ATTACHMENT 2 to AEP:NRC:1137

PROPOSED, REVISED TECHNICAL SPECIFICATIONS PAGES

FOR DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant average temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for 4 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

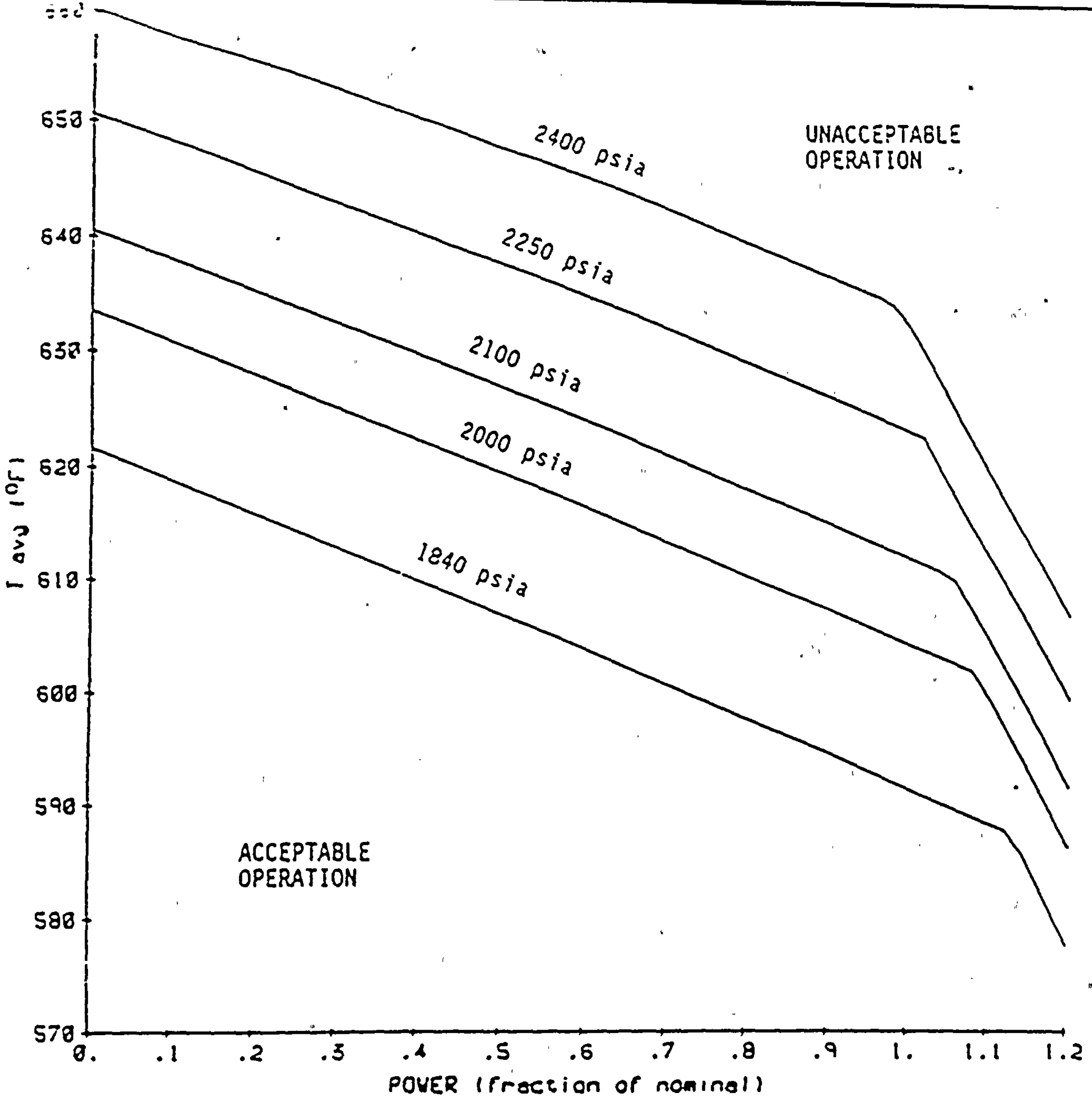
ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.



PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T-AVG IN DEGREES F)
1840	(0.0, 622.1), (1.13, 587.3), (1.20, 577.5)
2000	(0.0, 633.8), (1.08, 601.4), (1.20, 586.0)
2100	(0.0, 640.8), (1.06, 609.8), (1.20, 591.3)
2250	(0.0, 650.7), (1.02, 621.9), (1.20, 598.9)
2400	(0.0, 660.1), (0.98, 633.7), (1.20, 606.2)

FIGURE 2.1-1 REACTOR CORE SAFETY LIMITS

[illegible]

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level-Low-Low	Greater than or equal to 17% of narrow range instrument span - each steam generator	Greater than or equal to 16% of narrow range instrument span - each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Less than or equal to 0.71×10^6 lb/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 25% of narrow range instrument span - each steam generator	Less than or equal to 0.73×10^6 lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 24% of narrow range instrument span - each steam generator
15. Undervoltage - Reactor Coolant Pumps	Greater than or equal to 2750 volts - each bus	Greater than or equal to 2725 volts - each bus
16. Underfrequency - Reactor Coolant Pumps	Greater than or equal to 57.5 Hz - each bus	Greater than or equal to 57.4 Hz - each bus
17. Turbine Trip		
A. Low Fluid Oil Pressure	Greater than or equal to 800 psig	Greater than or equal to 750 psig
B. Turbine Stop Valve Closure	Greater than or equal to 1% open	Greater than or equal to 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident With Steam Line Pressure-Low	
a. Safety Injection (ECCS)	Less than or equal to 13.0#/23.0##
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation-Phase "A"	Less than or equal to 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Less than or equal to 14.0#/48.0##
h. Steam Line Isolation	Less than or equal to 11.0
7. Containment Pressure--High-High	
a. Containment Spray	Less than or equal to 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	Less than or equal to 10.0
d. Containment Air Recirculation Fan	Less than or equal to 600.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	Less than or equal to 2.5
b. Feedwater Isolation	Less than or equal to 11.0
9. Steam Generator Water Level--Low-Low	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
10. 4160 volt Emergency Bus Loss of Voltage	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
11. Loss of Main Feedwater Pumps	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
12. Reactor Coolant Pump Bus Undervoltage	
a. Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURED ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODE IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	M(1)	1,2,3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1,2,3
c. Containment Pressure--High-High	S	R	M(3)	1,2,3
d. Steam Flow in Two Steam Lines--High Coincident with Tavg--Low-Low	S	R	M	1,2,3
e. Steam Line Pressure--Low	S	R	M	1,2,3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	M	1,2,3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R	M	1,2,3
b. 4 kv Bus Loss of Voltage	S	R	M	1,2,3
c. Safety Injection	N.A.	N.A.	M(2)	1,2,3
d. Loss of Main Feed Pumps	N.A.	N.A.	R	1,2

TABLE 3.3-6 (Continued)
TABLE NOTATION

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per day.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirements, comply with the ACTION requirements of Specification 3.9.9. This ACTION is not required during the performance of containment integrated leak rate test.

ACTION 22A- With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements:

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3. Technical Specification Sections 3.0.3 and 3.0.4 Not Applicable.

ACTION 22B- With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements.

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3. In the event of an accident involving radiological releases initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours.

4. Technical Specification Sections 3.0.3 and 3.0.4 Not Applicable.

TABLE 3.3-10

Unit 1 and Common Area Fire Detection Systems

Detector System Location	Total Number of Detectors Heat (x/y)*	Flame (x/y)*	Smoke (x/y)*
Auxiliary Building			
a) Elevation 573			23/0C
b) Elevation 587			55/0C
c) Elevation 609			41/0C
d) Elevation 633			41/0C
e) Elevation 650			34/0C
f) New Fuel STGE Area			4/0C
g) RP Access Control & Chem Labs			25/0
U1 East Main Steam Valve Enclosure			28/0**
U1 Main Steam Line Area El. 612 (Around Containment)			13/0**
U1 NESW Valve Area El. 612			2/0
U1 4KV Switchgear (AB)		0/3	0/2
U1 4KV Switchgear (CD)		0/3	0/2
U1 Engr. Safety System Switchgear & XFMR. Rm.		0/5	0/9
U1 CRD, XFMR. & Switchgear Rm. Inverter & Bttry. Rms.		0/5	0/8
U1 Pressurizer Heater XFMR. Rm.			12/0
U1 Diesel Fuel Oil Transfer Pump Rm.	0/1		
U1 Diesel Generator Rm. 1AB	0/2		
U1 Diesel Generator Rm. 1CD	0/2		
U1 Diesel Generator Ramp Corr.			4/0
U1&2 AFWP Vestibule			2/0C
U1 Control Room			45/0
U1 Switchgear Cable Vault		0/10***	0/13
U1 Control Room Cable Vault			0/65****
U1 Aux. Cable Vault			0/6
U1&2 ESW Basement Area			4/0C
U1 ESW Pump & MCC Rms.			9/0

C System protects area common to both Units 1 and 2

*(x/y) x is number of Function A (early warning fire detection and notification only) instruments.
y is number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

** circuit contains both smoke and flame detectors

*** two circuits of five detectors each

**** two circuits of 32 and 33 detectors each

TABLE 3.3-11

POST-ACCIDENT MONITORING INTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2
2. Reactor Coolant Outlet Temperature- T_{HOT} (Wide Range)	2
3. Reactor Coolant Inlet Temperature- T_{COLD} (Wide Range)	2
4. Reactor Coolant Pressure-Wide Range	2
5. Pressurizer Water Level	2
6. Steam Line Pressure	2/steam generator
7. Steam Generator Water Level-Narrow Range	1/steam generator
8. Refueling Water Storage Tank Water Level	2
9. Boric Acid Tank Solution Level	1
10. Auxiliary Feedwater Flow Rate	1/steam generator*
11. Reactor Coolant System Subcooling Margin Monitor	1**
12. PORV Position Indicator -- Limit Switches***	1/Valve
13. PORV Block Valve Position Indicator -- Limit Switches	1/Valve
14. Safety Valve Position Indicator -- Acoustic Monitor	1/Valve
15. Incore Thermocouples (Core Exit Thermocouples)	2/Core Quadrant
16. Reactor Coolant Inventory Tracking System (Reactor Vessel Level Indication)	One Train (3 Channels/Train)
17. Containment Sump Level	1****
18. Containment Water Level	2****

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* Steam Generator Water Level Channels can be used as a substitute for the corresponding auxiliary feedwater flow rate channel instrument.

** PPC subcooling margin readout can be used as a substitute for the subcooling monitor instrument.

*** Acoustic monitoring of PORV position (1 channel per three valves - headered discharge) can be used as a substitute for the PORV Position Indicator - Limit Switches instruments.

****The requirements for these instruments will become effective after the level transmitters are modified or replaced and become operational. The schedule for modification or replacement of the transmitters is described in the Bases.

LIMITING CONDITION FOR OPERATION (Continued)

2. With two or more block valves inoperable,

within 1 hour either (1) restore a total of at least two block valves to OPERABLE status, or (2) close the block valves and remove power from the block valves, or (3) close the associated PORVs and remove power from their associated solenoid valves; and apply the portions of ACTION a.2 or a.3 above for inoperable PORVs, relating to OPERATIONAL MODE, as appropriate.

c. With PORVs and block valves not in the same line inoperable,*

within 1 hour either (1) restore the valves to OPERABLE status or (2) close and de-energize the other valve in each line. Apply the portions of ACTION a.2 or a.3 above, relating to OPERATIONAL MODE, as appropriate for two or three lines unavailable.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each of the three PORVs shall be demonstrated OPERABLE:

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

b. At least once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.11.2 Each of the three block valves shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel. The block valve(s) do not have to be tested when ACTION 3.4.11.a or 3.4.11.c is applied.

4.4.11.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by operating the valves through a complete cycle of full travel while the emergency buses are energized by the onsite diesel generators and onsite plant batteries. This testing can be performed in conjunction with the requirements of Specifications 4.8.2.3.2.d and 4.8.1.1.2.e.

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* PORVs isolated to limit RCS leakage through their seats and the block valves shut to isolate this leakage are not considered inoperable.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 12 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With an air lock inoperable, restore the air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By visual inspection after each opening to verify that the seal has not been damaged.

b. *Within 72 hours following each closing, perform an air leakage test without a simulated pressure force on the door by pressurizing the gap between the seals to 12 psig and verifying a seal leakage of no greater than 0.5 L_a.

*Exemption to Appendix "J" of 10 CFR 50.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 6 months, perform an air leakage test without a simulated pressure force on the door per 4.6.1.3.b., then perform an air leakage test with a simulated pressure force on the door, by pressurizing the volume between the seals to 12 psig and verifying a seal leakage of no greater than 0.0005 L_a.

d. At least once per 6 months by conducting an overall air lock leakage test at P_a (12 psig) and by verifying that the overall air lock leakage rate is within its limit.

e. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each isolation valve specified in Table 3.6-1 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 valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-1 (Continued)

VALVE NUMBER	FUNCTION	ISOLATION TIME IN SECONDS
C. CONTAINMENT PURGE EXHAUST** (Continued)		
12. VCR-205	UPPER COMP. PURGE AIR INLET	5
13. VCR-206	UPPER COMP. PURGE AIR OUTLET	5
14. VCR-207*	CONT. PRESS RELIEF FAN ISOLATION	5
D. MANUAL ISOLATION VALVES(1)		
1. ICM-111	RHR TO RC COLD LEGS	NA
2. ICM-129	RHR INLET TO PUMPS	NA
3. ICM-250	BORON INJECTION OUTLET	NA
4. ICM-251	BORON INJECTION OUTLET	NA
5. ICM-260	SAFETY INJECTION OUTLET	NA
6. ICM-265	SAFETY INJECTION OUTLET	NA
7. ICM-305	RHR/CTS SUCTION FROM SUMP	NA
8. ICM-306	RHR/CTS SUCTION FROM SUMP	NA
9. ICM-311	RHR TO RC HOT LEGS	NA
10. ICM-321	RHR TO RC HOT LEGS	NA
11. NPX 151 VI	DEAD WEIGHT TESTER	NA
12. PA 343	CONTAINMENT SERVICE AIR	NA
13. SF-151	REFUELING WATER SUPPLY	NA
14. SF-153	REFUELING WATER SUPPLY	NA
15. SF-159	REFUELING CAVITY DRAIN TO PURIFICATION SYSTEM	NA
16. SF-160	REFUELING CAVITY DRAIN TO PURIFICATION SYSTEM	NA
17. SI-171	SAFETY INJECTION TEST LINE	NA
18. SI-172	ACCUMULATOR TEST LINE	NA

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2

a. At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

1. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

2. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

b. At least one auxiliary feedwater flowpath in support of Unit 2 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.1.2.a - MODES 1, 2, 3.
Specification 3.7.1.2.b - At all times when Unit 2 is in MODES 1, 2, or 3.

ACTIONS:

When Specification 3.7.1.2.a is applicable:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

When Specification 3.7.1.2.b is applicable:

With no flow path 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.

REFUELING OPERATIONS

CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING*

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2,500 pounds shall be prohibited from travel over fuel assemblies in the storage pool. Loads carried over the spent fuel pool and the heights at which they may be carried over racks containing fuel shall be limited in such a way as to preclude impact energies over 24,240 in.-lbs., if the loads are dropped from the crane.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 Crane interlocks which prevent crane travel with loads in excess of 2,500 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

4.9.7.2 The potential impact energy due to dropping the crane's load shall be determined to be less than or equal to 24,240 in.-lbs. prior to moving each load over racks containing fuel.

* Shared system with Cook Nuclear Plant - Unit 2.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

Site Boundary For Gaseous and Liquid Effluents

5.1.3 The SITE BOUNDARY for gaseous and liquid effluents shall be as shown in Figure 5.1-3.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 115 feet.

b. Nominal inside height = 160 feet.*

c. Minimum thickness of concrete walls = 3'6".

d. Minimum thickness of concrete roof = 2'6".

e. Minimum thickness of concrete floor pad = 10 feet.

f. Nominal thickness of steel liner, side and dome = 3/8 inches.

g. Nominal thickness of steel liner, bottom = 1/4 inch.

h. Net free volume = 1.24×10^6 cubic feet.

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*From grade (Elev. 608') to inside of dome.

(l) Deleted by Amendment 63.

(m) Deleted by Amendment 19.

(n) Deleted by Amendment 28.

(o) <u>Fire Protection</u>

Amendment No. 12

The licensee may proceed with and is required to complete the modifications identified in Table 1 of the Fire Protection Safety Evaluation Report for the Donald C. Cook Nuclear Plant dated June 4, 1979. These modifications shall be completed in accordance with the dates contained in Table 1 of that SER or Supplements thereto. Administrative controls for fire protection as described in the licensee's submittals dated January 31, 1977 and October 27, 1977 shall be implemented and maintained.

Amendment No. 64, 121

(p) Deleted by Amendment 121

DEFINITIONS

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of radioactive liquid, resin and sludge wastes from liquid systems into a form that meets shipping and burial site requirements.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.30 The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints and the conduct of environmental radiological monitoring program.

GASEOUS RADWASTE TREATMENT SYSTEM

1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

VENTILATION EXHAUST TREATMENT SYSTEM

1.32 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

PURGE-PURGING

1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

VENTING

1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant average temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for 4 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

3.4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band about a targe flux difference. The target band is specified in the COLR.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference and with THERMAL POWER:

   1. Above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER, within 15 minutes:

      a) Either restore the indicated AFD to within the target band limits, or

      b) Reduce THERMAL POWER to less than 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER.

   2. Between 50% and 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER:

      a) POWER OPERATION may continue provided:

         1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

         2) The indicated AFD is within the limits specified in the COLR. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

      b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limit specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

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* See Special Test Exception 3.10.2

TABLE 3.3-1
REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHAN-NELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1,2 and *	12
2. Power Range, Neutron Flux	4	2	3	1, 2 and *	2#
3. Power Range, Neutron Flux, High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2## and *	4
B. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature Delta T					
Four Loop Operation	4	2	3	1, 2	6#
8. Overpower Delta T					
Four Loop Operation	4	2	3	1, 2	6#

TABLE 3.3-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRI	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Pressurizer Pressure - Low	4	2	3	1, 2	6#
10. Pressurizer Pressure - High	4	2	3	1, 2	6#
11. Pressurizer Water Level -- High	3	2	2	1, 2	7#
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	7#
14. Steam Generator Water Level-Low-Low	3/loop	2/loop in any operating loop	2/loop each operating loop	1, 2	7#
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water	2/loop - level and 2/loop-flow mismatch in same loop	1/loop - level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1, 2	7#

TABLE 3.3-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHAN-NELS	CHANNELS TO TRI	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Undervoltage-Reactor Coolant Pumps	4-1/bus	2	3	1	6#
17. Underfrequency-Reactor Coolant Pumps	4-1/bus	2	3	1	6#
18. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1	7#
B. Turbine Stop Valve Closure	4	4	3	1	6#
19. Safety Injection Input from ESF	2	1	2	1, 2	1
20. Reactor Coolant Pump Breaker Position Trip					
Above P-7	1/breaker	2	1/breaker per operating loop	1	11
21. Reactor Trip Breakers	2	1	2	1,2, 3*,4*,5*	1,13, 14
22. Automatic Trip Logic	2	1	2	1,2, 3*,4*,5*	1 14

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per day.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, comply with the ACTION requirements of Specification 3.9.9. This ACTION is not required during the performance of containment integrated leak rate test.

ACTION 22A- With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements:

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or
2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.
3. Technical Specification Sections 3.0.3 and 3.0.4 Not Applicable.

ACTION 22B- With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements.

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or
2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.
3. In the event of an accident involving radiological releases initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours.
4. Technical Specification Sections 3.0.3 and 3.0.4 Not Applicable.

TABLE 3.3-10
POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2
4. Reactor Coolant Pressure - Wide Range	2
5. Pressurizer Water Level	2
6. Steam Line Pressure	2/steam generator
7. Steam Generator Water Level - Narrow Range	1/steam generator
8. Refueling Water Storage Tank Water Level	2
9. Boric Acid Tank Solution Level	1
10. Auxiliary Feedwater Flow Rate	1/steam generator*
11. Reactor Coolant System Subcooling Margin Monitor	1**
12. PORV Position Indicator - Limit Switches***	1/valve
13. PORV Block Valve Position Indicator - Limit Switches	1/valve
14. Safety Valve Position Indicator - Acoustic Monitor	1/valve
15. Incore Thermocouples (Core Exit Thermocouples)	2/core quadrant
16. Reactor Coolant Inventory Tracking System (Reactor Vessel Level Indication)	one-train(3 channels/train)
17. Containment Sump Level	1****
18. Containment Water Level	2****

* Steam Generator Water Level Channels can be used as a substitute for the corresponding auxiliary feedwater flow rate channel instrument.

** PPC subcooling margin readout can be used as a substitute for the subcooling monitor instrument.

*** Acoustic monitoring of PORV position (1 channel per three valves - headered discharge) can be used as a substitute for the PORV Position Indicator - Limit Switches instruments.

**** The requirements for these instruments will become effective after the level transmitters are modified or replaced and become operational. The schedule for modification or replacement of the transmitters is described in the Bases.

TABLE 3.3-11

Unit 2 and Common Area Fire Detection Systems

Detection System Location	Heat (x/y)*	Flame (x/y)*	Smoke (x/y)*
		Total Number of Detectors	
Auxiliary Building			
a) Elevation 573			23/0C
b) Elevation 587			55/0C
c) Elevation 609			41/0C
d) Elevation 633			41/0C
e) Elevation 650			34/0C
f) New Fuel STGE Area			4/0C
U2 East Main Steam Valve Enclosure			28/0**
U2 Main Steam Line Area El. 612 (Around Containment)			13/0**
U2 NESW Valve Area El. 612			2/0
U2 4KV Switchgear (AB)		0/3	0/2
U2 4KV Switchgear (CD)		0/3	0/2
U2 Engr. Safety System Switchgear & XFMR. Rm.		0/5	0/14
U2 CRD, XFMR & Switchgear Rm. Inverter & AB Bttry. Rms.		0/5	0/17
U2 Pressurizer Heater XFMR. Rm.			12/0
U2 Diesel Fuel Oil XFMR. Rm.	0/1		
U2 Diesel Generator Rm. 2AB	0/2		
U2 Diesel Generator Rm. 2CD	0/2		
U2 Diesel Generator Ramp Corr.			4/0
U1&2 AFWP Vestibule			2/0C
U2 Control Room			42/0
U2 Switchgear Cable Vault		0/10***	0/13
U2 Control Rm. Cable Vault			0/76****
U2 Aux. Cable Vault			0/6
U1&2 ESW Basement Area			4/0C
U2 ESW Pump & MCC Rms.			9/0

C System protects area common to both Units 1 and 2

*(x/y) x is number of Function A (early warning fire detection and notification only) instruments.
y is number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

** circuit contains both smoke and flame detectors

*** two circuits of five detectors each

**** two circuits of 38 detectors each

[illegible]

LIMITING CONDITION FOR OPERATION (Continued)

2. With two or more block valves inoperable,

   Within 1 hour either (1) restore a total of at least two block valves to OPERABLE status, or (2) close the block valves and remove power from the block valves, or (3) close the associated PORVs and remove power from their associated solenoid valves; and apply the portions of ACTION a.2 or a.3 above for inoperable PORVs, relating to OPERATIONAL MODE, as appropriate.

c. With PORVs and block valves not in the same line inoperable,*

   within 1 hour either (1) restore the valves to OPERABLE status or (2) close and de-energize the other valve in each line. Apply the portions of ACTION a.2 or a.3 above, relating to OPERATIONAL MODE, as appropriate for two or three lines unavailable.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each of the three PORVs 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 performance of a CHANNEL CALIBRATION.

4.4.11.2 Each of the three block valves shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel. The block valve(s) do not have to be tested when ACTION 3.4.11.a or 3.4.11.c is applied.

4.4.11.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by operating the valves through a complete cycle of full travel while the emergency buses are energized by the onsite diesel generators and onsite plant batteries. This testing can be performed in conjunction with the requirements of Specifications 4.8.1.1.2.e and 4.8.2.3.2.d.

*PORVs isolated to limit RCS leakage through their seats and the block valves shut to isolate this leakage are not considered inoperable.

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 the control power locked out:

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. IMP-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.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 12 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With an air lock inoperable, maintain at least one door closed; restore the air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. *After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by performing an air leakage test without a simulated pressure force on the door by pressurizing the volume between the door seals to 12 psig and verifying a seal leakage rate of no greater than 0.5 L_a.

b. *Within 72 hours following each closing, perform an air leakage test without a simulated pressure force on the door per Specification 4.6.1.3.a.; then by performing an air leakage with a simulated pressure force on the door by pressurizing the volume between the door seals to 12 psig and verifying a seal leakage rate of no greater than 0.0005 L_a.

*Exemption to Appendix "J" of 10 CFR 50.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 6 months by conducting an overall air lock leakage test at P_a (12 psig) and by verifying that the overall air lock leakage rate is within its limit.

d. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

TABLE 3.6-1 (Cont'd)
CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME IN SECONDS
D. MANUAL ISOLATION VALVES(1) (Cont'd)		
3. ICM-250	BORON INJECTION OUTLET	NA
4. ICM-251	BORON INJECTION OUTLET	NA
5. ICM-260	SAFETY INJECTION OUTLET	NA
6. ICM-265	SAFETY INJECTION OUTLET	NA
7. ICM-305	RHR/CTS SUCTION FROM SUMP	NA
8. ICM-306	RHR/CTS SUCTION FROM SUMP	NA
9. ICM-311#	RHR TO RC HOT LEGS	NA
10. ICM-321#	RHR TO RC HOT LEGS	NA
E. OTHER		
1. CS-442-1	SEAL WTR. TO RCP #1	NA
2. CS-442-2	SEAL WTR. TO RCP #2	NA
3. CS-442-3	SEAL WTR. TO RCP #3	NA
4. CS-442-4	SEAL WTR. TO RCP #4	NA

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2

a. At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

   1. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

   2. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

b. At least one auxiliary feedwater flow path in support of Unit 1 shutdown function shall be available.

APPLICABILITY: Specification 3.7.1.2.a - MODES 1, 2, 3.
Specification 3.7.1.2.b - At all times when Unit 1 is in MODES 1, 2, or 3.

ACTIONS:

When Specification 3.7.1.2.a is applicable:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT STANDBY within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

When Specification 3.7.1.2.b is applicable:

With no flow path to Unit 1 available, return at least one flow path 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.

3/4.8.3 Alternative A.C. Power Sources

LIMITING CONDITION FOR OPERATION

3.8.3.1 The steady state bus voltage for the manual alternate reserve source* shall be greater than or equal to 90% of the nominal bus voltage.

APPLICABILITY: Whenever the manual alternate reserve source (69 kV) is connected to more than two buses.

ACTION: With bus voltage less than 90% nominal, adjust load on the remaining buses to maintain steady state bus voltage greater than or equal to 90% limit.

SURVEILLANCE REQUIREMENTS

4.8.3.1 No additional surveillance requirements other than those required by Specifications 4.8.1.1.1 and 4.8.1.2.

*Shared with Cook Nuclear Plant Unit 1.

CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING*

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2,500 pounds shall be prohibited from travel over fuel assemblies in the storage pool. Loads carried over the spent fuel pool and the heights at which they may be carried over racks containing fuel shall be limited in such a way as to preclude impact energies over 24,240 in.-lbs., if the loads are dropped from the crane.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 Crane interlocks which prevent crane travel with loads in excess of 2,500 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

4.9.7.2 The potential impact energy due to dropping the crane's load shall be determined to be less than or equal to 24,240 in.-lbs. prior to moving each load over racks containing fuel.

* Shared system with Cook Nuclear Plant - Unit 1.

[illegible]

REFUELING OPERATIONS

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Exhaust isolation system shall be OPERABLE.

APPLICABILITY: During Core Alterations or movement of irradiated fuel within the containment.

ACTION:

With the Containment Purge and Exhaust isolation system inoperable, close each of the Purge and Exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Exhaust isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment Purge and Exhaust isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.

5.0 DESIGN FEATURES

5.1 SITE

Exclusion Area

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

Low Population Zone

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

Site Boundary For Gaseous and Liquid Effluents

5.1.3 The SITE BOUNDARY for gaseous and liquid effluents shall be as shown in Figure 5.1-3.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 115 feet.

b. Nominal inside height = 160 feet.

c. Minimum thickness of concrete walls = 3'6".

d. Minimum thickness of concrete roof = 2'6".

e. Minimum thickness of concrete floor pad = 10 feet.

f. Nominal thickness of steel liner = 3/8 inches.

g. Net free volume = 1.24×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained in accordance with the original design provisions contained in Section 5.2.2 of the FSAR.

ATTACHMENT 3 TO AEP:NRC:1137

EXISTING T/S PAGES MARKED TO REFLECT PROPOSED CHANGES

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant average temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for 4 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig

APPLICABILITY: MODES 1, 2, 3, 4 and 5

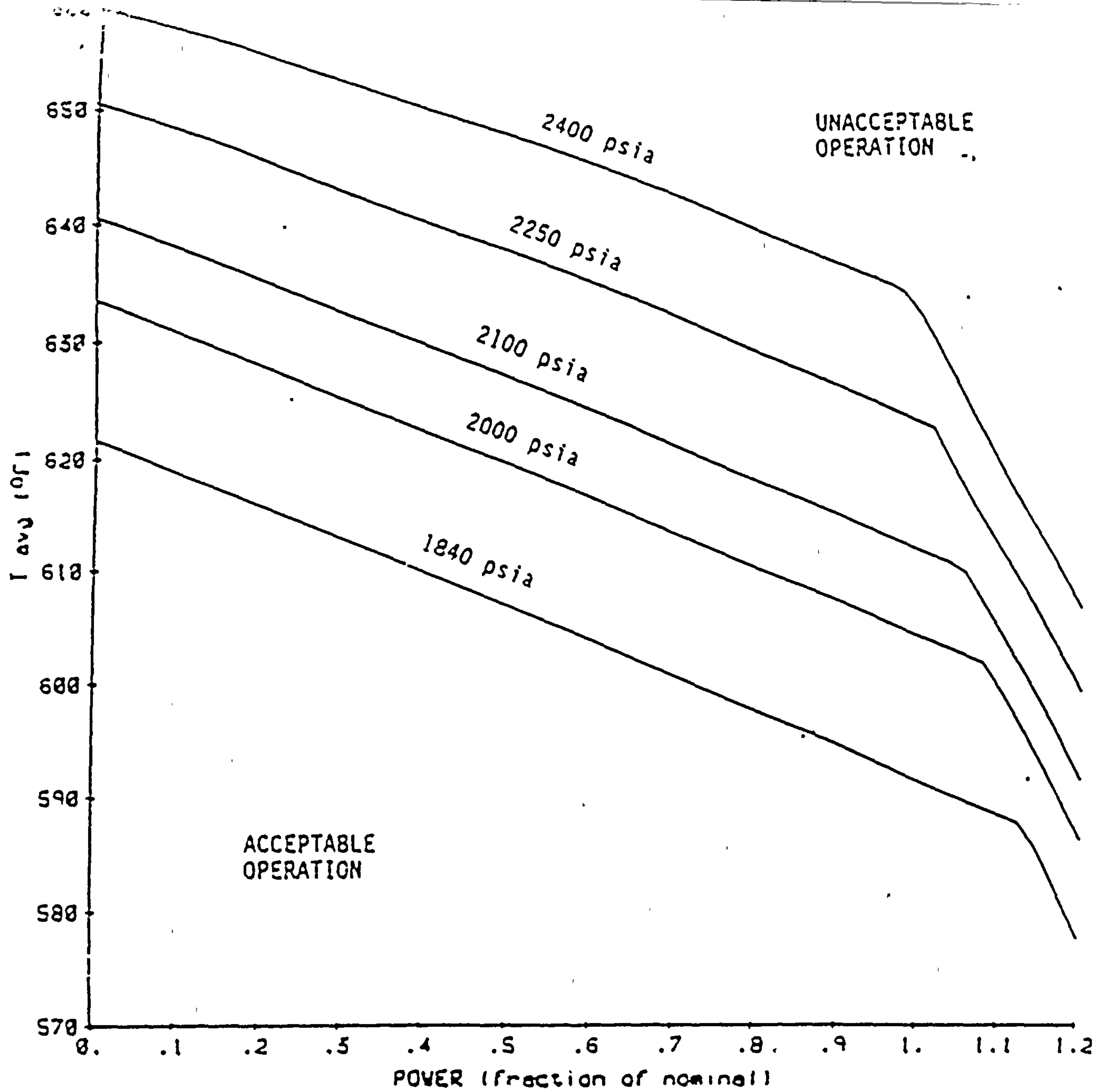
ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.



PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T-AVG IN DEGREES F)
1840	(0.0, 622.1), (1.13, 587.3). (1.20, 577.5)
2000	(0.0, 633.8), (1.08, 601.4). (1.20, 586.0)
2100	(0.0, 640.8), (1.06, 609.8). (1.20, 591.3)
2250	(0.0, 650.7), (1.02, 621.9). (1.20, 598.9)
2400	(0.0, 660.1), (0.98, 633.7), (1.20, 606.2)

~~RATED THERMAL POWER = 3413 MWT~~

FIGURE 2.1-1 REACTOR CORE SAFETY LIMITS

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level - Low-Low	$\geq$ 17% of narrow range instrument span - each steam generator	$\geq$ 16% of narrow range instrument span - each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	$\leq 0.71 \times 10^6$ lb/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 25% of narrow range instrument span - each steam generator	$\leq 0.73 \times 10^6$ lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 24% of narrow range instrument span - each steam generator
15. Undervoltage - Reactor Coolant Pumps	$\geq$ 2750 volts - each bus	$\geq$ 2725 volts - each bus
16. Underfrequency - Reactor Coolant Pumps	$\geq$ 57.5 Hz - each bus	$\geq$ 57.4 Hz - each bus
17. Turbine Trip		
A. ~~Low Trip System Pressure~~ Low Fluid Oil Pressure	$\geq$ 800 psig	$\geq$ 750 psig
B. Turbine Stop Valve Closure	$\geq$ 1% open	$\geq$ 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

Flip Orientation
Write out signs

TABLE 3.3-5 (Continued) Write out signs

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident With Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 13.0#/23.0##
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	≤ 14.0#/48.0##
h. Steam Line Isolation	≤ 11.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 10.0
d. Containment Air Recirculation Fan	≤ ~~660.0~~ 600
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 11.0
9. Steam Generator Water Level--Low-Low	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0
10. 4160 volt Emergency Bus Loss of Voltage	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
11. Loss of Main Feedwater Pumps	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
12. Reactor Coolant Pump Bus Undervoltage	
a. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURED ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODE IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	M(1)	1,2,3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1,2,3
c. Containment Pressure--High-High	S	R	M(3)	1,2,3
d. Steam Flow in Two Steam Lines--High Coincident with Tavg--Low-Low ~~Pressure--Low~~	S	R	M	1,2,3
e. Steam Line Pressure-Low	S	R	m	1,2,3
TURBINE TRIP & FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	M	1,2,3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R	M	1,2,3
b. 4 kv Bus Loss of Voltage	S	R	M	1,2,3
c. Safety Injection	N.A.	N.A.	M(2)	1,2,3
d. Loss of Main Feed Pumps	N.A.	N.A.	R	1,2

[illegible]

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per day.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirements, comply with the ACTION requirements of Specification 3.9.9. This ACTION is not required during the performance of containment integrated leak rate test.

ACTION 22A - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements:

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3. Technical Specification Sections 3.0.3, and 3.0.4 ~~and [illegible]~~ Not Applicable.

ACTION 22B - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements.

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3. In the event of an accident involving radiological releases initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours.

4. Technical Specification Sections 3.0.3, " 3.0.4 ~~and [illegible]~~ Not Applicable.

TABLE 3.3-10
Unit 1 and Common Area Fire Detection Systems

Detector System Location	Total Number of Detectors Heat (x/y)*	Flame (x/y)*	Smoke (x/y)*
Auxiliary Building			
a) Elevation ~~[illegible]~~ 573			23/0C
b) Elevation 587			55/0C
c) Elevation 609			41/0C
d) Elevation 633			41/0C
e) Elevation 650			34/0C
f) New Fuel STGE Area			4/0C
g) RP Access Control & Chem Labs			25/0
U1 East Main Steam Valve Enclosure			28/0**
U1 Main Steam Line Area El. 612 (Around Containment)			13/0**
U1 NESW Valve Area El. 612			2/0
U1 4KV Switchgear (AB)		0/3	0/2
U1 4KV Switchgear (CD)		0/3	0/2
U1 Engr. Safety System Switchgear & XFMR. Rm.		0/5	0/9
U1 CRD, XFMR. & Switchgear Rm. Inverter & Bttry. Rms.		0/5	0/8
U1 Pressurizer Heater XFMR. Rm.			12/0
U1 Diesel Fuel Oil Transfer Pump Rm.	0/1		
U1 Diesel Generator Rm. 1AB	0/2		
U1 Diesel Generator Rm. 1CD	0/2		
U1 Diesel Generator Ramp Corr.			4/0
U1&2 AFWP Vestibule			2/0C
U1 Control Room			45/0
U1 Switchgear Cable Vault		0/10***	0/13
U1 Control Room Cable Vault			0/65****
U1 Aux. Cable Vault			0/6
U1&2 ESW Basement Area			4/0C
U1 ESW Pump & MCC Rms.			9/0

C System protects area common to both Units 1 and 2

*(x/y) x is number of Function A (early warning fire detection and notification only) instruments.
y is number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

** circuit contains both smoke and flame detectors
*** two circuits of five detectors each
**** two circuits of 32 and 33 detectors each

Flip Orientation

TABLE 3.3-11
POST-ACCIDENT MONITORING INSTRUMENTATION

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

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* Steam Generator Water Level Channels can be used as a substitute for the corresponding auxiliary feedwater flow rate channel instrument

PPC

** ~~PRODAC 250~~ subcooling margin readout can be used as a substitute for the subcooling monitor instrument.

*** Acoustic monitoring of PORV position (1 channel per three valves - headered discharge) can be used as a substitute for the PORV Indicator - Limit Switches instruments.

**** The requirements for these instruments will become effective after the level transmitters are modified or replaced and become operational. The schedule for modification or replacement of the transmitters is described in the Bases.

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~~Amendment No. 108, 112 (Effective before start-up following refueling outage currently scheduled in 2/89)~~

LIMITING CONDITION FOR OPERATION (Continued)

2. With two or more block valves inoperable,

within 1 hour either (1) restore a total of at least two block valves to OPERABLE status, or (2) close the block valves and remove power from the block valves, or (3) close the associated PORVs and remove power from their associated solenoid valves; and apply the portions of ACTION a.2 or a.3 above for inoperable PORVs, relating to OPERATIONAL MODE, as appropriate.

c. With PORVs and block valves not in the same line inoperable,*

within 1 hour either (1) restore the valves to OPERABLE status or (2) close and de-energize the other valve in each line. Apply the portions of ACTION a.2 or a.3 above, relating to OPERATIONAL MODE, as appropriate for two or three lines unavailable.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1. Each of the three PORVs 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 performance of a CHANNEL CALIBRATION.

4.4.11.2 Each of the three block valves shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel. The block valve(s) do not have to be tested when ACTION 3.4.11.a or 3.4.11.c is applied.

4.4.11.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by operating the valves through a complete cycle of full travel while the emergency buses are energized by the onsite diesel generators and onsite plant batteries. This testing can be performed in conjunction with the requirements of Specifications ~~4.8.1.1.2.b and 4.8.2.3.2.d.~~
4.8.2.3.2.d and 4.8.1.1.2.e

* PORVS isolated to limit RCS leakage through their seats and the block valves shut to isolate this leakage are not considered inoperable.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of $\leq 0.05\ L_a$ at P_a, 12 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With an air lock inoperable, restore the air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By visual inspection after each opening to verify that the seal has not been damaged.

b. ~~*After each opening, except when the air lock is being used for multiple entries, when it shall be done at least once per 3 days, by performing an air leakage test~~ without a simulated pressure force on the door by pressurizing the gap between the seals to 12 psig and verifying a seal leakage of no greater than $0.5\ L_a$.

~~*Exemption to Appendix "J" of 10 CFR 50.~~

Within 72 hours following each closing, perform an air leakage test

[illegible]

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 6 months, perform an air leakage test without a simulated pressure force on the door per 4.6.1.3.b., then perform an air leakage test with a simulated pressure force on the door, by pressurizing the ~~gas~~ between the seals to 12 psig and verifying a seal leakage of no greater than 0.0005 L_a.
volume

d. At least once per 6 months by conducting an overall air lock leakage test at P_a (12 psig) and by verifying that the overall air lock leakage rate is within its limit.

e. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

~~valve or~~ its associated actuator, control or power circuit by performance ~~of the cycling test,~~ above, and verification of isolation ~~time.~~

4.6.3.1.2 Each isolation valve specified in Table 3.6-1 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 valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-1 (Continued)

	VALVE NUMBER	FUNCTION	ISOLATION TIME IN SECONDS
C.	CONTAINMENT PURGE EXHAUST (Continued)**		
12.	VCR-205	Upper Comp. Purge Air Inlet	5
13.	VCR-206	Upper Comp. Purge Air Outlet	5
14.	VCR-207*	Cont. Press Relief Fan Isolation	5
D.	MANUAL ISOLATION VALVES(1)		
1.	ICM-111	RHR to RC Cold Legs	NA
2.	ICM-129	RHR Inlet to Pumps	NA
3.	ICM-250	Boron Injection ~~Inlet~~ Outlet	NA
4.	ICM-251	Boron Injection I~~nlet~~	NA
5.	ICM-260	Safety Injection I~~nlet~~	NA
6.	ICM-265	Safety Injection I~~nlet~~	NA
7.	ICM-305	~~RHR~~ Suction from Sump RHR/CTS	NA
8.	ICM-306	~~RHR~~ Suction from Sump	NA
9.	ICM-311	RHR to RC Hot Legs	NA
10.	ICM-321	RHR to RC Hot Legs	NA
11.	NPX 151 VI	Dead Weight Tester	NA
12.	PA-343	Containment Service Air	NA
13.	SF-151	Refueling Water Supply	NA
14.	SF-153	Refueling Water Supply	NA
15.	SF-159	Refueling Cavity Drain to Purification System	NA
16.	SF-160	Refueling Cavity Drain to Purification System	NA
17.	SI-171	Safety Injection Test Line	NA
18.	SI-172	Accumulator Test Line	NA

Flip orientation

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2

a. At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

   1. Two motor-driven feedwater pumps, each capable of being powered from separate emergency busses, and

   2. One steam turbine feedwater pump capable of being powered from an OPERABLE steam supply system.

b. At least one auxiliary feedwater flowpath in support of Unit 2 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.1.2.a - MODES 1, 2, 3.
Specification 3.7.1.2.b - At all times when Unit 2 is in MODES 1, 2, or 3.

ACTIONS:

When Specification 3.7.1.2.a is applicable:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

When Specification 3.7.1.2.b is applicable:

With no flow path 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.

CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING*

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2,500 pounds shall be prohibited from travel over fuel assemblies in the storage pool. Loads carried over the spent fuel pool and the heights at which they may be carried over racks containing fuel shall be limited in such a way as to preclude impact energies over 24,240 in.-lbs., if the loads are dropped from the crane.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 Crane interlocks ~~and physical stops~~ which prevent crane travel with loads in excess of 2,500 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

4.9.7.2 The potential impact energy due to dropping the crane's load shall be determined to be $\leq$ 24,240 in.-lbs. prior to moving each load over racks containing fuel.

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

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

Site Boundary For Gaseous and Liquid Effluents

5.1.3 The site boundary for gaseous and liquid effluents shall be shown in Figure 5.1-3.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 115 feet.

b. Nominal inside height = 160 feet.*

c. Minimum thickness of concrete walls = 3'6".

d. Minimum thickness of concrete roof = 2'6".

e. Minimum thickness of concrete floor pad = 10 feet.

f. Nominal thickness of steel liner, side and dome = 3/8 inches.

g. Nominal thickness of steel liner, bottom = 1/4 inch.

h. Net free volume = 1.24×10^6 cubic feet.

* From grade (Elev. 608') to inside of dome.

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 position, except for (1) the Radiation Protection Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (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.

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 Coordinator and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and ~~Appendix~~ "A" of 10 CFR Part 55.

Chpt

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 Plant Manager on all matters related to nuclear safety.

RESPONSIBILITIES

6.5.1.6 The PNSRC shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Chairman of the NSDRC.

f. Review of all REPORTABLE EVENTS.

g. Review of facility operations to detect potential safety hazards.

h. Performance of special reviews, investigations of analyses and reports thereon as requested by the Chairman of the NSDRC.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the NSDRC.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the NSDRC.

k. Review of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the NSDRC.

l. Review of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste treatment system.

(l) Deleted by Amendment 63.

(m) Deleted by Amendment 19.

(n) Deleted by Amendment 28.

(o) <u>Fire Protection</u>

Amendment No. 12

The licensee may proceed with and is required to complete the modifications identified in Table 1 of the Fire Protection Safety Evaluation Report for the Donald C. Cook Nuclear Plant dated June 4, 1979. These modifications shall be completed in accordance with the dates contained in Table 1 of that SER or Supplements thereto. Administrative controls for fire protection as described in the licensee's submittals dated January 31, 1977 and October 27, 1977 shall be implemented and maintained.

Amendment No. 64, 121

(p) Deleted by Amendment 121

[illegible]

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of radioactive liquid, resins and sludge wastes from liquid systems into a form that meets shipping and burial site requirements.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.30 The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints and the conduct of environmental radiological monitoring program.

GASEOUS RADWASTE TREATMENT SYSTEM

1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

VENTILATION EXHAUST TREATMENT SYSTEM

1.32 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

PURGE-PURGING

1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

VENTING

1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

average

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for 4 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level-Low-Low	Greater than or equal to 21% of narrow range instrument span - each steam generator	Greater than or equal to 19.2% of narrow range instrument span - each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Less than or equal to 1.47 x 10^6 lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 25% of narrow range instrument span - each steam generator	Less than or equal to 1.56 x 10^6 lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 24% of narrow range instrument span - each steam generator
15. Undervoltage - Reactor Coolant Pumps	Greater than or equal to 2905 volts - each bus	Greater than or equal to 2870 volts - each bus
16. Underfrequency - Reactor Coolant Pumps	Greater than or equal to 57.5 Hz - each bus	Greater than or equal to 57.4 Hz - each bus
17. Turbine Trip		
A. Low ~~Trip System~~ Fluid Oil Pressure	Greater than or equal to 58 psig	Greater than or equal to 57 psig
B. Turbine Stop Valve Closure	Greater than or equal to 1% open	Greater than or equal to 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band about a target flux difference. The target ban is specified in the COLR.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference and with THERMAL POWER:

  1. Above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER, within 15 minutes:

     a) Either restore the indicated AFD to within the target band limits, or

     b) Reduce THERMAL POWER to less than 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER.

  2. Between 50% and 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER:

     a) POWER OPERATION may continue provided:

        1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

        2) The indicated AFD is within the limits specified in the COLR. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

     b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limit specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

* See Special Test Exception 3.10.2

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	12
2. Power Range, Neutron Flux	4	2	3	1, 2 and *	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2## and *	4
B. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature ΔT Four Loop Operation	4	2	3	1, 2	6#
8. Overpower ΔT Four Loop Operation	4	2	3	1, 2	6#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Pressurizer Pressure-Low	4	2	3	1, 2	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1, 2	7#
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	7#
14. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2	7#
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch in same loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1, 2	7#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Undervoltage-Reactor Coolant Pumps	4-1/bus	2	3	1	6 #
17. Underfrequency-Reactor Coolant Pumps	4-1/bus	2	3	1	6 #
18. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1	7 #
B. Turbine Stop Valve Closure	4	4	3	1	6 #
19. Safety Injection Input from ESF	2	1	2	1, 2	1
20. Reactor Coolant Pump Breaker Position Trip					
Above P-7	1/breaker	2	1/breaker per operating loop	1	11
21. Reactor Trip Breakers	2	1	2	1, 2, 3*, 4*, 5*	1, 13, 14
22. Automatic Trip Logic	2	1	2	1, 2, 3~~*, 4*, 5*~~ 3* 4*, 5*	1, ~~11~~ 14

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per day.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels Operable requirements, comply with the ACTION requirements of Specification 3.9.9. This ACTION is not required during the performance of containment integrated leak rate test.

ACTION 22A - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements:

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3. Technical Specification Sections 3.0.3, and 3.0.4 ~~and 6.9.1.13~~ Not Applicable.

ACTION 22B - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements.

1. either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2. prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3. In the event of an accident involving radiological releases initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours.

4. Technical Specification Sections 3.0.3, 3.0.4 ~~and 6.9.1.13~~ Not Applicable.

TABLE 3.3-11

Unit 2 and Common Area Fire Detection Systems

Detection System Location	Total Number of Detectors Heat (x/y)*	Flame (x/y)*	Smoke (x/y)*
Auxiliary Building			
a) Elevation 587			55/0C
b) Elevation 609			41/0C
c) Elevation 633			41/0C
d) Elevation 573			23/0C
e) Elevation 650			34/0C
f) New Fuel STGE Area			4/0C
U2 East Main Steam Valve Enclosure			28/0**
U2 Main Steam Line Area El. 612 (Around Containment)			13/0**
U2 NESW Valve Area El. 612			2/0
U2 4KV Switchgear (AB)		0/3	0/2
U2 4KV Switchgear (CD)		0/3	0/2
U2 Engr. Safety System Switchgear & XFMR. Rm.		0/5	0/14
U2 CRD, XFMR & Switchgear Rm. Inverter & AB Bttry. Rms.		0/5	0/17
U2 Pressurizer Heater XFMR. Rm.			12/0
U2 Diesel Fuel Oil XFMR. Rm.	0/1		
U2 Diesel Generator Rm. 2AB	0/2		
U2 Diesel Generator Rm. 2CD	0/2		
U2 Diesel Generator Ramp Corr.			4/0
U1&2 AFWP Vestibule			2/0C
U2 Control Room			42/0
U2 Switchgear Cable Vault		0/10***	0/13
U2 Control Rm. Cable Vault			0/76****
U2 Aux. Cable Vault			0/6
U1&2 ESW Basement Area			4/0C
U2 ESW Pump & MCC Rms.			9/0

C System protects area common to both Units 1 and 2

*(x/y) x is number of Function A (early warning fire detection and notification only) instruments.
y is number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

** circuit contains both smoke and flame detectors
*** two circuits of five detectors each
**** two circuits of 38 detectors each

TABLE 3.3-10
POST-ACCIDENT MONITORING INSTRUMENTATION

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

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* Steam Generator Water Level Channels can be used as a substitute for the corresponding auxiliary feedwater flow rate channel instrument

** ~~PRODAC 250~~ subcooling margin readout can be used as a substitute for the subcooling monitor instrument.

*** Acoustic monitoring of PORV position (1 channel per three valves - headered discharge) can be used as a substitute for the PORV Indicator - Limit Switches instruments.

**** The requirements for these instruments will become effective after the level transmitters are modified or replaced and become operational. The schedule for modification or replacement of the transmitters is described in the Bases. 95

~~Amendment No. 92, 95 (Effective before start up following refueling outage currently scheduled in early 1988)~~

LIMITING CONDITION FOR OPERATION (Continued)

2. With two or more block valves inoperable,

Within 1 hour either (1) restore a total of at least two block valves to OPERABLE status, or (2) close the block valves and remove power from the block valves, or (3) close the associated PORVs and remove power from their associated solenoid valves; and apply the portions of ACTION a.2 or a.3 above for inoperable PORVs, relating to OPERATIONAL MODE, as appropriate.

c. With PORVs and block valves not in the same line inoperable,*

within 1 hour either (1) restore the valves to OPERABLE status or (2) close and de-energize the other valve in each line. Apply the portions of ACTION a.2 or a.3 above, relating to OPERATIONAL MODE, as appropriate for two or three lines unavailable.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each of the three PORVs 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 performance of a CHANNEL CALIBRATION.

4.4.11.2 Each of the three block valves shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel. The block valve(s) do not have to be tested when ACTION 3.4.11.a or 3.4.11.c is applied.

4.4.11.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by operating the valves through a complete cycle of full travel while the emergency buses are energized by the onsite diesel generators and onsite plant batteries. This testing can be performed in conjunction with the requirements of Specifications ~~4.8.1.1.2.c~~ and 4.8.2.3.2.d.

*PORVs isolated to limit RCS leakage through their seats and the block valves shut to isolate this leakage are not considered inoperable.

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. IMP-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 otherswise 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.

[illegible]

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of $\leq 0.05\ L_a$ at P_a, 12.0 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With an air lock inoperable, maintain at least one door closed; restore the air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. *After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by performing an air leakage test without a simulated pressure force on the door by pressurizing the volume between the door seals to 12 psig and verifying a seal leakage rate of no greater than $0.5\ L_a$.

b. At ~~least once per 6 months by performing~~ an air leakage test without a simulated pressure force on the door per Specification 4.6.1.3.a; then by performing an air leakage with a simulated pressure force on the door by pressurizing the volume between the door seals to 12 psig and verifying a seal leakage rate of no greater than $0.0005\ L_a$.

*Exemption to Appendix "J" of 10 CFR 50.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 6 months by conducting an overall air lock leakage test at P_a (12 psig) and by verifying that the overall air lock leakage rate is within its limit.

d. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME IN SECONDS
D. MANUAL ISOLATION VALVES[1] (Continued)			
3.	1CM-250	Boron Injection ~~Inlet~~	NA
4.	1CM-251	Boron Injection ~~Inlet~~	NA
5.	1CM-260	Safety Injection ~~Inlet~~	NA
6.	1CM-265	Safety Injection ~~Inlet~~	NA
7.	1CM-305	RHR Suction From Sump	NA
8.	1CM-306	RHR Suction From Sump	NA
9.	1CM-311#	RHR to RC Hot Legs	NA
10.	1CM-321#	RHR to RC Hot Legs	NA
E. OTHER			
1.	CS-442-1	Seal Wtr. to RCP #1	NA
2.	CS-442-2	Seal Wtr. to RCP #2	NA
3.	CS-442-3	Seal Wtr. to RCP #3	NA
4.	CS-442-4	Seal Wtr. to RCP #4	NA

LIMITING CONDITION FOR OPERATION

3.7.1.2

a. At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

1. Two feedwater pumps, each capable of being powered from separate emergency busses, and

2. One feedwater pump capable of being powered from an OPERABLE steam supply system.

b. At least one auxiliary feedwater flow path in support of Unit 1 shutdown functions shall be available.

APPLICABILITY: Specification 3.7.1.2.a - MODES 1, 2, 3.
Specification 3.7.1.2.b - At all times when Unit 1 is in MODES 1, 2, or 3.

ACTIONS:

Specification 3.7.1.2.a is applicable:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

When Specification 3.7.1.2.b is applicable:

With no flow path to Unit 1 available, return at least one flow path 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.

3/4.8.3 Alternative A.C. Power Sources

LIMITING CONDITION FOR OPERATION

3.8.3.1 The steady state bus voltage for the manual alternate reserve source* shall be greater than or equal to 90% of the nominal bus voltage.

APPLICABILITY: Whenever the manual alternate reserve source (69 kV) is connected to more than two buses.

ACTION: With bus voltage less than 90% nominal, adjust load on the remaining buses to maintain steady state bus voltage greater than or equal to 90% limit.

SURVEILLANCE REQUIREMENTS

4.8.3.1 No additional surveillance requirements other than those required by Specifications 4.8.1.1.1 and 4.8.1.2.

*Shared with D. C. Cook Unit ~~2~~ 1.

CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING*

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2,500 pounds shall be prohibited from travel over fuel assemblies in the storage pool. Loads carried over the spent fuel pool and the heights at which they may be carried over racks containing fuel shall be limited in such a way as to preclude impact energies over 24,240 in.-lbs., if the loads are dropped from the crane.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 Crane interlocks ~~and physical~~ stops which prevent crane travel with loads in excess of 2,500 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

4.9.7.2 The potential impact energy due to dropping the crane's load shall be determined to be $\leq$ 24,240 in.-lbs. prior to moving each load over racks containing fuel.

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

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Exhaust isolation system shall be OPERABLE.

APPLICABILITY: During Core Alterations or movement of irradiated fuel within the containment.

ACTION:

With the Containment Purge and Exhaust isolation system inoperable, close each of the Purge and Exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Exhaust isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment Purge and Exhaust isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.

5.0 DESIGN FEATURES

5.1 SITE

Exclusion Area

5.1.1 The exclusion area shall be shown in Figure 5.1-1.

Low Population Zone

5.1.2 The low population zone shall be shown in Figure 5.1-2.

Site Boundary For Gaseous and Liquid Effluents

5.1.3 The site boundary for gaseous and liquid effluents shall be shown in Figure 5.1-3.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 115 feet.

b. Nominal inside height = 150 feet.

c. Minimum thickness of concrete walls = 3' 6".

d. Minimum thickness of concrete roof = 2' 6".

e. Minimum thickness of concrete floor pad = 10 feet.

f. Nominal thickness of steel liner = 3/8 inches.

g. Net free volume = 1.24×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained in accordance with the original design provisions contained in Section 5.2.2 of the FSAR.

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 Radiation Protection Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (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 transient and accidents.

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 Coordinator and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and ~~Appendix~~ "A" of 10 CFR Part 55.

Subpart

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY REVIEW COMMITTEE (PNSRC)

6.5.1.1 The PNSRC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PNSRC shall be composed of the:

Chairman:	Plant Manager or Designee
Member:	Assistant Plant Manager - Maintenance
Member:	Assistant Plant Manager - Operations
Member:	Operations Superintendent
Member:	Technical Superintendent - Engineering
Member:	Technical Superintendent - Physical Sciences
Member:	Maintenance Superintendent
Member:	Plant Radiation Protection Supervisor
Member:	QC Superintendent