TABLE 3.3-1 (Continued)

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-7	With 2 of 4 Power Range Neutron Flux Channels greater than or equal to 11% of RATED THERMAL POWER or 1 of 2 Turbine First Stage Pressure channels greater than or equal to 37 psig.	P-7 prevents or defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump under-voltage and under-frequency, turbine trip, pressurizer low pressure, and pressurizer high level. Low flow in a particular loop can be evidenced by either a detected low flow or by the opening of the reactor coolant pump breaker.
P-8	With 2 of 4 Power Range Neutron Flux channels greater than or equal to 31% of RATED THERMAL	P-8 prevents or defeats the automatic block of reactor trip caused by a low ~~POWER~~ coolant flow condition in a single loop. ~~or a reactor coolant pump breaker trip on a single loop.~~
P-10	With 3 of 4 Power range neutron flux channels less than 9% of RATED THERMAL POWER.	P-10 prevents or defeats the manual block of: Power range low setpoint reactor trip, Intermediate range reactor trip, and intermediate range rod stops. Provides input to P-7.

TABLE 3.3-10 (Continued)

Unit 1 and Common Area Fire Detection Systems

Detector System Location	Total Number of Detectors Heat (x/y)*	Flame (x/y)*	Smoke (x/y)*
U1 Cable Tunnels			
a) Quad 1 Cable Tunnel		0/3	0/4
b) Quad 2 Cable Tunnel		0/4	0/7
c) Quad 3N		0/3	0/4
d) Quad 3S		0/3	0/3
e) Quad 3M		0/3	0/4
f) Quad 4		0/5	0/6
U1 Charcoal Filter Ventilation Units			
a) 2-HV-AES-1	0/1*****		
b) 2-HV-AES-2	0/1*****		
1 c) 2-HV-ACRF	0/1*****		
d) 2-HV-CIPX	0/1*****		
e) 2-HV-CPR	0/1*****		
f) 12-HV-AFX	0/1*****C		
U1 Containment******			
a) RCP-1	1/0		
b) RCP 2	1/0		
c) RCP 3	1/0		
d) RCP 4	1/0		
e) Cable Trays	58/0*******		

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.

***** — Originally installed to automatically deluge charcoal filters. However, manual actions are now necessary.

****** — The fire detection instruments located within the Containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate tests.

******* — Thermistors are located within all cable trays which contain combustible cables, in both upper and lower containment throughout quadrants 1-4.

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~~****~~

* 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 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.~~

~~***** Pressurizer safety valve (SV-45A) position indicator acoustic monitor QR-107A is exempted from the above requirements until the end of Cycle 12.~~

TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

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

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

~~** 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 3.7-6

LOW PRESSURE CARBON DIOXIDE SYSTEMS

17-TON CAPACITY

LOCATION	ACTUATION PERIOD
Diesel Generator 1AB Room	Cross-zoned Heat
Diesel Generator 1CD Room	Cross-zoned Heat
Diesel Generator Fuel Oil Pump Room	Heat
4 KV Switchgear Rooms	Manual
Control Rod Drive, Transf. Switchgear Rooms	Manual
Engineered Safety Switchgear Room	Manual
Switchgear Room Cable Vault	Cross-zoned Ionization and Infrared
Auxiliary Cable Vault	Ionization
Control Room Cable Vault (Backup)*	Manual
Penetration Cable Tunnel Quadrant 1	Manual
Penetration Cable Tunnel Quadrant 2	Manual
Penetration Cable Tunnel Quadrant 3N	Manual
Penetration Cable Tunnel Quadrant 3M	Manual
Penetration Cable Tunnel Quadrant 3S	Manual
Penetration Cable Tunnel Quadrant 4	Manual

*Control Room Cable Vault CO_2 System is only required to be operable when the Cable Vault Halon System is inoperable.

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

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The gaseous radwaste treatment system and the ventilation exhaust treatment system shall be used to reduce the radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases to unrestricted areas (See Figure 5.1-3) when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases to unrestricted areas (See Figure 5.1-3) when averaged over 31 days would exceed 0.3 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

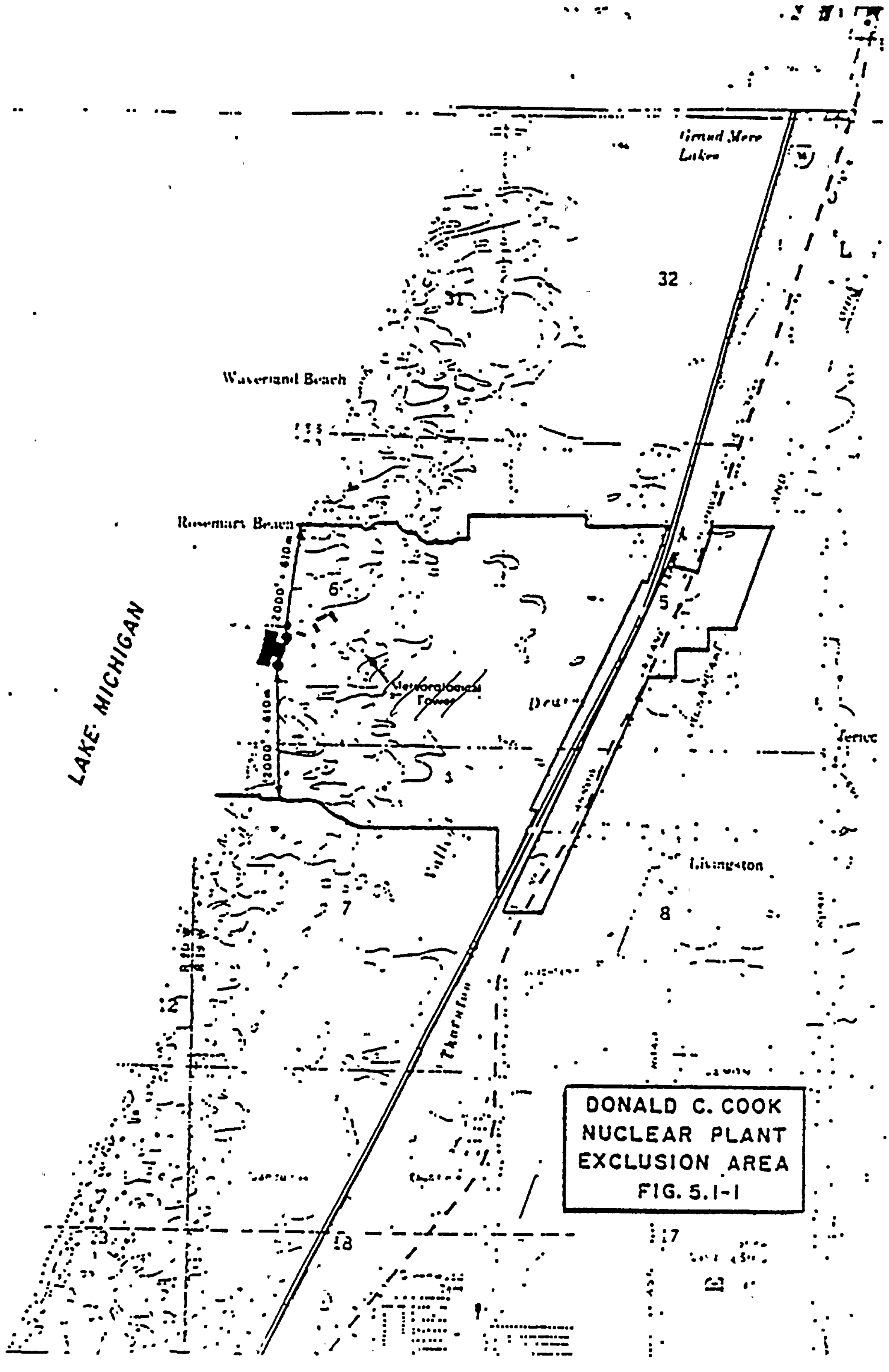
   1. Identification of the inoperable equipment or subsystems and the reason for inoperability.

   2. Action(s) taken to restore the inoperable equipment to operable status.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4 Doses due to gaseous releases to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the ODCM, whenever the gaseous waste treatment system or ventilation exhaust treatment system is not operational.


Grand Mere Lakes
32
31
Waverland Beach
Rosemary Beach
6
5
2000'- 610m
2000'- 610m
LAKE MICHIGAN
Meteorological Tower
Livingston
7
8
Thornton
18
17
DONALD C. COOK
NUCLEAR PLANT
EXCLUSION AREA
FIG. 5.1-1

DESIGN FEATURES

CAPACITY

5.6.4 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3613 fuel assemblies.

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as Category I Items in the FSAR shall be designed and maintained to the original design provisions contained in the FSAR with allowance for normal degradation pursuant to the applicant Surveillance Requirements.

5.8 METEOROLOGICAL TOWER LOCATION

5.8.1 The meteorological tower shall be located as shown on Figure 5.1-~~1~~3.

5.9 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.9.1 The components identified in Table 5.9-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.9-1.

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the control room complex, a designated individual) shall be responsible for the control room command function. A management directive to this effect signed by the Vice President - Nuclear Operations shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATIONS

6.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the ~~FSAR~~ UFSAR and updated in accordance with 10 CFR 50.71(e).

b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President - Nuclear Operations shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

FACILITY STAFF

6.2.2 The Facility organization shall be subject to the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents and, (3) the Operations Superintendent, who must hold or have held a Senior Operator License as specified in Section 6.2.2.h.

6.4 TRAINING

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

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY REVIEW COMMITTEE (PNSRC)

FUNCTION

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

COMPOSITION

6.5.1.2 The PNSRC shall be composed of Assistant Plant Managers, Department Superintendents, or supervisory personnel reporting directly to the Plant Manager, Assistant Plant Managers or Department Superintendents from the functional areas listed below:

Licensing Activities	Technical Support
Safety & Assessment	Radiation Protection
Operations	Maintenance

The Chairman, his alternate and other members and their alternates of the PNSRC shall be designated by the Plant Manager. In addition to the Chairman, the PNSRC membership shall consist of one individual from each of the areas designated above.

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

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.5.1 APPENDIX R REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the Appendix R remote shutdown instrumentation ensures that sufficient instrumentation is available to permit shutdown of the facility to COLD SHUTDOWN conditions at the local shutdown indication (LSI) panel. In the event of a fire, normal power to the LSI panels may be lost. As a result, capability to repair the LSI panels from Unit 2 has been provided. If the alternate power supply is not available, fire watches will be established in those fire areas where loss of normal power to the LSI panels could occur in the event of fire. This will consist of either establishing continuous fire watches or verifying OPERABILITY of fire detectors per Specification 4.3.3.7 and establishing hourly fire watches. The details of how these fire watches are to be implemented are included in a plant procedure.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION (SYSTEMS/DETECTORS)

OPERABILITY of the fire detection systems/detectors ensures that adequate detection capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of the fires will reduce the potential for damage to safety related systems or components in the areas of the specified systems and is an integral element in the overall facility fire protection program. In the event that a portion of the fire detection systems is inoperable, the ACTION statements provided maintain the facility's fire protection program and allows for continued operation of the facility until the inoperable system(s)/detector(s) are restored to OPERABILITY. However, it is not our intent to rely upon the compensatory action for an extended period of time and action will be taken to restore the minimum number of detectors to OPERABLE status within a reasonable period.

3/4.3.3.8 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident.

~~The containment water level and containment sump level transmitters will be modified or replaced and OPERABLE by the end of the refueling outage to begin in February 1989.*~~

~~*Amendment 112 (Effective before startup following refueling outage currently scheduled in 2/89).~~

3/4.9.10 AND 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis. Water level above the vessel flange in MODE 6 will vary as the reactor vessel head and the system internals are removed. The 23 feet of water are required before any subsequent movement of fuel assemblies or control rods."

3/4.9.12 STORAGE POOL VENTILATION SYSTEM

The limitations on the storage pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

The 1980 version of ANSI N510 is used as a testing guide. This standard, however, is intended to be rigorously applied only to systems which, unlike the storage pool ventilation system, are designed to ANSI N509 standards. For the specific case of the air-aerosol mixing uniformity test required by ANSI N510 as a prerequisite to in-place leak testing of charcoal and HEPA filters, the air-aerosol uniform mixing test acceptance criteria were not rigorously met. For this reason, a statistical correction factor will be applied to applicable surveillance test results where required.

In order to maintain the minimum negative pressure required by Technical Specifications (1/8 inch W.G.) during movement of fuel within the storage pool or during crane operation with loads over the pool, the crane bay roll-up door and the drumming room roll-up door, located on the 609-foot elevation of the auxiliary building, must be closed. However, they may be opened during these operations under administrative control. If the crane bay door needs to be opened during fuel movement, an example of an administrative control might be to station an individual at the door who would be in communication with personnel in the spent fuel pool area and could ~~open~~ close the door when passage was completed or in the event of an emergency. For the drumming room door, an example of an administrative control might be to require the door to be reclosed after normal ingress and egress of personnel or material, or to station an individual at the door if the door needs to remain open for an extended period of time.

Should the doors become blocked or stuck open while under administrative control, Technical Specification requirements will not be considered to be violated provided the Action Statement requirements of Specification 3.9.12 are expeditiously followed, i.e., movement of fuel within the storage pool or crane operation with loads over the pool is expeditiously suspended.

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, ~~14~~ 14

TABLE 3.3-1 (Continued)

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-7	With 2 of 4 Power Range Neutron Flux Channels ≥ 11% of RATED THERMAL POWER or 1 of 2 Pressure Before the First Stage channels ≥ 51 psig.	P-7 prevents or defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump under-voltage and under-frequency, turbine trip, pressurizer low pressure, and pressurizer high level. Low flow in a particular loop can be evidenced by either a detected low flow or by the opening of the reactor coolant pump breaker.
P-8	With 2 of 4 Power Range Neutron Flux channels ≥ 31% of RATED THERMAL POWER.	P-8 prevents or defeats the automatic block of reactor trip caused by ~~either~~ a low coolant flow condition in a single loop~~, or a reactor coolant pump breaker trip on a single loop.~~
P-10	With 3 of 4 Power Range Neutron Flux channels < 9% of RATED THERMAL POWER.	P-10 prevents or defeats the manual block of: Power range low setpoint reactor trip, Intermediate range reactor trip, and intermediate range rod stops. Provides input to P-7.

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 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.~~

~~*****Pressurizer safety valve (SV-45C) position indicator acoustic monitor QR-107C is exempted from the above requirements until the end of Cycle 8.~~

TABLE 4.3-10
POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

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

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

~~(4) The core exit thermocouples will not be installed until the 1988 refueling outage; therefore, surveillances will not be required until that time. See license amendment dated April 10, 1987.~~

~~* 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	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
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.~~ Transfer Pump 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

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

3. Verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of greater than or equal to 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1980 (ASTM D 3803-1979, 30°C, 95% R.H.). The carbon samples not obtained from test canisters shall be prepared by either:

   (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or.

   (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of ~~30,0000~~ 30,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 30,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

   1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of greater than or equal to 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1980 (ASTM D 3803-1979, 30°C, 95% R.H.)

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The gaseous radwaste treatment system and the ventilation exhaust treatment system shall be used to reduce the radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases to unrestricted areas (See Figure 5.1-3) when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases to unrestricted areas (See Figure 5.1-3) when averaged over 31 days would exceed 0.3 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

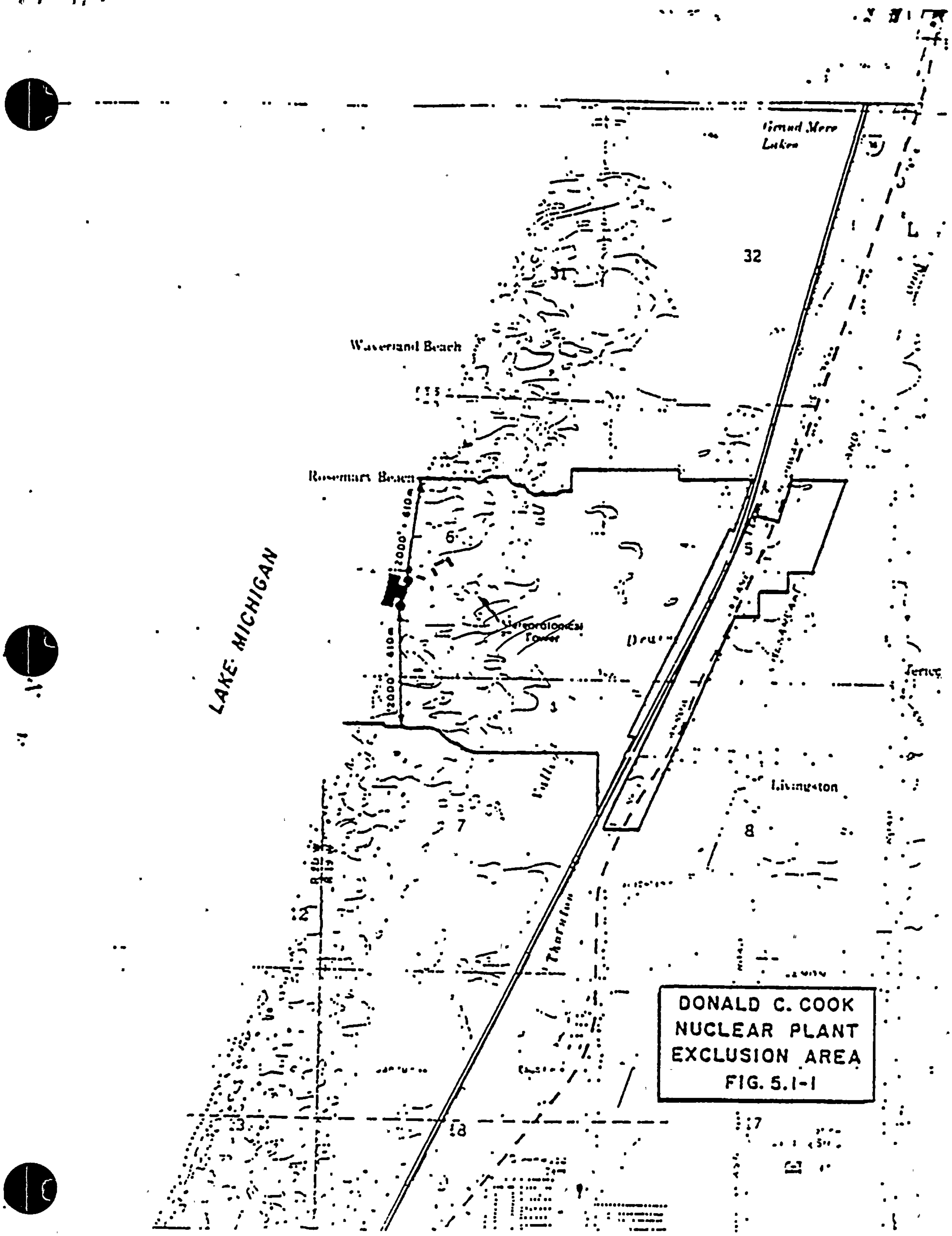
   1. Identification of the inoperable equipment or subsystems and the reason for inoperability.

   2. Action(s) taken to restore the inoperable equipment to operable status.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4 Doses due to gaseous releases to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the ODCM, whenever the gaseous waste treatment system or ventilation exhaust treatment system is not operational.


Grand Mere Lakes
32
31
Waverland Beach
Rosemary Beach
6
5
LAKE MICHIGAN
2000' • 610m
Meteorological Tower
7
8
Livingston
Thornton
DONALD C. COOK
NUCLEAR PLANT
EXCLUSION AREA
FIG. 5.1-1

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,612 plus or minus 100 cubic feet at a nominal Tavg of 70°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-~~2~~ 3.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than 0.95 when flooded with unborated water,

b. A nominal 8.97-inch center-to-center distance between fuel assemblies, placed in the storage racks.

c. The fuel assemblies will be classified as acceptable for Region 1, Region 2, or Region 3 storage based upon their assembly average burnup versus initial nominal enrichment. Cells acceptable for Region 1, Region 2, and Region 3 assembly storage are indicated in Figures 5.6-1 and 5.6-2. Assemblies that are acceptable for storage in Region 1, Region 2, and Region 3 must meet the design criteria that define the regions as follows:

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

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

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

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the control room complex, a designated individual) shall be responsible for the control room command function. A management directive to this effect signed by the Vice President - Nuclear Operations shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATIONS

6.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the ~~FSAR~~ UFSAR and updated in accordance with 10 CFR 50.71(e).

b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President - Nuclear Operations shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

FACILITY STAFF

6.2.2 The Facility organization shall be subject to the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

ADMINISTRATIVE CONTROLS

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents and, (3) the Operations Superintendent, who must hold or have held a Senior Operator License as specified in Section 6.2.2.h.

6.4 TRAINING

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

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY REVIEW COMMITTEE (PNSRC)

FUNCTION

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

COMPOSITION

6.5.1.2 The PNSRC shall be composed of Assistant Plant Managers, Department Superintendents, or supervisory personnel reporting directly to the Plant Manager, Assistant Plant Managers or Department Superintendents from the functional areas listed below:

Licensing Activities	Technical Support
Safety & Assessment	Radiation Protection
Operations	Maintenance

The Chairman, his alternate and other members and their alternates of the PNSRC shall be designated by the Plant Manager. In addition to the Chairman, the PNSRC membership shall consist of one individual from each of the areas designated above.

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

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident.

3/4.3.3.7 ~~AXIAL POWER DISTRIBUTION MONITORING SYSTEM (APDMS)~~ Deleted

~~OPERABILITY of the APDMS ensures that sufficient capability is available for the measurement of the neutron flux spatial distribution within the reactor core. This capability is required to 1) monitor the core flux patterns that are representative of the peak core power density and 2) limit the core average axial power profile such that the total power peaking factor F_Q is maintained within acceptable limits.~~

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY. Use of containment temperature monitoring is allowed once per hour if containment fire detection is inoperable.

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

The radioactive liquid effluent instrumentation is provide to monitor and control, as applicable, the release of radioactive material in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approval methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria specified in Section 11.3 of the Final Safety Analysis Report for the Donald C. Cook Nuclear Plant.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 AND 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis. Water level above the vessel flange in MODE 6 will vary as the reactor vessel head and the system internals are removed. The 23 feet of water are required before any subsequent movement of fuel assemblies or control rods.

3/4.9.12 STORAGE POOL VENTILATION SYSTEM

The limitations on the storage pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

The 1980 version of ANSI N510 is used as a testing guide. This standard, however, is intended to be rigorously applied only to systems which, unlike the storage pool ventilation system, are designed to ANSI N509 standards. For the specific case of the air-aerosol mixing uniformity test required by ANSI N510 as a prerequisite to in-place leak testing of charcoal and HEPA filters, the air-aerosol uniform mixing test acceptance criteria were not rigorously met. For this reason, a statistical correction factor will be applied to applicable surveillance test results where required.

In order to maintain the minimum negative pressure required by Technical Specifications (1/8 inch W.G.) during movement of fuel within the storage pool or during crane operation with loads over the pool, the crane bay roll-up door and the drumming room roll-up door, located on the 609-foot elevation of the auxiliary building, must be closed. However, they may be opened during these operations under administrative control. If the crane bay door needs to be opened during fuel movement, an example of an administrative control might be to station an individual at the door who would be in communication with personnel in the spent fuel pool area and could ~~open~~ close the door when passage was completed or in the event of an emergency. For the drumming room door, an example of an administrative control might be to require the door to be reclosed after normal ingress and egress of personnel or material, or to station an individual at the door if the door needs to remain open for an extended period of time.

ATTACHMENT 3 TO AEP:NRC:1137B

PROPOSED REVISED
TECHNICAL SPECIFICATION PAGES

TABLE 3.3-1 (Continued)

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-7	With 2 of 4 Power Range Neutron Flux Channels greater than or equal to 11% of RATED THERMAL POWER or 1 of 2 Turbine First Stage Pressure channels greater than or equal to 37 psig.	P-7 prevents or defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump under-voltage and under-frequency, turbine trip, pressurizer low pressure, and pressurizer high level. Low flow in a particular loop can be evidenced by either a detected low flow or by the opening of the reactor coolant pump breaker.
P-8	With 2 of 4 Power Range Neutron Flux channels greater than or equal to 31% of RATED THERMAL POWER	P-8 prevents or defeats the automatic block of reactor trip caused by a low coolant flow condition in a single loop.
P-10	With 3 of 4 Power range neutron flux channels less than 9% of RATED THERMAL POWER.	P-10 prevents or defeats the manual block of: Power range low setpoint reactor trip, Intermediate range reactor trip, and intermediate range rod stops. Provides input to P-7.

TABLE 3.3-10 (Continued)

Unit 1 and Common Area Fire Detection Systems

Detector System Location	Heat (x/y)*	Flame (x/y)*	Smoke (x/y)*
		Total Number of Detectors	
U1 Cable Tunnels			
a) Quad 1 Cable Tunnel		0/3	0/4
b) Quad 2 Cable Tunnel		0/4	0/7
c) Quad 3N		0/3	0/4
d) Quad 3S		0/3	0/3
e) Quad 3M		0/3	0/4
f) Quad 4		0/5	0/6
U1 Charcoal Filter Ventilation Units			
a) 1-HV-AES-1	0/1*****		
b) 1-HV-AES-2	0/1*****		
c) 1-HV-ACRF	0/1*****		
d) 1-HV-CIPX	0/1*****		
e) 1-HV-CPR	0/1*****		
f) 12-HV-AFX	0/1*****C		
U1 Containment******			
a) RCP 1	1/0		
b) RCP 2	1/0		
c) RCP 3	1/0		
d) RCP 4	1/0		
e) Cable Trays	1/0		

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.

***** Originally installed to automatically deluge charcoal filters. However, manual actions are now necessary.

****** The fire detection instruments located within the Containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate tests.

******* Thermistors located within cable trays which contain combustible cables, in both upper and lower containment throughout quadrants 1-4.

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

* 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.

TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

---

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

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

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

TABLE 3.7-6

LOW PRESSURE CARBON DIOXIDE SYSTEMS

17-TON CAPACITY

LOCATION	ACTUATION PERIOD
Diesel Generator 1AB Room	Cross-zoned Heat
Diesel Generator 1CD Room	Cross-zoned Heat
Diesel Generator Fuel Oil Pump Room	Heat
4 KV Switchgear Rooms	Manual
Control Rod Drive, Transf. Switchgear Rooms	Manual
Engineered Safety Switchgear Room	Manual
Switchgear Room Cable Vault	Cross-zoned Ionization and Infrared
Auxiliary Cable Vault	Ionization
Control Room Cable Vault (Backup)*	Manual
Penetration Cable Tunnel Quadrant 1	Manual
Penetration Cable Tunnel Quadrant 2	Manual
Penetration Cable Tunnel Quadrant 3N	Manual
Penetration Cable Tunnel Quadrant 3M	Manual
Penetration Cable Tunnel Quadrant 3S	Manual
Penetration Cable Tunnel Quadrant 4	Manual

*Control Room Cable Vault CO_2 System is only required to be operable when the Cable Vault Halon System is inoperable.

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 $\leq$ 24,240 in.-lbs. prior to moving each load over racks containing fuel.

* Shared system with Cook Nuclear Plant - Unit 2.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The gaseous radwaste treatment system and the ventilation exhaust treatment system shall be used to reduce the radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases to unrestricted areas (See Figure 5.1-3) when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases to unrestricted areas (See Figure 5.1-3) when averaged over 31 days would exceed 0.3 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability.

   2. Action(s) taken to restore the inoperable equipment to operable status.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4 Doses due to gaseous releases to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the ODCM, whenever the gaseous waste treatment system or ventilation exhaust treatment system is not operational.

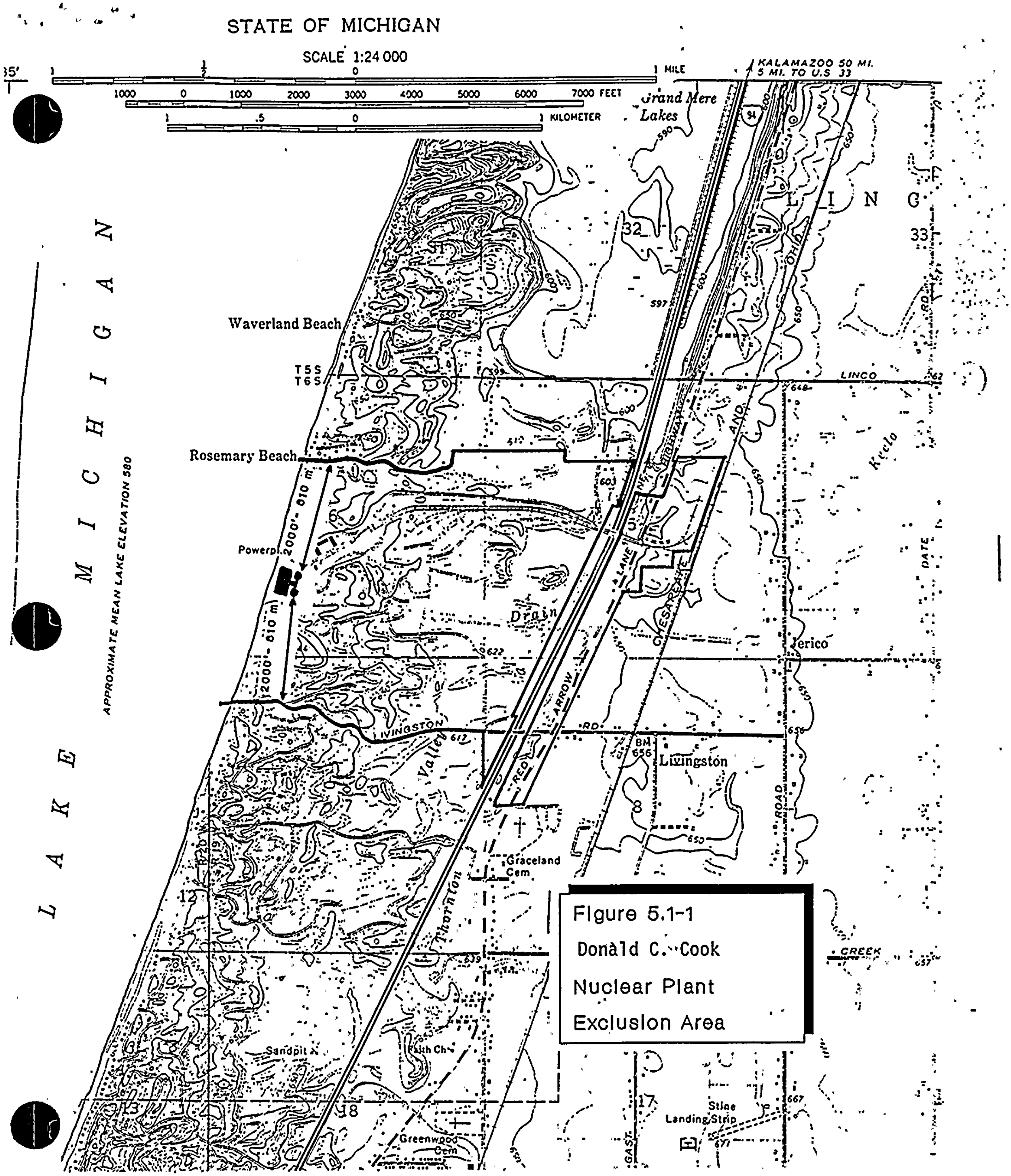


Figure 5.1-1
Donald C. Cook
Nuclear Plant
Exclusion Area

DESIGN FEATURES

CAPACITY

5.6.4 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3613 fuel assemblies.

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as Category I Items in the FSAR shall be designed and maintained to the original design provisions contained in the FSAR with allowance for normal degradation pursuant to the applicant Surveillance Requirements.

5.8 METEROLOGICAL TOWER LOCATION

5.8.1 The meterological tower shall be located as shown on Figure 5.1-3.

5.9 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.9.1 The components identified in Table 5.9-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.9-1.

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the control room complex, a designated individual) shall be responsible for the control room command function. A management directive to this effect signed by the Vice President - Nuclear Operations shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATIONS

6.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management level through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the UFSAR and updated in accordance with 10 CFR 50.71(e).

b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President - Nuclear Operations shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

FACILITY STAFF

6.2.2 The Facility organization shall be subject to the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents and, (3) the Operations Superintendent must hold or have held a Senior Operator License as specified in Section 6.2.2.h.

6.4 TRAINING

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

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY REVIEW COMMITTEE (PNSRC)

FUNCTION

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

COMPOSITION

6.5.1.2 The PNSRC shall be composed of Assistant Plant Managers, Department Superintendents, or supervisory personnel reporting directly to the Plant Manager, Assistant Plant Managers or Department Superintendents from the functional areas listed below:

Licensing Activities	Technical Support
Safety & Assessment	Radiation Protection
Operations	Maintenance

The Chairman, his alternate and other members and their alternates of the PNSRC shall be designated by the Plant Manager. In addition to the Chairman, the PNSRC membership shall consist of one individual from each of the areas designated above.

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

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.5.1 APPENDIX R REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the Appendix R remote shutdown instrumentation ensures that sufficient instrumentation is available to permit shutdown of the facility to COLD SHUTDOWN conditions at the local shutdown indication (LSI) panel. In the event of a fire, normal power to the LSI panels may be lost. As a result, capability to repair the LSI panels from Unit 2 has been provided. If the alternate power supply is not available, fire watches will be established in those fire areas where loss of normal power to the LSI panels could occur in the event of fire. This will consist of either establishing continuous fire watches or verifying OPERABILITY of fire detectors per Specification 4.3.3.7 and establishing hourly fire watches. The details of how these fire watches are to be implemented are included in a plant procedure.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION (SYSTEMS/DETECTORS)

OPERABILITY of the fire detection systems/detectors ensures that adequate detection capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of the fires will reduce the potential for damage to safety related systems or components in the areas of the specified systems and is an integral element in the overall facility fire protection program. In the event that a portion of the fire detection systems is inoperable, the ACTION statements provided maintain the facility's fire protection program and allows for continued operation of the facility until the inoperable system(s)/detector(s) are restored to OPERABILITY. However, it is not our intent to rely upon the compensatory action for an extended period of time and action will be taken to restore the minimum number of detectors to OPERABLE status within a reasonable period.

3/4.3.3.8 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis. Water level above the vessel flange in MODE 6 will vary as the reactor vessel head and the system internals are removed. The 23 feet of water are required before any subsequent movement of fuel assemblies or control rods.

3/4.9.12 STORAGE POOL VENTILATION SYSTEM

The limitations on the storage pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

The 1980 version of ANSI N510 is used as a testing guide. This standard, however, is intended to be rigorously applied only to systems which, unlike the storage pool ventilation system, are designed to ANSI N509 standards. For the specific case of the air-aerosol mixing uniformity test required by ANSI N510 as a prerequisite to in-place leak testing of charcoal and HEPA filters, the air-aerosol uniform mixing test acceptance criteria were not rigorously met. For this reason, a statistical correction factor will be applied to applicable surveillance test results where required.

In order to maintain the minimum negative pressure required by Technical Specifications (1/8 inch W.G.) during movement of fuel within the storage pool or during crane operation with loads over the pool, the crane bay roll-up door and the drumming room roll-up door, located on the 609-foot elevation of the auxiliary building, must be closed. However, they may be opened during these operations under administrative control. If the crane bay door needs to be opened during fuel movement, an example of an administrative control might be to station an individual at the door who would be in communication with personnel in the spent fuel pool area and could close the door when passage through the door was completed or in the event of an emergency. For the drumming room door, an example of an administrative control might be to require the door to be reclosed after normal ingress and egress of personnel or material, or to station an individual at the door if the door needs to remain open for an extended period of time.

Should the doors become blocked or stuck open while under administrative control, Technical Specification requirements will not be considered to be violated provided the Action Statement requirements of Specification 3.9.12 are expeditiously followed, i.e., movement of fuel within the storage pool or crane operation with loads over the pool is expeditiously suspended.

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*	1 14

TABLE 3.3-1 (Continued)

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-7	With 2 of 4 Power Range Neutron Flux Channels ≥ 11% of RATED THERMAL POWER or 1 of 2 Pressure before the First Stage channels ≥ 51 psig.	P-7 prevents or defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump under-voltage and under-frequency, turbine trip, pressurizer low pressure, and pressurizer high level. Low flow in a particular loop can be evidenced by either a detected low flow or by the opening of the reactor coolant pump breaker.
P-8	With 2 of 4 Power Range Neutron Flux channels ≥ 31% of RATED THERMAL POWER.	P-8 prevents or defeats the automatic block of reactor trip caused by a low coolant flow condition in a single loop.
P-10	With 3 of 4 Power Range Neutron flux channels < 9% of RATED THERMAL POWER..	P-10 prevents or defeats the manual block of: Power range low setpoint reactor trip, Intermediate range reactor trip, and intermediate range rod stops. Provides input to P-7.

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 Indicator - Limit Switches instruments.

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

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

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

TABLE 3.3-11

Unit 2 and Common Area Fire Detection Systems

Detection System Location	Heat (x/y)*	Total Number of Detectors Flame (x/y)*	Smoke (x/y)*
Auxiliary Building			
a) Elevation 573			23/0 C
b) Elevation 587			55/0 C
c) Elevation 609			41/0 C
d) Elevation 633			41/0 C
e) Elevation 650			34/0 C
f) New Fuel STGE Area			4/0 C
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 Transfer Pump 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			2/0 C
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			4/0 C
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

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 ≤ 24,240 in.-lbs. prior to moving each load over racks containing fuel.

* Shared system with Cook Nuclear Plant - Unit 1.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of greater than or equal to 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1980 (ASTM D 3803-1979, 30°C, 95% R.H.). The carbon samples not obtained from test canisters shall be prepared by either:

   (a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   (b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 30,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

   1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of greater than or equal to 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1980 (ASTM D 3803-1979, 30°C, 95%,R.H.).

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The gaseous radwaste treatment system and the ventilation exhaust treatment system shall be used to reduce the radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases to unrestricted areas (See Figure 5.1-3) when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad. for beta radiation. The ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases to unrestricted areas (See Figure 5.1-3) when averaged over 31 days would exceed 0.3 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability.

   2. Action(s) taken to restore the inoperable equipment to operable status.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4 Doses due to gaseous releases to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the ODCM, whenever the gaseous waste treatment system or ventilation exhaust treatment system is not operational.

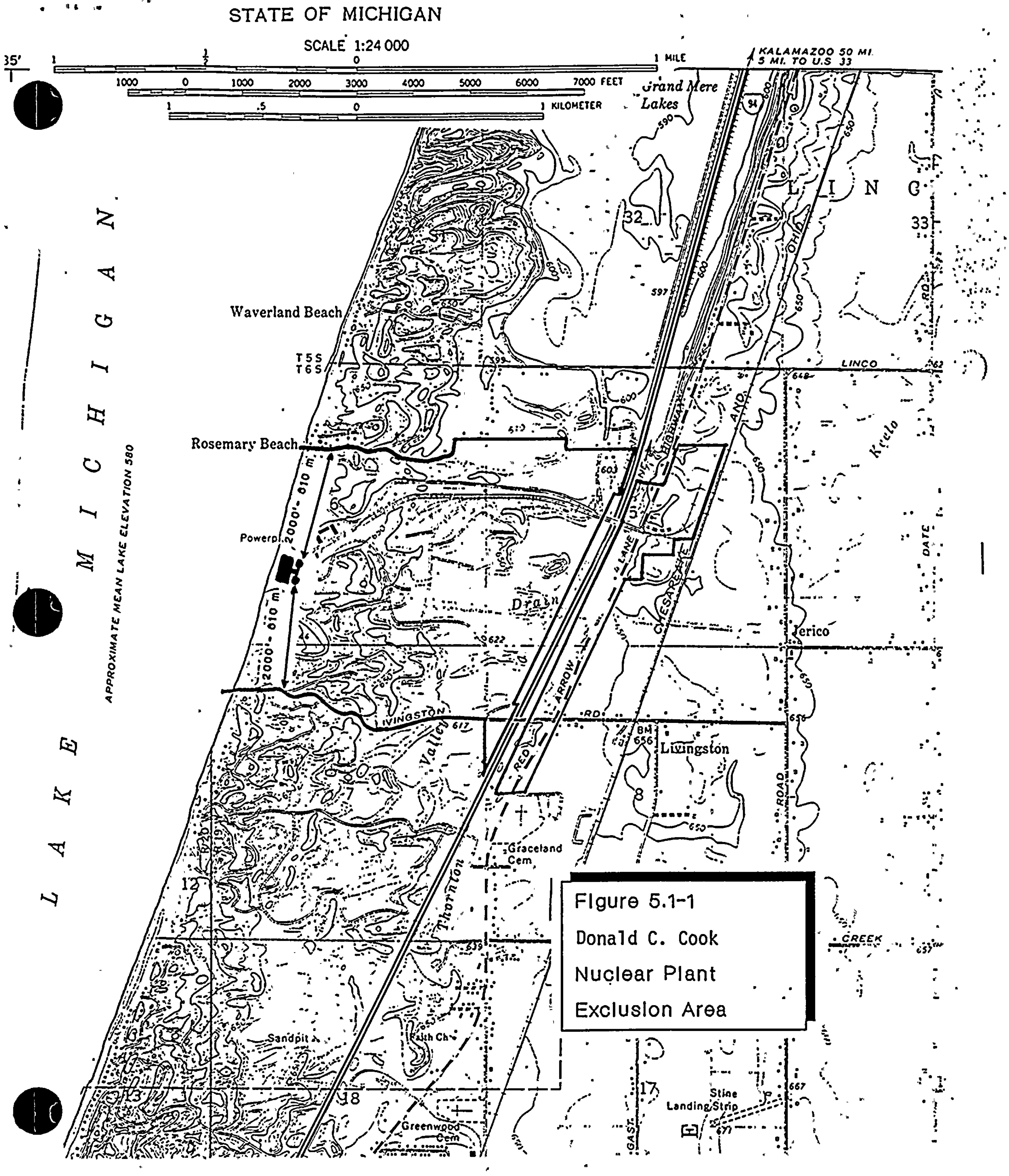

Figure 5.1-1
Donald C. Cook
Nuclear Plant
Exclusion Area

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,612 plus or minus 100 cubic feet at a nominal Tavg of 70°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-3.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than 0.95 when flooded with unborated water,

b. A nominal 8.97-inch center-to-center distance between fuel assemblies, placed in the storage racks.

c. The fuel assemblies will be classified as acceptable for Region 1, Region 2, or Region 3 storage based upon their assembly burnup versus initial nominal enrichment. Cells acceptable for Region 1, Region 2, and Region 3 assembly storage are indicated in Figures 5.6-1 and 5.6-2. Assemblies that are acceptable for storage in Region 1, Region 2, and Region 3 must meet the design criteria that define the regions as follows:

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

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

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

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the control room complex, a designated individual) shall be responsible for the control room command function. A management directive to this effect signed by the Vice President - Nuclear Operations shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATIONS

6.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management level through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the UFSAR and updated in accordance with 10 CFR 50.71(e).

b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President - Nuclear Operations shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

FACILITY STAFF

6.2.2 The Facility organization shall be subject to the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Plant Radiation Protection Manager, who shall meet or exceed qualifications of Regulatory Guide 1.8, September 1975, (2) the Shift Technical Advisor, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents and, (3) the Operations Superintendent, who must hold or have held a Senior Operator License as specified in Section 6.2.2.h.

6.4 TRAINING

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

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY REVIEW COMMITTEE (PNSRC)

FUNCTION

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

COMPOSITION

6.5.1.2 The PNSRC shall be composed of Assistant Plant Managers, Department Superintendents, or supervisory personnel reporting directly to the Plant Manager, Assistant Plant Managers or Department Superintendents from the functional areas listed below:

Licensing Activities	Technical Support
Safety & Assessment	Radiation Protection
Operations	Maintenance

The Chairman, his alternate and other members and their alternates of the PNSRC shall be designated by the Plant Manager. In addition to the Chairman, the PNSRC membership shall consist of one individual from each of the areas designated above.

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

INSTRUMENTATION

BASES

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident.

3/4.3.3.7 DELETED

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY. Use of containment temperature monitoring is allowed once per hour if containment fire detection is inoperable.

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the release of radioactive material in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approval methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria specified in Section 11.3 of the Final Safety Analysis Report for the Donald C. Cook Nuclear Plant.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 AND 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis. Water level above the vessel flange in MODE 6 will vary as the reactor vessel head and the system internals are removed. The 23 feet of water are required before any subsequent movement of fuel assemblies or control rods.

3/4.9.12 STORAGE POOL VENTILATION SYSTEM

The limitations on the storage pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

The 1980 version of ANSI N510 is used as a testing guide. This standard, however, is intended to be rigorously applied only to systems which, unlike the storage pool ventilation system, are designed to ANSI N509 standards. For the specific case of the air-aerosol mixing uniformity test required by ANSI N510 as a prerequisite to in-place leak testing of charcoal and HEPA filters, the air-aerosol uniform mixing test acceptance criteria were not rigorously met. For this reason, a statistical correction factor will be applied to applicable surveillance test results where required.

In order to maintain the minimum negative pressure required by Technical Specifications (1/8 inch W.G.) during movement of fuel within the storage pool or during crane operation with loads over the pool, the crane bay roll-up door and the drumming room roll-up door, located on the 609-foot elevation of the auxiliary building, must be closed. However, they may be opened during these operations under administrative control. If the crane bay door needs to be opened during fuel movement, an example of an administrative control might be to station an individual at the door who would be in communication with personnel in the spent fuel pool area and could close the door when passage through the door was completed or in the event of an emergency. For the drumming room door, an example of an administrative control might be to require the door to be reclosed after normal ingress and egress of personnel or material, or to station an individual at the door if the door needs to remain open for an extended period of time.