August 5, 1985

Docket Nos. 50-315
and 50-316

Distribution
~~Docket file~~
ORB#1 RDG
HThompson
DWigginton
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TBarnhart (8)
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NRC PDR
L PDR
CParrish
OELD
EJordan
JPartlow
WJones
ACRS (10)
RDiggs

Mr. John Dolan, Vice President
Indiana and Michigan Electric Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Dolan:

The Commission has issued the enclosed Amendment No. 87 to Facility Operating License No. DPR-58 and Amendment No. 73 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 transmitted by letter dated December 17, 1984, as supplemented by letter dated June 4, 1985.

These amendments revise the Technical Specification that update the offsite organization chart, organization and responsibilities of the Plant Nuclear Safety Review Committee (PNSRC) and Nuclear Safety and Design Review Committee (NSDRC), to update the reporting requirements addressed by the recent revision to 10 CFR 50.73, to revise the containment isolation valve listing, correct battery electrolyte surveillance temperature, and make a number of editorial changes.

Proposed changes number 3 and 8 from the December 17, 1984 letter have not been acted upon pending further discussions agreed to between the staff and the licensee. These involve the membership changes of the PNSRC and NSDRC prior to NRC approval and deletion of meeting minutes review by the committee.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

/s/DLWigginton

David L. Wigginton, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 87 to DPR-58
2. Amendment No. 73 to DPR-74
3. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:Dl	ORB#1:DL	~~BC-ORB~~#1:DL	OELD	~~AD:OR~~:DL
CParrish	DWigginton	SVarga		GLainas
07/23/85	07/2?/85	07/2?/85	07/29/85	07/2?/85

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

August 5, 1985

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and 50-316

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Indiana and Michigan Electric Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
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cc: w/enclosures
See next page

Mr. John Dolan
Indiana and Michigan Electric Company

Donald C. Cook Nuclear Plant

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Division of Radiological Health
Department of Public Health
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Lansing, Michigan 48909

The Honorable John E. Grotberg
United States House of Representatives
Washington, DC 20515

Regional Administrator, Region III
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Glen Ellyn, Illinois 60137

J. Feinstein
American Electric Power
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. DPR-58

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

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated December 17, 1984, supplemented by letter dated June 4, 1985, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 87, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The change to the Technical Specifications is to become effective within 45 days of issuance of this amendment.

4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 5, 1985

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

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 73
License No. DPR-74

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

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated December 17, 1984, supplemented by letter dated June 4, 1985, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 73 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The change to the Technical Specifications is to become effective within 45 days of issuance of this amendment.

4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 5, 1985

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 87 FACILITY OPERATING LICENSE NO. DPR-58

AMENDMENT NO. 73 FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NOS. 50-315 AND 50-316

Revise Appendix A as follows:

	Remove Pages	Insert Pages
Unit 1		
	1-2	1-2
	3/4 0-1	3/4 0-1
	3/4 3-12	3/4 3-12
	3/4 3-14	3/4 3-14
	3/4 3-26a	3/4 3-26a
	3/4 6-18	3/4 6-18
	3/4 6-21	3/4 6-21
	3/4 8-9	3/4 8-9
	3/4 8-14	3/4 8-14
	6-1	6-1
	6-2	6-2
	6-6	6-6
	6-7	6-7
	6-9	6-9
	6-10	6-10
	6-12	6-12
	6-14	6-14
	6-15	6-15
	6-18 thru 27	6-18 thru 24
Unit 2		
	1-2	1-2
	3/4 1-1	3/4 1-1
	3/4 3-11	3/4 3-11
	3/4 3-13	3/4 3-13
	3/4 3-25a	3/4 3-25a
	3/4 3-35	3/4 3-35
	3/4 3-37	3/4 3-37
	3/4 8-9	3/4 8-9
	6-1	6-1
	6-2	6-2
	6-5	6-5
	6-6	6-6
	6-7	6-7
	6-9	6-9
	6-10	6-10
	6-12	6-12
	6-14	6-14
	6-15	6-15
	6-18 thru 28	6-18 thru 25

REPORTABLE EVENT

1.7 REPORTABLE EVENT shall be any of those conditions specified in 10 CFR 50.73.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

1.8.2 All equipment hatches are closed and sealed.

1.8.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3, and

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Limiting Conditions for Operation and ACTION requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for each specification.

3.0.2 Adherence to the requirements of the Limiting Condition for Operation and/or associated ACTION within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in a MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

3.0.4 Entry into an OPERATIONAL MODE or other specified applicability condition shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION statements unless otherwise excepted. This provision shall not prevent passage through OPERATIONAL MODES as required to comply with ACTION statements.

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, within 2 hours action shall be initiated to place the unit in a MODE in which the applicable Limiting Condition for Operation does not apply by placing it as applicable in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This Specification is not applicable in MODES 5 or 6.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - Compare incore to excore axial imbalance above 15% of RATED THERMAL POWER. Recalibrate if absolute difference ≥ 3 percent.

(4) - Manual ESF functional input check every 18 months.

(5) - Each train tested every other month.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (BLOCK OF SOURCE RANGE REACTOR TRIP) setpoint.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	$\geq$ 17% of narrow range instrument span each steam generator	$\geq$ 16% of narrow range instrument span each steam generator
b. 4 kv Bus Loss of Voltage	3196 volts with a 2-second delay	3196, +18, -36 volts with a 2$\pm$.2 second delay
c. Safety Injection	Not Applicable	Not Applicable
d. Loss of Main Feedwater Pumps	Not Applicable	Not Applicable
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	$\geq$ 17% of narrow range instrument span each steam generator	$\geq$ 16% or narrow range instrument span each steam generator
b. Reactor Coolant Pump Bus Undervoltage	$\geq$ 2750 Volts--each bus	$\geq$ 2725 Volts--each bus
8. LOSS OF POWER		
a. 4 kv Bus Loss of Voltage	3196 volts with a 2-second delay	3196, +18, -36 volts with a 2$\pm$.2 second delay
b. 4 kv Bus Degraded Voltage	3596 volts with a 2.0 min. time delay	3596, +36, -18 volts with a 2.0 minute $\pm$ 6 second time delay

TABLE 3.6-1 (Continued)

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION	ISOLATION TIME IN SECONDS
A. PHASE "A" ISOLATION (Continued)			
63. NCR-107	PRZ Liquid Sample	Yes	10
64. NCR-108	PRZ Liquid Sample	Yes	10
65. NCR-109	PRZ Steam Sample	Yes	10
66. NCR-110	PRZ Steam Sample	Yes	10
67. NCR-252	Primary Water to Pressure Relief Tank	Yes	10
68. PCR-40	Containment Service Air	Yes	10
69. QCM-250	RCP Seal Water Discharge	No	15
70. QCM-350	RCP Seal Water Discharge	No	15
71. QCR-300	Letdown to Letdown Hx.	No	10
72. QCR-301	Letdown to Letdown Hx.	No	10
73. QCR-919	Demineralized Water Supply for Refueling Cavity	Yes	10
74. QCR-920	Demineralized Water Supply for Refueling Cavity	Yes	10
75. RCR-100	PRZ Relief Tank to Gas Anal.	Yes	10
76. RCR-101	PRZ Relief Tank to Gas Anal.	Yes	10
77. VCR-10	Glycol Supply to Fan Cooler	Yes	10
78. VCR-11	Glycol Supply to Fan Cooler	Yes	10
79. VCR-20	Glycol Supply from Fan Cooler	Yes	10
80. VCR-21	Glycol Supply from Fan Cooler	Yes	10
81. XCR-100	Control Air to Containment	No	10
82. XCR-101	Control Air to Containment Isolation	No	10
83. XCR-102	Control Air to Containment Isolation	No	10
84. XCR-103	Control Air to Containment	No	10
B. PHASE "B" ISOLATION			
1. CCM-451	CCW from RCP Oil Coolers	No	60
2. CCM-452	CCW from RCP Oil Coolers	No	60
3. CCM-453	CCW from RCP Thermal Barrier	No	30
4. CCM-454	CCW from RCP Thermal Barrier	No	30
5. CCM-458	CCW to RCP Oil Coolers & Thermal Barrier	No	60
6. CCM-459	CCW to RCP Oil Coolers & Thermal Barrier	No	60
7. ECR-31	Containment Airborne Radiation Monitor	No	10
8. ECR-32	Containment Airborne Radiation Monitor	No	10
9. ECR-33	Containment Airborne Radiation Monitor	No	10
10. ECR-35	Containment Airborne Radiation Monitor	No	10
11. ECR-36	Containment Airborne Radiation Monitor	No	10

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

TABLE 3.6-1 (Continued)

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION	ISOLATION TIME IN SECONDS
C. CONTAINMENT PURGE EXHAUST (Continued)**			
12. VCR-205	Upper Comp. Purge Air Inlet	Yes	5
13. VCR-206	Upper Comp. Purge Air Outlet	Yes	5
14. VCR-207*	Cont. Press Relief Fan Isolation	Yes	5
D. MANUAL ISOLATION VALVES(1)			
1. ICM-111	RHR to RC Cold Legs	Yes	NA
2. ICM-129	RHR Inlet to Pumps	No	NA
3. ICM-250	Boron Injection Inlet	Yes	NA
4. ICM-251	Boron Injection Inlet	Yes	NA
5. ICM-260	Safety Injection Inlet	Yes	NA
6. ICM-265	Safety Injection Inlet	Yes	NA
7. ICM-305	RHR Suction from Sump	Yes	NA
8. ICM-306	RHR Suction from Sump	Yes	NA
9. ICM-311	RHR to RC Hot Legs	Yes	NA
10. ICM-321	RHR to RC Hot Legs	Yes	NA
11. NPX 151 VI	Dead Weight Tester	Yes	NA
12. PA-343	Containment Service Air	No	NA
13. SF-151	Refueling Water Supply	Yes	NA
14. SF-153	Refueling Water Supply	Yes	NA
15. SF-159	Refueling Cavity Drain to Purification System	Yes	NA
16. SF-160	Refueling Cavity Drain to Purification System	Yes	NA
17. SI-171	Safety Injection Test Line	Yes	NA
18. SI-172	Accumulator Test Line	Yes	NA

SURVEILLANCE REQUIREMENTS (Continued)

2. The pilot cell specific gravity, corrected to 77°F and full electrolyte level, is ≥1.200.

3. The pilot cell voltage is ≥2.10 volts, and

4. The overall battery voltage is ≥250 volts.

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

1. The voltage of each connected cell is ≥2.10 volts under float charge and has not decreased more than 0.05 volts from the value observed during the original acceptance test, and

2. The specific gravity, corrected to 77°F and full electrolyte level, of each connected cell is ≥1.200 and has not decreased more than 0.03 from the value observed during the previous test, and

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

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

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

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

3. The battery is capable of supplying the following emergency loads for the specified times with the battery charger disconnected. The battery terminal voltage shall be maintained ≥210 volts throughout the entire test.

2. The pilot cell specific gravity, corrected to 77°F and full electrolyte level, is ≥1.200.

3. The pilot cell voltage is ≥2.10 volts, and

4. The overall battery voltage is ≥250 volts.

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

1. The voltage of each connected cell is ≥2.10 volts under float charge and has not decreased more than 0.05 volts from the value observed during the original acceptance test, and

2. The specific gravity, corrected to 77°F and full electrolyte level, of each connected cell is ≥1.200 and has not decreased more than 0.03 from the value observed during the previous test, and

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

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

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

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

3. The battery charger will supply at least 10 amperes at ≥250 volts for at least 4 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status the emergency loads for the specified times of Table 4.8-2 with the battery charger disconnected. The battery terminal voltage shall be maintained ≥210 volts throughout the entire test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test shall be performed subsequent to the satisfactory completion of the required battery service test.

6.0 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.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

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

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. ALL CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. A site Fire Brigade of at least 5 members shall be maintained onsite at all times. The fire Brigade shall not include 3 members of the minimum shift crew necessary for safe shutdown of the unit or any personnel required for other essential functions during a fire emergency.

g. The amount of overtime worked by plant staff members performing safety-related functions must be limited in accordance with NRC Policy Statement on working hours (Generic Letter No. 82-12).

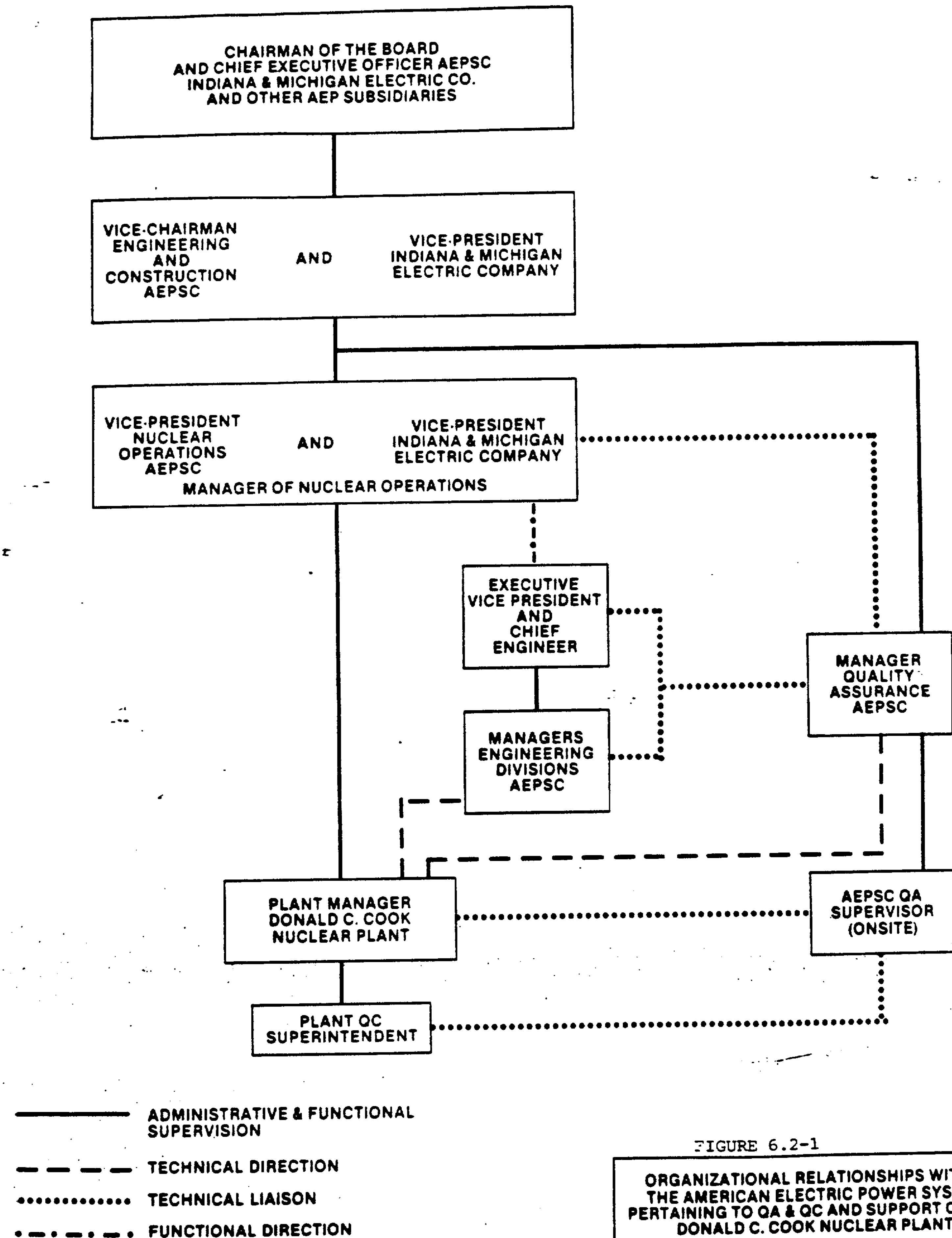

FIGURE 6.2-1

ORGANIZATIONAL RELATIONSHIPS WITHIN THE AMERICAN ELECTRIC POWER SYSTEM PERTAINING TO QA & QC AND SUPPORT OF THE DONALD C. COOK NUCLEAR PLANT

COMPOSITION

6.5.1.2 The PNSRC shall be composed of the:

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

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PNSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PNSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PNSRC shall meet at least once per calendar month and as convened by the PNSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the PNSRC shall consist of the Chairman or his designated alternate and sufficient members, including alternates, to constitute a majority.

RESPONSIBILITIES

6.5.1.6 The PNSRC shall be responsible for:

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

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

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

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

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

f. Review of all REPORTABLE EVENTS.

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

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

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

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

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

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

COMPOSITION

6.5.2.2 The NSDRC shall be composed of the following Regular Members:

1. Vice Chairman, Engineering and Construction
2. Executive Assistant to the President, I&MECo
3. Executive Vice President and Chief Engineer
4. Senior Vice President, Electrical Engineering and Deputy Chief Engineer
5. Assistant Vice President, Mechanical Engineering
6. Vice President, Engineering Administration
7. Vice President, Nuclear Operations (NSDRC Chairman)
8. Assistant Vice President, Environmental Engineering
9. Plant Manager, Donald C. Cook Nuclear Plant
10. Design Division Manager
11. Manager, Quality Assurance
12. Consulting Engineer, Nuclear Operations Division
13. Nuclear Safety and Licensing Section Manager, Nuclear Operations Division (NSDRC Secretary)
14. Vice President, Fossil Plant Operations

ALTERNATE MEMBERS

6.5.2.3 Designated Alternate Members shall be appointed by the Vice Chairman, Engineering and Construction or such other person as he shall designate. In addition, Temporary Alternate Members may be appointed by the NSDRC Chairman to serve on an interim basis, as required. Temporary Alternate members are empowered to act on the behalf of the Regular or Designated Alternate members for whom they substitute.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NSDRC Chairman to provide expert advice to the NSDRC.

MEETING FREQUENCY

6.5.2.5 The NSDRC shall meet at least once per six months.

QUORUM

6.5.2.6 A quorum, the minimum number of regular members and alternates required to hold a NSDRC meeting, shall be eight (8) members, of whom no more than two (2) shall be Designated or Temporary Alternates. The Chairman or Acting Chairman shall be present for all NSDRC meetings. If the number of members present* is greater than a quorum, then the majority participating and voting at the meeting shall not have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The NSDRC is responsible for assuring that independent** reviews of the following are performed:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations form normal and expected performance of plant equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, and components.

i. Reports and meeting minutes of the PNSRC.

* Regular NSDRC members are expected to attend the meeting whenever possible, and alternates may attend as voting members only on an irregular basis. If both a regular member and his alternate attend a meeting, only the regular member may participate as a voting member, and the alternate is considered a guest.

** Independent reviews may be performed by groups which report directly to the NSDRC and which must have NSDRC membership participation.

m. The PROCESS CONTROL PROGRAM and implementing procedures for solidification of radioactive wastes at least once per 24 months.

n. The performance of activities required by the Quality Assurance Program to meet the criteria of Regulatory Guide 1.21, Rev. 1, June 1974 and Regulatory Guide 4.1, Rev. 1, April 1975 at least once per 12 months.

AUTHORITY

6.5.2.9 The NSDRC shall report to and advise the Vice Chairman, Engineering and Construction, AEPSC, on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of NSDRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NSDRC meeting shall be prepared, approved and forwarded to the Vice Chairman, Engineering and Construction, AEPSC, within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Vice Chairman, Engineering and Construction, AEPSC, within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Vice Chairman, Engineering and Construction, AEPSC, and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE EVENT ACTION

6.6.1 Each REPORTABLE EVENT requiring notification to the Commission shall be reviewed by the PNSRC and submitted to the NSDRC and the Vice President, Nuclear Operations.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PNSRC and approved by the Plant Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[2] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

b. The complete results of steam generator tube inservice inspections performed during the report period (reference Specification 4.4.5.5.b).

c. Documentation of all challenges to the pressurizer power operated relief valves (PORVs) or safety valves.

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1 A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2 This tabulation supplements the requirements of 20.407 of 10 CFR Part 20.

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The radioactive effluent release report shall include the following information for each type of solid waste shipped offsite during the report period:

a. Volume (cubic meters),

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidifcation agent (e.g., cement).

The radioactive effluent release report shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluent on a quarterly basis.

The radioactive effluent release reports shall include any change to the PROCESS CONTROL PROGRAM (PCP) and the OFFSITE DOSE CALCULATION MANUAL (ODCM) made during the reporting period.

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office Of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office no later than the 15th of each month following the calendar month covered by the report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference Specifications:

a. Inservice Inspection Program Review, Specification 4.4.10.

b. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

c. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

d. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

e. Seismic event analysis, Specification 4.3.3.3.2.

f. Sealed Source leakage in excess of limits, Specification 4.7.7.1.3.

g. Fire Detection Instrumentation, Specification 3.3.3.7.

h. Fire Suppression Systems, Specifications 3.7.9.1, 3.7.9.2, 3.7.9.3 and 3.7.9.4.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.
b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
c. All REPORTABLE EVENTS submitted to the Commission.
d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
e. Records of changes made to the procedures required by Specification 6.8.1.
f. Records of sealed source leak tests and results.
g. Records of annual physical inventory of all sealed source material on record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.
b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.
c. Records of radiation exposure for all individuals entering radiation control areas.
d. Records of gaseous and liquid radioactive material released to the environs.
e. Records of radioactive shipments.
f. Records of transient or operational cycles for those facility components identified in Table 5.9-1.
g. Records of training and qualification for current members of the Plant staff.
h. Records of in-service inspections performed pursuant to these Technical Specifications.
i. Records of Quality Assurance activities required by the QA Manual.
j. Records of reviews performed for changes made to procedures or equipment or review of tests and experiments pursuant to 10 CFR 50.59.
k. Records of meetings of the PNSRC and the NSDRC.
l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.
m. Records of reactor tests and experiments.
n. Records of the service lives of hydraulic snubbers listed on Table 3.7-4 including the date at which service life commences and associated installation and maintenance records.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. A High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.12.1.a above, and in addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Operating Engineer on duty.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982, all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors "Guidelines for Evaluating Environment Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License No. DPR-74, dated October 24, 1980.

6.13.2 By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 The PCP shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the PNSRC.

2. Shall become effective upon review and acceptance by the PNSRC.

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.15.1 The ODCM shall be approved by the Commission prior to implementation.

6.15.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semi-Annual Radioactive Effluent Release Report in the next report after the report period the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the PNSRC.

2. Shall become effective upon review and acceptance by the PNSRC.

6.15.3 Commission initiated changes:

1. Shall be determined by the PNSRC to be applicable to the facility after consideration of facility design.

2. The licensee shall provide the Commission with written notification of their determination of applicability including any necessary revisions to reflect facility design.

6.16 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and Solid)

6.16.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Annual Operating Report for the period in which the evaluation was reviewed by the (PNSRC). The discussions of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

   e. An evaluation of the change which shows the expected maximum exposure to individuals in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

   f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the PNSRC.

2. Shall become effective upon review and acceptance by the PNSRC.

6.16.2 Commission initiated changes:

1. The applicability of the change to the facility shall be determined by the (PNSRC) after consideration of the facility design.

2. The licensee shall provide the Commission with written notification of its determination of applicability including any necessary revisions to reflect facility design.

REPORTABLE EVENT

1.7 A REPORTABLE EVENT shall be any of those conditions specified in 10 CFR 50.73.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

1.8.2 All equipment hatches are closed and sealed.

1.8.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.8.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{avg} > 200^{\circ}F$

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq 1.6\% \Delta k/k$.

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

ACTION:

With the SHUTDOWN MARGIN $< 1.6\% \Delta k/k$, immediately initiate and continue boration at ≥ 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq 1.6\% \Delta k/k$:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

*See Special Test Exception 3.10.1

\# With $K_{eff} \geq 1.0$

\#\# With $K_{eff} < 1.0$

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference > 2 percent.

(3) - Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference $\geq$ 3 percent.

(4) - Manual ESF functional input check every 18 months.

(5) - Each train tested every other month.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (BLOCK OF SOURCE RANGE REACTOR TRIP) setpoint.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	≥ 21% of narrow range instrument span each steam generator	≥ 20% of narrow range instrument span each steam generator
b. 4 kv Bus Loss of Voltage	3196 volts with a 2 second delay	3196, +18, -36 volts with a 2 ± 0.2 second delay
c. Safety Injection	Not Applicable	Not Applicable
d. Loss of Main Feedwater Pumps	Not Applicable	Not Applicable
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	≥ 21% of narrow range instrument span each steam generator	≥ 20% of narrow range instrument span each steam generator
b. Reactor Coolant Pump Bus Undervoltage	≥ 2750 Volts--each bus	≥ 2725 Volts--each bus
8. LOSS OF POWER		
a. 4 kv Bus Loss of Voltage	3196 volts with a 2 second delay	3196, +18, -36 volts with a 2 ± 0.2 second delay
b. 4 kv Bus Degraded Voltage	3596 volts with a 2.0 minute time delay	3596, +36, -18 volts with a 2.0 minute ± 6 second time delay

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

OPERATION MODE/INSTRUMENT	MINIMUM CHANNELS OPERABLE	HIGH ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. MODES 1, 2, 3, & 4				
a. AREA MONITOR				
i. Upper Containment	1	$\leq$ 2 x normal channel reading	10^{-1} to 10^{4} mR/hr	19
b. PROCESS MONITORS				
i. Particulate	1	$\leq$ 2 x normal channel reading	1.5×10^{-4} to 1.5 uCi	20
ii. Noble Gas	1	$\leq$ 2 x normal channel reading	10^{-7} to 10^{-2} uCi/cc	20
2. MODE 6				
a. TRAIN A	any 2/3 Channels			22
i. Containment Area Radiation Channel-VRS-2101		$\leq$ 2 x normal channel reading	10^{-1} to 10^{4} mR/hr	
ii. Particulate Channel-ERS-2301		$\leq$2 x normal channel reading	1.5×10^{-4} to 1.5 uCi	
iii. Noble Gas Channel-ERS-2305		$\leq$2 x normal channel reading	10^{-7} to 10^{-2} uCi/cc	
b. TRAIN B	any 2/3 Channels			22
i. Containment Area Radiation Channel-VRS-2201		Same as 2.a	Same as 2.a	
ii. Particulate Channel-ERS-2401		Same as 2.a	Same as 2.a	
iii. Noble Gas Channel ERS-2405		Same as 2.a	Same as 2.a	
3. *				
a. Spent Fuel Storage	1	$\leq$15 mR/hr	10^{-1} to 10^{4} mR/hr	19

* With fuel in storage pool or building.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

OPERATING MODE/INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. MODES 1, 2, 3, & 4				
a. AREA MONITOR				
i. Upper Containment	S	R	M	1, 2, 3 & 4
b. PROCESS MONITORS				
i. Particulate	S	R	M	1, 2, 3 & 4
ii. Noble Gas	S	R	M	1, 2, 3 & 4
2. MODE 6				
a. TRAIN A				
i. Containment Area Radiation Channel	S	R	M	6
ii. Particulate Channel	S	R	M	6
iii. Noble Gas Channel	S	R	M	6
b. TRAIN B				
i. Containment Area Radiation Channel	S	R	M	6
ii. Particulate Channel	S	R	M	6
iii. Noble Gas Channel	S	R	M	6
3. *				
a. SPENT FUEL STORAGE	S	R	M	*

* With fuel in the storage pool or building.

SURVEILLANCE REQUIREMENTS (Continued)

2. The pilot cell specific gravity, corrected to 77°F and full electrolyte level, is ≥ 1.200,

3. The pilot cell voltage is ≥ 2.10 volts, and

4. The overall battery voltage is ≥ 250 volts.

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

1. The voltage of each connected cell is ≥ 2.10 volts under float charge and has not decreased more than 0.05 volts from the value observed during the original acceptance test,

2. The specific gravity, corrected to 77°F and full electrolyte level, of each connected cell is ≥ 1.200 and has not decreased more than 0.03 from the value observed during the previous test, and

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

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

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

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

3. The battery charger will supply at least 140 amperes at ≥ 250 volts for at least 4 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status the emergency loads for the specified times of Table 4.8-1A with the battery charger disconnected. The battery terminal voltage shall be maintained ≥ 210 volts throughout the entire test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test shall be performed subsequent to the satisfactory completion of the required battery service test.

6.0 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.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown in Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

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

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. ALL CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. A site Fire Brigade of at least 5 members shall be maintained onsite at all times. The Fire Brigade shall not include 3 members of the minimum shift crew necessary for safe shutdown of the unit or any personnel required for other essential functions during a fire emergency.

g. The amount of overtime worked by plant staff members performing safety-related functions must be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).

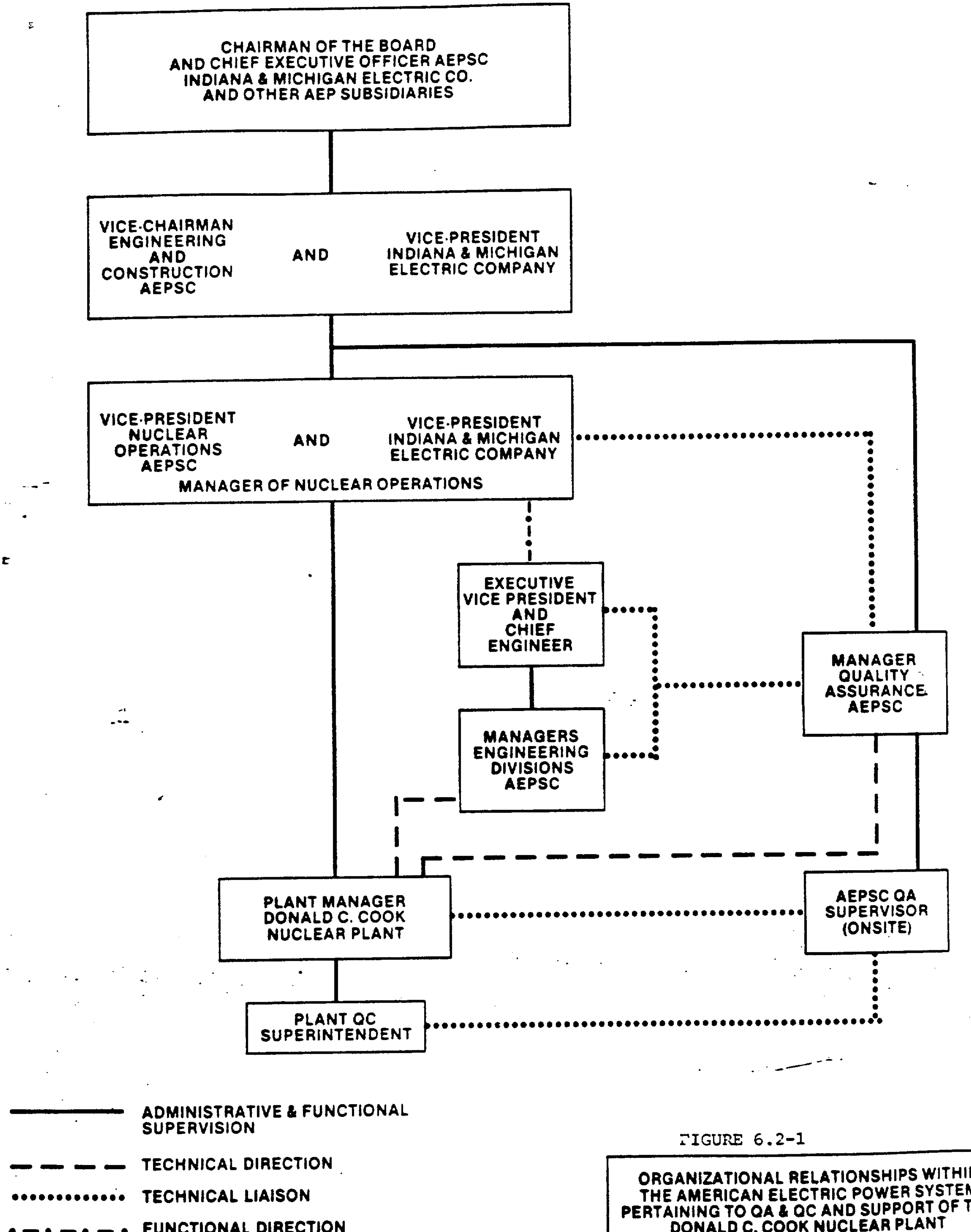

FIGURE 6.2-1

ORGANIZATIONAL RELATIONSHIPS WITHIN THE AMERICAN ELECTRIC POWER SYSTEM PERTAINING TO QA & QC AND SUPPORT OF THE DONALD C. COOK NUCLEAR PLANT

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Radiation Protection Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

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

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Plant Manager and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976.

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY REVIEW COMMITTEE (PNSRC)

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

COMPOSITION

6.5.1.2 The PNSRC shall be composed of the:

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

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PNSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PNSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PNSRC shall meet at least once per calendar month and as convened by the PNSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the PNSRC shall consist of the Chairman or his designated alternate and sufficient members, including alternates, to constitute a majority.

RESPONSIBILITIES

6.5.1.6 The PNSRC shall be responsible for:

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

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

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

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

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

f. Review of all REPORTABLE EVENTS.

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

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

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

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

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

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

AUTHORITY

6.5.1.7 The PNSRC shall:

a. Recommend to the Plant Manager written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the NSDRC of disagreement between the PNSRC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

COMPOSITION

6.5.2.2 The NSDRC shall be composed of the following Regular Members:

1. Vice Chairman, Engineering and Construction
2. Executive Assistant to the President, I&MECo
3. Executive Vice President and Chief Engineer
4. Senior Vice President, Electrical Engineering and Deputy Chief Engineer
5. Assistant Vice President, Mechanical Engineering
6. Vice President, Engineering Administration
7. Vice President, Nuclear Operations (NSDRC Chairman)
8. Assistant Vice President, Environmental Engineering
9. Plant Manager, Donald C. Cook Nuclear Plant
10. Design Division Manager
11. Manager, Quality Assurance
12. Consulting Engineer, Nuclear Operations Division
13. Nuclear Safety and Licensing Section Manager, Nuclear Operations Division (NSDRC Secretary)
14. Vice President, Fossil Plant Operations

ALTERNATE MEMBERS

6.5.2.3 Designated Alternate Members shall be appointed by the Vice Chairman, Engineering and Construction or such other person as he shall designate. In addition, Temporary Alternate Members may be appointed by the NSDRC Chairman to serve on an interim basis, as required. Temporary Alternate members are empowered to act on the behalf of the Regular or Designated Alternate members for whom they substitute.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NSDRC Chairman to provide expert advice to the NSDRC.

MEETING FREQUENCY

6.5.2.5 The NSDRC shall meet at least once per six months.

QUORUM

6.5.2.6 A quorum, the minimum number of regular members and alternates required to hold a NSDRC meeting, shall be eight (8) members, of whom no more than two (2) shall be Designated or Temporary Alternates. The Chairman or Acting Chairman shall be present for all NSDRC meetings. If the number of members present* is greater than a quorum, then the majority participating and voting at the meeting shall not have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The NSDRC is responsible for assuring that independent** reviews of the following are performed:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations form normal and expected performance of plant equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, and components.

i. Reports and meeting minutes of the PNSRC.

* Regular NSDRC members are expected to attend the meeting whenever possible, and alternates may attend as voting members only on an irregular basis. If both a regular member and his alternate attend a meeting, only the regular member may participate as a voting member, and the alternate is considered a guest.

** Independent reviews may be performed by groups which report directly to the NSDRC and which must have NSDRC membership participation.

m. The PROCESS CONTROL PROGRAM and implementing procedures for solidification of radioactive wastes at least once per 24 months.

n. The performance of activities required by the Quality Assurance Program to meet the criteria of Regulatory Guide 1.21, Rev. 1, June 1974 and Regulatory Guide 4.1, Rev. 1, April 1975 at least once per 12 months.

AUTHORITY

6.5.2.9 The NSDRC shall report to and advise the Vice Chairman, Engineering and Construction, AEPSC, on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of NSDRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NSDRC meeting shall be prepared, approved and forwarded to the Vice Chairman, Engineering and Construction, AEPSC, within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Vice Chairman, Engineering and Construction, AEPSC, within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Vice Chairman, Engineering and Construction, AEPSC, and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE EVENT ACTION

6.6.1 Each REPORTABLE EVENT requiring notification to the Commission shall be reviewed by the PNSRC and submitted to the NSDRC and the Vice President, Nuclear Operations.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PNSRC and approved by the Plant Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[2] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

b. The complete results of steam generator tube inservice inspections performed during the report period (reference Specification 4.4.5.5.b).

c. Documentation of all challenges to the pressurizer power operated relief valves (PORVs) or safety valves.

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1 A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2 This tabulation supplements the requirements of 20.407 of 10 CFR Part 20.

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The radioactive effluent release report shall include the following information for each type of solid waste shipped offsite during the report period:

a. Volume (cubic meters),

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidifcation agent (e.g., cement).

The radioactive effluent release report shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluent on a quarterly basis.

The radioactive effluent release reports shall include any change to the PROCESS CONTROL PROGRAM (PCP) and the OFFSITE DOSE CALCULATION MANUAL (ODCM) made during the reporting period.

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office Of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office no later than the 15th of each month following the calendar month covered by the report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Unit No. 1, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Unit No. 1, Specification 3.3.3.4.

d. Fire Detection Instrumentation, Specification 3.3.3.8.

e. Fire Suppression Systems, Specifications, 3.7.9.1, 3.7.9.2, 3.7.9.3 and 3.7.9.4.

f. Seismic Event Analysis, Specification 4.3.3.3.2.

g. Sealed Source leakage in excess of limits, Specification 4.7.8.1.3.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.
b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
c. ALL REPORTABLE EVENTS submitted to the Commission.
d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
e. Records of changes made to Operating Procedures.
f. Records of sealed source and fission detection leak tests and results.
g. Records of annual physical inventory of all sealed source material on record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.
b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.
c. Records of radiation exposure for all individuals entering radiation control areas.
d. Records of gaseous and liquid radioactive material released to the environs.
e. Records of transient or operational cycles for those facility components identified in Table 5.7-1.
f. Records of reactor tests and experiments.
g. Records of training and qualification for current members of the Plant staff.
h. Records of in-service inspections performed pursuant to these Technical Specifications.
i. Records of Quality Assurance activities required by the QA Manual.
j. Records of reviews performed for changes made to procedures or equipment or review of tests and experiments pursuant to 10 CFR 50.59.
k. Records of meetings of the PNSRC and the NSDRC.
l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.
m. Records of radioactive shipments.
n. Records of the service lives of hydraulic snubbers listed on Table 3.7-9 including the date at which service life commences and associated installation maintenance records.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

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

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

6.12.2 The requirements of 6.12.1, above shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Plant Health Physicist.

* Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982, all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environment Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License No. DPR-74, dated October 24, 1980.

6.13.2 By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 The PCP shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the PNSRC.

2. Shall become effective upon review and acceptance by the PNSRC.

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.15.1 The ODCM shall be approved by the Commission prior to implementation.

6.15.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semi-Annual Radioactive Effluent Release Report in the next report after the report period the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the PNSRC.

2. Shall become effective upon review and acceptance by the PNSRC.

6.15.3 Commission initiated changes:

1. Shall be determined by the PNSRC to be applicable to the facility after consideration of facility design.

2. The licensee shall provide the Commission with written notification of their determination of applicability including any necessary revisions to reflect facility design.

6.16 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and Solid)

6.16.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Annual Operating Report for the period in which the evaluation was reviewed by the (PNSRC). The discussions of each change shall contain:

    a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

    b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

    c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

    d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

    e. An evaluation of the change which shows the expected maximum exposure to individuals in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

    f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the PNSRC.

2. Shall become effective upon review and acceptance by the PNSRC.

6.16.2 Commission initiated changes:

1. The applicability of the change to the facility shall be determined by the (PNSRC) after consideration of the facility design.

2. The licensee shall provide the Commission with written notification of its determination of applicability including any necessary revisions to reflect facility design.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA AND MICHIGAN ELECTRIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

DOCKET NOS. 50-315 AND 50-316

Introduction

By letter dated December 17, 1984, the Indiana and Michigan Electric Company (IMEC) submitted an application to amend Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The amendments would change the Technical Specifications to update the offsite organization chart, and organization and responsibilities of the Plant Nuclear Safety Review Committee (PNSRC) and the Nuclear Safety and Design Review Committee (NSDRC), to update the reporting requirements addressed by the recent revision to 10 CFR 50.73, to revise the containment isolation valve listing, to correct an error in reference to the battery electrolyte temperature for surveillance, and to make a number of editorial changes.

By letter dated June 4, 1985, IMEC submitted additional information as additional clarification of the proposed technical specification changes. The June 4, 1985 letter also included an amended technical specification to increase the committee meeting quorums in conjunction with the previously proposed increased committee membership and to include additional clarifying language on committee meeting attendance and sub-committee reviews. The June 4, 1985 proposed technical specifications do not represent a significant departure from the December 17, 1984 submittal in that the quorum requirements are set by the approved number of committee members and the additional footnotes proposed by the licensee on committee attendance and sub-committee reviews was provided at NRC request but is not required to be in the technical specifications.

Evaluation

A) Administrative Controls

A brief description and our evaluation of the proposed changes are discussed separately below:

1. Figure 6.2-1 (Offsite Organization Chart) for Units 1 and 2 is changed to reflect the reorganization that occurred as a result of the quality assurance enhancements that were made pursuant to NRC recommendations and as a result of the corporate move to Columbus, Ohio. These changes are acceptable as they do not diminish the level of corporate support to the nuclear plant.

2. Sections 6.5.1.2 and 6.5.1.5 are changed to reflect title changes and changes in the membership of the Plant Nuclear Safety Review Committee (PNSRC). The following management positions are added:

   - Assistant Plant Manager - Maintenance
   - Assistant Plant Manager - Operations
   - Technical Superintendent - Engineering
   - Technical Superintendent - Physical Sciences
   - Quality Control Superintendent

   The following positions are deleted from PNSRC membership:

   - Instrument and Control Engineer
   - Nuclear Engineer
   - Chemical Supervisor
   - Performing Supervisory Engineer

The above changes are reflected in a new organizational chart enclosed with the June 4, 1985, submittal (Attachment 3, p. 6). Page 4 of Attachment 3 provides a background sketch of the individuals filling the new positions.

This change would enhance management participation in and knowledge of specified PNSRC responsibilities by having higher level management personnel as PNSRC members. It is, therefore, acceptable.

3. The licensee proposes that a footnote be added to Sections 6.5.1.2 and 6.5.2.2 for Units 1 and 2 which would specify that PNSRC and NSDRC membership changes resulting from title and reorganization changes could be made without prior NRC approval, but that the licensee would notify NRR of the change within 30 days. This proposal is unacceptable as stated since it could result in a change to the license without prior NRC approval. We propose to withhold a decision on this item until after further discussion with the licensee to clarify the licensee's and the staff's interpretation of this requirement. It may not be necessary for the licensee to specify titles of members of the Committee and, therefore, be required to notify NRC when a member's title changes or the member is replaced; rather the licensee might specify the numbers and qualifications of the membership such that notification of the NRC would be required only when the numbers or qualifications of members change. The licensee is in agreement on withholding a decision at this time on the proposed change.

4. Section 6.5.2.2 for Units 1 and 2 is changed to reflect a title change and membership changes for the NSDRC as follows:

- Manager, Quality Assurance (added)
- Consulting Engineer, Nuclear Operations Division (added)
- V.P., Fossil Plant Operations (former V.P., Mechanical Engineering)
- Executive Assistant to President and Chief Operating Officer of I&MECo. (replacing the President and Chief Operating Officer)

Descriptions of the above positions (with the exception of the Manager, QA, which is in the FSAR) and a new organizational chart is enclosed with the licensee's June 4, 1985, submittal (Attachment 3, pp. 2 and 3).

The proposed changes do not detract from the capabilities of the NSRDC with the exception of the deletion of the President and Chief Operating Officer of I&MECo. We agree with this deletion, however, since this individual would normally be much too occupied with other duties to serve on the NSRDC. We, therefore, find the proposed changes acceptable.

5. Section 6.5.2.5 is changed for both Units 1 and 2 to delete the condition related to the NSDRC meeting frequency for the initial year of facility operation. Since the initial year has passed, this change is acceptable.

6. Section 6.5.2.6 for Units 1 and 2 is changed to clarify the definition of "quorum" as follows:

> A quorum, the minimum number of regular members and alternates required to hold a NSDRC meeting, shall be eight (8) members, of whom no more than two (2) shall be "Designated" or "Temporary" Alternates. The Chairman or Acting Chairman shall be present for all NSDRC meetings. If the number of members present is greater than a quorum, then the majority participating and voting at the meeting shall not have line responsibility for operation of the facility.

In addition, the licensee proposes that a footnote be added to Section 6.5.2.6 for Units 1 and 2 which states that regular NSDRC members are expected to attend the meeting whenever possible, and that alternates may attend as voting members only on an irregular basis. If both a regular member and his alternate attend a meeting, only the regular member may participate as a voting member, and the alternate is considered a guest.

We find this revised definition of a quorum, with the explanatory footnote, acceptable.

7. The licensee proposes that a footnote be added to Section 6.5.2.7 for Units 1 and 2 which states that independent reviews may be performed by groups which report directly to the NSDRC and which must have NSDRC membership participation. The Technical Specifications require that the NSRDC be responsible to assure that independent reviews are performed. This requirement is met by having groups that report to the NSRDC perform the actual reviews. Therefore, we find this clarifying footnote acceptable.

8. The licensee proposes to delete Section 6.5.2.7.i for both Units 1 and 2. These deletions would eliminate the requirement for the NSDRC to review the reports and meeting minutes of the PNSRC. The licensee contends that these reviews are redundant and do not result in any increase in plant safety. The staff does not agree. A principal function of corporate safety review groups traditionally has been to maintain an oversight over the activities of plant safety review groups. We have discussed this matter with representatives of the licensee and we propose to withold a decision on this item pending the outcome of further discussions. We believe the licensee's request for change stems from a difference in interpretation of the extent to which the NSDRC reviews matters previously reviewed by the PNSRC. The licensee is in agreement on withholding a decision at this time on the proposed change.

9. The licensee proposes (1) to change the spelling of the word "Chairman" in Section 6.5.1.6.g of the Unit 2 Technical Specifications; (2) to change the word "modification" to "solidification" in Section 6.5.2.8.m of the Unit 1 Technical Specifications; and (3) to change "Recommended"

to "Recommend" in Section 6.5.1.7.a of the Unit 2 Technical Specifications. These changes are purely administrative in nature, correct obvious errors in the Technical Specifications, and are acceptable.

10. The licensee proposes to change the wording in the Technical Specifications, Sections 6.5.1.6.f, 6.5.2.7.g, and 6.6 for Units 1 and 2, from "Reportable Occurrences" to "Reportable Events." This change is in response to Generic Letter 83-43, which relates to the reporting requirements of 10 CFR 50.73, and is acceptable.
The licensee also proposes to delete Sections 6.9.1.11, 6.9.1.12, and 6.9.1.13 in response to guidance provided in Generic Letter 83-43. Section 6.10.1.c has been revised to "all Reportable Events" and Section 6.6 changed to reduce the redundancy between the Technical Specifications and 10 CFR 50.73. We find these changes acceptable.

B). Containment air service penetration isolation barriers.

The licensee proposes to use an automatic isolation valve (PCR-40), in lieu of a blind flange, outside containment, and a check valve (PA-343), in lieu of the manual valve (PA-145), inside containment as the containment isolation barriers for the containment service air line. The automatic isolation valve is actuated upon receipt of a Phase A isolation signal. The purpose of this change is to permit the use of the containment air service penetration above MODE 5. We find that the change in isolation barriers meets the isolation requirements of General Design Criterion 56, and, therefore, is acceptable. Accordingly, the proposed revision of Table 3.6-1 of the Technical Specifications to reflect the above design change is acceptable.

C) Shutdown Margin - Reference to Control Rod Insertion Limits

The licensee notes that Surveillance Requirements 4.1.1.1.1 b and c improperly reference limits of Specification 3.1.3.5 for Unit 2. The Westinghouse Standard Specifications and the Unit 1 Technical Specifications properly reference the Control Rod Insertion Limits which in 3.1.3.5 for those instances, however, the Control Rod Insertion Limit for Unit 2 is 3.1.3.6. The licensee's request is a correction brought to our attention and is acceptable.

D) Battery Surveillance Temperature

The IMEC proposed a change to the existing Unit 1 Technical Specifications 4.8.2.3.2.a.2, 4.8.2.3.2.b.2 (Page 3/4 8-9), and 4.8.2.5.2.b.2 (Page 3/4 8-14) and Unit 2 Technical specification 4.8.2.3.2.b.2. The proposed change will increase the temperature that specific gravity is measured at or corrected to from the present 70°F to 77°F. The licensee states that the 70°F temperature shown in the above technical specifications is due to a typographical error and is inconsistent with the temperature used in other portions of the technical specifications. In addition, the new temperature is more conservative and is consistent with standard practice. We concur with the licensee and find the proposed change acceptable.

E) Administrative Changes

The licensee has requested a number of changes to correct references to amendment numbers, correct instrument identification numbers, correct reference indication marks, and delete obsolete footnotes. The references to the amendment numbers is to properly identify the amendment to the applicable Unit. The Unit 2 Technical Specifications incorrectly listed the Unit 1 instrument numbers in Table 3.3-6. The Unit 2 footnote reference on Table 3.3-6 should apply to item 3 only and the additional footnote on that Table regarding the 1982 refueling outage is no longer applicable. All of these changes are proper and correct and the change will have no bearing on safety or safe operation. These changes are acceptable.

Environmental Consideration

These amendments involve a change in the installation or use of the facilities components located within the restricted areas as defined in 10 CFR 20. The staff has determined that these 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec. 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 these amendments.

Conclusion

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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