Docket No. 50-315 ✓ JAN 4 1978

JMcGough
KKniel
MMlynczak
WButler
GZwetzig
JWetmore
RPSnaider
JRBuchanan
TBAbernathy
ACRS (16)

DISTRIBUTION:
Docket
NRC PDR
Local PDR
ORB-2 Reading
VStello
DEisenhut
KRGoller
TJCarter
DKDavis
RMDiggs
MFletcher
Attorney, OELD
BScharf (10)
BJones (4)
CMiles, OPA
DRoss
BHarless

Indiana & Michigan Electric Company
Indiana & Michigan Power Company
ATTN: Mr. John Tillinghast
Vice President
P. O. Box 18
Bowling Green Station
New York, New York 10004

Gentlemen:

In response to your requests dated June ~~22~~ 18, 1976, and February 9, July 13, September 15 and 22, October 13, and November 3, 1977, the Commission has issued the enclosed Amendment No. 23 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant Unit No. 1.

The amendment involves Technical Specification modifications to incorporate requirements for use of the Cask Drop Protection System, to add surveillance requirements for certain ECCS throttle valves, to change the maximum specified pressurizer heatup rate, to exclude certain large hydraulic shock suppressors from functional testing, and to incorporate NRC recommended changes into the Administrative Controls section. A clarification of the requirements for the emergency diesel generator load sequencing times is also included in the amendment.

This amendment concludes our review of the Cask Drop Protection System for the Donald C. Cook Nuclear Plant for fuel shipping casks weighing up to 110 tons.

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

Sincerely,

BCB 12/15/77

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Const 1

DOR:STS	DOR:TA/ORs	PSB			DOR:ORB-2
JMcGough	JCarter	WButler	GZwetzig	JWetmore	RPSnaider
12/13/77	~~12/~~ 1/3/78	12/15/77	12/13/77	12/13/77	12/8/77

OFFICE ➤	DOR:ORB-2	DOR:ORB-2	DPM:LWR-2	DPM:LWR-2	OELD	DOR:ORB-2
SURNAME ➤	RMDiggs	MFletcher/esp	MMlynczak	KKniel	R Black	DKDavis
DATE ➤	12/8/77	12/6/77	12/5/77	12/6/77	12/12/77	12/3/77

Enclosures:
1. Amendment No. 23 to
   License No. DPR-58
2. Safety Evaluation
3. Notice

cc w/enclosures:
Mr. Robert Hunter
Vice President
American Electric Power Service
  Corporation
2 Broadway
New York, New York 10004

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

Donald Cook Nuclear Plant
ATTN: Mr. D. Shaller
Plant Manager
P. O. Box 458
Bridgman, Michigan 49106

David Dinsmore Comey
Executive Director
Citizens for a Better Environment
59 East Van Buren Street
Chicago, Illinois 60605

Maude Reston Palenske Memorial Library
500 Market Street
St. Joseph, Michigan 49085

cc w/enclosures and cy of
  I&MECo filings referenced
  in first paragraph of
  letter:
Mr. William R. Rustem (2)
Office of the Governor
Room 1 - Capitol Building
Lansing, Michigan 48913

cc w/enclosures:
Mr. Wade Schuler, Supervisor
Lake Township
Baroda, Michigan 49101

Honorable W. Mabry, Mayor
City of Bridgman, Michigan 49106

Chief, Energy Systems Analyses
  Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 Dearborn Street
Chicago, Illinois 60604

OFFICE➤						
SURNAME➤						
DATE➤						

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

INDIANA & MICHIGAN ELECTRIC COMPANY

INDIANA & MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
License No. DPR-58

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

   A. The applications for amendment by Indiana & Michigan Electric Company and Indiana & Michigan Power Company (the licensees) dated June 22, 1976 and February 9, July 13, September 15 and 22, October 13, and November 3, 1977, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 23, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 30 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 4, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 23

FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

IX
XV
XVIII
1-2
3/4 4-30
3/4 5-6
3/4 5-6a (added)
3/4 7-29
3/4 8-4
3/4 9-17 (added)
3/4 9-18 (added)
B 3/4 5-2
B 3/4 9-3
6-14
6-15
6-16
6-17
6-18
6-19
6-20
6-21
6-22 (deleted)

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

SECTION | Page

BASES

INDEX

DESIGN FEATURES

INDEX

ADMINISTRATIVE CONTROLS

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, electric power, 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).

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.8 and 6.9.1.9.

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.

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

SPECIMEN	REMOVAL INTERVAL
Capsule 1	Replacement of 1st region of core
Capsule 2	10 years
Capsule 3	20 years
Capsule 4	30 years
Capsule 5, 6, 7, 8	Extra capsules for complementary duplicate testing

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

SURVEILLANCE REQUIREMENTS (Continued)

Valve Number	Valve Function	Valve Position
d. IMO 262*	d. Mini flow line	d. Open
e. IMO 263*	e. Mini flow line	e. Open
f. IMO 261*	f. SI Suction	f. Open
g. ICM 305*	g. Sump line	g. Closed
h. ICM 306*	h. Sump line	h. Closed

5. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

6. 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. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing containment integrity, and

2. Of the areas affected within containment at the completion of each containment entry when containment integrity is established.

c. 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.

*These valves must change position during the switchover from injection to recirculation flow following LOCA.

SURVEILLANCE REQUIREMENTS (Continued)

d. 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 Safety Injection signal.

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

a) Centrifugal charging pump

b) Safety injection pump

c) Residual heat removal pump

e. By verifying the correct position of each mechanical stop for 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.

2. At least once per 18 months.

Boron Injection Throttle Valves	Safety Injection Throttle Valves
Valve Number	Valve Number
1. 1-SI-141 L1	1. 1-SI-121 N
2. 1-SI-141 L2	2. 1-SI-121 S
3. 1-SI-141 L3	
4. 1-SI-141 L4	

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 135 ± 3 gpm	Loop 1 and 4 Cold Leg Flow 300 ± 3 gpm
Loop 2 Boron Injection Flow 135 ± 3 gpm	Loop 2 and 3 Cold Leg Flow 300 ± 3 gpm
Loop 3 Boron Injection Flow 135 ± 3 gpm	
Loop 4 Boron Injection Flow 135 ± 3 gpm	

HYDRAULIC SNUBBERS (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

4.7.8.1.3 At least once per 18 months during shutdown, a representative sample of at least 10 hydraulic snubbers or at least 10% of all snubbers listed in Table 3.7-4, whichever is less, shall be selected and functionally tested to verify correct piston movement, lock up and bleed. Snubbers greater than 50,000 lb capacity may be excluded from functional testing requirements. Snubbers selected for functional testing shall be selected on a rotating basis. Snubbers identified as either "Especially Difficult to Remove" or in "High Radiation Zones" may be exempted from functional testing provided these snubbers were demonstrated OPERABLE during previous functional tests. Snubbers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each snubber found inoperable during these functional tests, an additional minimum of 10% of all snubbers or 10 snubbers, whichever is less, shall also be functionally tested until no more failures are found or all snubbers have been functionally tested.

TABLE 3.7-4

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
1	Chemical Control.Az 290° Elev. 613'. In Containment	I	Yes	No
2	Reactor Coolant.Az 255°, Elev. 620'. In Containment	I	Yes	No
3	Reactor Coolant.Az 315° Elev. 671'. In Pressurizer Enclosure	I	Yes	Yes
4	Reactor Coolant.Az 320°, Elev. 670'. In Pressurizer Enclosure.	I	Yes	Yes
5	Reactor Coolant.Az 315°, Elev. 681'. In Pressurizer Enclosure.	I	Yes	Yes
6	Reactor Coolant.Az 305°, Elev. 681'. In Pressure Enclosure.	I	Yes	Yes
7	Reactor Coolant.Az 300° Elev. 683'. In Pressurizer Enclosure	I	Yes	Yes
8	Reactor Coolant.Az 305° Elev. 684'. In Pressurizer Enclosure.	I	Yes	Yes
9	Reactor Coolant.Az 310°, Elev. 681'. In Pressurizer Enclosure.	I	Yes	Yes
10	Reactor Coolant.Az 310°, Elev. 686'. In Pressurizer Enclosure.	I	Yes	Yes
11	Reactor Coolant.Az 300°, Elev. 684'. In Pressurizer Enclosure.	I	Yes	Yes

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the fuel level in the fuel storage tank,

3. Verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water and sediment,

4. Verifying the fuel transfer pump can be started from the control panel and transfers fuel from the storage system to the day tank,

5. Verifying the diesel starts from ambient condition,

6. Verifying the generator is synchronized, loaded to $\geq$ 1750 kw, and operates for $\geq$ 60 minutes, and

7. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2. Verifying the generator capability to reject a load of $\geq$ 600 kw without tripping,

3. Simulating a loss of offsite power in conjunction with a safety injection signal, and:

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected emergency loads through the load sequencer and operates for $\geq$ 5 minutes while its generator is loaded with the emergency loads.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying the diesel generator operates for $\geq$ 60 minutes while loaded to $\geq$ 3500 kw.

5. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 3650 kw.

6. Verifying that the automatic sequence timing relays are OPERABLE with each load sequence time within $\pm$ 5% of its required value and that each load is sequenced on within the design allowable time limit.

SPENT FUEL CASK MOVEMENT

LIMITING CONDITION FOR OPERATION

3.9.13 Movement of the spent fuel cask above elevation 620 feet shall be done with the spent fuel cask handling crane operating in the Controlled Path Mode of operation.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.13 Crane interlocks which prevent raising the bottom of the spent fuel cask more than 6 inches above the top of the Cask Drop Protection System cyclinder and restrict the crane's movement to the Controlled Path shall be demonstrated OPERABLE within 7 days prior to crane operation in the Controlled Path Mode and at least once per 7 days thereafter during crane operation in the Controlled Path Mode.

SPENT FUEL CASK DROP PROTECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.14 The maximum weight of a spent fuel cask used with the Cask Drop Protection System shall be limited to 110 tons (nominal).

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.14 The weight of a spent fuel cask shall be verified to be $\leq$ 110 tons (nominal) prior to its use with the Cask Drop Protection System.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each RCS accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

BASES

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 BORON INJECTION SYSTEM

The OPERABILITY of the boron injection system as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam line rupture.

The limits on injection tank minimum volume and boron concentration ensure that the assumptions used in the steam line break analysis are met.

The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 21000 ppm boron.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

3/4.9.12 STORAGE POOL VENTILATION SYSTEM

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

3/4.9.13 SPENT FUEL CASK MOVEMENT

The limitations of this specification ensures that, during insertion or removal of spent fuel casks from the spent fuel pool, fuel cask movement will be constrained to the path and lift height assumed in the Cask Drop Protection System safety analysis. Restricting the spent fuel cask movement within these requirements provides protection for the spent fuel pool and stored fuel from the effects of a fuel cask drop accident.

3/4.9.14 SPENT FUEL CASK DROP PROTECTION SYSTEM

The limitations on the use of spent fuel casks weighing in excess of 110 tons (nominal) provides assurance that the spent fuel pool would not be damaged by a dropped fuel cask since this weight is consistent with the assumptions used in the safety analysis for the performance of the Cask Drop Protection System.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission and to the Chairman of the NSDRC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PNSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the Chairman of the NSDRC and the Vice President, Nuclear Engineering within 10 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the PNSRC and approved by the Plant Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

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 AND REPORTABLE OCCURRENCES

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 Director of the Regional Office of Inspection and Enforcement 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).

[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.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, 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.

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.8 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

g. Conditions arising from natural or man-made events that, as a direct result of the event require unit shutdown, operation of safety systems, or other protective measures required by technical specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specification bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

THIRTY DAY WRITTEN REPORTS

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above designed to contain radioactive material resulting from the fission process.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office 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. 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.

6.9 REPORTING REQUIREMENTS (Continued)

e. Seismic event analysis, Specification 4.3.3.3.2.

f. Sealed Source leakage on excess of limits, Specification 4.7.9.1.3.

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 facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

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 reviews of tests and experiments pursuant to 10 CFR 50.59.

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

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 6.13.1.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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING APPROVAL OF FACILITY MODIFICATIONS TO REDUCE THE CONSEQUENCES OF A FUEL CASK DROP ACCIDENT TO AN ACCEPTABLY LOW LEVEL

AND

AMENDMENT NO. 23 TO LICENSE NO. DPR-58

INDIANA AND MICHIGAN ELECTRIC COMPANY

INDIANA AND MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-315

INTRODUCTION

This Safety Evaluation supports NRC staff acceptance of the D. C. Cook Nuclear Plant refueling cask drop protection system and five areas of modification to the Appendix A Technical Specifications for License No. DPR-58 for D. C. Cook Unit No. 1. The Technical Specification modifications involve incorporating new requirements for the use of the refueling cask drop protection system, adding new requirements for the surveillance of certain emergency core cooling system (ECCS) valve throttle settings, changing the specified maximum pressurizer heatup rate, excluding certain large hydraulic shock suppressors from functional testing, and incorporating NRC staff recommended changes into the Administrative Controls section of the Technical Specifications. Each of these modifications is addressed in a separate section of this evaluation.

In addition to these modifications, a minor administrative change was incorporated into the Technical Specification requirements for emergency diesel generator load sequencing times. The change, which was agreed to by the licensee, does not alter the intent of the present requirements and results in a clarification of the meaning of this Technical Specification. Therefore, we find it acceptable.

CASK DROP PROTECTION SYSTEM

Background

By letters dated June 22, 1976 and February 9 and July 13, 1977, Indiana & Michigan Electric Company and Indiana & Michigan Power Company (the licensee) submitted Amendments 70, 73 and 76 to the Final Safety Analysis Report (FSAR) for the Donald C. Cook Nuclear Plant Unit Nos. 1 and 2. The amendments provided a description and analysis of the MPR Associates Cask Drop Protection System (CDPS) which will be installed at the D. C. Cook Plant to mitigate the consequences of dropping a spent fuel shipping cask during its insertion or removal from the plant fuel pool. The use of the CDPS represents a modification to the licensee's previous commitment, described in the Commission's Safety Evaluation of the Donald C. Cook Nuclear Plant Units 1 and 2 of September 10, 1973, to install a metallic honey-comb energy absorber on the floor of the spent fuel pool.

Discussion/Evaluation

At the Donald C. Cook Nuclear Plant, the handling of nuclear fuel would require that heavy spent fuel shipping casks be lowered into and raised from the spent fuel storage pool by an overhead crane. A potential problem with this operation is that the cask could be dropped into the pool causing damage to the pool or to the spent fuel stored in the pool. The licensee has considered a number of cask drop protection concepts, and has concluded that the Cask Drop Protection System "dashpot" developed by MPR Associates provides more effective protection for the spent fuel storage pool and the stored fuel assemblies. This "dashpot" system is essentially the same as those installed at Oyster Creek and Nine Mile Point, Unit 1 and that being fabricated for the FitzPatrick Nuclear Power Plant. The dashpot is a passive system, i.e., no moving parts, that guides and restrains a falling cask and limits the resulting impact loads imposed on the spent fuel pool floor to acceptable values. The "dashpot" consists of a cylindrical structure located and anchored to the southeast corner of the spent fuel pool. The top surface of the structure is essentially flush with the top of the pool wall (Elevation 650'-0"). The dashpot piston consists of a circular plate to which the cask will be attached before the combined assembly is lowered into the pool. Since the "dashpot" does not extend above the top of the pool wall, it will not interfere with the handling of the cask or other equipment in the Auxiliary Building at floor elevation 650'-0". The cylinder, fabricated from stainless steel, is tapered so that the flow area of the annular orifice, formed by it and the circular plate attached to the bottom of the shipping cask, decreases as the cask approaches the floor of the spent fuel storage pool. The assembly acts as a dashpot in that the water, entrapped within the cylinder and circular plate attached to the bottom of the shipping cask, is required to pass through the annular orifice as the cask travels the length of the cylinder. The magnitude of the differential pressure, developed across the orifice, is a function of the velocity of the cask and the flow area of the annular orifice. The differential pressure acts on the circular plate to limit the descent velocity of the cask.

We have reviewed the analyses provided by the licensee to show that the worst case dynamic load on the fuel pool floor resulting from a dropped cask is within the structural capability of the fuel pool. The "dashpot" is designed to accommodate the drop of a 110-ton cask and to limit its terminal velocity to 0.7 feet per second. For casks of lesser weight, i.e., the 70 or 25 ton casks, the terminal velocity will be less than 0.7 feet per second. Sections of horizontal stainless steel pipes are placed at the bottom of the cylinder to absorb the kinetic energy associated with the terminal velocity by plastic deformation. The resulting maximum dynamic load on the pool floor will be 1,610,000 pounds while the maximum allowable dynamic load is 3,375,000 pounds. We find this to be acceptable.

To prevent damage to the fuel stored in the fuel pool in the event of a cask drop, measures have been taken to preclude the tipping of a dropped cask onto the stored fuel. These measures are:

(a) the carrying height of the cask will be limited to less than six inches above the top of the cylinder;

(b) the structural design of the cylinder and the approach path are such that if the cask drop were to occur at any point along the travel path, the cask will enter the cylinder, be tipped away from the stored fuel and toward the walls of the spent fuel pool, or will remain upright on the cylinder top plate.

In addition, if a cask drop into the CDPS cylinder were to occur, the water passing through the annular orifice in the CDPS cylinder would be directed upward and would not damage or disturb the spent fuel elements stored in the pool.

We have reviewed the control system for positively controlling the fuel cask during its travel within the controlled transfer path and limiting its height to acceptable limits. The system consists of a mode selector switch on the crane. When the switch is placed in the "controlled transfer path" position, limit switches will prevent the bridge, trolley, and hoist from traveling outside the acceptable path. To permit the crane from traveling beyond this path (such as when the crane is handling other loads) the mode switch will be placed in "bypass". The transfer path was selected so that, during lateral movement of the cask to the CDPS cylinder, the center of gravity of the cask is always in a position so that if the cask was dropped either (1) there would be no tendency for the cask to tip, or (2) the cask would be tipped in a direction away from the pool, or (3) the cask would enter the CDPS cylinder. Based on our review, we have found the crane transfer path control system to be acceptable.

We have reviewed, in accordance with the criteria described in Sections 3.7 and 3.8 of the Standard Review Plan, the supporting arrangements for the CDPS including (1) restraints, design, fabrication, installation procedures, (2) the structural analysis for all loads including seismic and impact loadings, (3) load combinations, (4) structural acceptance criteria, and (5) applicable industry codes. Analysis of the elastic dynamic response of the CDPS was accomplished using response spectral curves for the Donald C. Cook auxiliary building and a one degree of freedom analytical model. The behavior of the water external to and inside the cask drop protection system was examined to establish the fluid forces inside and outside of the CPDS cylinder. The use of 300 series stainless steel materials for the fabrication of the CDPS structure and its performance requirements during the service life were reviewed for consistency with the requirements identified in Section 9.1.2 of the Standard Review Plan.

We have concluded that the analysis, design, fabrication, and installation of the CDPS is in accordance with accepted criteria for seismic Category I equipment. Therefore, we find that the proposed system is acceptable.

In the course of our review, we have determined that additional technical specification requirements are necessary for the use of the CDPS. These specifications require (1) testing of the crane interlocks for the controlled transfer path prior to and during fuel cask transfers, (2) limiting the maximum weight of a fuel cask used with the CDPS to a nominal 110 ton, and (3) placing the crane in the "controlled path" mode when handling fuel casks above the 620' level of the auxiliary building. These Technical Specification requirements have been discussed with and agreed to by the licensee.

Based on our review of the information submitted by the licensee and implementation of the above-described technical specification requirements, we have concluded that the CDPS together with the fuel cask transfer path control system provides adequate protection from the effects of a fuel cask drop accident for the spent fuel pool and stored fuel in the pool and is therefore acceptable.

ECCS THROTTLE VALVE SETTINGS

Background

A large number of pressurized water reactor High Pressure and Low Pressure Safety Injection Systems (HPSI and LPSI) utilize a common low pressure and a common high pressure header to feed the several cold (and in some cases hot) leg injection points. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration; (2) provide a proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA* analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. Many plants have either manual or motor operated valve(s) in the lines to each injection point that have electrical or mechanical stops which have been adjusted during preoperational testing of the plant to ensure that these flow requirements are satisfied.

Discussion/Evaluation

In a letter dated July 18, 1977, the NRC requested the licensee to determine if throttle valves were used to obtain the required flow distribution in the HPSI or LPSI ECCS. We requested that, if throttle valves were used, the licensee propose Technical Specification changes to incorporate surveillance requirements for the verification of throttle valve position. By letter dated September 22, 1977, the licensee proposed Technical Specifications which require the periodic (at least once per 18 months) and post-maintenance verification of the throttle position for six ECCS related valves and flow balance testing following ECCS modifications that alter system flow characteristics.

We have reviewed the licensee's proposal and concluded that, since the Technical Specification will provide greater assurance that the

*Loss-of-Coolant Accident

ECCS will perform as required and analyzed by ECCS-LOCA calculations, the implementation of the Technical Specifications will improve the level of safety at D. C. Cook Unit No. 1 and is acceptable.

PRESSURIZER HEATUP RATE

Discussion/Evaluation

By letter dated September 15, 1977, the licensee proposed to change the maximum pressurizer heatup rate in the Technical Specifications from 200°F in any one hour period to 100°F in any one hour period. This change had been recommended to the licensee by Westinghouse Electric Corporation, the nuclear steam system supplier for D. C. Cook Unit No. 1, and is based on the Westinghouse Design Transient Document and Pressurizer Equipment Specifications.

We have reviewed the proposed pressurizer heatup rate change and concluded that it corrects an error in the present Technical Specification requirements. Therefore, it represents an improvement in the operating limits which affect pressurizer integrity and is acceptable.

HYDRAULIC SHOCK SUPPRESSORS

Discussion/Evaluation

By letter dated September 15, 1977, the licensee proposed that a statement be included in the Technical Specifications governing hydraulic shock suppressor (snubber) surveillance. The statement would allow the exclusion of snubbers with greater than 50,000 lb. capacity from functional testing.

We have concluded that the proposed Technical Specification is acceptable because our present staff position is that functional testing of such large capacity snubbers is impracticable.

ADMINISTRATIVE CONTROLS CHANGES

Background

In October 1974, we initiated a program to update the reporting requirements specified in Technical Specifications of power reactor licensees. Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications" was the basis for the reporting requirements. Licensees were issued specifications that either refer to Regulatory Guide 1.16 or utilize words from certain sections of Regulatory Guide 1.16. Indiana & Michigan Electric Company and Indiana & Michigan Power Company was one of a few licensees that referenced the Regulatory Guide. By letter dated August 12, 1977, we requested that the licensee propose Technical Specification changes to replace the reference to Regulatory Guide 1.16 with words from the Regulatory Guide that are appropriate and applicable. Also, in a letter dated September 19, 1977, we requested that the licensee (1) delete from the Technical Specifications the requirement for an Annual Operating Report provided that certain information presently reported in the Annual Operating Report continues to be reported and (2) modify the content of the Monthly Operating Report. By letters dated October 13 and November 3, 1977, the licensee proposed the above changes to the reporting requirement Technical Specifications.

On November 29, 1976, the Commission published in the Federal Register an amended Section 20.103 of 10 CFR 20, which became effective on December 29, 1976. One effect of this revision is that in order to receive credit for limiting the inhalation of airborne radioactive material, respiratory protective equipment must be used as stipulated in Regulatory Guide 8.15. Another requirement of the amended regulation is that licensees authorized to make allowance for use of respiratory protective equipment prior to December 29, 1976, must bring the use of their respiratory protective equipment into conformance with Regulatory Guide 8.15 by December 29, 1977. Because the respiratory protective program described in the Administrative Controls section of the D. C. Cook Unit No. 1 Technical Specifications differs from that in Regulatory Guide 8.15, we informed the licensee, by letter dated August 12, 1977, that the respiratory protection program portion of the Technical Specifications would be deleted in a future license amendment.

Evaluation

We have evaluated the proposed changes to the facility reporting requirements. Some modifications to the licensee's proposals were necessary to meet our requirements. The modifications include a requirement to report certain steam generator tube inspection data on an annual basis and deletion of the following notes derived from Regulatory Guide 1.16:

Note: Routine surveillance testing, instrument calibration, or preventive maintenance which require system configurations as described in items a and b need not be reported except where test results themselves reveal a degraded mode as described above.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

The steam generator tube inspection data is required to be reported by section 4.4.5.5 of the D. C. Cook Unit No. 1 Technical Specifications. And, although we agree with the above notes from Regulatory Guide 1.16, we do not believe that sufficient explicative benefit is derived from them to warrant their inclusion in the Technical Specifications. Based on our evaluation of the proposed changes, as modified, we have concluded that they are acceptable because:

(1) deletion of a reference to Regulatory Guide 1.16 allows the use of an improved format for reporting Licensee Event Reports (LERs). The new format will provide better information for our use in evaluating operational events;

(2) the Annual Operating Report in the present Technical Specifications can be deleted and sufficient information will still be available to meet the NRC reporting objectives; and

(3) the modifications to the Monthly Operating Report will provide more timely information for our evaluation.

We have also concluded that the elimination of the respiratory protection section of the D. C. Cook Unit No. 1 Technical Specifications is acceptable because, with the change in the regulations (10 CFR 20.103), there is no longer a need for specific authorization for use of respiratory protection equipment.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does 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 amendment involves 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 environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: January 4, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-315

INDIANA & MICHIGAN ELECTRIC COMPANY

INDIANA & MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 23 to Facility Operating License No. DPR-58, issued to Indiana & Michigan Electric Company and Indiana & Michigan Power Company (the licensees), which revised the Technical Specifications for operation of the Donald C. Cook Nuclear Plant Unit No. 1 (the facility), located in Berrien County, Michigan. The amendment is effective 30 days from the date of its issuance.

The amendment revised the Technical Specifications to incorporate requirements for use of the Cask Drop Protection System, to add surveillance requirements for certain emergency core cooling system throttle valves, to change the maximum specified pressurizer heatup rate, to exclude certain large hydraulic shock suppressors from functional testing, to incorporate NRC recommended changes into the Administrative Controls section of the Technical Specifications relating to reporting requirements, and to clarify the requirements for emergency diesel generator load sequencing timers.

The applications for the amendment 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 amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment 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 this amendment.

For further details with respect to this action, see (1) the June 22, 1976 and February 9, July 13, September 9 and 15, October 13, and November 3, 1977 applications for amendment, (2) Amendment No. 23 to License No. DPR-58, 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 single 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 Operating Reactors.

Dated at Bethesda, Maryland, this 4th day of January, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors