in position.
- (10) Uses contacts from scram system. Tested and calibrated in accordance with Tables 4.1.1 and 4.1.2.

Bases 3.2 (Continued):

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

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

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

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

Bases 4.2:

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

3.0 LIMITING CONDITIONS FOR OPERATION

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

D. Control Rod Accumulators

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

1. In the Startup or Run Mode, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:
 - (a) Inoperable accumulator, or
 - (b) Directional control valve electrically disarmed while in a non-fully inserted position.

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

4.0 SURVEILLANCE REQUIREMENTS

D. Control Rod Accumulators

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

Bases 3.3/4.3 (Continued):

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

C. Scram Insertion Times

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

Bases 3.4/4.4:

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

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

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

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

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

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

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

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

3.0 LIMITING CONDITIONS FOR OPERATION

3. One of the following conditions of inoperability may exist for the period specified:
 - a. One Core Spray subsystem may be inoperable for 7 days, or
 - b. One RHR pump may be inoperable for 30 days, or
 - c. One low pressure pump or valve (Core Spray or RHR) may be inoperable with an ADS valve inoperable for 7 days, or
 - d. One of the two LPCI injection paths may be inoperable for 7 days, or
 - e. Two RHR pumps may be inoperable for 7 days, or
 - f. Both of the LPCI injection paths may be inoperable for 72 hours, or
 - g. HPCI may be inoperable for 14 days, provided RCIC is operable, or
 - h. One ADS valve may be inoperable for 14 days, or
 - i. Two or more ADS valves may be inoperable for 12 hours.
4. If the requirements or conditions of 3.5.A.1, 2 or 3 cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be placed in a condition in which the affected equipment is not required to be operable within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

4. Perform the following tests:

<u>Item</u>	<u>Frequency</u>
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
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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
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5. Perform the following test on the Core Spray Δp Instrumentation:

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

3.0 LIMITING CONDITIONS FOR OPERATION

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

Conductivity	5 μ mho/cm
Chloride ion	0.1 ppm

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

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

Conductivity	5 μ mho/cm
Chloride ion	0.5 ppm

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

4.0 SURVEILLANCE REQUIREMENTS

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

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

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

3.0 LIMITING CONDITIONS FOR OPERATION

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, reactor coolant system leakage, based on sump monitoring, shall be limited to:
 - a. 5 gpm Unidentified Leakage
 - b. 2 gpm increase in Unidentified Leakage within any 24 hour period
 - c. 20 gpm Identified Leakage
 - d. no pressure boundary leakage
2. With reactor coolant system leakage greater than 3.6.D.1.a or 3.6.D.1.c above, reduce the leakage rate to within acceptable limits within four hours or initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.
3. With an increase in Unidentified Leakage in excess of the rate specified in 3.6.D.1.b, identify the source of increased leakage within four hours or initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.
4. If any Pressure Boundary Leakage is detected when the corrective actions outlined in 3.6.D.2 and 3.6.D.3 above are taken, initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.

3.6/4.6

4.0 SURVEILLANCE REQUIREMENTS

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, the following surveillance program shall be carried out:
 - a. Unidentified and Identified Leakage rates shall be recorded once per 12 hours using primary containment floor and equipment drain sump monitoring equipment.
2. The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:
 - a. Primary containment atmosphere particulate monitoring systems-performance of a sensor check once per 12 hours, a channel functional test at least monthly and a channel calibration at least once per cycle.
 - b. Primary containment sump leakage measurement system-performance of a sensor check once per 12 hours and a channel calibration test at least once per cycle.

126

Amendment No. 15, 17, 87

3.0 LIMITING CONDITIONS FOR OPERATION

4. Pressure Suppression Chamber-Drywell Vacuum Breakers
- a. When primary containment integrity is required, all eight drywell-suppression chamber vacuum breakers shall be operable and positioned in the closed position as indicated by the position indication system, except during testing and except as specified in 3.7.A.4.b through 3.7.A.4.d below.
 - b. Any drywell-suppression chamber vacuum breaker may be nonfully closed as indicated by the position indication and alarm system provided that drywell to suppression chamber differential pressure decay does not exceed that shown on Figure 3.7.1
 - c. Up to two drywell-suppression chamber vacuum breakers may be inoperable provided that: (1) the vacuum breakers are determined to be fully closed and at least one position alarm circuit is operable or (2) the vacuum breaker is secured in the closed position or replaced by a blank flange.
 - d. Drywell-suppression chamber vacuum breakers may be cycled, one at a time, during containment inerting and deinerting operations to assist in purging air or nitrogen from the suppression chamber vent header.

4.0 SURVEILLANCE REQUIREMENTS

4. Pressure Suppression Chamber-Drywell Vacuum Breakers
- a. Operability and full closure of the drywell-suppression chamber vacuum breakers shall be verified by performance of the following:
 - (1) Monthly each operable drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle.
 - (2) Once each operating cycle, drywell to suppression chamber leakage shall be demonstrated to be less than that equivalent to a one-inch diameter orifice and each vacuum breaker shall be visually inspected. (Containment access required)
 - (3) Once each operating cycle, vacuum breaker position indication and alarm systems shall be calibrated and functionally tested. (Containment access required)
 - (4) Once each operating cycle, the vacuum breakers shall be tested to determine that the force required to operate each valve from fully closed to fully open does not exceed that equivalent to 0.5 psi acting on the suppression chamber face of the valve disc. (Containment access required.)

Bases 4.7 (Continued):

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

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

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

Bases 4.7 (Continued):

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

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

E. Combustible Gas Control System

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

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

Notes:

- a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal. Note (a) of Table 4.8.3 is applicable.
- b. Grab samples taken at the discharge of the plant stack and reactor building vent are generally below minimum detectable levels for most nuclides with existing analytical equipment. For this reason, isotopic analysis data, corrected for holdup time, for samples taken at the steam jet air ejector may be used to calculate noble gas ratios.
- c. Whenever the steady state radioiodine concentration is greater than 10 percent of the limit of Specification 3.6.C.1, daily sampling of reactor coolant for radioactive iodines of I-131 through I-135 is required. Whenever a change of 25% or more in calculated Dose Equivalent I-131 is detected under these conditions, the iodine and particulate collection devices for all release points shall be removed and analyzed daily until it is shown that a pattern exists which can be used to predict the release rate. Sampling may then revert to weekly. When samples collected for one day are analyzed, the corresponding LLD's may be increased by a factor of 10. Samples shall be analyzed within 48 hours after removal.
- d. To be representative of the average quantities and concentrations of radioactive materials in particulate form in gaseous effluents, samples should be collected in proportion to the rate of flow of the effluent streams.
- e. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.
- f. Nuclides which are below the LLD for the analyses shall be reported as "less than" the LLD of the nuclide and should not be reported as being present at the LLD level for that nuclide. The "less than" values shall not be used in the required dose calculations. When unusual circumstances result in LLD's higher than reported, the reasons shall be documented in the semiannual effluent report.
- g. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period sampled.
- h. H³ analysis shall not be required prior to purging if the limits of 3.8.B.1 are satisfied for other nuclides. However, the H³ analysis shall be completed within 24 hours after sampling.
- i. In lieu of grab samples, continuous monitoring with bi-weekly analysis using silica-gel samplers may be provided.

3.0 LIMITING CONDITIONS FOR OPERATION

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 service providing both the emergency diesel generators are operable.

4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

3.13 FIRE DETECTION AND PROTECTION SYSTEMS

Applicability:

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

Objective:

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

Specification:

A. Fire Detection Instrumentation

1. Except as specified below, the minimum fire detection instrumentation for each fire detection zone shown in Table 3.13.1 shall be operable whenever equipment in that fire detection zone is required to be operable.
2. If specification 3.13.A.1 cannot be met, within one hour establish a fire watch patrol to inspect the zone(s) with inoperable instruments once per hour (+ 25%). Restore the minimum number of instruments to operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of the inoperability and the plans and schedule for restoring the instruments to operable status.

3.13/4.13

4.0 SURVEILLANCE REQUIREMENTS

4.13 FIRE DETECTION AND PROTECTION SYSTEMS

Applicability:

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

Objective:

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

Specification:

A. Fire Detection Instrumentation

1. Fire detection instrumentation in each of the zones in Table 3.13.1 shall be demonstrated operable every six months by performance of functional tests.
2. Alarm circuitry associated with the fire detector instruments in each of the zones in Table 3.13.1 shall be demonstrated operable every six months.

3.0 LIMITING CONDITIONS FOR OPERATION

F. Halon Systems

1. The cable spreading room Halon system shall be operable with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure.
2. If specification 3.13.F.1 cannot be met, within one hour establish a continuous fire watch with backup fire suppression equipment in the cable spreading room. Restore the system to operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of the inoperability and the plans and schedule for restoring the system to operable status.

G. Penetration Fire Barriers

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

3.13/4.13

4.0 SURVEILLANCE REQUIREMENTS

F. Halon Systems

1. The cable spreading room Halon system shall be demonstrated operable as follows:
 - a. Each valve (manual, power operated, or automatic) in the flow path that is not electrically supervised, locked, sealed or otherwise secured in position, shall be verified to be in its correct position every month.
 - b. Verify Halon storage tank weight and pressure every six months.
 - c. Perform a system functional test every 18 months which includes verifying the system, including associated ventilation dampers, actuates manually and automatically, upon receipt of a test signal.
 - d. Perform an air flow test every 3 years through headers and nozzles to assure no blockage.
 - e. Visually examine headers and nozzles every 18 months. An air flow test shall be performed upon evidence of obstructions of any Halon system nozzle.

G. Penetration Fire Barriers

1. A visual inspection of penetration fire barriers in fire area boundaries protecting safe shutdown equipment shall be conducted every 18 months.
2. Following repair or maintenance of a penetration fire barrier a visual inspection of the seal shall be conducted.

227b

Amendment No. 7, 46, 61

3.0 LIMITING CONDITIONS FOR OPERATION

H. Alternate Shutdown System

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

4.0 SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.13.1 (Continued)
SAFETY RELATED FIRE DETECTION INSTRUMENTS

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

Table 3.14.1

Instrumentation for Accident Monitoring

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

* Required Conditions

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

Table 3.14.1 (Continued)
Instrumentation for Accident Monitoring

* Required Conditions (continued)

- B. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, the minimum number of channels shall be restored to operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours and Cold Shutdown within the following 24 hours.
- C. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, the torus temperature shall be monitored once per 12 hours (+25%) to observe any unexplained temperature increase which might be indicative of an open SRV; the minimum number of channels shall be restored to operable status within 30 days or be in at least Hot Shutdown within the next 12 hours and Cold Shutdown within the following 24 hours.
- D. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, initiate the preplanned alternate method of monitoring the appropriate parameters in addition to submitting the report required in (A) above.

3.0 LIMITING CONDITIONS FOR OPERATION

4.0 SURVEILLANCE REQUIREMENTS

B. Inservice Testing

1. Inservice Testing of Quality Group A, B, and C pumps and valves shall be performed in accordance with the requirements for ASME Code Class 1, 2 and 3 pumps and valves, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g) except where relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55(a)(g)(6)(i), or where alternate testing is justified in accordance with Generic Letter 89-04.
2. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

3.0 LIMITING CONDITIONS FOR OPERATION

4.0 SURVEILLANCE REQUIREMENTS

3. Deviations are permitted from the required sampling schedule if samples are unobtainable due to hazardous conditions, seasonable unavailability, or to malfunction of automatic sampling equipment. If the latter occurs, every effort shall be made to complete corrective action prior to the end of the next sampling period.
4. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of Table 4.16.3 when averaged over any calendar quarter, submit a special report to the Commission within 30 days from the end of the affected calendar quarter pursuant to Specification 6.7.C.2. When more than one of the radionuclides in Table 4.16.3 are detected in the sampling medium, this report shall be submitted if:

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

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

3.0 LIMITING CONDITIONS FOR OPERATION

3.17 CONTROL ROOM HABITABILITY

Applicability:

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

Objective:

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

Specification:

A. Control Room Ventilation System

1. Except as specified in 3.17.A.2 and 3.17.A.3 below, both trains of the control room ventilation system shall be operable, whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, or during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel.
- 2.a With one control room ventilation train inoperable, restore the inoperable train to operable status within 30 days.
- 2.b If 2.a is not met, then be in hot shutdown within the next 12 hours following the 30 days and in cold shutdown within 24 hours following the 12 hours.
- 2.c If 2.a is not met during movement of irradiated fuel assemblies in the secondary containment, core alterations or activities having the potential for draining the reactor vessel then immediately place the operable control room ventilation train in operation or immediately suspend these activities.

3.17/4.17

4.0 SURVEILLANCE REQUIREMENTS

4.17 CONTROL ROOM HABITABILITY

Applicability:

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

Objective:

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

Specification:

A. Control Room Ventilation System

1. Once per 12 hours check control room temperature.

229u

Amendment No. 65, 89

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

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

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

B. Offsite and Onsite Organizations

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

1. Lines of authority, responsibility and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, function descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements are documented in corporate and plant procedures, or the Updated Safety Analysis Report or the Operational Quality Assurance Plan.
2. The President, NSP Nuclear Generation shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining and providing technical support to the plant to ensure nuclear safety. This position has the responsibility for the Fire Protection Program.
3. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

C. Plant Staff

1. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.1.1.
2. At least one licensed operator shall be in the control room when fuel is in the reactor.
3. At least two licensed operators shall be present in the control room during cold startup, scheduled reactor shutdown, and during recovery from reactor trips.
4. An individual qualified in radiation protection procedures shall be onsite when fuel is in the reactor.
5. All alterations of the reactor core shall be directly supervised by a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.
6. A fire brigade of at least five members shall be maintained onsite at all times.* The fire brigade shall not include the three members of the shift organization required for safe shutdown of the reactor from outside the control room.
7. The General Superintendent, Operations shall be formerly licensed as a Senior Reactor Operator or hold a current Senior Reactor Operator License.
8. At least one member of plant management holding a current Senior Reactor Operator License shall be assigned to the plant operations group on a long term basis (approximately two years). This individual will not be assigned to a rotating shift.

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

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

- E. A training program for individuals serving in the fire brigade shall be maintained under the direction of a designated member of Northern States Power management. This program shall meet the requirement of Section 27 of the NFPA Code - 1976 with the exception of training scheduling. Fire brigade training shall be scheduled as set forth in the training program.
- F. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel. Procedures shall include the following provisions:
 - 1. Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8 or 12-hour day, nominal 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:
 - a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.
 - b. Overtime should be limited for all nuclear plant staff personnel so that total work time does not exceed 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, not more than 84 hours in any seven day period, all excluding shift turnover time. Individuals should not be required to work more than 15 consecutive days without two consecutive days off.
 - c. A break of at least eight hours including shift turnover time should be allowed between work periods.
 - d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

6.2 Review and Audit

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

A. Safety Audit Committee (SAC)

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

1. Membership

- a. The SAC shall consist of at least five (5) persons.
- b. The SAC Chairman shall be an NSP representative, not having line responsibility for operation of the plant, appointed by the President, NSP Nuclear Generation. Other members shall be appointed by the President, NSP Nuclear Generation or by such other person as he may designate. The Chairman shall appoint a Vice Chairman from the SAC membership to act in his absence.
- c. No more than two members of the SAC shall be from groups holding line responsibility for operation of the plant.
- d. A SAC member may appoint an alternate to serve in his absence, with concurrence of the Chairman. No more than one alternate shall serve on the SAC at any one time. The alternate member shall have voting rights.

2. Qualifications

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

- f. Investigation of all Reportable Events and Events requiring Special Reports to the Commission.
 - g. Revisions to the Facility Emergency Plan, the Facility Security Plan, and the Fire Protection Program.
 - h. Operations Committee minutes to determine if matters considered by that Committee involve unreviewed or unresolved safety questions.
 - i. Other nuclear safety matters referred to the SAC by the Operations Committee, plant management or company management.
 - j. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components.
 - k. Reports of special inspections and audits conducted in accordance with specification 6.3.
 - l. Changes to the Offsite Dose Calculation Manual (ODCM).
 - m. Review of investigative reports of unplanned releases of radioactive material to the environs.
6. Audit - The operation of the nuclear power plant shall be audited formally under the cognizance of the SAC to assure safe facility operation.
- a. Audits of selected aspects of plant operation, as delineated in ANSI N18.7-1976 as modified by the Operational Quality Assurance Plan, shall be performed with a frequency commensurate with their nuclear safety significance and in a manner to assure that an audit of all nuclear safety-related activities is completed within a period of two years. The audits shall be performed in accordance with appropriate written instructions and procedures.
 - b. Audits of aspects of plant radioactive effluent treatment and radiological environmental monitoring shall be performed as follows:
 - 1. Implementation of the Offsite Dose Calculation Manual and quality controls for effluent monitoring at least once every two years.
 - 2. Implementation of the Process Control Program for solidification of radioactive waste at least once every two years.
 - 3. The Radiological Environmental Monitoring Program and the results thereof, including quality controls, at least once every year.
 - c. Periodic review of the audit program should be performed by the SAC at least twice a year to assure its adequacy.
 - d. Written reports of the audits shall be reviewed by the President, NSP Nuclear Generation, by the SAC at a scheduled meeting, and by members of Management having responsibility in the areas audited.

7. Authority

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

8. Records

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

9. Procedures

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

- a. Subjects within the purview of the group.
- b. Responsibility and authority of the group.
- c. Mechanisms for convening meetings.
- d. Provisions of use of specialists or subgroups.
- e. Authority to obtain access to the nuclear power plant operating record files and operating personnel when assigned audit functions.
- f. Requirements for distribution of reports and minutes prepared by the group to others in the NSP Organization.

B. Operations Committee (OC)

1. Membership

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

2. Meeting Frequency

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

3. Quorum

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

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

- a. Proposed tests and experiments and their results.
- b. Modifications to plant systems or equipment as described in the Updated Safety Analysis Report and having nuclear safety significance or which involve an unreviewed safety question as defined in 10 CFR 50.59.
- c. Proposals which would effect permanent changes to normal and emergency operating procedures and any other proposed changes or procedures that are determined by the Plant Manager to affect nuclear safety.
- d. Proposed changes to the Technical Specifications or operating license.
- e. All reported or suspected violations of Technical Specifications, operating license requirements, administrative procedures, or operating procedures. Results of investigations, including evaluation and recommendations to prevent recurrence, will be reported, in writing, to the President, NSP Nuclear Generation and to the Chairman of the Safety Audit Committee.

- f. Investigation of all Reportable Events and Events requiring Special Reports to the Commission.
- g. Drills on emergency procedures (including plant evacuation) and adequacy of communication with off-site support groups.
- h. All procedures required by these Technical Specifications, including implementing procedures of the Emergency Plan and the Security Plan (except as exempted in Section 6.5.F), shall be reviewed with a frequency commensurate with their safety significance but at an interval of not more than two years.
- i. Perform special reviews and investigations, as requested by the Safety Audit Committee.
- j. Review of investigative reports of unplanned releases of radioactive material to the environs.
- k. All changes to the Process Control Program (PCP) and the Offsite Dose Calculation Manual (ODCM).

5. Authority

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

6. Records

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

7. Procedures

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

- a. Responsibility and authority of the group.
- b. Content and method of submission of presentations to the Operations Committee.

- c. Mechanism for scheduling meetings
- d. Meeting agenda
- e. Use of subcommittee
- f. Review and approval, by members, of OC actions
- g. Distribution of minutes

6.3 Special Inspections and Audits

- A. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite Northern States Power Company personnel or an outside fire protection consultant.
- B. An inspection and audit by an outside qualified fire protection consultant shall be performed at intervals no greater than three years.

6.4 Action to be Taken if a Safety Limit is Exceeded

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

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

B. Radiological

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

- (1) A Radiation Protection Plan, which shall be a complete definition of radiation protection policy and program
- (2) Procedures which implement the requirements of the Radiation Protection Plan

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

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

- (1) A radiation monitoring device that continuously indicates the radiation dose rate in the area.
- (2) A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been determined and personnel have been made knowledgeable of them.
- (3) An individual qualified in radiation protection procedures with a radiation dose rate monitoring device. This individual is responsible for providing positive radiation protection control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation protection procedures or the applicable Radiation Work Permit.

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

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 be submitted to the Commission with the Semi-Annual Radioactive Effluent release report for the period in which the change(s) were made effective. This submittal shall contain:
 - a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with a revision date, together with appropriate analyses or evaluations justifying the change(s).
 - b. a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and
 - c. documentation of the fact that the change has been reviewed and found acceptable by the Operations Committee.
2. Shall become effective upon review and acceptance by the Operations Committee.

F. 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. Records Retained for Plant Life (continued)

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

B. Reportable Events

The following actions shall be taken for Reportable Events:

- a. The Commission shall be notified by a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50 and,
- b. Each Reportable Event shall be reviewed by the Operations Committee and the results of this review shall be submitted to the Safety Audit Committee and the President, NSP Nuclear Generation.