ATTACHMENT 2A TO AEP:NRC:0433Q

TECHNICAL SPECIFICATIONS PAGES
MARKED TO SHOW PROPOSED CHANGES

UNIT 1

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3/4.0 APPLICABILITY

SURVEILLANCE REQUIREMENTS

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler And Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing criteria	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

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

4.0.6 Deleted

~~Amendments 100, 107 and 108 granted extensions for certain surveillances required to be performed on or before July 31, 1987, and until the end of the Cycle 9-10 refueling outage. For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 1 1987 refueling outage.~~

4.0.7 Deleted

~~Amendment 121 granted extensions for certain surveillances required to be performed on or before April 1, 1989, until the end of the Cycle 10-11 refueling outage. For these specific surveillances under this extension, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 1 1989 refueling outage.~~

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level--Low-Low	3/Stm. Gen.	2/Stm. Gen. any Stm.Gen.	2/Stm. Gen.	1,2,3	14*
b. 4 kv Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1,2,3	14*
Pump Start		2/bus (T11A-Train B; T11D-Train A)			
Valve Actuation (Both trains)		2/bus on (T11A & T11B or 2/busses T11C & T11D)			
c. Safety Injection	2	1	2	1, 2, 3	18*
d. Loss of Main Feedwater Pumps	2	2	2	1, 2	18*
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level--Low-Low	3/Stm. Gen.	2/Stm. Gen. any 2 Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. Reactor Coolant Pump Bus Undervoltage	4-1/Bus	2	3	1, 2, 3	19*
8. LOSS OF POWER					
a. 4 kv Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*
b. 4 kv Bus Degraded Voltage	3/Bus (T11A-Train B; T11D-Train A)	2/Bus (T11A-Train B; T11D-Train A)	2/Bus (T11A-Train B; T11D-Train A)	1, 2, 3, 4	14*

REACTOR COOLANT SYSTEM

REACTOR COOLANT VENT SYSTEM

REACTOR VESSEL HEAD VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.1 Both Reactor Vessel head vent paths shall be demonstrated OPERABLE# at least once per 18 months by:

1. Verifying the common manual isolation valve in the Reactor vessel head vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Reactor Vessel head vent paths during venting operation, while in Modes 5 or 6.

~~# Surveillance requirements to demonstrate the operability of each Reactor Vessel head vent path will be performed the next time the unit enters MODES 5 or 6 following the issuance of this Technical Specification, and after the appropriate Plant procedures have been written.~~

REACTOR COOLANT SYSTEM

REACTOR COOLANT VENT SYSTEM

PRESSURIZER STEAM SPACE VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.2 Both Pressurizer steam space vent paths shall be demonstrated OPERABLE# at least once per 18 months by:

1. Verifying the common manual isolation valve in the Pressurizer steam space vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Pressurizer steam space vent paths during venting operation, while in Modes 5 or 6.

~~# Surveillance requirements to demonstrate the operability of each Pressurizer steam space vent path will be performed the next time the unit enters MODES 5 or 6 following the issuance of this Technical Specification, and after the appropriate Plant procedures have been written.~~

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1

a. At least two independent component cooling water loops shall be OPERABLE.

b. At least ~~once~~ one component cooling water flowpath in support of Unit 2 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.3.1.a - MODES 1, 2, 3 and 4.
Specification 3.7.3.1.b - At all times when Unit 2 is in MODES 1, 2, 3, or 4.

ACTION:

~~With~~ When Specification 3.7.3.1.a is applicable:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

When Specification 3.7.3.1.b is applicable:

With no flowpath to Unit 2 available, return at least one flow path to available status within 7 days, or provide equivalent shutdown capability in Unit 2 and return at least one flow path to available status within the next 60 days, or have Unit 2 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours. The requirements of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection test signal.

c. By verifying pump performance pursuant to Specification 4.0.5.

d. At least once per 18 months during shutdown, by verifying that the Unit cross-tie valves can cycle full travel. Following cycling, the valves will be verified to be in their closed positions.

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2400 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes** and initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boric acid solution or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2400 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least ~~3 times per 7 days with a maximum time interval between samples of~~ once per 72 hours.

* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

** For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

REFUELING OPERATIONS

STORAGE POOL VENTILATION SYSTEM**

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel storage pool exhaust ventilation system shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no fuel storage pool exhaust ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool+ until at least one spent fuel storage pool exhaust ventilation system is restored to OPERABLE status.*

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel storage pool ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

   1. Deleted.

   2. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm $\pm$ 10%.

* The crane bay roll-up door and the South door of the auxiliary building crane bay ~~drumming room roll-up door~~ may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool.

** Shared system with D. C. COOK - UNIT 2.

\+ This does not include the main load block. For purposes of this specification, a deenergized main load block need not be considered a load.

5.6 FUEL STORAGE (Continued)

1. Region 1 is designed to accommodate new fuel with a maximum nominal enrichment of 4.95 wt% U-235, or spent fuel regardless of the discharge fuel burnup.

2. Region 2 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 50,000 MWD/MtU, or fuel of other enrichments with equivalent reactivity.

3. Region 3 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 38,000 MWD/MtU, or fuel of other enrichments with equivalent reactivity.

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations: ~~and graphically depicted in Figure 5.6-3.~~

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-22{,}670 + 22{,}220\,E - 2{,}260\,E^2 + 149\,E^3$$

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-26{,}745 + 18{,}746\,E - 1{,}631\,E^2 + 98.4\,E^3$$

Where E = Initial Peak Enrichment

5.6.1.2: Fuel stored in the spent fuel storage racks shall have a maximum nominal fuel assembly enrichment as follows:

	Description		Maximum Nominal Fuel Assembly Enrichment Wt. % U-235
1)	Westinghouse	15 x 15 STD 15 x 15 OFA	4.95
2)	Exxon/ANF	15 x 15	4.95
3)	Westinghouse	17 x 17 STD 17 x 17 OFA 17 x 17 V5	4.95
4)	Exxon/ANF	17 x 17	4.95

~~Figure 5.6-3: Acceptable Burnup Domain in Regions 2 & 3~~

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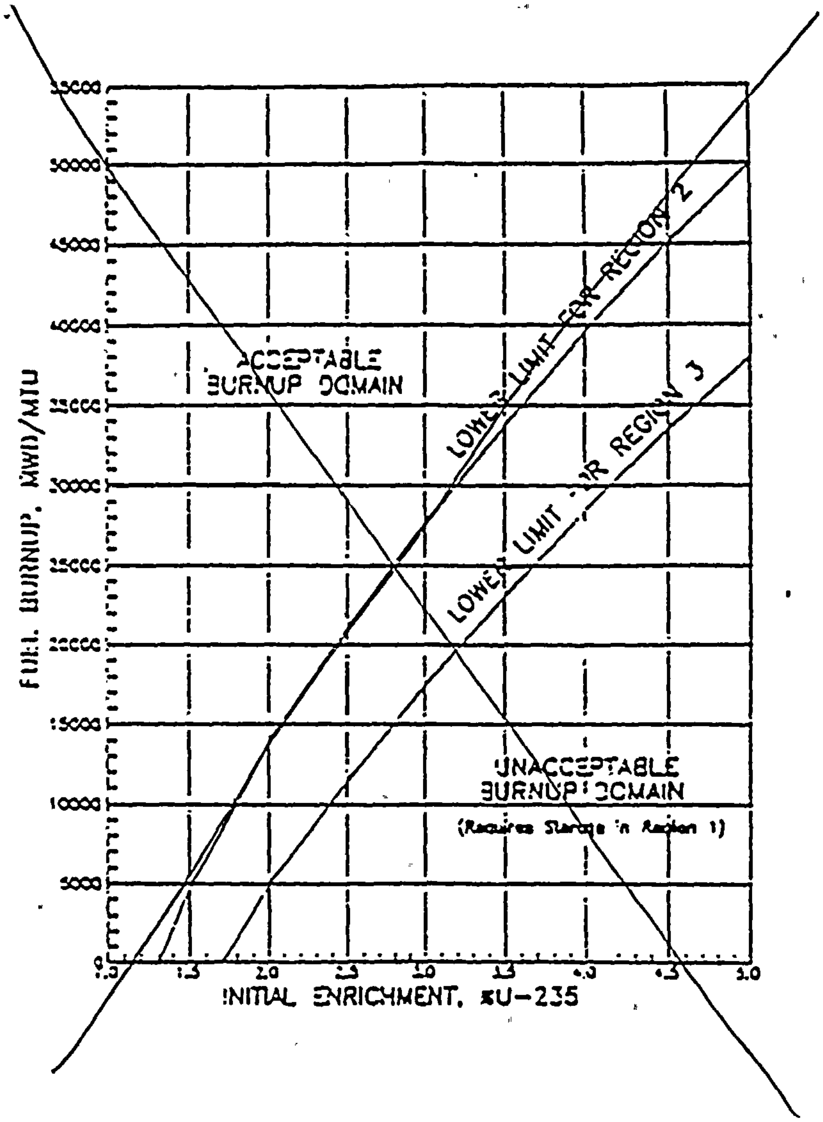

6.0 ADMINISTRATIVE CONTROLS

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed 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 Operations Superintendent, who must ~~hold or have held a Senior Operator License~~ be qualified as specified in Section 6.2.2.~~f~~g.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY REVIEW COMMITTEE (PNSRC)

FUNCTION

6.5.1.1 The PNSRC shall function to advise the Site Vice President/Plant Manager, or designee, on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PNSRC shall be composed of Assistant Plant Managers, Department Superintendents, or supervisory personnel reporting directly to the Site Vice President/Plant Manager, Assistant Plant Managers or Department Superintendents. The membership shall represent the functional areas of the plant, including, but not limited to Operations, Technical Support, Licensing, Maintenance and Radiation Protection

The PNSRC membership shall consist of at least one individual from each of the areas designated. All members, including the Chairman and his alternates, the members and their alternates, shall be designated by the Site Vice President/Plant Manager.

PNSRC members and alternates shall meet or exceed the minimum qualifications of ANSI N18.1-1971 Section 4.4 for comparable positions. The nuclear power plant operations individual shall meet the qualifications of Section 4.2.2 of ANSI N18.1-1971 except for the requirement to hold a current Senior Operator License. The operations individual must hold or have held a Senior Operator License or have been certified for equivalent senior operator knowledge at Cook Nuclear Plant or a similar reactor. The maintenance individual shall meet the qualifications of Section 4.2.3 of ANSI N18.1-1971.

ATTACHMENT 2B TO AEP:NRC:0433Q

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MARKED TO SHOW PROPOSED CHANGES

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3/4.0 APPLICABILITY

SURVEILLANCE REQUIREMENTS

b. Surveillance Intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler And Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing criteria	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

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

4.0.6 Deleted

~~Amendment 78 granted extensions for certain surveillances required to be performed on or before March 31, 1986, until the end of the Cycle 5-6 refueling outage. For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 2 1986 refueling outage.~~

4.0.7 Deleted

~~Amendments 97 and 99 granted extensions for certain surveillances required to be performed on or before July 1, 1988, until the end of the Cycle 6-7 refueling outage. For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 2 1988 refueling outage.~~

3/4.0 APPLICABILITY

SURVEILLANCE REQUIREMENTS

Deleted

4.0.8 ~~By specific reference to this section, those surveillances which must be performed on or before August 13, 1994, and are designed as 18-month or 36-month surveillances (or required as outage-related surveillances under the provisions of Specification 4.0.5) may be delayed until the end of the cycle 9-10 refueling outage. For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 2 1994 refueling outage.~~

Deleted

4.0.9 ~~By specific reference to this section, those surveillances which must be performed on or before September 7, 1994, and are designated as 18-month surveillances may be delayed until just prior to core reload in the Unit 2 Cycle 9-10 refueling outage.~~

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip				
A. Shunt Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
B. Undervoltage Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux	S	D(2,8),M(3,8) and Q(6,8)	M and S/U(1)	1, 2 and *
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	M	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(6)	M	1, 2
5. Intermediate Range, Neutron Flux	S	R(6,8)	S/U(1)	1, 2 and *
6. Source Range, Neutron Flux	S	R(6,14)	M(14) and S/U(1)	2(7), 3(7), 4 and 5
7. Overtemperature ΔT	S	R(9)†	M	1, 2
8. Overpower ΔT	S	R(9)†	M	1, 2
9. Pressurizer Pressure--Low	S	R†	M	1, 2
10. Pressurizer Pressure--High	S	R†	M	1, 2
11. Pressurizer Water Level--High	S	R†	M	1, 2
12. Loss of Flow - Single Loop	S	R(8)	M	1

~~† The provisions of Technical Specification 4.0.8 are applicable.~~

TABLE 3.3-3 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level--Low-Low	3/Stm. Gen.	2/Stm. Gen. any 2 Stm.Gen.	2/Stm. Gen.	1, 2, 3	14*
b. Reactor Coolant Pump Bus Undervoltage	4-1/Bus	2	3	1, 2, 3	19*
8. LOSS OF POWER					
a. 4 kV Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*
b. 4 kV Bus Degraded Voltage	3/Bus (T21A-Train B; T21D-Train A)	2/Bus (T21A-Train B; T21D-Train A)	2/Bus (T21A-Train B; T21D-Train A)	1, 2, 3, 4	14*
9. MANUAL					
a. Safety Injection (ECCS) Feedwater Isolation Reactor Trip (SI) Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation Auxiliary Feedwater Pumps Essential Service Water System	2/train	1/train	2/train	1, 2, 3, 4	18
b. Containment Spray Containment Isolation - Phase "B" Containment Purge and Exhaust Isolation Containment Air Recirculation Fan	1/train	1/train	1/train	1, 2, 3, 4	18
c. Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation	1/train	1/train	1/train	1, 2, 3, 4	18
d. Steam Line Isolation	2/steam line (1 per train)	2/steam line (1 per train)	2/operating steam line (1 per train)	1, 2, 3	20

TABLE 4.3-2
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Manual Initiation	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure – High	S	R	M(3)	N.A.	1, 2, 3
d. Pressurizer Pressure – Low	S	R~~†~~	M	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines – High	S	R	M	N.A.	1, 2, 3
f. Steam Line Pressure – Low	S	R	M	N.A.	1, 2, 3
2. CONTAINMENT SPRAY					
a. Manual Initiation	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure – High-High	S	R	M(3)	N.A.	1, 2, 3
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual	See Functional Unit 9				
2) From Safety Injection Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
b. Phase "B" Isolation					
1) Manual	See Functional Unit 9				
2) Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
3) Containment Pressure– High-High	S	R	M(3)	N.A.	1, 2, 3

~~†The provisions of Technical Specification 4.0.8 are applicable.~~

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
c. Purge and Exhaust Isolation					
1) Manual	See Functional Unit 9				
2) Containment Radioactivity -- High	S	R	Q	N.A.	1, 2, 3, 4
4. STEAM LINE ISOLATION					
a. Manual	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3
c. Containment Pressure -- High-High	S	R	M(3)	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines -- High Coincident with T_{avg} -- Low-Low	S	R	M	N.A.	1, 2, 3
e. Steam Line Pressure -- Low	S	R	M	N.A.	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION					
a. Steam Generator Water Level -- High-High	S	R	M	N.A.	1, 2, 3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	S	R	M	N.A.	1, 2, 3
b. 4 kV Bus Loss of Voltage	S	R	M	N.A.	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	N.A.	1, 2, 3
d. Loss of Main Feed Pumps	N.A.	N.A.	R	N.A.	1, 2

~~†The provisions of Technical Specification 4.0.8 are applicable.~~

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

~~This Technical Specification will not be effective until after the 1982 refueling outage.~~

TABLE 4.3-6A

APPENDIX R REMOTE SHUTDOWN MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

INSTRUMENT	LOCATION	CHANNEL CHECK	CHANNEL CALIBRATION
1. Steam Generators 1 and 4 Level	LSI Cabinet 1 and LSI Cabinet 4	M	R
2. Steam Generators 2 and 3 Level	LSI Cabinet 2 and LSI Cabinet 4	M	R
3. Steam Generators 1 and 4 Pressure	LSI Cabinet 4 and LSI Cabinet 5	M	R
4. Steam Generators 2 and 3 Pressure	LSI Cabinet 4 and LSI Cabinet 6	M	R
5. Reactor Coolant Loop 4 Temperature (Cold)	LSI Cabinet 4 and LSI Cabinet 5	M	R
6. Reactor Coolant Loop 4 Temperature (Hot)	LSI Cabinet 4 and LSI Cabinet 5	M	R
7. Reactor Coolant Loop 2 Temperature (Cold)	LSI Cabinet 4 and LSI Cabinet 6	M	R
8. Reactor Coolant Loop 2 Temperature (Hot)	LSI Cabinet 4 and LSI Cabinet 6	M	R
9. Pressurizer Level	LSI Cabinet 3	M	R
10. Reactor Coolant System Pressure	LSI Cabinet 3	M	R
11. Charging Cross-Flow Between Units	Corridor Elev. 587'	n/a	R*
12. Source Range Neutron Detector (N-23)	LSI Cabinet 4	n/a	R

* Charging Cross-Flow between Units is an instrument common to both Unit 1 and 2. This surveillance will only be conducted on an interval consistent with Unit 1 refueling.

~~† The provisions of Technical Specification 4.0.8 are applicable.~~

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Containment Pressure	M	R
2.	Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3.	Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4.	Reactor Coolant Pressure - Wide Range	M	R
5.	Pressurizer Water Level	M	R
6.	Steam Line Pressure	M	R
7.	Steam Generator Water Level - Narrow Range	M	R
8.	RWST Water Level	M	R
9.	Boric Acid Tank Solution Level	M	R
10.	Auxiliary Feedwater Flow Rate	M	R
11.	Reactor Coolant System Subcooling Margin Monitor	M	R
12.	PORV Position Indicator - Limit Switches	M	R
13.	PORV Block Valve Position Indicator - Limit Switches	M	R
14.	Safety Valve Position Indicator - Acoustic Monitor	M	R
15.	Incore Thermocouples (Core Exit Thermocouples)	M	R(1)
16.	Reactor Coolant Inventory Tracking System (Reactor Vessel Level Indication)	M(2)	R(3)
17.	Containment Sump Level	M	R
18.	Containment Water Level	M	R

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(1) Partial range channel calibration for sensor to be performed below P-12 in MODE 3.

(2) With one train of Reactor Vessel Level Indication inoperable, Subcooling Margin Indication and Core Exit Thermocouples may be used to perform a CHANNEL CHECK to verify the remaining Reactor Vessel Indication train OPERABLE.

(3) Completion of channel calibration for sensors to be performed below P-12 in MODE 3.

~~† The provisions of Technical Specification 4.0.8 are applicable.~~

~~Technical Specification will not be effective until after the 1982 refueling outage.~~

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G. Prompt notification to NRC pursuant to specification 6.9.1	All other S. G.s are C-1	None	N/A	N/A
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.1	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. One of the containment atmosphere particulate radioactivity monitoring channels (ERS-2301 or ERS-2401),

b. The containment sump level and flow monitoring system, and

c. Either the containment humidity monitor or one of the containment atmosphere gaseous radioactivity monitoring channels (ERS-2305 or ERS-2405).

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring channels are inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous (if being used) monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months,~~†~~

c. Containment humidity monitor (if being used) - performance of CHANNEL CALIBRATION at least once per 18 months.

~~† The provisions of Technical Specification 4.0.8 are applicable.~~

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

g. With PORVs and block valves not in the same line inoperable due to causes other than excessive seat leakage, within 1 hour restore the valves to OPERABLE status or close and de-energize the associated block valve and place the associated PORV in manual control in each respective line. Apply the portions of ACTION c or d above, relating to the OPERATIONAL MODE, as appropriate for two or three lines unavailable.

h. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE:

a. At least once per 31 days by performance of a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. At least once per 18 months by operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

c. At least once per 18 months by operating solenoid air control valves and check valves in PORV control systems through one complete cycle of full travel, and

d. At least once per 18 months by performing a CHANNEL CALIBRATION of the actuation instrumentation.~~‡~~

4.4.11.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b, c, or d in Specification 3.4.11.

4.4.11.3 Deleted.

~~‡The provisions of Technical Specification 4.0.8 are applicable.~~

REACTOR COOLANT SYSTEM

REACTOR COOLANT VENT SYSTEM

REACTOR VESSEL HEAD VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.1 Both Reactor Vessel head vent paths shall be demonstrated OPERABLE# at least once per 18 months by:

1. Verifying the common manual isolation valve in the Reactor vessel head vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Reactor Vessel head vent paths during venting operation, while in Modes 5 or 6.

~~# Surveillance requirements to demonstrate the operability of each Reactor Vessel head vent path will be performed the next time the unit enters MODES 5 or 6 following the issuance of this Technical Specification, and after the appropriate Plant procedures have been written.~~

REACTOR COOLANT SYSTEM

REACTOR COOLANT VENT SYSTEM

PRESSURIZER STEAM SPACE VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.2 Both Pressurizer steam space vent paths shall be demonstrated OPERABLE# at least once per 18 months by:

1. Verifying the common manual isolation valve in the Pressurizer steam space vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Pressurizer steam space vent paths during venting operation, while in Modes 5 or 6.

~~# Surveillance requirements to demonstrate the operability of each Pressurizer steam space vent path will be performed the next time the unit enters MODES 5 or 6 following the issuance of this Technical Specification, and after the appropriate Plant procedures have been written.~~

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
a. IMO-390	a. RWST to RHR	a. Open
b. IMO-315	b. Low head SI to Hot Leg	b. Closed
c. IMO-325	c. Low head SI to Hot Leg	c. Closed
d. IMO-262*	d. Mini flow line	d. Open
e. IMO-263*	e. Mini flow line	e. Open
f. IMO-261*	f. SI Suction	f. Open
g. ICM-305*	g. Sump Line	g. Closed
h. ICM-306*	h. Sump Line	h. Closed

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

*These valves must change position during the switchover from injection to recirculation flow following LOCA.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.†

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE charging pump, shall be demonstrated inoperable, by verifying that the motor circuit breakers have been removed from their electrical power supply circuits, at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F as determined at least once per hour when any RCS cold leg temperature is between 152°F and 200°F.

~~† The provisions of Technical Specification 4.0.8 are applicable.~~

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure--High-High test signal.

d. At least once per 5 years by verifying a water flow rate of at least 20 gpm (greater than or equal to 20 gpm) but not to exceed 50 gpm (less than or equal to 50 gpm) from the spray additive tank test line to each containment spray system with the spray pump operating on recirculation with a pump discharge pressure greater than or equal to 255 psig.

~~† The provisions of Technical Specification 4.0.8 are applicable.~~

[illegible]

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:~~†~~

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Containment Purge and Exhaust isolation signal, each Purge and Exhaust valve actuates to its isolation position.

4.6.3.1.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5

~~† The provisions of Technical Specification 4.0.8 are applicable.~~

CONTAINMENT SYSTEMS

DIVIDER BARRIER SEAL

LIMITING CONDITION FOR OPERATION

3.6.5.9 The divider barrier seal shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the divider barrier seal inoperable, restore the seal to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.9 The divider barrier seal shall be determined OPERABLE at least once per 18 months during shutdown by:†

a. Removing two divider barrier seal test coupons and verifying that the physical properties of the test coupons are within the acceptable range of values shown in Table 3.6-2.

b. Visually inspecting at least 95 percent of the seal's entire length and:

   1. Verifying that the seal and seal mounting bolts are properly installed, and

   2. Verifying that the seal material shows no visual evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearances.

~~†. The provisions of Technical Specification 4.0.8 are applicable.~~

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1

a. At least two independent component cooling water loops shall be OPERABLE.

b. At least one component cooling water flow path in support of Unit 1 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.3.1.a. - MODES 1, 2, 3, 4.
Specification 3.7.3.1.b. - At all times when Unit 1 is in MODES 1, 2, 3, or 4.

ACTION:

When Specification 3.7.3.1.a is applicable:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

When Specification 3.7.3.1.b is applicable:

With no flowpath ~~in~~ to Unit 1 available, return at least one flowpath to available status within 7 days, or provide equivalent shutdown capability in Unit 1 and return at least one flow path to available status within the next 60 days, or have Unit 1 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours. The requirements of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection test signal.X

c. By verifying pump performance pursuant to Specification 4.0.5.

d. ~~4.7.3.2~~ At least once per 18 months during shutdown, verify that the unit cross-tie valves can cycle full travel. Following cycling, the valves will be verified to be in their closed positions.

~~† The provisions of Technical Specification 4.0.8 are applicable.~~

X

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1

a. At least two independent essential service water loops shall be OPERABLE.

b. At least one essential service water flowpath associated with support of Unit 1 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.4.1.a. - MODES 1, 2, 3, and 4.
Specification 3.7.4.1.b. - At all times when Unit 1 is in MODES 1, 2, 3, or 4.

ACTION:

When Specification 3.7.4.1.a is applicable:

With only one essential service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

When Specification 3.7.4.1.b is applicable:

With no essential service water flow path available in support of Unit 1 shutdown functions, return at least one flow path to available status within 7 days or provide equivalent shutdown capability in Unit 1 and return the equipment to service within the next 60 days, or have Unit 1 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours. The requirements of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two essential service water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection test signal. ~~†~~

~~† The provisions of Technical Specification 4.0.8 are applicable.~~

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

2. a. Verifying that on a Safety Injection Signal from Unit 1, the system automatically operates in the pressurization/cleanup mode.~~†~~

   b. Verifying that on a Safety Injection Signal from Unit 2, the system automatically operates in the pressurization/cleanup mode.~~†~~

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/16 inch W. G. relative to the outside atmosphere at a system flow rate of 6000 cfm plus or minus 10%.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

~~†The provisions of Technical Specification 4.0.8 are applicable.~~

PLANT SYSTEMS

3/4.7.7 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.7.1 All safety-related snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.7.1.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Inspection~~†~~

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 3.7-9. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 3.7-9 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment No. 156.

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified as acceptable for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that

~~† The provisions of Technical Specification 4.0.8 are applicable.~~

ELECTRICAL POWER SYSTEMS

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. A day fuel tank containing a minimum of 70 gallons of fuel,

2. A fuel storage system containing a minimum indicated volume of 46,000 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2.a.5. †

* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

† ~~The provisions of Technical Specification 4.0.8 are applicable.~~

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is greater than or equal to 2.13 volts under float charge.

2. The specific gravity, corrected to 77°F, and full electrolyte level (fluid at the bottom of the maximum level indication mark), of each connected cell is greater than or equal to 1.200 and has not decreased more than 0.03 from the value observed during the previous test, and

3. The electrolyte level of each connected cell is between the top of the minimum level indication mark and the bottom of the maximum level indication mark.

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,

3. The battery charger will supply at least 140 amperes at greater than or equal to 250 volts for at least 4 hours.

d. At least once per 18 months, perform a battery service test during shutdown (MODES 5 or 6), by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status the actual or simulated emergency loads for the design duty cycle which is based on the composite load profile. The composite load profile envelopes both the LOCA/LOOP and Station Blackout profiles and provides the basis for the times listed in Table 4.8-2. The battery charger will be disconnected throughout the test. The battery terminal voltage shall be maintained greater than or equal to 210 volts throughout this test.

e. At least once per 60 months, conduct a performance test of battery capacity during shutdown (MODES 5 or 6), by verifying that the battery capacity is at least 80% of the manufacturer's rating. When this test is performed in place of a battery service test, a modified performance test shall be conducted.

Annual performance tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% from its capacity on the previous performance test, or is below 90% of the manufacturer's rating. If the battery has reached 85% of service life, delivers a capacity of 100% or greater of the manufacturer's rated capacity, and has shown no signs of degradation, performance testing at two year intervals is acceptable until the battery shows signs of degradation.

~~The provisions of Specification 4.0.9 are applicable to the 2-CD battery and charger.~~

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

1 - 250-volt D.C. bus, and

1 - 250-volt battery bank and charger associated with the above D.C. bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and bus OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 250-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 250-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

~~*The provisions of Specification 4.0.9 are applicable to the Surveillance Requirements 4.8.2.3.2.c.3 and 4.8.2.3.2.d for the 2-CD battery and charger.~~

STORAGE POOL VENTILATION SYSTEM**

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel storage pool exhaust ventilation system shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no fuel storage pool exhaust ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool+ until at least one spent fuel storage pool exhaust ventilation system is restored to OPERABLE status.*

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel storage pool ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

1. Deleted.

2. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm $\pm$ 10%.

* The crane bay roll-up door and the south door of the auxiliary building crane bay ~~drumming room roll-up door~~ may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool.

** Shared system with D. C. COOK - UNIT 1.

\+ This does not include the main load block. For purposes of this specification, a deenergized main load block need not be considered a load.

5.6 FUEL STORAGE (Continued)

CRITICALITY - SPENT FUEL (Continued)

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations. ~~and graphically depicted in Figure 5.6-3.~~

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-22,670 + 22,220\ E - 2,260\ E^2 + 149\ E^3$$

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-26,745 + 18,746\ E - 1,631\ E^2 + 98.4\ E^3$$

Where E = Initial Peak Enrichment

5.6.1.2 Fuel stored in the spent fuel storage racks shall have a nominal fuel assembly enrichment as follows:

	Description		Maximum Nominal Fuel Assembly Enrichment Wt. % U-235
1)	Westinghouse	15 x 15 STD 15 x 15 OFA	4.95
2)	Exxon/ANF	15 x 15	4.95
3)	Westinghouse	17 x 17 STD 17 x 17 OFA 17 x 17 V5	4.95
4)	Exxon/ANF	17 x 17	4.95

~~Figure 5.6-3: Acceptable Burnup Domain in Regions 2 &~~ 3

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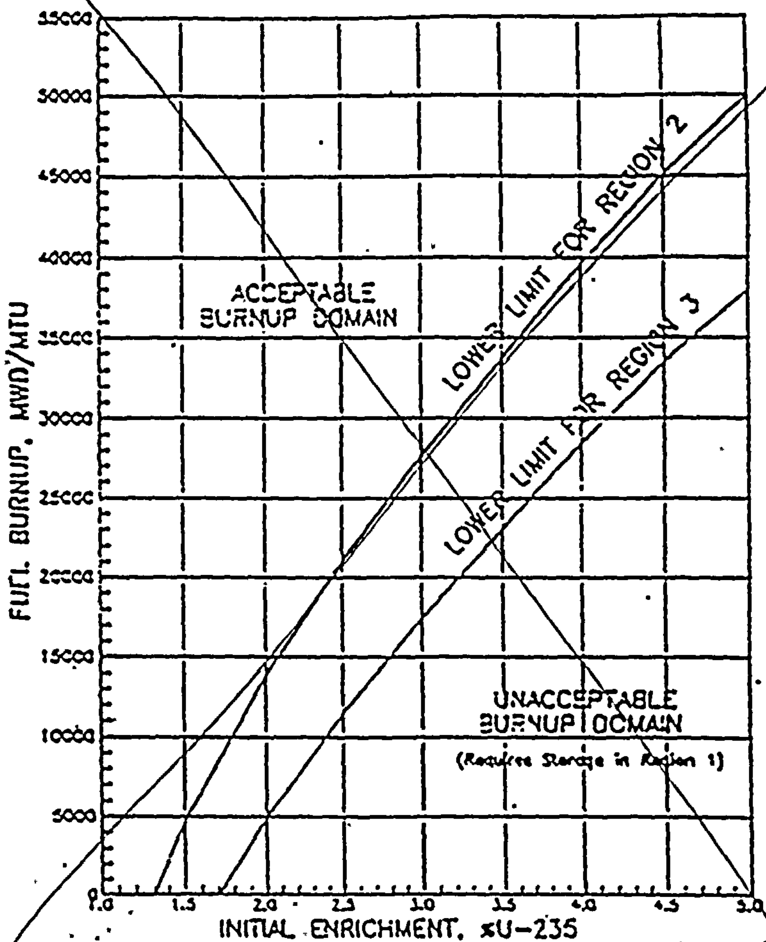

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed 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 Operations Superintendent, who must ~~hold or have held a Senior Operator License~~ be qualified as specified in Section 6.2.2.~~h~~ g.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY REVIEW COMMITTEE (PNSRC)

FUNCTION

6.5.1.1 The PNSRC shall function to advise the Site Vice President/Plant Manager, or designee, on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PNSRC shall be composed of Assistant Plant Managers, Department Superintendents, or supervisory personnel reporting directly to the Site Vice President/Plant Manager, Assistant Plant Managers or Department Superintendents. The membership shall represent the functional areas of the plant, including, but not limited to Operations, Technical Support, Licensing, Maintenance and Radiation Protection.

The PNSRC membership shall consist of at least one individual from each of the areas designated. All members, including the Chairman and his alternates, the members and their alternates, shall be designated by the Site Vice President/Plant Manager.

PNSRC members and alternates shall meet or exceed the minimum qualifications of ANSI N18.1-1971 Section 4.4 for comparable positions. The nuclear power plant operations individual shall meet the qualifications of Section 4.2.2 of ANSI N18.1-1971 except for the requirement to hold a current Senior Operator License. The operations individual must hold or have held a Senior Operator License or have been certified for equivalent senior operator knowledge at Cook Nuclear Plant or a similar reactor. The maintenance individual shall meet the qualifications of Section 4.2.3 of ANSI N18.1-1971.

ATTACHMENT 3A TO AEP:NRC:0433Q

PROPOSED TECHNICAL SPECIFICATIONS PAGES

REVISED PAGES

UNIT 1

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3/4 3-21a
3/4 4-38
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3/4 9-13
5-6
5-7b
6-4

SURVEILLANCE REQUIREMENTS

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing criteria	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

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

4.0.6 Deleted.

4.0.7 Deleted.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level--Low-Low	3/Stm. Gen.	2/Stm. Gen. any Stm.Gen.	2/Stm. Gen.	1,2,3	14*
b. 4 kv Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1,2,3	14*
Pump Start		2/bus (T11A-Train B; T11D-Train A)			
Valve Actuation (Both trains)		2/bus on (T11A & T11B or 2/busses T11C & T11D)			
c. Safety Injection	2	1	2	1, 2, 3	18*
d. Loss of Main Feedwater Pumps	2	2	2	1, 2	18*
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level--Low-Low	3/Stm. Gen.	2/Stm. Gen. any 2 Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. Reactor Coolant Pump Bus Undervoltage	4-1/Bus	2	3	1, 2, 3	19*
8. LOSS OF POWER					
a. 4 kv Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*
b. 4 kv Bus Degraded Voltage	3/Bus (T11A-Train B; T11D-Train A)	2/Bus (T11A-Train B; T11D-Train A)	2/Bus (T11A-Train B; T11D-Train A)	1, 2, 3, 4	14*

REACTOR COOLANT VENT SYSTEM

REACTOR VESSEL HEAD VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.1 Both Reactor Vessel head vent paths shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the common manual isolation valve in the Reactor vessel head vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Reactor Vessel head vent paths during venting operation, while in Modes 5 or 6.

REACTOR COOLANT VENT SYSTEM

PRESSURIZER STEAM SPACE VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.2 Both Pressurizer steam space vent paths shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the common manual isolation valve in the Pressurizer steam space vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Pressurizer steam space vent paths during venting operation, while in Modes 5 or 6.

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1

a. At least two independent component cooling water loops shall be OPERABLE.

b. At least one component cooling water flowpath in support of Unit 2 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.3.1.a - MODES 1, 2, 3 and 4.
Specification 3.7.3.1.b - At all times when Unit 2 is in MODES 1, 2, 3, or 4.

ACTION:

When Specification 3.7.3.1.a is applicable:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

When Specification 3.7.3.1.b is applicable:

With no flowpath to Unit 2 available, return at least one flow path to available status within 7 days, or provide equivalent shutdown capability in Unit 2 and return at least one flow path to available status within the next 60 days, or have Unit 2 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours. The requirements of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection test signal.

c. By verifying pump performance pursuant to Specification 4.0.5.

d. At least once per 18 months during shutdown, by verifying that the unit cross-tie valves can cycle full travel. Following cycling, the valves will be verified to be in their closed positions.

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2400 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes** and initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boric acid solution or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2400 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

---

*The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

**For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

REFUELING OPERATIONS

STORAGE POOL VENTILATION SYSTEM**

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel storage pool exhaust ventilation system shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no fuel storage pool exhaust ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool+ until at least one spent fuel storage pool exhaust ventilation system is restored to OPERABLE status.*

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel storage pool ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

   1. Deleted.

   2. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm ± 10%.

*The crane bay roll-up door and the south door of the auxiliary building crane bay may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool.

**Shared system with D. C. COOK - UNIT 2.

+This does not include the main load block. For purposes of this specification, a de-energized main load block need not be considered a load.

5.6 FUEL STORAGE (Continued)

1. Region 1 is designed to accommodate new fuel with a maximum nominal enrichment of 4.95 wt% U-235, or spent fuel regardless of the discharge fuel burnup.

2. Region 2 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 50,000 MWD/MtU, or fuel of other enrichments with equivalent reactivity.

3. Region 3 is designed to accommodate fuel of 4.95% initial nominal enrichment burned to at least 38,000 MWD/MtU, or fuel of other enrichments with equivalent reactivity.

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations:

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-22{,}670 + 22{,}220\,E - 2{,}260\,E^2 + 149\,E^3$$

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-26{,}745 + 18{,}746\,E - 1{,}631\,E^2 + 98.4\,E^3$$

Where E = Initial Peak Enrichment

5.6.1.2: Fuel stored in the spent fuel storage racks shall have a maximum nominal fuel assembly enrichment as follows:

	Description		Maximum Nominal Fuel Assembly Enrichment Wt. % U-235
1)	Westinghouse	15 x 15 STD 15 x 15 OFA	4.95
2)	Exxon/ANF	15 x 15	4.95
3)	Westinghouse	17 x 17 STD 17 x 17 OFA 17 x 17 V5	4.95
4)	Exxon/ANF	17 x 17	4.95

Figure 5.6-3 intentionally deleted.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed 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 Operations Superintendent, who must be qualified as specified in Section 6.2.2.g.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY REVIEW COMMITTEE (PNSRC)

FUNCTION

6.5.1.1 The PNSRC shall function to advise the Site Vice President/Plant Manager, or designee, on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PNSRC shall be composed of Assistant Plant Managers, Department Superintendents, or supervisory personnel reporting directly to the Site Vice President/Plant Manager, Assistant Plant Managers or Department Superintendents. The membership shall represent the functional areas of the plant, including, but not limited to Operations, Technical Support, Licensing, Maintenance and Radiation Protection.

The PNSRC membership shall consist of at least one individual from each of the areas designated. All members, including the Chairman and his alternates, the members and their alternates, shall be designated by the Site Vice President/Plant Manager.

PNSRC members and alternates shall meet or exceed the minimum qualifications of ANSI N18.1-1971 Section 4.4 for comparable positions. The nuclear power plant operations individual shall meet the qualifications of Section 4.2.2 of ANSI N18.1-1971 except for the requirement to hold a current Senior Operator License. The operations individual must hold or have held a Senior Operator License or have been certified for equivalent senior operator knowledge at Cook Nuclear Plant or a similar reactor. The maintenance individual shall meet the qualifications of Section 4.2.3 of ANSI N18.1-1971.

ATTACHMENT 3B TO AEP:NRC:0433Q

PROPOSED TECHNICAL SPECIFICATIONS PAGES

REVISED PAGES

UNIT 2

3/4 0-3
3/4 0-4
3/4 3-11
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3/4 3-44d
3/4 3-47
3/4 4-13
3/4 4-14
3/4 4-33
3/4 4-35
3/4 4-37
3/4 5-4
3/4 5-8
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3/4 8-9
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3/4 9-12
5-6
5-8
6-4

SURVEILLANCE REQUIREMENTS

b. Surveillance Intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing criteria	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 monoths	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

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

4.0.6 Deleted.

4.0.7 Deleted.

SURVEILLANCE REQUIREMENTS

4.0.8 Deleted.

4.0.9 Deleted.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODE IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip				
A. Shunt Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
B. Undervoltage Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux	S	D(2,8), M(3,8) and Q(6,8)	M and S/U(1)	1, 2 and *
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	M	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(6)	M	1, 2
5. Intermediate Range, Neutron Flux	S	R(6,8)	S/U(1)	1, 2, and *
6. Source Range, Neutron Flux	S	R(6,14)	M(14) and S/U(1)	2(7), 3(7), 4 and 5
7. Overtemperature ΔT	S	R(9)	M	1, 2
8. Overpower ΔT	S	R(9)	M	1, 2
9. Pressurizer Pressure -- Low	S	R	M	1, 2
10. Pressurizer Pressure -- High	S	R	M	1, 2
11. Pressurizer Water Level -- High	S	R	M	1, 2
12. Loss of Flow-Single Loop	S	R(8)	M	1

TABLE 3.3-3 (Continued)
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen.	2/Stm. Gen. any 2 Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. Reactor Coolant Pump Bus Undervoltage	4-1/Bus	2	3	1, 2, 3	19*
8. LOSS OF POWER					
a. 4 kV Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*
b. 4 kV Bus Degraded Voltage	3/Bus (T21A - Train B; T21D - Train A)	2/Bus (T21A - Train B; T21D-Train A)	2/Bus (T21A-Train B; T21D-Train A)	1, 2, 3, 4	14*
9. MANUAL					
a. Safety Injection (ECCS) Feedwater Isolation Reactor Trip (SI) Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation Auxiliary Feedwater Pumps Essential Service Water System	2/train	1/train	2/train	1, 2, 3, 4	18
b. Containment Spray Containment Isolation - Phase "B" Containment Purge and Exhaust Isolation Containment Air Recirculation Fan	1/train	1/train	1/train	1, 2, 3, 4	18
c. Containment Isolation - Phase "A" Containment Purge and Exhaust Isolation	1/train	1/train	1/train	1, 2, 3, 4	18
d. Steam Line Isolation	2/steam line (1 per train)	2/steam line (1 per train)	2/operating steam line (1 per train)	1, 2, 3	20

TABLE 4.3-2
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Manual Initiation	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure -- High	S	R	M(3)	N.A.	1, 2, 3
d. Pressurizer Pressure -- Low	S	R	M	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines -- High	S	R	M	N.A.	1, 2, 3
f. Steam Line Pressure -- Low	S	R	M	N.A.	1, 2, 3
2. CONTAINMENT SPRAY					
a. Manual Initiation	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
c. Containment Pressure -- High-High	S	R	M(3)	N.A.	1, 2, 3
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual	See Functional Unit 9				
2) From Safety Injection Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
b. Phase "B" Isolation					
1) Manual	See Functional Unit 9				
2) Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3, 4
3) Containment Pressure-- High-High	S	R	M(3)	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
c. Purge and Exhaust Isolation					
1) Manual	See Functional Unit 9				
2) Containment Radioactivity -- High	S	R	Q	N.A.	1, 2, 3, 4
4. STEAM LINE ISOLATION					
a. Manual	See Functional Unit 9				
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	N.A.	1, 2, 3
c. Containment Pressure -- High-High	S	R	M(3)	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines -- High Coincident with T_{avg} -- Low-Low	S	R	M	N.A.	1, 2, 3
e. Steam Line Pressure -- Low	S	R	M	N.A.	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION					
a. Steam Generator Water Level -- High-High	S	R	M	N.A.	1, 2, 3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	S	R	M	N.A.	1, 2, 3
b. 4 kV Bus Loss of Voltage	S	R	M	N.A.	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	N.A.	1, 2, 3
d. Loss of Main Feed Pumps	N.A.	N.A.	R	N.A.	1, 2

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

TABLE 4.3-6A
APPENDIX R REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT		LOCATION	CHANNEL CHECK	CHANNEL CALIBRATION
1.	Steam Generators 1 and 4 Level	LSI Cabinet 1 and LSI Cabinet 4	M	R
2.	Steam Generators 2 and 3 Level	LSI Cabinet 2 and LSI Cabinet 4	M	R
3.	Steam Generators 1 and 4 Pressure	LSI Cabinet 4 and LSI Cabinet 5	M	R
4.	Steam Generators 2 and 3 Pressure	LSI Cabinet 4 and LSI Cabinet 6	M	R
5.	Reactor Coolant Loop 4 Temperature (Cold)	LSI Cabinet 4 and LSI Cabinet 5	M	R
6.	Reactor Coolant Loop 4 Temperature (Hot)	LSI Cabinet 4 and LSI Cabinet 5	M	R
7.	Reactor Coolant Loop 2 Temperature (Cold)	LSI Cabinet 4 and LSI Cabinet 6	M	R
8.	Reactor Coolant Loop 2 Temperature (Hot)	LSI Cabinet 4 and LSI Cabinet 6	M	R
9.	Pressurizer Level	LSI Cabinet 3	M	R
10.	Reactor Coolant System Pressure	LSI Cabinet 3	M	R
11.	Charging Cross-Flow Between Units	Corridor Elev. 587'	n/a	R*
12.	Source Range Neutron Detector (N-23)	LSI Cabinet 4	n/a	R

*Charging Cross-Flow between Units is an instrument common to both Unit 1 and 2. This surveillance will only be conducted on an interval consistent with Unit 1 refueling.

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Containment Pressure	M	R
2.	Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3.	Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4.	Reactor Coolant Pressure - Wide Range	M	R
5.	Pressurizer Water Level	M	R
6.	Steam Line Pressure	M	R
7.	Steam Generator Water Level - Narrow Range	M	R
8.	RWST Water Level	M	R
9.	Boric Acid Tank Solution Level	M	R
10.	Auxiliary Feedwater Flow Rate	M	R
11.	Reactor Coolant System Subcooling Margin Monitor	M	R
12.	PORV Position Indicator - Limit Switches	M	R
13.	PORV Block Valve Position Indicator - Limit Switches	M	R
14.	Safety Valve Position Indicator - Acoustic Monitor	M	R
15.	Incore Thermocouples (Core Exit Thermocouples)	M	R[1]
16.	Reactor Coolant Inventory Tracking System (Reactor Vessel Level Indication)	M[2]	R[3]
17.	Containment Sump Level	M	R
18.	Containment Water Level	M	R

[1]Partial range channel calibration for sensor to be performed below P-12 in MODE 3.
[2]With one train of Reactor Vessel Level Indication inoperable, Subcooling Margin Indication and Core Exit Thermocouples may be used to perform a CHANNEL CHECK to verify the remaining Reactor Vessel Indication train OPERABLE.
[3]Completion of channel calibration for sensors to be performed below P-12 in MODE 3.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Prompt notification to NRC pursuant to specification 6.9.1	All other S.G.s are C 1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3.	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.1.	N/A	N/A

S=3(N+n)% Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. One of the containment atmosphere particulate radioactivity monitoring channels (ERS-2301 or ERS-2401),

b. The containment sump level and flow monitoring system, and

c. Either the containment humidity monitor or one of the containment atmosphere gaseous radioactivity monitoring channels (ERS-2305 or ERS-2405).

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring channels are inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous (if being used) monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3.

b. Containment sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months.

c. Containment humidity monitor (if being used) - performance of CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

g. With PORVs and block valves not in the same line inoperable due to causes other than excessive seat leakage, within 1 hour restore the valves to OPERABLE status or close and de-energize the associated block valve and place the associated PORV in manual control in each respective line. Apply the portions of ACTION c or d above, relating to the OPERATIONAL MODE, as appropriate for two or three lines unavailable.

h. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE:

a. At least once per 31 days by performance of a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. At least once per 18 months by operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

c. At least once per 18 months by operating solenoid air control valves and check valves in PORV control systems through one complete cycle of full travel, and

d. At least once per 18 months by performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.11.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b, c, or d in Specification 3.4.11.

4.4.11.3 Deleted.

REACTOR COOLANT VENT SYSTEM

REACTOR VESSEL HEAD VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.1 Both Reactor Vessel head vent paths shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the common manual isolation valve in the Reactor vessel head vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Reactor Vessel head vent paths during venting operation, while in Modes 5 or 6.

REACTOR COOLANT VENT SYSTEM

PRESSURIZER STEAM SPACE VENTS

SURVEILLANCE REQUIREMENTS

4.4.12.2 Both Pressurizer steam space vent paths shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the common manual isolation valve in the Pressurizer steam space vent is sealed in the open position.

2. Cycling each of the remotely operated valves in each path through at least one complete cycle of full travel from the Control Room while in Modes 5 or 6.

3. Verifying flow through both of the Pressurizer steam space vent paths during venting operation, while in Modes 5 or 6.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
a. IMO-390	a. RWST to RHR	a. Open
b. IMO-315	b. Low head SI to Hot Leg	b. Closed
c. IMO-325	c. Low head SI to Hot Leg	c. Closed
d. IMO-262*	d. Mini flow line	d. Open
e. IMO-263*	e. Mini flow line	e. Open
f. IMO-261*	f. SI Suction	f. Open
g. ICM-305*	g. Sump Line	g. Closed
h. ICM-306*	h. Sump Line	h. Closed

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

*These valves must change position during the switchover from injection to recirculation flow following LOCA.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE charging pump, shall be demonstrated inoperable, by verifying that the motor circuit breakers have been removed from their electrical power supply circuits, at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F as determined at least once per hour when any RCS cold leg temperature is between 152°F and 200°F.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure—High-High test signal.

d. At least once per 5 years by verifying a water flow rate of at least 20 gpm (greater than or equal to 20 gpm) but not to exceed 50 gpm (less than or equal to 50 gpm) from the spray additive tank test line to each containment spray system with the spray pump operating on recirculation with a pump discharge pressure greater than or equal to 255 psig.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each containment isolation valve specified shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Containment Purge and Exhaust isolation signal, each Purge and Exhaust valve actuates to its isolation position.

4.6.3.1.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

DIVIDER BARRIER SEAL

LIMITING CONDITION FOR OPERATION

3.6.5.9 The divider barrier seal shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the divider barrier seal inoperable, restore the seal to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.9 The divider barrier seal shall be determined OPERABLE at least once per 18 months during shutdown by:

a. Removing two divider barrier seal test coupons and verifying that the physical properties of the test coupons are within the acceptable range of values shown in Table 3.6-2.

b. Visually inspecting at least 95 percent of the seal's entire length and:

1. Verifying that the seal and seal mounting bolts are properly installed, and

2. Verifying that the seal material shows no visual evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearances.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1

a. At least two independent component cooling water loops shall be OPERABLE.

b. At least one component cooling water flow path in support of Unit 1 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.3.1.a. - MODES 1, 2, 3, 4.
Specification 3.7.3.1.b. - At all times when Unit 1 is in MODES 1, 2, 3, or 4.

ACTION:

When Specification 3.7.3.1.a is applicable:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

When Specification 3.7.3.1.b is applicable:

With no flowpath to Unit 1 available, return at least one flowpath to available status within 7 days, or provide equivalent shutdown capability in Unit 1 and return at least one flow path to available status within the next 60 days, or have Unit 1 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours. The requirements of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection test signal.

c. By verifying pump performance pursuant to Specification 4.0.5.

d. At least once per 18 months during shutdown, verify that the unit cross-tie valves can cycle full travel. Following cycling, the valves will be verified to be in their closed positions.

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1

a. At least two independent essential service water loops shall be OPERABLE.

b. At least one essential service water flowpath associated with support of Unit 1 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.4.1.a. - MODES 1, 2, 3, and 4.
Specification 3.7.4.1.b. - At all times when Unit 1 is in MODES 1, 2, 3, or 4.

ACTION:

When Specification 3.7.4.1.a is applicable:

With only one essential service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

When Specification 3.7.4.1.b is applicable:

With no essential service water flow path available in support of Unit 1 shutdown functions, return at least one flow path to available status within 7 days or provide equivalent shutdown capability in Unit 1 and return the equipment to service within the next 60 days, or have Unit 1 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours. The requirements of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two essential service water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection test signal.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

2. a. Verifying that on a Safety Injection Signal from Unit 1, the system automatically operates in the pressurization/cleanup mode.

   b. Verifying that on a Safety Injection Signal from Unit 2, the system automatically operates in the pressurization/cleanup mode.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/16 inch W. G. relative to the outside atmosphere at a system flow rate of 6000 cfm plus or minus 10%.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm plus or minus 10%.

3/4.7.7 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.7.1 All safety-related snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.7.1.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Inspection

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 3.7-9. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 3.7-9 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment No. 156.

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified as acceptable for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that

ELECTRICAL POWER SYSTEMS

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. A day fuel tank containing a minimum of 70 gallons of fuel.

2. A fuel storage system containing a minimum indicated volume of 46,000 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

Will less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2.a.5.

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* For purpose of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is greater than or equal to 2.13 volts under float charge.

2. The specific gravity, corrected to 77°F, and full electrolyte level (fluid at the bottom of the maximum level indication mark), of each connected cell is greater than or equal to 1.200 and has not decreased more than 0.03 from the value observed during the previous test, and

3. The electrolyte level of each connected cell is between the top of the minimum level indication mark and the bottom of the maximum level indication mark.

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,

3. The battery charger will supply at least 140 amperes at greater than or equal to 250 volts for at least 4 hours.

d. At least once per 18 months, perform a battery service test during shutdown (MODES 5 or 6), by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status the actual or simulated emergency loads for the design duty cycle which is based on the composite load profile. The composite load profile envelopes both the LOCA/LOOP and Station Blackout profiles and provides the basis for the times listed in Table 4.8-2. The battery charger will be disconnected throughout the test. The battery terminal voltage shall be maintained greater than or equal to 210 volts throughout this test.

e. At least once per 60 months, conduct a performance test of battery capacity during shutdown (MODES 5 or 6), by verifying that the battery capacity is at least 80% of the manufacturer's rating. When this test is performed in place of a battery service test, a modified performance test shall be conducted.

Annual performance tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% from its capacity on the previous performance test, or is below 90% of the manufacturer's rating. If the battery has reached 85% of service life, delivers a capacity of 100% or greater of the manufacturer's rated capacity, and has shown no signs of degradation, performance testing at two year intervals is acceptable until the battery shows signs of degradation.

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

1 - 250-volt D.C. bus, and

1 - 250-volt battery bank and charger associated with the above D.C. bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and bus OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 250-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 250-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

STORAGE POOL VENTILATION SYSTEM**

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel storage pool exhaust ventilation system shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no fuel storage pool exhaust ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storagepool+ until at least one spent fuel storage pool exhaust ventilation system is restored to OPERABLE status*

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel storage pool ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:

1. Deleted.

2. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm± 10%.

*The crane bay roll-up door and the south door of the auxiliary building crane bay may be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool.

**Shared system with D. C. Cook - Unit 1.

+This does not include the main load block. For purposes of this specification, a deenergized main load block need not be considered a load.

5.6 FUEL STORAGE (Continued)

CRITICALITY - SPENT FUEL (Continued)

The equivalent reactivity criteria for Region 2 and Region 3 is defined via the following equations:

For Region 2 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-22{,}670 + 22{,}220\,E - 2{,}260\,E^2 + 149\,E^3$$

For Region 3 Storage

Minimum Assembly Average Burnup in MWD/MTU =

$$-26{,}745 + 18{,}746\,E - 1{,}631\,E^2 + 98.4\,E^3$$

Where E = Initial Peak Enrichment

5.6.1.2 Fuel stored in the spent fuel storage racks shall have a nominal fuel assembly enrichment as follows:

	Description		Maximum Nominal Fuel Assembly Enrichment Wt. % U-235
1)	Westinghouse	15 x 15 STD 15 x 15 OFA	4.95
2)	Exxon/ANF	15 x 15	4.95
3)	Westinghouse	17 x 17 STD 17 x 17 OFA 17 x 17 V5	4.95
4)	Exxon/ANF	17 x 17	4.95

Figure 5.6-3 intentionally deleted.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed 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 Operations Superintendent, who must be qualified as specified in Section 6.2.2.g.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY REVIEW COMMITTEE (PNSRC)

FUNCTION

6.5.1.1 The PNSRC shall function to advise the Site Vice President/Plant Manager, or designee, on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PNSRC shall be composed of Assistant Plant Managers, Department Superintendents, or supervisory personnel reporting directly to the Site Vice President/Plant Manager, Assistant Plant Managers or Department Superintendents. The membership shall represent the functional areas of the plant, including, but not limited to Operations, Technical Support, Licensing, Maintenance and Radiation Protection.

The PNSRC membership shall consist of at least one individual from each of the areas designated. All members, including the Chairman and his alternates, the members and their alternates, shall be designated by the Site Vice President/Plant Manager.

PNSRC members and alternates shall meet or exceed the minimum qualifications of ANSI N18.1-1971 Section 4.4 for comparable positions. The nuclear power plant operations individual shall meet the qualifications of Section 4.2.2 of ANSI N18.1-1971 except for the requirement to hold a current Senior Operator License. The operations individual must hold or have held a Senior Operator License or have been certified for equivalent senior operator knowledge at Cook Nuclear Plant or a similar reactor. The maintenance individual shall meet the qualifications of Section 4.2.3 of ANSI N18.1-1971.

ATTACHMENT 4 TO AEP:NRC:0433Q

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Evaluation of Significant Hazards Consideration

The Licensee has evaluated this proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;

2. create the possibility of a new or different kind of accident from any previously analyzed; or

3. involve a significant reduction in a margin of safety.

The Licensee proposes to make administrative changes to several technical specifications (T/S) for Donald C. Cook Nuclear Plant unit 1 and unit 2. The proposed changes include: (1) revising boron sampling requirements in mode 6; (2) deleting a reference to obsolete equipment in a footnote; (3) deleting a redundant figure; (4) correcting a reference to another requirement; (5) deleting obsolete notes; (6) adding to surveillance requirements; (7) clarifying instrumentation configuration; and (8) correcting typographical errors. These changes are proposed to remove obsolete information, provide consistency between unit 1 and unit 2, provide consistency with the Standard Technical Specifications, provide clarification, and correct typographical errors.

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change for boron sampling requirements in mode 6 does not affect the probability of a fuel handling accident. The unlikely event of a fuel assembly being misloaded is independent of the sampling frequency for fuel pool boron concentration. It has no impact on the event initiator, which is a human error while positioning a fuel assembly. The change has no impact on the assumptions for a fuel handling accident. The boron concentration requirement is not changed; there is sufficient boron in the fuel storage pool to maintain k_{eff} below 0.95 to preclude an inadvertent criticality. Therefore, the consequences of the accident will be mitigated as previously evaluated. The 72-hour maximum interval between samples is maintained. Operating experience has shown 72 hours to be adequate. Removing the additional limitation of sampling at least three times per week would allow the sample to be collected two or three times per week, consistent with the maximum 72-hour interval. This is acceptable because boron concentration changes occur slowly due to the large

volume of water in the system and relatively small volumes of dilution sources. The consequences are not increased because there are no changes to the spent fuel, shielding (water), or systems used to mitigate the consequences of an accident. Additionally, there is no change in the types or significant increase in the amounts of any effluents released offsite.

Deleting the redundant figure for equivalent reactivity criteria for regions in the spent fuel storage racks does not impact the storage requirements because the equations provide equivalent requirements. The unlikely event of a fuel assembly being misloaded is independent of the characteristics of the spent fuel in the pool. It has no impact on the event initiator, which is a human error while positioning a fuel assembly. The change has no impact the assumptions for a fuel handling accident because the fuel storage requirements are not changed. The consequences of an accident are not increased because the fuel storage requirements are not changed and no other changes are made to systems that mitigate the consequences of an accident.

The proposed changes to correct a reference to another requirement, delete obsolete notes, revise the name of drumming room roll-up door, and correct typographical errors are considered administrative. The reference leads to a section that no longer exists; the proposed change corrects the error. The notes permitted exceptions to requirements, and they are no longer required. The normal requirements have applied since the provisions expired. Deleting them eliminates extraneous information. The revised description of the door reflects the current use of the installed door. Correcting the typographical errors improves readability. The corrections are not intended to change the meaning. These changes do not affect accidents described in the UFSAR.

Adding new surveillance requirements to test the unit 2 pump performance pursuant to T/S 4.0.5 does not affect accident initiators or precursors. The change reflects ASME code requirements. Including the requirements in the corresponding section provides assurance that the pumps will operate as assumed in the accident analyses. As such, the probability and consequences of previously evaluated accidents is unchanged.

The proposed change to the description of instrumentation configuration is considered administrative because the configuration had been reviewed and approved by the NRC Staff, as documented in the Safety Evaluation Report for amendment 39 for DPR-58 and amendment 22 for DPR-74. There are no changes to the actual plant configuration. The change is intended to describe the installed equipment more clearly. The change does not affect the probability and consequences of previously evaluated accidents because the equipment is installed and operated as described in the correspondence related to the previous amendments.

Based on this review, it is concluded that the proposed changes do not involve a significant increase in the

probability of occurrence or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes remove obsolete information, provide consistency between unit 1 and unit 2, provide consistency with the Standard Technical Specifications, provide clarification, and correct typographical errors. These changes are considered administrative because they do not affect the design or operation of any system, structure, or component in the plant. The accident analysis assumptions and results are unchanged. No new failures or interactions have been created. Based on this review, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

   The proposed changes are considered administrative in nature. They do not affect any safety limits or T/S parameter limits. The proposed changes do not introduce new equipment, equipment modifications, or new or different modes of plant operation. These changes do not affect the operational characteristics of any equipment or systems. Based on this review, it is concluded that no reduction in the margin of safety will occur as a result of the changes.

In summary, based upon the above evaluation, the Licensee has concluded that these changes involve no significant hazards consideration.

ATTACHMENT 5 TO AEP:NRC:0433Q

ENVIRONMENTAL ASSESSMENT

Environmental Assessment

The Licensee has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. The Licensee has determined that this license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in attachment 4, this proposed amendment does not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

As documented in attachment 4, there will be no change in the types or significant increase in the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.