DISTRIBUTION
Docket File
NRC PDR
Local PDR
NSIC
TERA
ORB 1 File
D. Eisenhut
C. Parrish
R. Cilimberg
OELD
OI&E (2)
T. Barnhart (8)
L. Schneider
D. Brinkman
ACRS (10)
OPA (Clare Miles)
R. Diggs
R. Ballard
ASLAB

October 4, 1982

Docket Nos. 50-315
and 50-316

Mr. John Dolan, Vice President
Indiana and Michigan Electric Company
Post Office Box 18
Bowling Green Station
New York, New York 10004

Dear Mr. Dolan:

The Commission has issued the enclosed Amendment No. 63 to Facility Operating License No. DPR-58 and Amendment No. 45 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your applications transmitted by letters dated December 22, 1978, February 13, 1979, February 22, 1980 and May 26, 1981.

These amendments revise the Technical Specifications to (1) change the surveillance requirements for the Boron Injection System, the Safety Injection System, the Containment Return Air Fan and the Spray Additive System, (2) change the requirements for updating the core flux mapping, (3) revise Reactor System Interlock P-8 Rated Thermal Power for N-1 loop operation for Unit No. 2, (4) administratively revises a number of technical specifications to correct editorial errors, (5) administratively deletes the special operability requirements for certain cabinets, motor control centers and switchgear since the permanent seismic supports have been installed, and (6) administratively adds surveillance requirements and limited conditions for operation (LCO) for seismic monitoring instruments in Unit 2. These instruments are shared between Unit 1 and Unit 2 and the surveillance requirements and LCO's are currently only covered in the Unit 1 Technical Specifications.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by:
S. A. Varga

for
Ramon L. Cilimberg, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

8210250340 821004
PDR ADOCK 05000315
P PDR

Enclosures:
1. Amendment No. 63 to DPR-58
2. Amendment No. 45 to DPR-74
3. Safety Evaluation
4. Notice of Issuance

OFFICE	cc w/enclosures:		ORB 1	ORB 1	ORB 1	AD:OR	OELD
SURNAME	See next page		CParrish	RCilimberg/rs	SVarga	GLainas	
DATE			9/15/82	9/15/82	9/ /82	9/ /82	9/ /82

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

Mr. John Dolan
Indiana and Michigan Electric Company

cc: Mr. Robert W. Jurgensen
Chief Nuclear Engineer
American Electric Power
Service Corporation
2 Broadway
New York, New York 10004

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, D. C. 20036

W. G. Smith, Jr., Plant Manager
Donald C. Cook Nuclear Plant
P. O. Box 458
Bridgman, Michigan 49106

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

Mr. Wade Schuler, Supervisor
Lake Township
Baroda, Michigan 49101

Mr. William R. Rustem (2)
Office of the Governor
Room 1 - Capitol Building
Lansing, Michigan 48913

Honorable James Bemenek, Mayor
City of Bridgman, Michigan 49106

Regional Radiation Representative
EPA Region V
230 South Dearborn Street
Chicago, Illinois 60604

Maurice S. Reizen, M.D.
Director
Department of Public Health
P.O. Box 30035
Lansing, Michigan 48109

William J. Scanlon, Esquire
2034 Pauline Boulevard
Ann Arbor, Michigan 48103

The Honorable Tom Corcoran
United States House of Representatives
Washington, D. C. 20515

James G. Keppler
Regional Administrator - Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

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. 63
License No. DPR-58

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

   A. The applications for amendment by Indiana and Michigan Electric Company (the licensee) dated December 22, 1978, February 22, 1980 and May 26, 1981 comply 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 applications, 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. 63, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 4, 1982

1.0 DEFINITIONS

DEFINED TERMS -

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3250 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

INSTRUMENTATION

AXIAL POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.6 The axial power distribution monitoring system (APDMS) shall be OPERABLE with:

a. At least two detector thimbles available for which $\bar{R}$ has been determined from full incore flux maps. These two thimbles shall be those having the lowest uncertainty, σ, covering the full configuration of permissible rod patterns permitted at RATED THERMAL POWER.

b. At least two movable detectors, with associated devices and readout equipment, available for mapping $F_j(Z)$ in the above required thimbles.

APPLICABILITY: When the APDMS is used for monitoring the axial power distribution*#.

ACTION: With the APDMS inoperable, do not use the system for determining the Axial Power Distribution. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6.1 The full incore flux maps used to determine $\bar{R}$ and for monitoring $F_j(Z)$ shall be updated at least once per 31 EFPD. The continued accuracy and representativeness of the selected thimbles shall be verified by using their latest flux maps to update the $\bar{R}$ for each representative thimble. The original uncertainty, σ, shall not be updated, except as follows:

*Except as provided in Specification 4.2.6.1.b.
#The APDMS may be out of service: 1) when incore maps are being taken as part of the Augmented Startup Test Program, or 2) when surveillance for determining power distribution maps is being performed.

a. If the absolute value of $\frac{R_{ij}-\overline{R}_j}{\overline{R}_j}$ is greater than $2\sigma_j$, another map shall be completed to verify the new $\overline{R}_j$. If the second map shows the first to be in error, the first map shall be disregarded. If the second map confirms the new $\overline{R}_j$, four more maps (including rodded configurations allowed by the insertion limits) will be completed so that a new $\overline{R}_j$ and σ_j can be defined from the six new maps.

4.3.3.6.2 The APDMS shall be demonstrated OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST within 7 days prior to its use and at least once per 31 days thereafter when used for monitoring $F_j(Z)$.

b. At least once per 18 months, during shutdown or below 5% of RATED THERMAL POWER, by performance of a CHANNEL CALIBRATION.

SURVEILLANCE REQUIREMENTS (Continued)

f. By performing a flow balance test during shutdown following completion of modifications to the ECCS subsystem that alter the subsystem flow characteristics and verifying the following flow rates:

Boron Injection System Single Pump*	Safety Injection System Single Pump**
Loop 1 Boron Injection Flow 117.5 gpm	Loop 1 and 4 Cold Leg Flow $\geq$ 300 gpm
Loop 2 Boron Injection Flow 117.5 gpm	Loop 2 and 3 Cold Leg Flow $\geq$ 300 gpm
Loop 3 Boron Injection Flow 117.5 gpm	
Loop 4 Boron Injection Flow 117.5 gpm	

*The flow rate in each Boron Injection (BI) line should be adjusted to provide 117.5 gpm (nominal) flow into each loop. Under these conditions there is zero miniflow and 80 gpm simulated RCP seal injection line flow. The actual flow in each BI line may deviate from the nominal so long as the difference between the highest and lowest flow is 10 gpm or less and the total flow to the four branch lines does not exceed 470 gpm. Minimum flow (total flow) required is 345.8 gpm to the three most conservative (lowest flow) branch lines.

**Total SIS (single pump) flow, including miniflow, shall not exceed 650 gpm.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown, by:

   1. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

   2. Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure -- High-High signal.

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

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE:

a. At least once per 92 days by cycling each OPERABLE power operated or automatic valve testable during plant operation through at least one complete cycle of full travel.

b. Immediately prior to returning the valve to service after maintenance, repair or replacement work is performed on the

CONTAINMENT AIR RECIRCULATION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.5.6 Two independent containment air recirculation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment air recirculation system inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.6 Each containment air recirculation system shall be demonstrated OPERABLE at least once per 3 months on a STAGGERED TEST BASIS by:

a. Verifying that the return air fan starts on an auto-start signal after a 10 $\pm$ 1 minute delay and operates for at least 15 minutes,

b. Verifying that with the return air fan discharge backdraft damper locked closed and the fan motor energized, the static pressure between the fan discharge and the backdraft damper is $\geq$ 4.0 inches, water gauge.

c. Verifying that with the fan off, the return air fan damper opens when a force of $\leq$ 11 lbs is applied to the counter-weight, and

d. Verifying that the motor operated valve in the suction line to the containment's lower compartment opens after a 9 $\pm$ 1 minute delay.

FLOOR DRAINS

LIMITING CONDITION FOR OPERATION

3.6.5.7 The ice condenser floor drains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the ice condenser floor drain inoperable, restore the floor drain to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.7 Each ice condenser floor drain shall be demonstrated OPERABLE at least once per 18 months during shutdown by:

a. Verifying that valve gate opening is not impaired by ice, frost or debris,

b. Verifying that the valve seat is not damaged,

c. Verifying that the valve gate opens when a force of $\leq$ 100 lbs is applied, and

d. Verifying that the 12 inch drain line from the ice condenser floor to the containment lower compartment is unrestricted.

COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8 At least one residual heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 A residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of $\geq$ 3000 gpm at least once per 24 hours.

REFUELING OPERATIONS

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS- $F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than $\pm$ 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.51/1.08$. The 8% allowance is based on the following considerations:

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitations provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 52 GPM. This limitation is based on the maximum seal injection flow capability of the Reactor Coolant Pumps and ensures a maximum safety injection flow assumed in the accident analysis.

COMPOSITION

6.5.2.2 The NSDRC shall be composed of the:

Chairman:	Assistant Vice President, Nuclear Engineering
Member:	Vice Chairman, Engineering and Construction
Member:	President and Chief Operating Officer of I&MECo
Member:	Executive Vice President, Construction and New York Engineering
Member:	Vice President, Mechanical Engineering
Member:	Vice President, Electrical Engineering
Member:	Vice President, Engineering Administration
Member:	Assistant Vice President, Design Division
Member:	Assistant Vice President, Environmental Engineering Division
Member:	Plant Manager, D. C. Cook Plant
Member:	Manager, Nuclear Safety and Licensing Section
Alternate:	Assistant Chief Mechanical Engineer
Alternate:	Assistant Plant Manager, D. C. Cook Plant
Alternate:	Executive Assistant to the President of I&MECo
Alternate:	Assistant Division Manager, Nuclear Engineering

ALTERNATES

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

CONSULTANTS

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

MEETING FREQUENCY

6.5.2.5 The NSDRC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of NSDRC shall consist of the Chairman or his designated alternate and more than half the NSDRC membership including alternates or at least 5 members including alternates whichever is greater. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The NSDRC shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, 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 Section 50.59, 10 CFR.

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

d. Proposed changes in Technical Specifications or licenses.

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

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

g. REPORTABLE OCCURRENCES requiring 24 hour notification to the Commission.

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

i. Reports and meetings minutes of the PNSRC.

6.9 REPORTING REQUIREMENTS (Continued)

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 OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to the procedures required by Specification 6.8.1.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all sealed source material of 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 5.9-1.

f. Records of training and qualification for current members of the plant staff.

g. Records of in-service inspections performed pursuant to these Technical Specifications.

h. Records of Quality Assurance activities required by the QA Manual.

i. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

j. Records of meetings of the PNSRC and the NSDRC.

k. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

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.

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. 45
License No. DPR-74

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

   A. The applications for amendment by Indiana and Michigan Electric Company (the licensee) dated December 22, 1978, February 22, 1980 and May 26, 1981 comply 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 applications, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 4, 1982

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3391 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{avg} > 200°F$

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 $\geq$ 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 3.1.3.5.

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 3.1.3.5.

*See Special Test Exception 3.10.1

#With $K_{eff} \geq 1.0$

##With $K_{eff} < 1.0$

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure.

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

INSTRUMENTATION

SEISMIC INSTRUMENTATION*

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With the number of OPERABLE seismic monitoring instruments less than required by Table 3.3-7, restore the inoperable instrument(s) to OPERABLE status within 30 days.

b. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status and a CHANNEL CALIBRATION performed within 24 hours following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

*Shared System with D. C. Cook Unit 1.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. STRONG MOTION TRIAXIAL ACCELEROGRAPHS		
a. Reactor Pit FLoor	0-1 g	1
b. Top of Crane Wall	0-1 g	1
c. Free Field	0-1 g	1
2. PEAK RECORDING ACCELEROGRAPHS		
a. Containment Spring Line	0-2 g	1
b. Diesel Generator Room Floor	0-2 g	1
c. Spent Fuel Pool	0-2 g	1

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

AXIAL POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.7 The axial power distribution monitoring system (APDMS) shall be OPERABLE with:

a. At least two detector thimbles available for which $\bar{R}$ has been determined from full incore flux maps. These two thimbles shall be those having the lowest uncertainty, σ, covering the full configuration of permissible rod patterns permitted at RATED THERMAL POWER.

b. At least two movable detectors, with associated devices and readout equipment, available for mapping $F_j(Z)$ in the above required thimbles.

APPLICABILITY: When the APDMS is used for monitoring the axial power distribution*#.

ACTION: With the APDMS inoperable, do not use the system for determining the Axial Power Distribution. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 The full incore flux maps used to determine $\bar{R}$ and for monitoring $F_j(Z)$ shall be updated at least once per 31 EFPD. The continued accuracy and representativeness of the selected thimbles shall be verified by using their latest flux maps to update the $\bar{R}$ for each representative thimble. The original uncertainty, σ, shall not be updated, except as follows:

*Except as provided in Specification 4.2.6.1.b.
#The APDMS may be out of service when surveillance for determining power distribution maps is being performed.

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4.1 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one stop valve or one control valve per high pressure turbine steam lead inoperable or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam lead inoperable, operation may continue for up to 72 hours provided the inoperable valve(s) is restored to OPERABLE status or at least one valve in the affected steam lead is closed; otherwise, isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours either restore the system to OPERABLE status or isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.1.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE.

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position.

1. Four high pressure turbine stop valves.

2. Four high pressure turbine control valves.

3. Six low pressure turbine reheat stop valves.

4. Six low pressure turbine reheat intercept valves.

LIMITING CONDITION FOR OPERATION

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position.

c. At least once per 18 months by performance of a CHANNEL CALIBRATION on the turbine overspeed protection systems.

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS

NORMAL OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6.*

ACTION:

Above P-7, comply with either of the following ACTIONS:

a. With one reactor coolant loop and associated pump not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to less than 31% of RATED THERMAL POWER and the following ESF instrumentation channels associated with the loop not in operation, are placed in their tripped condition within 1 hour:

   1. T_{avg} -- Low-Low channel used in the coincidence circuit with Steam Flow - High for Safety Injection.

   2. Steam Line Pressure - Low channel used in the coincidence circuit with Steam Flow - High for Safety Injection.

   3. Steam Flow-High Channel used for Safety Injection.

   4. Differential Pressure Between Steam Lines - High channel used for Safety Injection (trip all bistables which indicate low active loop steam pressure with respect to the idle loop steam pressure).

b. With one reactor coolant loop and associated pump not in operation, subsequent STARTUP and POWER OPERATION above 31% of RATED THERMAL POWER may proceed provided:

   1. The following actions have been completed with the reactor in at least HOT STANDBY:

      a) Reduce the overtemperature ΔT trip setpoint to the value specified in Specification 2.2.1 for 3 loop operation.

*See Special Test Exception 3.10.4.

ACTION (Continued)

b) Place the following reactor trip system and ESF instrumentation channels, associated with the loop not in operation, in their tripped conditions:

1) Overpower ΔT channel.

2) Overtemperature ΔT channel.

3) T_{avg} -- Low-Low channel used in the coincidence circuit with Steam Flow - High for Safety Injection.

4) Steam Line Pressure - Low channel used in the coincidence circuit with Steam Flow - High for Safety Injection.

5) Steam Flow-High channel used for Safety Injection.

6) Differential Pressure Between Steam Lines - High channel used for Safety Injection (trip all bistables which indicate low active loop steam pressure with respect to the idle loop steam pressure).

c) Change the P-8 interlock setpoint from the value specified in Table 3.3-1 to $\leq$ 76% of RATED THERMAL POWER.

2. THERMAL POWER is restricted to $\leq$ 71% of RATED THERMAL POWER.

Below P-7:

a. With $K_{eff} \geq 1.0$, operation may proceed provided at least two reactor coolant loops and associated pumps are in operation.

b. With $K_{eff} < 1.0$, operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant or residual heat removal pump.*

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

---

* All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

a) Centrifugal charging pump

b) Safety injection pump

c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated discharge pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1. Centrifugal charging pump $\geq$ 2405 psig

2. Safety Injection pump $\geq$ 1445 psig

3. Residual heat removal pump $\geq$ 195 psig

g. By verifying the correct position of each mechanical stop for the the following Emergency Core Cooling System throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with:

a. A spray additive tank containing a volume of between 4000 and 4600 gallons of between 30 and 34 percent by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the spray additive system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the spray additive system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

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

b. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the NaOH solution by chemical analysis.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

CONTAINMENT SYSTEMS

DIVIDER BARRIER PERSONNEL ACCESS DOORS AND EQUIPMENT HATCHES

LIMTING CONDITION FOR OPERATION

3.6.5.5 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With a personnel access door or equipment hatch inoperable or open except for personnel transit entry and T_{avg} > 200°F, restore the door or hatch to OPERABLE status or to its closed position (as applicable) within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.5.1 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be determined closed by a visual inspection prior to increasing the Reactor Coolant System T_{avg} above 200°F and after each personnel transit entry when the Reactor Coolant System T_{avg} is above 200°F.

4.6.5.5.2 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be determined OPERABLE by visually inspecting the seals and sealing surfaces of these penetrations and verifying no detrimental misalignments, cracks or defects in the sealing surfaces, or apparent deterioration of the seal material:

a. Prior to final closure of the penetration each time it has been opened, and

b. At least once per 10 years for penetrations containing seals fabricated from resilient materials.

CONTAINMENT AIR RECIRCULATION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.5.6 Two independent containment air recirculation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment air recirculation system inoperable, restore the inoperable system to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.6 Each containment air recirculation system shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by:

a. Verifying that the return air fan starts on an auto-start signal after a 9 ± 1 minute delay and operates for at least 15 minutes.

b. Verifying that with the return air fan discharge backdraft damper locked closed and the fan motor energized, the static pressure between the fan discharge and the backdraft damper is $\geq$ 4.0 inches, water gauge.

c. Verifying that with the fan off, the return air fan damper opens when a force of $\leq$ 11 lbs is applied to the counterweight.

d. Verifying that the motor operated valve in the suction line to the containment's lower compartment opens after a 9 ± 1 minute delay.

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least, 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on Figures 3.2-3 and 3.2-4, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3 and 3.2-4. Measurement errors of 3.5% for RCS total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and normalizing its respective output.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

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

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

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

3/4.3.3.7 AXIAL POWER DISTRIBUTION MONITIORING SYSTEM (APDMS)

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

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain calculated DNBR above the design DNBR value during Condition I and II events. With one reactor coolant loop not in operation, THERMAL POWER is restricted to < 51 percent of RATED THERMAL POWER until the Overtemperature ΔT trip is reset. Either action ensures that the calculated DNBR will be maintained above the design DNBR value. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (31 percent of RATED THERMAL POWER).

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing a RHR loop into operation in the shutdown cooling mode if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

QUORUM

6.5.2.6 A quorum of NSDRC shall consist of the Chairman or his designated alternate and more than half the NSDRC membership including alternates or at least 5 members including alternates whichever is greater. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The NSDRC shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, 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 Section 50.59, 10 CFR.

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

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 from normal and expected performance of plant equipment that affect nuclear safety.

g. Events requiring 24 hour written notification to the Commission.

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

i. Reports and meetings minutes of the PNSRC.

6.10 RECORD RETENTION

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

a. Records and logs of facility 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 OCCURRENCES 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 radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

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

a. Records and drawing changes reflecting facility 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 of 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 reviews 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.

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.

*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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 45 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 letters dated December 22, 1978, February 13, 1979, February 22, 1980 and May 26, 1981, Indiana and Michigan Electric Company (the licensee) proposed changes to the Technical Specifications (TSs) appended to Facility Operating License Nos. DPR-58 and DPR-74 for Donald C. Cook Nuclear Plant Unit Nos. 1 & 2. The proposed changes consist of (1) editorial changes to correct inconsistencies, errors and obsolete position titles and (2) changes that involve a safety consideration. The acceptability of the changes that involve a safety consideration are discussed below.

DISCUSSION

1. Boron Injection System and Safety Injection Systems Surveillance Requirements

The change proposed by the licensee is a revision to the boron injection and safety injection system surveillance requirements of Section 4.5.2. The proposed minimum and maximum flow rates are lower than the existing Technical Specifications requirements.

The surveillance requirements were implemented to maintain ECCS flow within acceptable limits in order to: (1) prevent total pump flow from exceeding runout conditions, and (2) ensure that the minimum delivered ECCS flow is in accordance with the assumptions of the safety analysis.

(a) Boron Injection System

The maximum four line flow allowed by the proposed Technical Specifications for a single charging pump is 470 gpm, which includes provision for 80 gpm simulated RCP seal injection line flow without exceeding pump runout flow of 550 gpm. The previous limit was 540 gpm, so this change provides additional margin to pump runout.

The proposed Technical Specifications also requires a minimum flow of 345.8 gpm for three lines, which is a reduction from the 405 gpm required by present Technical Specifications. The safety analysis in the FSAR for the reactor coolant pipe break assumes only one charging pump is operable due to a single failure of a diesel. The minimum boron injection capability is therefore one charging pump delivering flow to three cold leg injection points. The fourth line spills to reactor coolant back-pressure due to the cold leg pipe rupture. The safety analysis flow rate as shown by figures in Section 14.3.2 of the FSAR is less than 34.5 gpm, so the Technical Specifications minimum requirements will ensure the safety analysis assumptions are satisfied.

(b) Safety Injection System

The minimum safety injection flow Technical Specifications requirement is presently 325 ± 3 gpm to each hot leg injection header, which is a total single pump flow to all loops of 650 gpm, the pump runout flow. The proposed Technical Specifications requires that the maximum single pump safety injection flow, including mini-flow, be limited to 650 gpm. This will prevent pump runout.

The proposed Technical Specifications requires a minimum flow of 300 gpm to each cold leg header (each header serves two injection lines) for a single pump.

The loss-of-coolant analysis in the FSAR assumes one safety injection pump is operable, with the flow from one cold leg header spilling to reactor coolant backpressure. The minimum flow rate assumed is less than 300 gpm, so the safety analysis assumptions are ensured by satisfying the Technical Specifications.

Since the proposed Technical Specifications provide margin to the pump runout flows, and include minimum flow rate requirements which will ensure that the flows assumed in the safety analysis are delivered by the boron injection and safety injection systems, the intent of the surveillance requirements is satisfied, and the Technical Specifications changes are acceptable.

2. Spray Additive System

The proposed revision to the Spray Additive System surveillance requirements would require that the flow rate from the Spray Additive Tank to each Containment Spray system be between 20 and 50 gpm.

For Cook Unit No. 1, this places an upper bound on the existing Technical Specifications which now require only that the flow be at least 20 gpm. For Cook Unit No. 2, a 20 gpm minimum at the drain connections has been replaced by the 20-50 gpm range described above.

Analyses have been performed to show that for flow rates between 20 and 50 gpm, the pH of the spray solution will be within the range assumed in the safety analysis.

The proposed Technical Specifications change imposes additional constraints on existing surveillance requirements, and thus does not reduce the safety margin. Therefore, since the required flows are consistent with the safety analysis, the change is considered acceptable.

3. Incore Flux Maps

The revision proposed by the licensee would change the requirements to update the incore flux maps used for R and Fj(z) determination from once every 31 days to once every 31 Effective Full Power Days (EFPD).

At regular effective full power monthly intervals (31 EFPD), power distribution maps are taken to verify that the core hot channel factors (e.g. F_Q) are within prescribed limits and to update the target axial flux difference. A 31 EFPD interval for APDMS file updating will allow these same flux maps to be used to update the Basic Data file as described in WCAP-8589, August 1975. This method will simplify operation.

Since the radial (R) and axial (Fj(z)) components of the hot channel factors depend on the flux, which is burnup dependent, a burnup-dependent surveillance interval will provide more meaningful information on changes in peaking factors.

Since the plant is typically operated as close to full power as possible, the difference between 31 calendar days and 31 EFPD will usually be small. In general, R decreases with burnup, so a slightly longer time between updates of the data file will not decrease the safety margin.

On this basis, the proposed change is considered acceptable.

4. Reactor System Interlock P-8 Rated Thermal Power for N-1 Loop Operation

The change proposed by the licensee makes the Reactor System Interlock P-8 Rated Thermal Power for N-1 loop operation 31% power. Presently, 46% power and 51% power are given in different places in the plant Technical Specifications.

The 31% power level is lower than the existing Technical Specifications limits, and thus conservative, and is consistent with the safety analysis for D. C. Cook 2. Therefore, this change is considered acceptable.

5. Containment Air Return Fan Surveillance Testing

The licensee proposed certain changes to Technical Specifications 4.6.5.6.a and 4.6.5.6.d concerning the delay times for containment return air fan auto-start and the opening of the valve in the suction line to the containment lower compartment (for hydrogen recirculation). The licensee proposes to change these delay times, which are measured from the time the auto-start signals occur until the components begin operation. The current values are 10 ± 1 minutes for both functions for Unit 1, and 10 + 0/-1 minutes for both functions for Unit 2. The licensee proposes to change these to a uniform value of 9 ± 1 minutes. The licensee has stated that a minumum value of seven minutes was used in the facility safety analysis for the fan auto-start delay time. The present hydrogen control analysis for Unit 2 assumes a maximum, fan auto-start delay time of ten minutes.

The minimum delay time for return air fan auto-start is set by core reflood considerations. If the fans start too soon, air pressure in the containment lower compartment will be reduced before core reflooding is complete, which will prolong the completion of reflood. Also, the return air fan, in conjunction with the suction line, serves to assure mixing of the hydrogen in the post-accident atmosphere, and must, according to the analysis, start with no more than a ten minute delay. In addition, the return air fans and suction line valves must have the same delay time in order to function properly. Therefore, it is appropriate to have a delay time between seven and ten minutes, and the proposed delay time of 9 ± 1 minutes is conservative and acceptable.

Fan operability is currently based on the measurement of fan motor current and fan speed; the licensee proposed to change the range of acceptable motor current for demonstrating fan operability. However, subsequent to its letter of May 26, 1981, the licensee proposed a different approach to demonstrating fan operability. The revised approach would require that, with the fan motor energized, the static pressure between the fan discharge and the closed backdraft damper should be equal to or greater than 4.0 inches (water gauge).

The staff concurs with the licensee's view that this revised approach provides for a more direct demonstration of the fan operability, and, therefore, is a better test. The staff concludes, therefore, that the proposed change to the Technical Specifications is acceptable.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and enviornmental impact appraisal need not be prepared in connection with the issuance of these amendments.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: October 4, 1982

Principal Reviewers:
R. Cilimberg
S. Miner
F. Allenspach
M. Virgilio

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-315 AND 50-316

INDIANA AND MICHIGAN ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 63 to Facility Operating License No. DPR-58, and Amendment No. 45 to Facility Operating License No. DPR-74 issued to Indiana and Michigan Electric Company (the licensee), which revised Technical Specificaitons for operation of Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 (the facilities) located in Berrien County, Michigan. The amendments are effective as of the date of issuance.

These amendments revises the Technical Specifications to (1) change the surveillance requirements for the Boron Injection System, the Safety Injection, the Containment Return Air Fan and spray additive system, (2) change the requirements for updating the core flux mapping, (3) revises the Reactor System Interlock P-8 Rated Thermal Power for N-1 loop operation for Unit No. 2, (4) administrative revises a number of technical specifications to correct editorial errors, (5) administratively deletes the special operability requirements for certain cabinets, motor control centers and switchgear since the permanent seismic supports have been installed and (6) administratively adds surveillance requirements and limited conditions for operation (LCO) for seismic monitoring instruments in Unit No. 2. These instruments are shared between Unit No. 1 and Unit No. 2 and the surveillance requirements and LCO's are currently only covered in the Unit No. 1 Technical Specification.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the applications for amendments dated December 22, 1978, February 13, 1979, February 22, 1980 and May 26, 1981. (2) Amendment Nos. 63 and 45 to License Nos. DPR-58 and DPR-74, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Maude Reston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 4th day of October 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing