December 30, 1994

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43215

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT RE: ADMINISTRATIVE AND EDITORIAL CHANGES (TAC NOS. M88266 AND M88267)

Dear Mr. Fitzpatrick:

The Commission has issued the enclosed Amendment No. ~~168~~ 186 to Facility Operating License No. DPR-58 and Amendment No. 172 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated November 15, 1993.

The amendments make various administrative and editorial changes to the Technical Specifications.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

John B. Hickman, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-315
and 50-316

Enclosures:
1. Amendment No. ~~168~~ 186 to DPR-58
2. Amendment No. 172 to DPR-74
3. Safety Evaluation

cc w/encl: See next page

DOCUMENT NAME: G:\WPDOCS\DCCOOK\CO88266.AMD

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	LA:PD31	E	PM:PD31	E	BC:SPLB	BC:SRXB	OGC	D:PD31
NAME	CJamerson		JHickman		CMcCracken	RJones	R Bachmann	JHannon
DATE	12/1/94		12/1/94		12/6/94	12/6/94	12/9/94	12/12/94

OFFICIAL RECORD COPY

9501090343 941230
PDR ADOCK 05000315
P PDR

DF01

DATED: December 30, 1994

AMENDMENT NO. ~~168~~ 186 TO FACILITY OPERATING LICENSE NO. DPR-58-D. C. COOK-UNIT 1
AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. DPR-74-D. C. COOK-UNIT 2

Docket File
PUBLIC
PDIII-1 Reading
J. Roe
C. Jamerson
J. Hannon
J. Hickman (2)
C. McCracken
R. Jones
OGC-WF
G. Hill (4)
C. Grimes, O-11F23
ACRS (4)
OPA
OC/LFDCB
W. Kropp, RIII
SEDB

cc: Plant Service list

060020

Mr. E. E. Fitzpatrick
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, Michigan 48913

Township Supervisor
Lake Township Hall
Post Office Box 818
Bridgman, Michigan 49106

Al Blind, Plant Manager
Donald C. Cook Nuclear Plant
Post Office Box 458
Bridgman, Michigan 49106

U.S. Nuclear Regulatory Commission
Resident Inspector Office
7700 Red Arrow Highway
Stevensville, Michigan 49127

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N. W.
Washington, DC 20037

Mayor, City of Bridgman
Post Office Box 366
Bridgman, Michigan 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, Michigan 48909

Nuclear Facilities and Environmental
Monitoring Section Office
Division of Radiological Health
Department of Public Health
3423 N. Logan Street
P. O. Box 30195
Lansing, Michigan 48909

Mr. S. Brewer
American Electric Power Service
Corporation
1 Riverside Plaza
Columbus, Ohio 43215

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 186
License No. DPR-58

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated November 15, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-58 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 186, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John N. Hannon

John N. Hannon, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 30, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 186

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3/4 3-9	3/4 3-9
3/4 3-53a	3/4 3-53a
3/4 3-55	3/4 3-55
3/4 3-56	3/4 3-56
3/4 7-41	3/4 7-41
3/4 9-8	3/4 9-8
3/4 11-12	3/4 11-12
B 3/4 3-6	B 3/4 3-6
B 3/4 9-3	B 3/4 9-3
5-2	5-2
5-9	5-9
6-1	6-1
6-4	6-4

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)*	Total Number of Detectors 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) 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	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 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.

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.

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 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.

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.

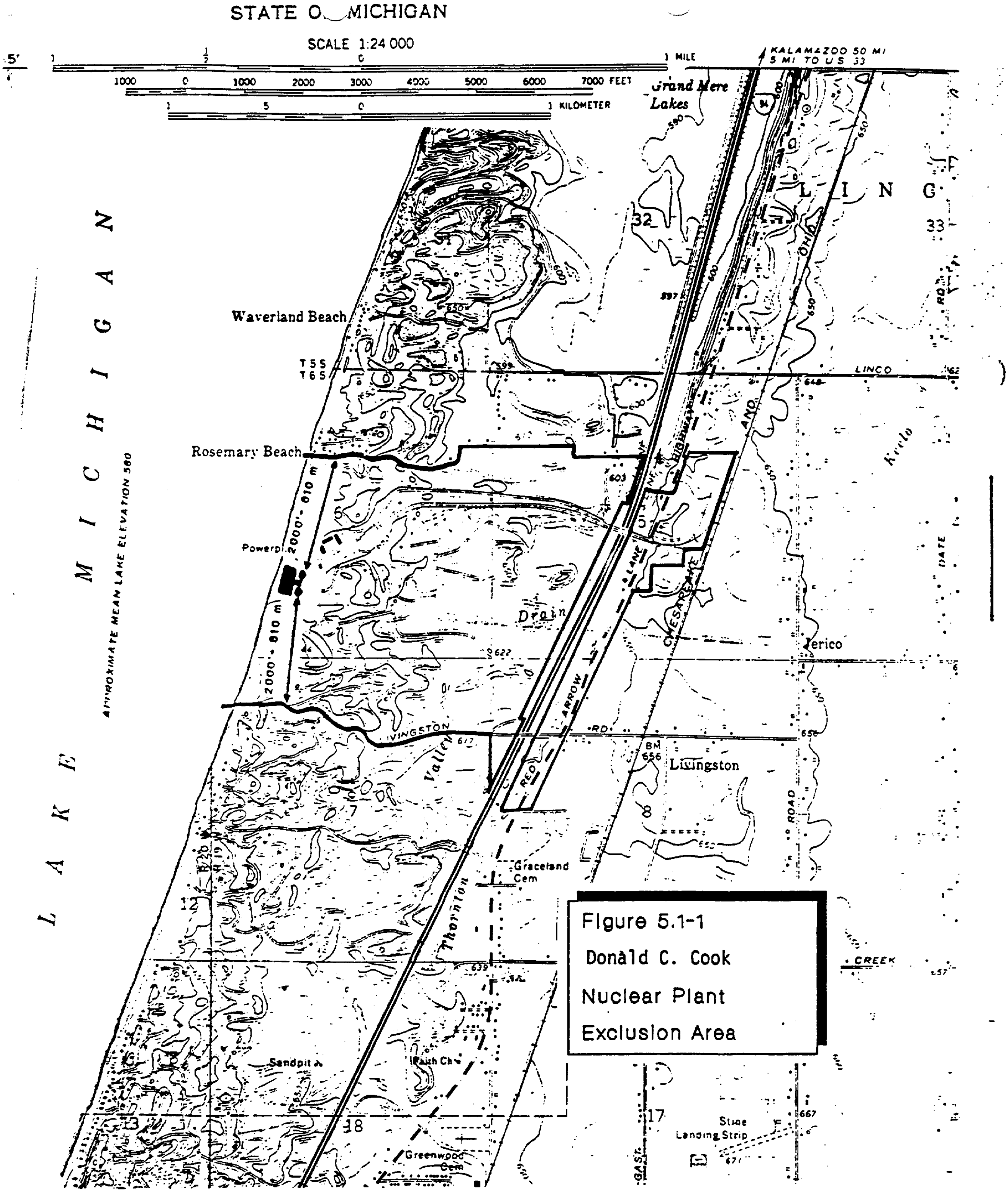

Figure 5.1-1
Donald C. Cook
Nuclear Plant
Exclusion Area

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 applicable 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.

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 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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
License No. DPR-74

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated November 15, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-74 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 172, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John N. Hannon

John N. Hannon, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
 Specifications

Date of Issuance: December 30, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 172

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3/4 3-4	3/4 3-4
3/4 3-8	3/4 3-8
3/4 3-46	3/4 3-46
3/4 3-47	3/4 3-47
3/4 3-52	3/4 3-52
3/4 9-7	3/4 9-7
3/4 9-13	3/4 9-13
3/4 11-12	3/4 11-12
B 3/4 3-3	B 3/4 3-3
B 3/4 9-3	B 3/4 9-3
5-2	5-2
5-5	5-5
6-1	6-1
6-4	6-4

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

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

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

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

† The provisions of Technical Specification 4.0.8 are applicable.

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

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.

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.

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.

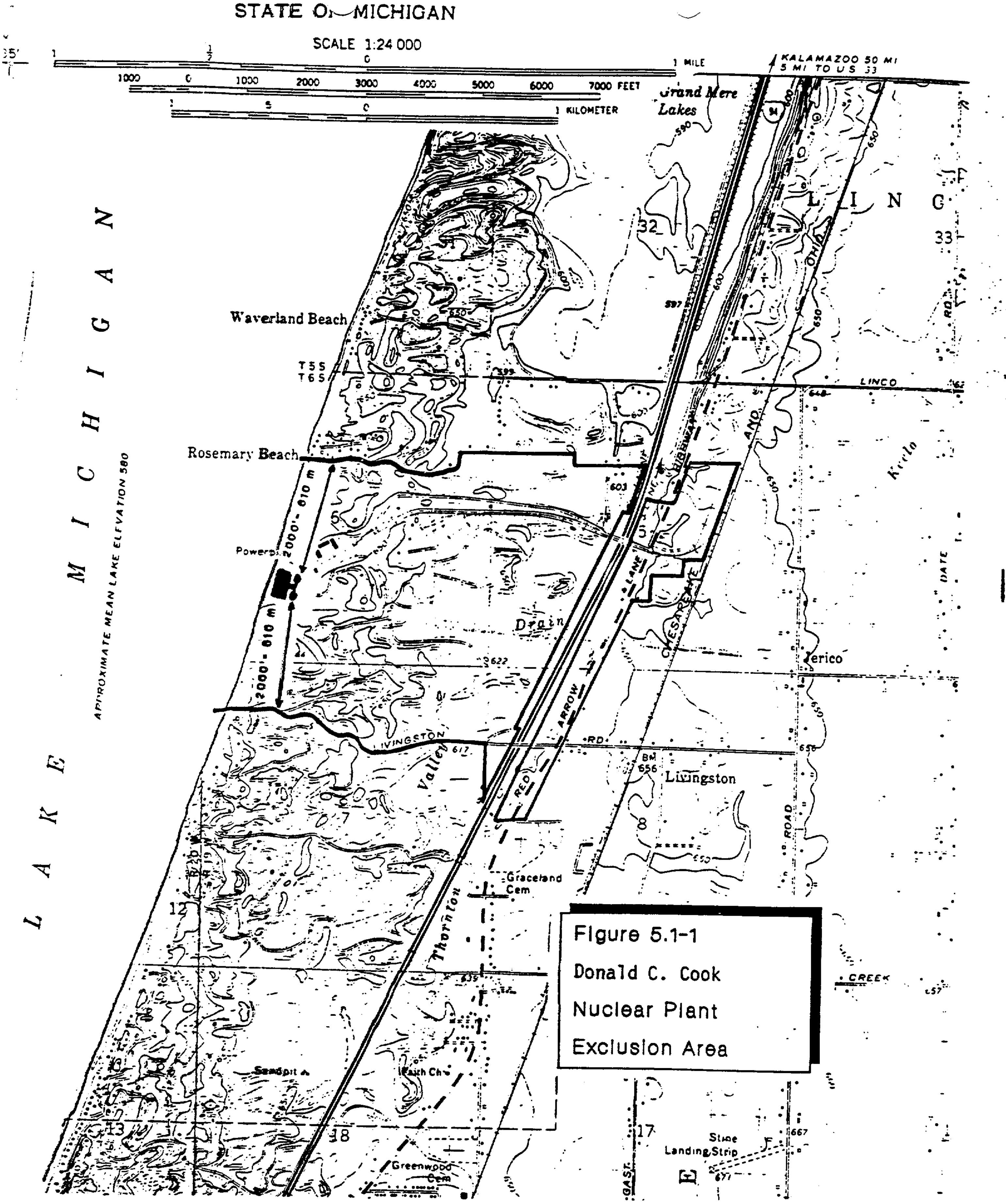

Figure 5.1-1
Donald C. Cook
Nuclear Plant
Exclusion Area

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DPR-58
AND AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. DPR-74
INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated November 15, 1993, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The proposed amendments would make various administrative or editorial changes. The changes are intended to correct various oversights or errors in the TS, or to update the TS to reflect current plant conditions.

2.0 EVALUATION

The TS changes requested by the licensee address 13 separate items, each of which is evaluated below.

2.01 ERRONEOUS DESCRIPTION OF OPERATOR ACTION FOR ROLLUP DOORS

In Amendment Numbers 124 and 111, for D.C. Cook Units 1 and 2, respectively, dated May 19, 1989, a footnote was added to TS 3.9.12 to allow the crane bay roll-up door and the drumming room roll-up door to be opened under administrative control during movement of fuel within the storage pool or crane operation with loads over the storage pool. As part of this change, an example of appropriate administrative controls was added to the bases. However, this example states that an individual could be stationed at the door to open it in the event of an emergency. This wording is contrary to the actual intended function of stationing an individual for compensatory action. As stated in the Safety Evaluation for Amendments 124 and 111, "... the person stationed at the crane bay door would be in communication with personnel in the spent fuel pool area so that he would be readily informed of an emergency and, if needed, actuate the door closure mechanism." Since the purpose of stationing an individual at the door would be to close the door to prevent radioactive gases from escaping through the door, the licensee has proposed to revise the wording in the bases to correctly reflect that. Based on the above, the proposed change is acceptable.

2.02 INCORRECT UNIT NUMBERS FOR FIRE DETECTION SYSTEMS AND INCORRECT ROOM DESCRIPTION

Table 3.3-10 in the Unit 1 TS lists fire detection system locations at the charcoal filter ventilation units and other locations. The listing of the charcoal filter ventilation units incorrectly has a number "2" prefix for the units indicating a Unit 2 piece of equipment when they are in fact, in Unit 1. The licensee has proposed to correct the listing to reflect the Unit 1 location. This change is acceptable. The corresponding list for Unit 2 is correct.

Table 3.3-11 in the Unit 2 TS lists fire detection system locations, with one noted at the U2 Diesel Fuel Oil XFMR. Rm. The room in which the detector is actually located contains the Diesel Fuel Oil Transfer Pump not the Diesel Fuel Oil Transformer. Therefore, the licensee has proposed to revise the listing to correctly list the location as the U2 Diesel Fuel Oil Transfer Pump. This change is acceptable. The corresponding listing for Unit 1 is correct.

2.03 CORRECTION OF FIGURE REFERENCE

In TS 3.11.2.4, for both units, the first sentence of the limiting condition for operation refers to Figure 5.1.3. This is incorrect and inconsistent with the standard numbering of technical specifications. The correct reference is to Figure 5.1-3. The licensee has proposed to correct the reference and the change is acceptable.

2.04 REFERENCE TO CORRECT VERSION OF FSAR

TS 6.2.1.a, for both units, states that organizational charts will be documented in the FSAR. However, the FSAR is not a current document as it has been superseeded by the UFSAR (Updated Final Safety Analysis Report) which is maintained up to date. The licensee has proposed to revise the reference to reflect the current document. The change is acceptable.

2.05 REFERENCE TO 10 CFR 55

TS 6.4, for both units, references Appendix A to 10 CFR 55. However, 10 CFR 55 has been rewritten and the material previously included in Appendix A is now part of the body of 10 CFR 55. Therefore, the licensee has proposed to revise the reference to simply refer to 10 CFR 55. The change is acceptable.

2.06 DESCRIPTION OF P-8 INTERLOCK

Amendment Numbers 140 and 127 for D.C. Cook Units 1 and 2, respectively, dated June 28, 1990, modified the TS for the reactor protection system to change the logic for the reactor coolant pump breaker position above permissive P-8 from one out of four breakers open to two out of four breakers open. This change was requested to avoid a spurious reactor trip on a false signal from a single reactor coolant pump breaker auxiliary contact. Although the change was properly incorporated in TS Table 3.3-1, "Reactor Trip System Instrumentation," the change was not incorporated in the description of the

P-8 interlock at the end of the table. Therefore, for consistency the licensee has proposed to revise the description at the end of the table to correctly describe the current configuration. In addition, the licensee proposed to delete the word "POWER" from the Unit 1 text to be consistent with the Unit 2 text. The proposed change to reflect the previously approved amendment is consistent with the change which was already considered and approved by the staff and is therefore acceptable. The deletion of the word "POWER" from the Unit 1 TS does not materially affect the description of the P-8 interlock and provides consistency, and is therefore acceptable.

2.07 CORRECTION OF METEOROLOGICAL TOWER LOCATION

Amendment Numbers 127 and 113 for D.C. Cook Units 1 and 2, respectively, dated July 5, 1989, modified the TSs to reflect the installation of a new meteorological monitoring system and provide proper reference to meteorological tower and associated instrumentation locations. As part of this amendment Figure 5.1-3, "Site Boundary for Liquid and Gaseous Effluents," was modified to show the location of the new meteorological tower. This amendment however, did not correct the reference in TS 5.8.1, for Unit 1, and TS 5.5.1, for Unit 2, which states that the location of the meteorological tower is shown on Figure 5.1-1. In addition Figure 5.5-1 was not corrected to show the location of the new meteorological tower. Therefore, the licensee has proposed to correct the reference in TS 5.8.1, for Unit 1, and TS 5.5.1, for Unit 2, to refer to Figure 5.1-3 for the meteorological tower location, and to delete the meteorological tower depiction from Figure 5.1-1. The proposed change reflects the new meteorological tower location previously approved by the staff and is therefore acceptable.

2.08 REMOVAL OF OUTDATED FOOTNOTES FOR POST-ACCIDENT INSTRUMENTATION

The first footnote the licensee has proposed to delete addresses the containment sump level and containment water level. The footnote states that the requirements for these instruments will become effective after the level transmitters are modified or replaced and become operational. The footnotes for both units specify specific refueling outages during which the equipment modification will be done. For both units these modifications have been done and the instruments have been in service for several years. Therefore the specifications have been applicable for some time, the footnote specifying the beginning of effectiveness is no longer necessary, and the deletion is acceptable. In addition, the note in the Unit 1 bases which provides the specific date reference for applicability is also no longer necessary and is acceptable to delete. The corresponding note in the Unit 2 bases was previously deleted by Amendment Number 119.

The second footnote the licensee has proposed to delete is the "*****" footnote for Table 3.3-11 for Unit 1 and Table 3.3-10 for Unit 2, associated with a specified safety valve position indicator acoustic monitor. The footnote exempts a specific acoustic monitor from its normal operability requirements for the remainder of a specified fuel cycle. The footnote addition was requested and approved by the staff due to inoperability of an acoustic monitor that would have otherwise necessitated a unit shutdown. Since the specified fuel cycles have passed and the monitors have since been restored to service, the deletion of the footnotes is acceptable.

The last footnote proposed to be deleted is footnote 4 on Table 4.3-10 for Unit 2. This footnote states that the core exit thermocouples will not be installed until the 1988 refueling outage and therefore, surveillances will not be required until that time. Since the installation has been completed and surveillances have been required since installation, the note is no longer necessary and the deletion is acceptable.

2.09 CORRECTION OF ERROR IN STORAGE POOL VENTILATION SYSTEM FLOW RATE

The last paragraph in TS 4.9.12.b.4 for Unit 2 requires verifying operability of charcoal adsorbers by testing "while operating the [spent fuel storage pool exhaust] ventilation system at a flow rate of 30,0000 cfm plus or minus 10%." (emphasis added) The flow rate required is obviously incorrect in either value or format. In fact, as listed elsewhere in TS 4.9.12 and in the Unit 1 TS, the correct flow rate is 30,000. The licensee has proposed to correct the value, and the staff finds the proposal acceptable.

2.10 CORRECTION OF CRANE TRAVEL SURVEILLANCE REQUIREMENTS

Currently Cook TS 4.9.7.1 requires a demonstration of operability of the auxiliary building crane interlocks and physical stops which prevent crane travel with loads in excess of 2,500 pounds over fuel assemblies. This text was taken essentially verbatim from NUREG-0452, Revision 4, "Standard Technical Specifications for Westinghouse PWRs." However, the physical stops on the crane trolley and bridge rails are not in locations that prevent crane travel over fuel assemblies. The protection against the transport of excessive loads over the fuel pool is provided only through the crane interlocks. Therefore, the licensee has proposed to delete reference to the physical stops as a feature which restricts crane travel. The use of a single method of protection from high load travel over the fuel pool is consistent with other staff decisions. Other facilities have only a requirement for manual verification of acceptable load before transport over the pool and the requirement is entirely deleted from the *improved* "Standard Technical Specifications Westinghouse Plants", NUREG-1431. Since the proposed change is consistent with existing plant design and staff positions, it is acceptable.

2.11 REMOVAL OF UNNECESSARY BASES MATERIAL ON APDMS

Amendment Number 82 to the Cook Unit 2 TS, dated May 21, 1986, removed the Axial Power Distribution Monitoring System (APDMS) from use to be replaced with Allowable Power Level control. The APDMS equipment was subsequently removed. However, bases section 3/4.3.3.7, which describes the function of the APDMS, was not removed at that time. The licensee has proposed to delete the bases section and the deletion is acceptable.

2.12 CORRECTION OF CONTROL ROOM CABLE VAULT FIRE PROTECTION REQUIREMENTS

Cook Unit 1 TS 3.7.9.3 requires the low pressure CO_2 systems located in the areas shown in Table 3.7-6 to be operable. Table 3.7-6 lists, among other locations, Control Room Cable Vault (Backup)*. The * note states that the Control Room Cable Vault CO_2 System is only required to be operable when the Cable Vault Halon System is operable. In fact, the CO_2 system is a backup to the Halon system and is required when the Halon system is inoperable. The footnote in the corresponding table for Unit 2 correctly states that the CO_2 system is required when the Halon system is inoperable. The licensee has proposed to correct the wording on the Unit 1 table and the staff considers the change acceptable.

2.13 CORRECTION OF AUTOMATIC TRIP LOGIC ACTION

Table 3.3-1 in the Unit 2 TS lists various reactor trip system instrumentation by functional units with associated operability requirements and actions for inoperability. Functional unit 22, "Automatic Trip Logic," is currently listed as being applicable in Modes 1, 2, 3, 4, and 5 with two separate action requirements. As currently listed it is not clear which action is applicable for which mode. In fact, as correctly listed in the Unit 1 TS, Action 1 applies to Modes 1 and 2, and Action 14 applies to Modes 3, 4, and 5. This confusing formatting was previously corrected by Amendment Number 107, but due to overlapping amendment submittals was reintroduced in Amendment Number 127. The licensee's proposal to clarify which actions apply to which modes is an administrative improvement and therefore acceptable.

2.14 ADMINISTRATIVE CHANGES

During the course of this review the staff identified a typographical error in the Unit 1 TS. The last line of Section 5.7.1 reads "applicant Surveillance Requirement", when it should read "applicable Surveillance Requirement". This error is corrected by this amendment. In addition, for consistency, the * footnotes on page 3/4 9-8 of the Unit 1 TS and page 3/4 9-7 of the Unit 2 TS are revised to read "Cook Nuclear Plant" rather than "D. C. Cook". These are administrative changes and acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (58 FR 67849). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: John B. Hickman, NRR

Date: December 30, 1994